INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1121 **Rev:** 0 **Rev Date:** 2/21/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 008 **System Title:** Pressurizer Vapor Space Accident

Description: . Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident:
Control of PZR level

K/A Number: AA1.06 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 1 **SRO:**

Unit 1 is at 100% power.

A leak occurs on the upper tap of a Pressurizer level transmitter sensing line, causing a PZR steam space leak.

Actual PZR level will ___________ and RCS makeup flow will _________.

A. Drop, Drop

B. Drop, Rise

C. Rise, Drop

D. Rise, Rise

Answer:

C. Rise, Drop

Notes:

"C" is correct since a steam space leak will cause actual level to rise. A leak on the upper tap will cause the differential pressure to decrease on the Pressurizer (PZR) level transmitter. The PZR level transmitter is reverse acting so a lowering DP will cause indicated level to rise. Since indicated level is rising RCS makeup flow will drop as indicated level rises above setpoint.

"A" is incorrect but plausible since the RCS makeup flow trend given is correct. However, actual PZR level will rise with a steam space leak, not drop.

"B" is incorrect, a steam space leak will cause actual level to rise. This answer (drop, rise) has trends which work together if applicant believe the loss of inventory (steam space leak) will cause indicated level to drop and makeup flow will rise to counteract.

"D" is incorrect but plausible since actual level trend is correct but makeup flow trend is wrong.

This is a modified version of QID 371. Added unit condition of 100% power and changed "actual PZR level" to "RCS makeup flow". This made "C" the correct answer (vs. D in 371).

This question matches the K/A since it involves a vapor space accident (upper PZR level transmitter tap) and evaluates applicant's knowledge on how the PZR level control system works in response to the steam leak.

References:

1304.022, Unit 1 Pressurizer Level & Temperature Channel Calibration
STM 1-69, Non-Nuclear Instrumentation System

History:

Modified QID 371 for 2017 re-exam.

QID: 0368 **Rev:** 2 **Rev Date:** 3/29/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 8 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 009 **System Title:** Small Break LOCA

Description: Knowledge of the interrelations between the small break LOCA and the following: S/Gs

K/A Number: EK2.03 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 2 **SRO:**

A reactor trip has occurred from 100% power due to low RCS pressure.

The following conditions have existed for three (3) minutes:

* CETs, 590 degrees F and rising
* RCS pressure, 1700 psig and lowering
* RCPs are running

Which of the following operator actions are required to be performed?

A. Trip all running RCPs.

B. Go to Overheating EOP (1202.004).

C. Verify EFW flow to each Steam Generator is 320 gpm.

D. Verify Reflux Boiling setpoint is selected on both EFIC trains.

Answer:

D. Verify Reflux Boiling setpoint is selected on both EFIC trains.

Notes:

"D" is correct, Core Exit Thermocouple (CET) indication and RCS pressure condition show a loss of subcooling margin. With subcooling margin lost, the Reflux Boiling setpoint is required to be selected per RT-5, Verify Proper EFW Actuation and Control.

"A" is incorrect, plausible since this action would be taken on a loss of subcooling margin but only if less than 2 minutes had expired. Question condition says three minutes have gone by making this choice incorrect.

"B" is incorrect, plausible since this is an entry condition for the Overheating EOP but a loss of subcooling margin exists which has a higher priority than Overheating (1015.043, ANO-1 EOP/AOP User Guide, p. 13).

"C" is incorrect, plausible since (per RT-5) EFW flow is verified to both SGs but the flow value given is similar to but less than the minimum flow rate of greater than or equal to 340 gpm, i.e., flow should be greater than or equal to 340 gpm, not 320.

Added "due to low RCS pressure" in first sentence, changed "RCS temperatures" to "CETs", and revised stem to be current with question development philosophy. Shuffled answer choices so they are short to long.

This matches the K/A since condtions are given for a SBLOCA and the applicant is required to know a related action for the SGs (Reflux Boiling Setpoint selected) due to the loss of SCM from the SBLOCA.

References:

1202.012, Repetitive Tasks, RT-5, Verify Proper EFW Actuation and Control

History:

Direct from regular exambank QID 3030.
Selected for use in 2002 SRO exam.
Modified for use in 2005 RO exam, replacement question. EK1.01
Selected for use in 2017 Re-take exam.

QID: 0491 **Rev:** 1 **Rev Date:** 2/21/17 **Source:** Bank **Originator:** Pullin

TUOI: A1LP-RO-EOP10 **Objective:** 6 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 011 **System Title:** Large Break LOCA

Description: Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions for throttling or stopping HPI.

K/A Number: EA2.11 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.3	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 3 **SRO:**

Given:
* Large break LOCA has occurred
* ESAS has actuated
* A3 bus is locked out and cannot be re-energized
* RCS pressure is 50 psig and stable
* BWST level, 12 ft. and lowering

LPI/HPI flow rates are as follows:

- "B" LPI flow 3150 gpm
- "B" HPI pump total flow is 275 gpm

Which of the following actions are required per ESAS EOP (1202.010) for these conditions?

A. Raise HPI flow on P-36B HPI pump.

B. Secure the P-36B HPI pump.

C. Open Decay Heat supply to Makeup Pump suction CV-1277.

D. Swap to RB sump recirculation using Attachment 1.

Answer:

B. Secure the P-36B HPI pump.

Notes:

"B" is correct, the HPI pump can be secured since LPI flow is greater than 3050 gpm with only one LPI pump (A3 bus is locked out so "A" LPI pump cannot run) per step 14 of 1202.010.

"A" is incorrect, this action would be taken if LPI flow was inadequate (<3050 gpm), but LPI flow is adequate.

"C" is incorrect, plausible if LPI flow was inadequate (or RCS pressure was higer than LPI discharge pressure) and thus HPI pump operation was to continue via "piggyback" mode following transfer to RB sump suction, but LPI flow is adequate.

Answer "D" is incorrect but plausible since for a large break LOCA BWST level would be dropping but transfer to RB recirculation is not performed until BWST level is less than 6 ft. and BWST level is 12 ft.

Rev. 1: added condition of RCS pressure at 50 psig and BWST level of 12 ft. Replaced "C" distractor since validators thought it was not plausible. Added component numbers.

This matches the K/A since condtions are given for a large break LOCA and LPI flow is greater than the value required to allow for stopping HPI.

References:

1202.010, ESAS

History:

New Question on 2013 RO Exam EK2.02
Selected for 2017 RO Re-exam

QID: 0396 **Rev:** 1 **Rev Date:** 2/21/17 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-ICS **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 015 **System Title:** Reactor Coolant Pump (RCP) Malfunctions

Description: Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Consequences of an RCPS failure

K/A Number: AK1.02 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 4 **SRO:**

The plant is steady state at 70% power per Dispatcher request.

Subsequently, the following indications are observed:

* "A" MFW flow 4.7 e 6 lbm/hr
* "B" MFW flow 2.3 e 6 lbm/hr

What event would cause this MFW flow discrepancy?

A. "B" MFW pump trip

B. "A" T cold failed high

C. "D" RCP trip

D. "A" RCP trip

Answer:

D. "A" RCP trip

Notes:

"D" is correct, "A" RCP trip (B loop) will cause ICS to re-ratio FW flow so that the highest MFW flow will be in the loop with the highest RCS flow (A loop, two RCPs running). This is to balance heat input with heat removal. With power unchanged (there would be no ICS runback on an RCP trip since power is less than 75%) 2/3 of the heat is now being removed from the A loop since it has 2/3 of the RCS flow and 1/3 of RCS flow will be going through the B loop so it will receive 1/3 of the total FW flow. Total FW flow will still be equivalent to 75% power.

"A" is incorrect, but plausible since MFW flow would drop to the B loop (until the crosstie valve opened fully) but a MFW pump trip would cause FW flow to decrease to both SGs (ICS runback to 60%) and not be skewed as shown.

"B" is incorrect, this is plausible as MFW would be affected by a Tcold failure but this would cause FW to decrease to the "A" SG to attempt to reduce "A" Tcold temperature, FW flow would also increase to the "B" SG making this choice incorrect.

"C" is incorrect, this is plausible as MFW would re-ratio but a "D" RCP trip (A loop) would cause FW to decrease to the "A" SG and increase to the "B" SG.

Changed order of correct answer so now "D" is correct. Deleted "instrument" from B so they are all about the same length

This question matches the K/A since re-ratioing of MFW is an operational implication, and consequence, of a RCP failure.

References:

STM 1-64, Integrated Control System

History:

New question created for 2001 RO/SRO Exam. AA1.05
Selected for 2013 Exam
Selected for 2017 RO Re-exam.

QID: 0610 **Rev:** 1 **Rev Date:** 3/27/17 **Source:** Bank **Originator:** Cork/Pullin

TUOI: A1LP-RO-MU **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 022 **System Title:** Loss of Reactor Coolant Makeup

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:
Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging

K/A Number: AK3.02 **CFR Reference:** 41.5, 41.10 / 45.6 /45.13

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 5 **SRO:**

Given:

- Plant at 100% power.
- Makeup pump P-36B has tripped.

What is the primary reason the AOP directs isolation of Letdown for this condition?

A. Prevent overheating of the Letdown Demineralizers.

B. Prevent overfilling the Makeup Tank.

C. Maintain RCS inventory.

D. Reduce heat load on Nuclear ICW system.

Answer:

C. Maintain RCS inventory.

Notes:

"C" is correct, RCS inventory is maintained by isolating letdown.

"A" is incorrect, this is plausible since this is the reason for automatic letdown isolation on high Letdown temperature.

"B" is incorrect, this is plausible since Makeup Tank level would rise with the Makeup Pump tripped and letdown still going but the primary reason is to maintain RCS inventory.

"D" is incorrect, this is plausible since isolating letdown will reduce heat load on Nuclear ICW but the reason is to maintain RCS inventory.

This question matches the K/A since it is about a loss of RCS makeup (makeup pumt trip) and it tests knowledge of the reason for an action from the loss of charging AOP.

References:

1203.026, Loss of Reactor Coolant Makeup

History:

New for 2005 RO exam, replacement question. AK3.04
Selected for 2017 RO Re-exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1122 **Rev:** 0 **Rev Date:** 2/22/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 025 **System Title:** Loss of RHR System

Description: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

K/A Number: 2.2.42 **CFR Reference:** 41.7 / 41.10 / 43.2 / 43.3 / 45.3

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 6 **SRO:**

Given:
* Plant Heatup to 200 °F in progress to prepare for starting RCPs
* Both Decay Heat pumps are in operation
* "A" DH outlet temperature of 185 °F was recorded at 0915
* RCS pressure 190 psig

At 0945 the ATC reports the following:
* Pressurizer level 95 inches
* "A" DH suction temperature 205 °F
* RCS pressure 215 psig

NOW
The P-34A Decay Heat pump trips.

Which of the following Technical Specifications LCO's is NOT met?

A. 3.4.3, Reactor Coolant Pressure/Temperature Limits

B. 3.4.6, RCS Loops - Mode 4

C. 3.4.9, Pressurizer

D. 3.5.3, ECCS - Shutdown

Answer:

D. 3.5.3, ECCS - Shutdown

Notes:

"D" is correct, with RCS temperature at 205 °F, Mode 4 has been entered (> 200 °F) and 3.4.3 requires two operable ECCS trains in Mode 4. With P-34A DH pump breaker tripped, operability is not satisfied.

"A" is incorrect but plausible, a temperature rise of 20 °F (205 - 185) has taken place over 30 minutes (0915 to 0945) but the 3.4.3 limit of 50 °F/hr has not yet been exceeded.

"B" is incorrect but plausible since 3.4.6 requires two operable loops with one loop in operation but RCS pressure is at the point (>200 psig where an RCP can be started and the other DH pump is running, so this spec is met for the given conditions.

"C" is incorrect but plausible since an 1102.002 procedural requirement of less than 95 inches in the Pressurizer is not met but 3.4.9 does not have a minimum Pressurizer level for operability. LCO 3.4.11, LTOP does have a surveillance requirement to verify the pressurizer does not represent a water solid condtion but that is not part of 3.4.9.

This question matches the K/A since it involves a loss of an RHR system (DH pump) with parameters requiring entry into TS action for 3.5.3.

References:

Technical Specificatlions 3.5.3

History:

New question for 2017 RO Re-take exam

QID: 1123 **Rev:** 0 **Rev Date:** 2/22/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 026 **System Title:** Loss of Component Cooling Water

Description: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Control of flow rates to components cooled by the CCWS.

K/A Number: AA1.06 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier:	1	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 7 **SRO:**

Given:
* Unit 1 is at 100% power
* P-33B ICW pump is tagged out for maintenance

The ATC reports the following:
* ICW FLOW LO (K12-B4) alarms
* P-33C Nuclear ICW pump has tripped and will not re-start.

Which of the following actions will be in compliance with 1203.052, Loss of Intermediate Cooling Water?

A. Total ICW flow is 3250 gpm and ICW isolated to one SFP Cooler.

B. Total ICW flow is 3050 gpm and Letdown re-established with one Letdown Cooler.

C. Total ICW flow is 3250 gpm and Letdown re-established with one Letdown Cooler.

D. Total ICW flow is 3050 gpm and ICW isolated to one SFP Cooler.

Answer:

B. Total ICW flow is 3050 gpm and Letdown re-established with one Letdown Cooler.

Notes:

"B" is correct, with the "B" ICW pump tagged out and the Nuclear ICW pump tripping there will only be the "A" ICW pump running so total ICW flow must be maintained less than or equal to 3100 gpm to prevent pump runout. Initiallly Letdown is isolated bu then is re-stablished following isolating ICW to one Letdown cooler.

"A" is incorrect but plausible, the total ICW flow is close to the required value but is too high. Also, ICW is isolated to both SFP Coolers, not just one.

"C" is incorrect but plausible, since the action for Letdown flow is correct. The total ICW flow value is close to the required value but is too high.

"D" is incorrect but plausible, the total ICW flow value is correct but ICW is isolated to both SFP Coolers, not just one.

This question matches the K/A as a loss of CCW has occurred (ICW pump trip) and the letdown flow rate must be controlled to be within the capacity of one Letdown cooler (cooled by ICW).

References:

1203.052, Loss of Intermediate Cooling Water

History:

New question for 2017 RO Re-take exam

QID: 0933 **Rev:** 0 **Rev Date:** 9/18/14 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 14 **Point Value:** 1

Section: 4.2 **Type:** Generic Abnormal Plant Evolutions

System Number: 027 **System Title:** Pressurizer Pressure Control Malfunction

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

K/A Number: AK2.03 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 2.8 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 8 **SRO:**

Given:
* Unit 1 is at 100% power
* A MFW pump trips.

Current plant conditions are
* Unit 1 is at 85% power and lowering
* RCS pressure has now lowered to 2110 psig.

Which of the following indicates a malfunction of the Pressurizer Pressure Control System?

A. Pressurizer spray valve OPEN

B. Heater bank 3 ON

C. Heater bank 4 ON

D. Heater bank 5 ON

Answer:

D. Heater bank 5 ON

Notes:

"D" is correct, Bank 5 should not be on until 2105 psig so if Bank 5 were ON at 2110 psig it would be a malfunction of the Pressurizer Pressure controller.

"A" is incorrect but plausible, the spray valve should normally close at 2155 psig. However a MFW Pump trip above 80% power reduces the 2205 open setpoint for the spray valve to 2080 psig and the close setpoint drops to 2030 psig so the spray valve should be open (it is NOT a malfunction).

"B" is incorrect but plausible since this is another heater bank but Bank 3 turns on at 2135 psig so it should be on at 2110 psig and thus is not a malfunction.

"C" is incorrect but plausible since this is another heater bank but Bank 4 turns on at 2120 psig so it should be on at 2110 psig and thus is not a malfunction.

This question matches the K/A since it involves the Pressurizer Pressure Control system and determines if applicant can recognize a malfunction of the Pressure Controller.

References:

1103.005, Pressurizer Operation

History:

New for 2014 Exam - NOT USED due to similarities with another question.
New for 2017 RO Re-exam.

QID: 1124 **Rev:** 0 **Rev Date:** 2/22/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ESAS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 040 **System Title:** Steam Line Rupture

Description: Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring ESFAS initiation.

K/A Number: AA2.04 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 4.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 9 **SRO:**

A steam line break has occurred in the Reactor Building.

When performing RT-10, What RB pressure would require actuation of ESAS Channels 1-10?

A. 15.2 psia

B. 18.7 psia

C. 30.0 psia

D. 44.7 psia

Answer:

D. 44.7 psia

Notes:

"D" is correct, RB pressure of 44.7 psia (30 psig) requires actuation of all ESAS channels.

"A" is incorrect, but plausible since this is an abnormal RB pressure and would require operator action to address, but no actuations would occur.

"B" is incorrect, but plausible since this is the actuation setpoint for channels 1-6, but not for channels 7-10.

"C" is incorrect, but plausible since this is similar to the actuation setpoint for channels 7-10, i.e., 30 psig.

This question matches the K/A since it requires the recognize the correct setpoint for ESFAS actuation following a major steam line rupture.

References:

1202.012, Repetitive Tasks, RT-10, Verify Proper ESAS Actuation

History:

New for 2017 RO Re-exam

QID: 0623 **Rev:** 0 **Rev Date:** 10/31/05 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-ICS **Objective:** 28 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 054 **System Title:** Loss of Main Feedwater (MFW)

Description: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Occurrence of reactor and/or turbine trip.

K/A Number: AA2.01 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 4.4 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 10 **SRO:**

Given:
* A plant startup is in progress.
* Per procedure, the second MFW pump, P-1B, is being placed in service at 350 MW.
* Startup of P-1B is complete to the point of placing the H/A station in AUTO when P-1B trips.

Subsequently both Startup and Low Load Control valves are observed to go closed.

What has occurred during this transient?

A. Loss of Instrument Air

B. Reactor trip

C. Loss of power to H1 bus

D. Turbine trip

Answer:

B. Reactor trip

Notes:

"B" is correct, on a reactor trip the ICS Rapid Feedwater Reduction (RFR) circuit will close the Startup and Low Load control valves. This ICS design feature closes these valves to totally stop Feedwater thereby preventing an overcooling. The RFR circuit will release the Startup and Low Load control valves when SG levels drop to the low level limit setpoint of 45" (40" is normal setpoint but a reactor trip will bias this setpoint by 5" to limit "undershoot").

"A" is incorrect, although plausible since both valves are actuated by Instrument Air, but the Low Load Control Valves fail "As-Is" on a loss of Inst. Air and so will not be fully closed.

"C" is incorrect, although plausible since a loss of power to the H1 bus means a loss of one RCP in each loop which will generate a ICS runback. A runback would close the block valves but will not close the control valves.

"D" is incorrect, although plausible since the Turbine trip signal is an input into several ICS control schemes, but this would have no direct effect on the FW system. The condition of 350 Mwe is equivalent to 39% power which is below the 43% threshold of generating a reactor trip due to a turbine trip.

Changed correct answer from "C" to "B".

This question matches the K/A since it involves a loss of Main Feedwater (trip of B pump) and requires interpretation of what could have closed the FW valves, i.e., a reactor trip.

References:

STM 1-64, Integrated Control System

History:

New for 2005 RO re-exam.
Selected for 2017 RO Re-exam.

QID: 1125 **Rev:** 0 **Rev Date:** 2/23/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-EOP08 **Objective:** 14 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 055 **System Title:** Station Blackout

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

K/A Number: EK1.02 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.4 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 11 **SRO:**

Given:
* Unit 1 tripped from 100% power due a loss of off site power
* Both EDG's have tripped and repair actions are in progress.
* Natural circulation has been established for one hour with Tave at 545°F and Thot 565°F.

In accordance with RT-5, Verify Proper EFW Actuation and Control, with primary to secondary heat transfer in progress, the operator should observe:

A. Tcold tracking SG Tsat temperature

B. Core delta T rising

C. Tcold tracking CET temperature

D. ADVs opening periodically to maintain SG pressure

Answer:

A. Tcold tracking SG Tsat temperature

Notes:

"A" is correct. if natural circulation is cooling the core, then Tcold should be tracking SG Tsat temeperature.

"B" is incorrect since Core delta T should be stable or dropping, but this is plausible since core delta T will rise early in the event before natural circulation is established, but this event has been going on for an hour.

"C" is incorrect but plausible as an RCS temperature should be tracking CETS but it will be Thot, not Tcold.

"D" is incorrect, as long as heat removal is continuing, secondary heat must be removed by steaming continuously. This is plausible since periodic ADV opening could be observed if forced circulation were in progress.

Modified version of QID 124: changed EDGs to "tripped" so this question would apply to Blackout, replaced Natural Circulation AOP with RT-5, replaced "D" distractor since TBVs aren't available, changed "B" to "rising", changed "A" to "Tcold" to make it the correct answer.

This question matches the K/A since it has Blackout condtions and tests the knowledge of an operational implication: how to verify natural circulation cooling is in progress.

References:

1202.008, Blackout
1202.012, Repetitive Tasks, RT-5, Verify Proper EFW Actuation and Control

History:

Modifed QID 124 for 2017 RO Re-exam

QID: 1145 **Rev:** 0 **Rev Date:** 3/22/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 11 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 056 **System Title:** Loss of Off-site Power

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

K/A Number: 2.4.4 **CFR Reference:** 41.10 / 43.2 / 45.6

Tier:	1	**RO Imp:**	4.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 12 **SRO:**

Given:
* Unit 1 tripped from 100% power
* A3 & A4 buses energized from EDGs, all other 4160v/6900v buses de-energized
* CETs 615 °F and rising slowly
* RCS pressure 2000 psig and rising slowly
* Both SG pressures 1000 psig and steady

Which EOP is required to be in use for the above conditions?

A. 1202.002, Loss of Subcooling Margin

B. 1202.004, Overheating

C. 1202.005, Inadequate Core Cooling

D. 1202.007, Degraded Power

Answer:

D. 1202.007, Degraded Power

Notes:

"D" is correct, since only EDGs are powering 4160V buses following a reactor trip, the entry condition for Degraded Power (1202.007) has been met. 1202.007 has sections designed to mitigate a loss of Subcooling Margin (SCM), overheating, and overcooling events.

"A" is incorrect but highly plausible since SCM does not exist in the conditions given, however, Degraded Power beginning at step 24 has actions to mitigate a loss of SCM.

"B" is incorrect but highly plausible since an overheating condition exists (CETs >610 °F with RCPs off), however Degraded Power beginning at step 55 has actions to mitigate overheating.

"C" is incorrect but plausible due to elevated RCS core exit conditions but entry conditions are not met for the ICC procedure (CETs are not moving to the right of the saturation line.)

This question matches the K/A since a loss of offsite power condition is given which is the entry condtion for the loss of offsite power EOP (actually entry conditions are met for serval EOPs).

References:

1202.007, Degraded Power
A1LP-RO-EOP07, Degraded Power lesson plan

History:

New for 2017 RO Re-exam

QID: 1163 **Rev:** 0 **Rev Date:** 4/8/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 21 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 057 **System Title:** Loss of Vital AC Instrument Bus

Description: Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Backup instrument indications.

K/A Number: AA1.05 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 13 **SRO:**

Power has been lost to 120VAC Panel RS-2.

One of the instruments that has lost power are RCS Flow transmitters.

Aside from the Plant Computer or SPDS, where can a backup indicator for RCS Flow be found?

A. Back panel C486-1 in the Control Room

B. Dasey Panel C166 in the Computer Room

C. Panel C18 in the Control Room

D. Panel C03 in the Control Room

Answer:

D. Panel C03 in the Control Room

Notes:

"D" is correct, although all four RPS Channels have an input from RCS Flow for protective trips, only A and B RPS channels provide RCS flow indication. Panel C03 in the Control Room has alternate indications for RCS Flow from "A" RPS channel.

"A" is incorrect but plausible since C486-1 has alternate indications for other important parameters from RPS, such as RB pressure, but it does not have RCS flow.

"B" is incorrect but plausible since the Dasey Panel alternate indications for other important parameters from RPS, such as RCS pressure from A RPS channel, but it does not have RCS flow.

"C" is incorrect, but plausible since panel C18 has indications for RB pressure, HPI flow, BWST level, and LPI flow but it does not have an indicator for RCS flow.

References:

STM 1-63, Reactor Protection System

History:

New for 2017 RO Re-exam

QID: 1156 **Rev:** 0 **Rev Date:** 4/4/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 058 **System Title:** Loss of DC Power

Description: Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems.

K/A Number: AA2.03 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 14 **SRO:**

Given:
* Reactor tripped from 100% power due to a loss of 125VDC bus D02.

Which of the following is a de-energized DC load and what is the impact on the plant?

A. DG2 control power,
DG2 will not start from the control room

B. Main Turbine trip block solenoid,
Main Turbine can not be tripped from the control room

C. Generator Lockout Relays,
all even buses de-energized

D. Vital AC panels RS2 and RS4,
inadvertent ESAS actuation

Answer:

A. Bus A4 bus de-energized,
DG2 will not start from the control room

Notes:

"A" is correct, control power for DG2 comes from D02 via distribution panel D21, without control power the EDG can not be started from the control room, it can only be started using the "No DC" start procedure.

"B" is incorrect, control power for Main Turbine trip block solenoid comes from D01, but plausible since it is powered from Vital DC.

"C" is incorrect, the Generator Lockout relays are powered from D01 via D11, but the plant impact is correct, all even buses will be de-energized but they will be de-energized due to the loss of control power to the breakers so they can not auto-transfer to a Startup Transformer.

"D" is incorrect, but plausible since RS2 and RS 4 are lost but only one of the ESAS Analog channels will be lost so there will not be an inadvertent ESAS actuation. An inadvertent ESAS actuation does occur on a loss of D01, but not D02.

This question matches the K/A since it involves a loss of DC power (D01) and evaluates knowledge of DC loads and the impact on the plant from the loss of some of those loads.

References:

1203.036, Loss of 125V DC

History:

New question for 2017 RO re-exam

QID: 0947 **Rev:** 0 **Rev Date:** 2/18/13 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-MSSS **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 062 **System Title:** Loss of Nuclear Service Water

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.

K/A Number: AK3.02 **CFR Reference:** 41.4, 41.8 / 45.7

Tier:	1	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 15 **SRO:**

The plant is operating normally at 100% power with the following conditions:

* Service water pumps P-4A, P-4B, and P-4C are in service

* The service water cross-tie valves are all open and in "AUTO"

* The P-4B MOD breaker is aligned to provide power from the A4 bus

Following an Engineered Safeguards Actuation System (ESAS) channels 1 and 2 actuation with offsite power available, how will the Service Water System respond and what is the reason for this response?

A. The crosstie valves between P-4B and P-4C will remain open;
to prevent deadheading the pump.

B. All the crosstie valves will close and P-4B will trip;
in order to maintain train separation.

C. All four crosstie valves will close and P-4B will continue to run;
to provide cooling water to the Aux Cooling Water loop.

D. All four crosstie valves will remain open as long as off-site power is not lost in order
to maximize system reliability.

Answer:

A. The crosstie valves between P-4B and P-4C will remain open;
to prevent deadheading the pump.

Notes:

"A" is correct, with the system aligned as described in the stem, the cross-tie valves between P-4B and C will remain open since all 3 SW pumps are running and offsite power is available. This prevents deadheading the pump.

"B" is incorrect yet plausible since all crosstie valves would close if only A and C SW pumps were running. This could be the case if offsite power were lost coincident with the ESAS actuation. SW pump P-4B would not be re-started after the EDG energizes the bus as long as P-4C was available.

"C" is incorrect yet plausible if the candidate recalls that all 3 SW pumps will remain runnning but cannot recall what happens with the ACW isolation. The ACW isolation can be powered from either B55 or B56, and thus will be affected by ESAS channels 1 or 2 depending on its power supply. The ACW isolation will remain open in certain situations but only if Channel 1 or 2 actuated (not both) with all 3 pumps runnning and offsite power available. This unique set of conditions would prevent deadheading the B pump due to inadvertent ESAS actuation of a single channel, but is incorrect for this question since two channels actuated.

"D" is incorrect since the cross-tie valves between P-4A and B will close, but plausible if the candidate can recall that all 3 pumps will remain running and cannot recall that train separation is required after ESAS

actuation. All four crosstie valves open is the normal system configuration.

Swapped "A" and "B" since this is a bank question. Separated automatic actions from reasons to make question more clear.

This question matches the K/A as the conditions are given for an ESAS actuation and the candidate is required to know how the Service Water system will re-align and the reason for that alignment.

References:

STM 1-42, Service and Auxiliary Cooling Water

History:

New for 2013 Exam
Selected for 2017 RO Re-exam

QID: 0108 **Rev:** 1 **Rev Date:** 2/24/17 **Source:** Modified **Originator:** JCork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 065 **System Title:** Loss of Instrument Air

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Knowing effects on plant operation of isolating certain equipment from instrument air.

K/A Number: AK3.03 **CFR Reference:** 41.5 / 41.10 / 45.6 / 45.13

Tier:	1	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 16 **SRO:**

Given:

Plant operating at 100% power.

An instrument air leak causes air to Turbine Bypass Valve (CV-6688) valve operator to be isolated.

Since the leak is downstream of the isolation point, the leak eventually causes the instrument air accumulators for CV-6688 to depressurize.

What effect will this have on CV-6688 operation and why?

A. CV-6688 will go OPEN due to system pressure.

B. CV-6688 will go OPEN due to spring pressure.

C. CV-6688 will remain CLOSED due to system pressure.

D. CV-6688 will remain CLOSED due to spring pressure.

Answer:

A. CV-6688 will go OPEN due to system pressure.

Notes:

"A" is correct. On a loss of instrument the accumulator for CV-6688 will maintain the valve closed for a period of time, but the given condition of a failure of the instrument air to the valve operator (with a leak) will result in a loss of accumulator pressure. At the given power level, the valve will be forced open by steam pressure.

"B" is incorrect but plausible since some air operated valves do fail open by spring pressure on loss of air but there is no spring.

"C" is incorrect and not very plausible but it balances the answer choices. All of the other choices are plausible however, so this does not make the question psychometrically flawed.

"D" is ncorrect but plausible if the candidate believes the actuator contains a spring and the valve failed closed on loss of air but there is no spring.

Modified question to be a "2 x 2" due to two implausible distractors in the original question. Two distractors revised and condition was changed from a severed line to Inst. Air lost with a small leak.

This question matches the K/A due to a loss of Instrument Air condition and the applicant must know the effect on the equipment and the reason for that effect.

References:

1203.024, Loss of Instrument Air, Attachment A

History:

Developed for the 1998 RO/SRO Exam.
Used in 98 RO Re-exam
Selected for 2002 RO/SRO exam.
Selected for 2007 RO Exam. Changed KA from 041 K6.03 to 065 AK3.03.
Modified for 2017 RO Re-exam.

QID: 0514 **Rev:** 1 **Rev Date:** 3/31/17 **Source:** Modified **Originator:** NRC

TUOI: A1LP-RO-EOP04 **Objective:** 14 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: E04 **System Title:** Inadequate Heat Transfer

Description: Knowledge of the interrelations between the (EOP Rules) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A Number: EK2.1 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 17 **SRO:**

Given:

* EOP 1202.004, Overheating, is in use.
* HPI Cooling, RT-4, is also in use.

What is the preferred method of operation of the ERV in this situation, AND why is the method preferred?

A. Manual; minimizes the cycles and reduces the chance of ERV failure.

B. Manual; ensures ES actuations occur to simplify the subsequent operator actions.

C. Automatic; minimizes the cycles and reduces the chance of ERV failure.

D. Automatic; ensures ES actuations occur to simplify the subsequent operator actions.

Answer:

A. Manual; minimizes the cycles and reduces the chance of ERV failure.

Notes:

"A" is correct, automatic operation would make the ERV open and close more often in reaction to RCS pressure, more cycling of the valve increases the chance of ERV failure, thus manual operation is preferred.

"B" is incorrect but plausible since opening the ERV manually is the preferred method per the EOP bases but the reason is incorrect since another reason for manual operation of the ERV is also to prevent ES actuations, not cause them as stated in this distractor.

"C" is incorrect but plausible since the reason give is the correct one but this choice lists automatic operation instead of manual operation.

"D" is incorrect, this distractor completes the 2x2 choices with both the incorrect mode of opeation and the incorrect reason for the preferred mode.

Modified all answers by making them into a 2x2 format, this eliminates implausible distractors present in the original question. Rephrased stem of question to be more specific.

This question matches the K/A due to an Overheating condition is given and it requires the applicant to know the interrelation between the event and the method of operation of a component, i.e., the ERV, per the EOP bases document for Inadequate Heat Transfer and Repetitive Task (Rule) for HPI Cooling

References:

GEOG bases, HPI Cooling
GEOG bases III.C, Inadequate Heat Transfer
1202.004, Overheating
1202.012, Repetitive Tasks

History:

Developed by NRC.
Taken from Crystal River, Unit 3, Date: 03/22/1996
Selected for 2017 RO Re-exam

QID: 1126 **Rev:** 0 **Rev Date:** 2/24/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-GEN **Objective:** 7 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 077 **System Title:** Generator Voltage and Electric Grid Disturbances

Description: Ability to interpret reference materials, such as graphs, curves, tables, etc.

K/A Number: 2.1.25 **CFR Reference:** 41.10 / 43.5 / 45.12

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 18 **SRO:**

***************REFERENCE PROVIDED****************************

Thunderstorms are in the area.

Due to plant issues, Unit 1 is operating at 720 MWe under-excited with a power factor of .98.

Main Generator Hydrogen pressure is 60 psig.

The Dispatcher calls the Control Room, says that lightning has tripped a capacitor bank, and requests Unit 1 reactive load be raised as much as possible.

The Dispatcher states that Main Generator electrical load and power factor must NOT change.

What is the MAXIMUM change in reactive load allowed for the above conditions per Power Operation (1102.004)?

A. 550 MVARs

B. 340 MVARs

C. 280 MVARs

D. 140 MVARs

Answer:

C. 280 MVARs

Notes:

"C" is correct, operating at a PF of .98 under-excited with a load of 720 MW means the Main Generator reactive load must be -140 MVARs. The most reactive load which could be raised is where the 720 MW line intersets the .98 PF line in the over-excited half of Att. N. This would b -140 to +140 for a total of 280 MVARs. This also meets an Ops log restriction of a max of +160 MVARs.

"A" is incorrect yet plausible if the candidate mistakenly starts at the 720 MW line at .98 PF and goes until the 720 MW line intersects the 60 psig limit line, then performs the calculation.

"B" is incorrect yet plausible if the candidate mistakenly starts where the .98 PF line intersects the 60 psig limit line on the under-excited side and draws a line straight up to where the 60 psig limit intersects the .98 PF on the over-excited side and performs the calculation.

"D" is incorrect but is plausible if the candidate mistakenly starts at the 720 MW line at 1.0 PF and draws as line to the .98 PF line.

This question matches the K/A since it involves a grid disturbance and requires the candidate to use the provided graph and conditions to arrive at the correct answer.

References:

1102.004, Power Operation

1102.004, Attachment N, must be in RO handout!!!

History:

New question for 2017 RO Re-exam

QID: 0397 **Rev:** 1 **Rev Date:** 3/31/17 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-RO-NOP **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 001 **System Title:** Continuous Rod Withdrawal

Description: Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place.

K/A Number: AA2.03 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 4.5 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.8 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 19 **SRO:**

Approach to criticality is in progress.
Reactor power is in the Source Range.
The CBOR commences sequential withdrawal of the regulating rods.

The following indications are observed:

- SR (Source Range) count rate rising
- Sustained SUR of 2.5 DPM
- Continued outward rod motion without a command.

What action is required to be taken (or verified to occur) per 1102.008, Approach to Criticality?

A. Select "JOG" on rod speed selector switch.

B. Trip the reactor and go to 1202.001, Reactor Trip.

C. Verify the Out Inhibit on high SUR stops rod motion automatically.

D. Commence Emergency Boration per RT-12 until SUR is negative.

Answer:

B. Trip the reactor and go to 1202.001, Reactor Trip.

Notes:

"B" is correct, the Source Range rod hold on high SUR of 2.5 DPM should have stopped outward rod motion. A system failure has occurred, control of reactivity has been lost, and reactor should be tripped.

"A" is incorrect, plausible since this is one of the steps to stop rod motion but not all of them.

"C" is incorrect, plausible since the SR High SUR inhibit should have stopped rod motion at 2 DPM but it would be imprudent to wait to see if the IR High SUR inhibit at 3 DPM would work.

"D" is incorrect, plausible since emergency boration will add negative reactivity but this method is too slow.

Revised question by adding procedure to stem and changing order of answers.

This matches the K/A since conditions are given for a continuous rod withdrawal event and to answer the question correctly requires knowledge of the proper procedural action.

References:

1102.008, Approach to Criticality

History:

New created for 2001 SRO Exam.
No longer SRO, selected for 2017 RO Re-exam.

QID: 1127 **Rev:** 0 **Rev Date:** 2/27/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ACRD **Objective:** 26 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 003 **System Title:** Dropped Control Rod

Description: Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod: Actions contained in EOP for a dropped control rod.

K/A Number: AK3.04 **CFR Reference:** 41.5, 41.10 / 45.6 / 45.13

Tier:	1	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 20 **SRO:**

Given:

* Plant is at 38% power.
* ICS is in full automatic.
* Annunciator K08-C2 "CONTROL ROD ASYMMETRIC" alarms.
* Rod 9 in Group 5 has dropped.
* All actions in response to the dropped rod have been completed.

Which of the following actions must be performed FIRST to recover the dropped rod, and why must the rod be withdrawn so it is level with its group?

A. Depress FAULT RESET on the Diamond panel;
to ensure power peaking limits and shutdown margin stay within design limits.

B. Latch the dropped rod using auxiliary power supply;
to ensure power peaking limits and shutdown margin stay within design limits.

C. Latch the dropped rod using auxiliary power supply;
to ensure operation remains within the accident analysis for an ejected rod.

D. Depress FAULT RESET on the Diamond panel;
to ensure operation remains within the accident analysis for an ejected rod.

Answer:

B. Latch the dropped rod using auxiliary power supply;
to ensure power peaking limits and shutdown margin stay within design limits.

Notes:

"B" is correct, the rod is in Group 5 which is the first group to be withdrawn and thus will be at the out limit, so per 1105.009 the dropped rod is first latched using the aux power supply, and per lesson plan (A1LP-RO-CRD) a misaligned control rod may cause increased power peaking and reduction in total available shutdown margin.

"A" is incorrect and plausible since the reason given is correct but FAULT RESET is not pressed until after the rod is leveled with its group.

"C" is incorrect and plausible since the action given is the first one to be performed per 1105.009 but the reason given is not correct per lesson plan.

"D" is incorrect as it contains both the incorrect action and the incorrect reason. The incorrect action is plausible if the student forgets CRD logic and assumes the fault must be reset before outward rod motion can take place (e.g., out inhibit). The incorrect reason is plausible since there is an accident analysis in chapter 14 of the SAR for an ejected rod accident.

This question matches the K/A since condtions are given for a dropped rod and the candidate must know how a dropped rod is recovered, and why a dropped rod is re-leveled with its group.

This question has ANO-1 specific OE where a fault occurred with a CRD at 100% power. The crew did not

lower power and depressed FAULT RESET. This removed the withdrawal inhibit and the rods immediately began pulling, causing power to go above 100% (no trip, just greater than 100%).

References:

1105.009, CRD System Operating Procedure

History:

New question for 2017 RO Re-exam.

QID: 1128 **Rev:** 0 **Rev Date:** 3/1/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NI **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 033 **System Title:** Loss of Intermediate Range NI

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Intermediate Range Nuclear Instrumentation: Effects of voltage changes on performance.

K/A Number: AK1.01 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier:	1	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.0	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 21 **SRO:**

Given:
* Plant startup (approach to criticality) in progress.
* The following NI readings are observed:
  - Channel 1 Source Range, NI-501 at 6 E4 cps
  - Channel 2 Source Range, NI-502 at 5 E4 cps
  - Intermediate Range Channel NI-3 at 2 E -11 amps
  - Intermediate Range Channel NI-4 at 6 E -10 amps
  - Power Range Channels NI-5 thru 8 at 0%

Which of the following conclusions is correct for the above indications?

A. NI-3 is reading low due to low compensating voltage (under-compensated)

B. NI-3 is reading low due to high compensating voltage (over-compensated)

C. NI-4 is reading high due to low compensating voltage (under-compensated)

D. NI-4 is reading high due to high compensating voltage (over-compensated)

Answer:

B. NI-3 is reading low due to high compensating voltage (over-compensated)

Notes:

"B" is correct, Intermediate Range Channel NI-3 is reading very low and does not exhibit one decade overlap from Source Range to Intermediate Range. The compensating voltage set too high would definitely cause a low reading by subtracting too much from the detector's overall output.

"A" is incorrect, while NI-3 is reading low (making this answer plausible), but a low compensating voltage will cause the detector to read high, not low.

"C" is incorrect, yet plausible since a low compensating voltage could cause NI-4 to read too high, NI-4 is exhibiting one decade of overlap with the Source Range channels and is thus reading correctly.

"D" is incorrect, yet plausible if the candidiate determines NI-4 is reading high (vs. NI-3 reading low), and believes compensating voltage and indication go together (high compensating voltage causes high indication).

This question matches the K/A since it concerns the affect of Intermediate Range detector performance as it is affected by a voltage change.

References:

STM 1-67, Nuclear Instrumentation

History:

New question for 2017 RO Re-take exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1129 **Rev:** 0 **Rev Date:** 3/2/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-ICS **Objective:** 13 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: A01 **System Title:** Plant Runback

Description: Knowledge of the interrelations between the (Plant Runback) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A Number: AK2.1 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 22 **SRO:**

Given:
* Plant is at 88% power for Main Turbine TV/GV testing.
* P-1A MFWP suddenly trips.

Assuming the reactor does NOT trip, what plant conditions would you expect to observe after the plant is again at steady state (without operator actions)?

A. Total MFW flow 4.9 x e6 lbm/hr

B. Total MFW flow 4.4 x e6 lbm/hr

C. Total MFW flow 8.2 x e6 lbm/hr

D. Total MFW flow 6.0 x e6 lbm/hr

Answer:

B. Total MFW flow 4.4 x e6 lbm/hr

Notes:

B" is correct, these are the indications for 40% power. There is a runback to this power for a loss of 1 MFWP or asymmetric rod. 100% power was assumed to be 11 x e6 lbm/hr total, 11 x 0.4 = 4.4

"A" is incorrect, these are the indications for 45% power. There is a runback to this power for a loss of 1 RCP in each loop which makes this plausible.

"C" is incorrect, ICS will runback the plant to 75% power for a loss of one RCP which makes this plausible..

"D" is incorrect, these are the indications for 55% power. The runback for a loss of 1 RCP in each loop was formally at this power level which makes this plausible. Also, this runback is only applicable when RX power is less than 55% so this value is credible.

This is a modified version of QID 635. The power level condition was changed to 88% power and a MFWP trip replaced the RCP trip. The answer changed from 8.2 x e6 to 4.4 x e6 lbm/hr.

This question matches the K/A since it requires the knowledge of the interrelation between MFW flow and the power level for the automatic runback feature in ICS for the trip of a MFWP.

References:

1105.004, Integrated Control System

History:

Modified QID 635 for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0020 **Rev:** 0 **Rev Date:** 7/6/98 **Source:** Bank **Originator:** GGiles

TUOI: A1LP-RO-NNI **Objective:** 6 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs/APEs

System Number: A02 **System Title:** Loss of NNI-X

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

K/A Number: AA1.1 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier:	1	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 23 **SRO:**

Given the following indications/alarms:

* (K07-B4) SASS MISMATCH in alarm (fast flash)

* (K07-E3) B OTSG BTU LIMIT in alarm (slow flash)

* SG "B" FW Temp signal select switch selected to SASS Enable (Auto) with the white indicating light off and the blue "Y" light on.

What operator action is procedurally required per the annunciator corrective actions for K07-B4, SASS MISMATCH?

A. Depress the Auto pushbutton for SG "B" FW Temp on the SASS panel in C47-2.

B. Place the SG "B" FW Temp signal select switch to the "Y" position.

C. No action necessary, SASS has automatically transferred to "X" NNI.

D. Place both MFW pump H/A stations in HAND.

Answer:

B. Place the SG "B" FW Temp signal select switch to the "Y" position.

Notes:

"B" is the correct response per procedure 1203.012F and 1105.006.

"A" is incorrect, but plausible since this action is performed inside the ICS cabinets when resetting a failed signal.

"C" is incorrect SASS has transferred to [Y], not [X], placing the selector switch to the input signal automatically transferred is a procedurally required action.

"D" is incorrect, but plausible as 1203.001, ICS Abnormal Operation, directs taking the AFFECTED pump to hand in Section 11, but not both. Also, this action is not necessary since the SASS system has transferred to a good signal and no ICS upset should occur.

Changed "D" from "both FW loop demands in Manual" to "both MFW pump H/A stations in HAND" since the former was also a correct answer per 1203.001 section 11. Also, swapped correct answer from A to B position.

This question matches the K/A since a loss of an NNI-X instrument has occurred and the candidate must know that he needs to operate a manual feature in response to the failure.

References:

1203.012F, Annunciator K07 Corrective Action

History:

Developed for 1998 RO/SRO Exam.
Modified QID 3127
Used in 2001 RO/SRO Exam.
Selected for use in 2002 RO/SRO exam.
Selected for 2007 RO Exam.
Selected for 2017 RO Re-exam.

QID: 0026 **Rev:** 0 **Rev Date:** 7/8/98 **Source:** Bank **Originator:** GGiles

TUOI: A1LP-RO-RPS **Objective:** 14 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs/APEs

System Number: A04 **System Title:** Turbine Trip

Description: Knowledge of system purpose and/or function.

K/A Number: 2.1.27 **CFR Reference:** 41.7

Tier: 1 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 24 **SRO:**

Which of the following describes the purpose of the Anticipatory Trip/Turbine Trip in the Reactor Protection System?

A. To limit plant cooldown following a loss of heat sink.

B. To limit reactor power to prevent exceeding linear heat rate limits.

C. To limit reactor power to prevent exceeding DNBR limits.

D. To limit plant heatup following a loss of heat sink.

Answer:

D. To limit plant heatup following a loss of heat sink.

Notes:

"D" is correct, tripping the Reactor when the Main Turbine trips is an Anticipatory Trip whenever Rx power is greater than 43%. This limits plant heatup due to the loss of secondary heat removal from operation of the Main Turbine at a power level which would exceed the capacity of the Turbine Bypass Valves. At power levels above 43% the Reactor would trip anyway from a Main Turbine trip due to the primary response to a loss of secondary heat sink, hence the term "aniticipatory".

"A" is incorrect yet plausible since this is the reason for tripping the Main Turbine on a Reactor Trip.

"B" is incorrect yet plausible since this is the purpose of most of the RPS trips.

"C" is incorrect yet plausible since this is the purpose of most of the RPS trips.

References:

STM-1-63, Reactor Protection System

History:

Developed for 1998 SRO Exam. This question is no longer SRO Only. JWC 3/2/17
Selected for 2017 RO Re-exam.

QID: 0780 **Rev:** 0 **Rev Date:** 9/09/2009 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs/APEs

System Number: A07 **System Title:** Flooding

Description: Ability to determine and interpret the following as they apply to the (Flooding): adherence to appropriate procedures and operation within the limitations in the facilities license and amendments.

K/A Number: AA.2.2 **CFR Reference:** 41.10 / 43.5 / 4 5.13

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 25 **SRO:**

Given:

* Plant power 100%
* P-34A Decay Heat pump OOS
* Dardanelle Lake Level 346 feet and rising due to heavy rains.
* Corps of Engineers predicts peak flood levels will reach 355 feet.

What action is required per Natural Emergencies (1203.025) section 6, Flood?

A. Perform Rapid Plant Shutdown (1203.045) and align "B" Decay Heat pump for Decay Heat removal.

B. Perform Rapid Plant Shutdown (1203.045) and make preparations to transfer plant auxiliaries to SU2 transformer.

C. Trip Reactor and perform cooldown per Forced Flow Cool Down (1203.040).

D. Trip Reactor and perform cooldown per Plant Shutdown and Cool Down (1102.010).

Answer:

B. Perform Rapid Plant Shutdown (1203.045) and make preparations to transfer plant auxiliaries to SU2 transformer.

Notes:

"B" is correct due to 1203.025 directing performance of a shutdown per 1203.045 when lake level is greater than 345 feet, and SU2 transformer is designed for flooding and auxiliaries will be transferred to it following some preparatory activities.

"A" is incorrect yet plausible since 1203.025 directs one to perform a shutdown per 1203.045, but 1203.025 states to ensure the operable DH loop is aligned for ES standby (LPI) if only one loop is available.

"C" is incorrect since the procedure does not call for a reactor trip on high lake level but plausible since the reactor is tripped on high ECP temperature in the loss of lake section of 1203.025. Also, 1203.040 Forced Flow Cooldown is an option in the Flooding section of 1203.025.

"D" is incorrect since the procedure does not call for a reactor trip on high lake level but plausible since the reactor is tripped on high ECP temperature in the loss of lake section of 1203.025. Also, 1102.010 Plant Shutdown and Cooldown is an option in the flooding section of 1203.025.

This question matches the K/A since the candidate must adhere to the flooding procedure by performing a rapid plant shutdown when lake level exceeds a threshold value (345 ft.) to ensure ANO-1 does not continue operating when flood waters could exceed our design flood level.

References:

1203.025 , Natural Emergencies

History:

New for 2010 RO/SRO exam
Selected for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1147 **Rev:** 0 **Rev Date:** 3/22/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ASDCD **Objective:** 4 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: E08 **System Title:** LOCA Cooldown

Description: Knowledge of the reasons for the following responses as they apply to the (LOCA Cooldown): Normal, abnormal and emergency operating procedures associated with (LOCA Cooldown).

K/A Number: EK3.2 **CFR Reference:** 41.5 / 41.10, 45.6, 45.13

Tier:	1	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 26 **SRO:**

Given:
* Unit 1 is in SBLOCA Cooldown (1203.041)
* SCM is adequate
* "A" SG has a leaking MSSV
* All RCPs are running

Considering the above conditions which RCP will be selected to trip and why?

A. Trip RCP P-32A to leave the combination running with the least stringent NPSH limitations

B. Trip RCP P-32D to leave the combination running with the least stringent NPSH limitations

C. Trip RCP P-32A due to degraded SG "A"

D. Trip RCP P-32D due to degraded SG "A"

Answer:

D. Trip RCP P-32D due to degraded SG "A"

Notes:

"D" is correct. RCP P-32D should be tripped per 1203.041 since this RCP is in the loop with a degraded SG and it does not provide PZR spray.

"A" is incorrect, but plausible since this is in the "B" loop and would normally be tripped so as to leave C & D RCPs running due to the less stringent NPSH requirements when another RCP is tripped at lower RCS pressures. However, A SG is degraded and an RCP should be tripped in that loop.

"B" is incorrect, but plausible since this is the correct RCP to trip but the reason given is incorrect.

"C" is incorrect, but plausible since this is the correct reason but the wrong RCP, the A RCP is in the B loop at ANO-1.

This question matches the K/A since this is a LOCA cooldown situtation and the question requires knowledge of the reason for a step in an abnormal operating procedure.

References:

1203.041, Small Break LOCA Cooldown

History:

New for 2017 RO Re-exam

QID: 0348 **Rev:** 2 **Rev Date:** 3/22/17 **Source:** Bank **Originator:** Goad

TUOI: A1LP-RO-MU **Objective:** 10 **Point Value:** 1

Section: 4.3 **Type:** B&W EOP/AOP

System Number: E13 **System Title:** EOP Rules and Enclosures

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

K/A Number: EK1.3 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 27 **SRO:**

Given:

* LOCA has occurred
* RCS pressure, 950 psig
* CETs, 525°F
* ESAS actuated on channels 1 through 4
* The "ES" HPI pump has failed

Flow from the "OP" HPI pump is as follows:

300 gpm to "A" HPI line
65 gpm to "B" HPI line
85 gpm to "C" HPI line
95 gpm to "D" HPI line

All valves are in ESAS actuated position.

Annunciator A HPI FLOW HI/LO (K11-A4) is in alarm.

What operator action is required per RT-10, Verify Proper ESAS Actuation?

A. Throttle all HPI line flows until they are within 20 gpm of each other.

B. Close the isolation for the "A" HPI line to stop inventory loss due to line break.

C. Throttle the "A" HPI valve until "A" line flow is within 20 gpm of "B" line flow to minimize inventory loss on a line break.

D. Throttle the "A" HPI valve until "A" line flow is within 20 gpm of "D" line flow to minimize inventory loss on a line break.

Answer:

D. Throttle the "A" HPI valve until "A" line flow is within 20 gpm of "D" line flow to minimize inventory loss on a line break.

Notes:

"D" is correct, conditions give only one HPI pump running, so in accordance with RT-10 the highest flow HPI line should be throttled to within 20 gpm of the next highest line, this is done in the event of an HPI line break. This will also clear alarm K11-A4 which must be done per RT-10. All 4 flows add up to 545 gpm total which is greater than alarm setpoint of 450 gpm. Throttling A to 115 gpm will put it within 20 gpm of D and bring total flow down to 360 gpm.

"A" is incorrect but plausible if applicant can recall 20 gpm value but does not recall how to apply it. If they are all within 20 gpm of each other, then this will clear the alarm but will not be in compliance with all procedure steps.

"B" is incorrect but plausible as the conditions are indicative of a break and normally breaks are isolated but the diagnosis of a break is not assured in this case, therefore procedural guidance has flow maintained but balanced.

"C" is incorrect but plausible if applicant can recall 20 gpm value but does not recall how to apply it and mistakenly applies it to be within 20 gpm of the lowest line flow. This will clear the alarm but will not be in compliance with all procedure steps.

Revised original question by adding HPI flow hi alarm to match K/A. Replaced distractor A since it said to do nothing. Added CET temperature of 525 °F to ensure conditions indicate a loss of Subcooling Margin (SCM). Made minor editorial changes.

This question matches the K/A since it involves an EOP Rule/Enclosure (RT-10 and Loss of SCM Rule) and an alarm with associated remedial actions.

References:

1202.012, Repetitive Tasks, RT-10, Verify Proper ESAS Actuation
123.012J, Annunciator K11 Corrective Action

History:

Used in 1999 exam.
Direct from ExamBank, QID# 1711 used in class exam
Selected for the 2008 RO Exam
Selected for 2017 RO Re-exam

QID: 1148 **Rev:** 0 **Rev Date:** 3/23/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ARCP **Objective:** 2 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal

System Number: 003 **System Title:** Reactor Coolant Pump

Description: Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow.

K/A Number: K5.04 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 28 **SRO:**

Given:
* Plant Power 75%
* ICS in full automatic
* RCP P-32A has been stopped due to vibration

Subsequently "A" MFWP trips.

When the plant stabilizes (assuming no operator action) Main Feedwater flow should be ________ for the "A" RCS loop and __________ for the "B" RCS loop?

A. 3.0 X e6 lbm/hr
 1.5 X e6 lbm/hr

B. 1.5 X e6 lbm/hr
 3.0 X e6 lbm/hr

C. 3.3 X e6 lbm/hr
 1.65 X e6 lbm/hr

D. 1.65 X e6 lbm/hr
 3.3 X e6 lbm/hr

Answer:

A. 3.0 X e6 lbm/hr
 1.5 X e6 lbm/hr

Notes:

"A" is correct, the ICS will ratio FW flow so the "A" loop has 2/3 flow and the "B" loop has 1/3 flow with a total of 40% flow which is power level for a MFW Pump trip runback.

"B" is incorrect but plausible, this is the correct FW ratio and total flow but to the wrong loops.

"C" is incorrect but plausible, this would be the correct ratio of FW but to the wrong runback power level of 45% which is the runback limit for a loss of 1 RCP in each loop.

"D" is incorrect but plausible since this is the correct ratio and this is a runback limit but this is the limit (45%) for a loss of 1 RCP in each loop. The proportion of FW flow is backwards, the "A" loop should be higher than the "B".

This question matches the K/A since it requires the knowledge of the operational implications of the secondary effects (proportion of FW flow) due to an RCP shutdown.

References:

1105.004, Integrated Control System
1203.022, Reactor Coolant Pump Trip

History:

New for 2017 RO Re-exam

QID: 1146 **Rev:** 0 **Rev Date:** 3/23/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RCS **Objective:** 23 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal

System Number: 003 **System Title:** Reactor Coolant Pump

Description: Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following:: Prevention of cold water accidents or transients.

K/A Number: K4.02 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 2.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 29 **SRO:**

What interlock and/or procedure limit prevents a cold water reactor power excursion when starting a Reactor Coolant Pump?

A. Reactor power < 22%

B. Cold Leg > 241 °F prior to stating the third RCP

C. SG Downcomer < 50 °F deltaT from cold leg temperature

D. Reactor power < 45% with no more than three RCPs running

Answer:

A. Reactor power < 22% to start any RCP

Notes:

"A" is correct per SAR 14.1.2.5 analysis. Allowing RCP starts only when power is less than 22% prevents a cooldown which, if the moderator coefficient is negative, will introduce positive reactivity into the core causing power to rise. An interlock of power less than 22% ensures RCS pressure rise will not exceed code pressure limits and the minimum DNBR will be greater than 1.3.

"B" is incorrect but plausible as this is the reason when starting the third RCP is allowed, this is a limit and precaution in 1103.006, Reactor Coolant Pump Operation.

"C" is incorrect but plausible as this is the reason when starting an RCP is allowed based on SG Downcomer and RCS temperature deltaT. This is a limit and precaution in 1103.006, Reactor Coolant Pump Operation.

"D" is incorrect but plausible as this is an ICS runback setpoint for only having one RCP running in each loop.

This question matches the K/A since it directly asks for the purpose of the RCP power interlock which is to prevent a cold water transient.

References:

1103.006, Reactor Coolant Pump Operation
ANO-1 SAR, Chapter 14

History:

New for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0417 **Rev:** 1 **Rev Date:** 4/2/17 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-EOP01 **Objective:** 10 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 004 **System Title:** Chemical and Volume Control System

Description: Knowledge of the effect of a loss or malfunction on the following CVCS components: Flow paths for emergency boration

K/A Number: K6.17 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	4.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 30 **SRO:**

Given:

* Rx tripped from 100% power
* P-36A OP HPI pump is running
* Three CRDMs indicate 100% withdrawn
* Boric Acid Pump P-39A is out of service

Emergency Boration is initiated per RT-12.

Boric Acid Pump P-39B discharge pressure indicates 12 psig
Boric Acid flow indicates 2 gpm

What operator action is procedurally required for these conditions?

A. Vent Makup Tank to lower pressure to 10 psig.

B. Raise Batch Controller setting to maximum batch size (999999).

C. Open BWST Outlet valve CV-1407.

D. Verify Batch Controller Flow Control valve, CV-1249, is 100% open.

Answer:

C. Open BWST Outlet valve CV-1407.

Notes:

"C" is correct per RT-12, if Boric Acid pumps or controller are not working (as evidenced by low flow rate on batch controller and low discharge pressure of only Boric Acid Pump), then boration from BWST is initiated.

"A" will lower MUT pressure to less than discharge pressure of Boric Acid Pump P-39B but this is not procedurally directed and thus incorrect.

"B" is incorrect, while this would appear to help, the Batch Controller should already be at maximum batch size prior to starting the Boric Acid Pump P-39B.

"D" is incorrect, this step would have been performed right after starting the Boric Acid Pump P-39B.

Revised question: added P-36A OP HPI pump running to conditions, changed correct answer, formerly said both BWST outlets, changed to only one BWST outlet, this agrees with current revision of 1202.012.

This matches the K/A since the situation requires emergency boration, the makeup and purification system is used for this purpose, and the question involves what action to take if the normal emergency boration flow path is not working.

References:

1202.012, Repetitive Tasks, RT-12 and RT-2

History:

New for 2002 RO/SRO exam.
Selected for 2017 RO Re-exam.

QID: 0700 **Rev:** 1 **Rev Date:** 4/2/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-ADHR **Objective:** 1 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal from Reactor Core

System Number: 005 **System Title:** Residual Heat Removal System

Description: Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: RCS.

K/A Number: K3.01 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 31 **SRO:**

Given:
* Plant is in mode 5
* "A" Decay Heat System is in service
* RCS is intact
* Steam Generators NOT available

A local loss of Instrument Air to the "A" Decay Heat Vault occurs.

What affect would this have on the Decay Heat valves and Reactor Coolant System parameters?

A. Cooler Outlet (CV-1428) stays "as-is"
Cooler Bypass (CV-1433) opens fully
RCS temperature rises
RCS pressure rises

B. Cooler Outlet (CV-1428) opens fully
Cooler Bypass (CV-1433) closes fully
RCS temperature lowers
RCS pressure lowers

C. Cooler Outlet (CV-1428) stays "as-is"
Cooler Bypass (CV-1433) closes fully
RCS temperature lowers
RCS pressure lowers

D. Cooler Outlet (CV-1428) opens fully
Cooler Bypass (CV-1433) opens fully
RCS temperature rises
RCS pressure rises

.

Answer:

C. Cooler Outlet (CV-1428) stays "as-is"
Cooler Bypass (CV-1433) closes fully
RCS temperature lowers
RCS pressure lowers

Notes:

"C" is correct. The Cooler Outlet valve (CV-1428) is an MOV and not be affected by the loss of instrument air. The Cooler Bypass valve (CV-1433) is air operated and will fail closed on loss of instrument air. The increased flow through the Decay Heat cooler will reduce DH outlet temperature and thus RCS temperature and pressure will lower.

"A" is incorrect. The Cooler Bypass valve (CV-1433) fails closed, not open. This distractor is plausible since the given RCS parameter effects correspond to increased bypass flow around the cooler, RCS temperature and pressure would rise.

"B" is incorrect. The Cooler Outlet valve (CV-1428) is an MOV and will not be affected by a loss of instrument

air. However, this distractor is plausible since the Cooler Bypass valve (CV-1433) does fail closed on loss of instrument air. The increased flow through the Decay Heat cooler will reduce DH outlet temperature and thus RCS temperature and pressure will lower.

"D" is incorrect. The Cooler Outlet valve (CV-1428) is an MOV and will not be affected by a loss of instrument air. Also, the Cooler Bypass valve (CV-1433) fails closed, not open. This distractor is plausible since the given RCS parameter effects correspond to increased bypass flow around the cooler, RCS temperature and pressure would rise.

Modified this question, previous version featured a trip of the DH pump with effects on RCS pressure, temperature, and Pressurizer level. There were only two possible answers that would correspond to the laws of physics and thus two distractors were not plausible. Revised this question to be "2x2" with a loss of instrument air to the vault with the answer choices being the corresponding effects on valve positions and RCS press/temp.

This matches the K/A since the conditions give a malfunction of the Decay Heat system (loss of air) and requires the applicant to determine the resulting effects on the RCS.

References:

1203.024, Loss of Instrument Air
STM 1-05, Decay Heat System

History:

New for the 2008 RO Exam.
Modified for 2017 RO Re-exam

QID: 1130 **Rev:** 0 **Rev Date:** 3/3/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-TS **Objective:** 4 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 006 **System Title:** Emergency Core Cooling

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration)

K/A Number: A1.13 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 32 **SRO:**

Unit is at 100% power.

The Core Flood system is properly aligned with the following parameters:

T-2A level	- 13.1 feet	T-2B level	- 12.5 feet
T-2A pressure	- 618 psig	T-2B pressure	- 581 psig

One of the above Core Flood Tank (CFT) parameters is unacceptable per Tech Specs because:

A. Level is too high causing N2 volume to be insufficient to fully inject the CFT contents into the vessel.

B. N2 pressure will cause RCS inventory to be lost out of the break in the event of a LOCA.

C. Level is not sufficient to reflood the vessel following a LOCA.

D. N2 pressure is not sufficient to fully inject the CFT contents into the vessel during a LOCA.

Answer:

C. Level is not sufficient to reflood the vessel following a LOCA.

Notes:

"C" is correct. B level is less than the Tech Spec limit of 12.6 ft and it takes both CFT to accomplish their design of re-flooding the vessel.

"A" is incorrect. This would be the case if either level were out of spec high, however, the levels are within specs, both less than 13.4 ft. This is plausible since T-2A level is close to the high limit

"B" is incorrect, A CFT pressure is less than the Tech Spec limit of 628 psig from 1104.001. This is plausible since A CFT pressure is high but within the admin limit in 1104.001 of 620 psig.

"D" is incorrect. N2 pressures for both are greater than the 572 psig TS limit from 1104.001. This is plausible since B CFT pressure is close to the admin limit in 1104.001 of 580 psig.

Modified QID 197 by changing all of the CFT parameters, most importantly changing T-2B level to 12.5 ft which is less than the T.S. level. This made "C" the correct answer. Answers were revised to remove vague terms like "may" or "could".

This question matches the K/A as it concerns Core Flood Tank parameters (accumulators) and requires the candidate to know the parameter values that define operability for the CFTs to ensure they are within design limits.

References:

1104.001, Core Flood System Operating Procedure
Technical Specifications, LCO 3.5.1 Bases

History:

Modified QID197 for 2017 RO Re-exam

QID: 0904 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** New **Originator:** Possage/Cork

TUOI: A1LP-RO-RCS **Objective:** 17 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 007 **System Title:** Pressurizer Relief Tank/Quench Tank System (PRTS)

Description: Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT

K/A Number: A3.01 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 33 **SRO:**

Given:
* Plant has tripped due to a secondary transient
* RCS pressure has dropped to 1010 psig
* Quench Tank pressure 25 psig
* Reactor Building pressure 15.7 psia

Which of the following Quench Tank temperatures would indicate that the ERV is lifting?

A. 215 °F

B. 239 °F

C. 267 °F

D. 547 °F

Answer:

C. 267 °F

Notes:

"C" is correct, this is saturation temperature for Quench Tank pressure of 40 psia (25 psig). The ERV will be relieving to the Quench Tank which is at ambient, but above normal, temperature. The indicated pressure for the Quench Tank is above the normal nitrogen cover gas pressure of ~3 psig. The relief piping within the Quench Tank is under a water level, so the steam from the ERV will pass through this water, cooling the steam. Therefore, one cannot use the Mollier diagram to derive the correct temperature using an isenthalpic process since the steam condition in the Quench Tank will not be superheated but saturated.

"A" is incorrect, this is saturation temperature for the Reactor Building which is at 15.7 psia. This is above normal but is now ambient temperature for the RB which is elevated due to a coolant leak, or a secondary steam leak. This is plausible but only if the ERV had been lifting continuously and this lifting broke the rupture disc of the Quench Tank, causing the Quench Tank to be at RB pressure but the Quench Tank pressure is given at 25 psig.

"B" is incorrect, this is saturation temperature for 25 psia which is a plausible error an applicant could make.

"D" is incorrect, this is saturation temperature for RCS pressure of 1010 psig but the steam conditions in the Quench Tank will not be the same as the RCS.

This question matches the K/A since to monitor automatic operation of the PRTS (Quench Tank), and the components which discharge to it, one has to deduce the expected temperature if a relief valve was lifting.

References:

Steam Tables
STM 1-03, Reactor Coolant System; Figure 3.21 Reactor Coolant Quench Tank

History:

written for 2014 but not used
New question for 2017 RO Re-exam

QID: 1132 **Rev:** 0 **Rev Date:** 3/6/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-MSSS **Objective:** 9 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 008 **System Title:** Component Cooling Water

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low surge tank level.

K/A Number: A2.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 34 **SRO:**

Given:
* Unit 1 is at 100% power
* RCS leakage has increased
* Inside AO dispatched to ICW Surge Tanks
* Inside AO reports ICW Surge Tank T-37B level has risen from 1.4 psid to 2.1 psid in five minutes

* RCP Seal Injection flows
A 9 gpm
B 8 gpm
C 7 gpm
D 8 gpm

* RCP seal bleedoff temperatures are stable

* Letdown Cooler E29A Outlet temperature is rising

Which of the following actions are required by 1203.039, Excess RCS Leakage, for the above conditions?

A. Isolate Letdown Cooler E29A by closing inlet (CV-1213) and outlet (CV-1214)

B. Trip reactor and perform 1202.001 in conjunction with 1203.039

C. Close Nuclear ICW RB Inlet (CV-2233) and both Nuclear ICW RB Outlets (CV-2214 and CV-2215)

D. Open ICW Surge Tank Crossconnect Isolation (ICW-165)

Answer:

A. Isolate Letdown Cooler E29A by closing inlet (CV-1213) and outlet (CV-1214)

Notes:

"A" is correct. RCS is leaking into ICW system as seen by Nuclear ICW Surge Tank T-37B level risiing. Letdown Cooler E29A outlet temperature is rising, indicating a leak into this cooler. 1203.039 isolates E29A letdown cooler to an attempt to stop the leak.

"B" is incorrect but plausible since 1203.039 does have a step which states to trip the Reactor if HPI is needed to maintain PZR inventory. However, leak rate via ICW surge tank is 46 gpm, this is less than the 50 gpm threshold normally used to make this determination. ICW surge tank contains 1000 gallons and DP indication is 0.5 to 2.7 psid and 1.0 psid is 333 gallons. An increase of 0.7 psid over five minutes means a leak rate of 46.6 gpm. 0.7 x 333 = 233 gallons/5 min = 46.6 gpm

"C" is incorrect but plausible since this action is taken if there is a leak into ICW from an RCP seal cooler. One indication of this is skewed Seal Injection flows. The Seal Injection flows given do vary but they are not significantly skewed.

"D" is incorrect but plausible since the purpose of the cross-connect is to allow the Surge Tanks to overflow from one to the other in case of a leak. However, 1203.039 directs closing, not opening, the cross-connect valve.

This question matches the K/A since it involves high CCW (ICW at ANO-1) surge tank level due to RCS leakage into the ICW system, predicting the impact on operations, and taking procedural action to mitigate the leakage. The applicant is required to calculate RCS leakage based on the rise in surge tank level (thus predicting the impact of this malfunction), determine approximate RCS leak rate to eliminate one course of action (tripping the reactor), diagnose the conditions to determine if it is a leak into the letdown cooler or into the RCP seal cooler, and then arrive at the proper action as stated in the abnormal operating procedure.

References:

1203.039, Excess RCS Leakage

History:

New for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0562 **Rev:** 0 **Rev Date:** 4/5/05 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-RO-MSSS **Objective:** 9 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 008 **System Title:** Component Cooling Water System (CCWS)

Description: Knowledge of the bus power supplies to the following: CCW pump, including emergency

K/A Number: K2.02 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 35 **SRO:**

Which of the following identifies the correct power supplies to the Intermediate Cooling Water Pumps (P-33A, P-33B, P-33C)?

A. P33A is powered from B-12 while P33B and P33C are powered from B-22.

B. P33A and P33B are powered from B-12 while P33C is powered from B-22.

C. P33A, P33B and P33C are powered from B-11, B-12 and B-13 respectively.

D. P33A, P33B and P33C are powered from B-12, B-22 and B-32 respectively.

Answer:

A. P33A is powered from B-12 while P33B and P33C are powered from B-22.

Notes:

"A" lists the correct power supplies for the ICW pumps.

"B" is incorrect since P33B is not powered from B-12 but plausible since the other power supplies are correct.

"C" is incorrect since P33A and P33C power supplies are incorrect but the power supply for the B pump is correct and they are presented in a logical order to enhance plausiblity.

"D" is incorrect since P33C is not powered from B-32 but plausible since the other power supplies are correct.

This question matches the K/A since it requires recall of the power supplies for the ICW (CCW) pumps.

Re-order distractors since this is a bank question. Swapped the order of A and B and also swapped the order of C and D.

References:

STM 1-43, Intermediate Cooling Water

History:

Direct from regular exam bank QID#4674
Selected for 2005 RO exam
Selected for 2011 RO exam.
Selected for 2017 RO Re-exam

QID: 1149 **Rev:** 0 **Rev Date:** 4/3/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NOP **Objective:** 4 **Point Value:** 1

Section: 3.3 **Type:** Reactor Pressure Control

System Number: 010 **System Title:** Pressurizer Pressure Control

Description: Ability to monitor automatic operation of the PZR PCS, including: PZR pressure.

K/A Number: A3.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 36 **SRO:**

Unit 1 is at 100% power.
The ATC observes RCS pressure rising.

Assuming no malfunctions, what is the pressure setpoint at which the ATC will observe the Pressurizer Spray valve automatically OPEN?

A. 2395 psig

B. 2205 psig

C. 2155 psig

D. 2080 psig

Answer:

B. 2205 psig

Notes:

"B" is correct, the Pressurizer Pressure controller sends a signal to open the Spray valve (CV-1008) at 2205 psig.

"A" is incorrect but plausible, the Electromatic Relief Valve (ERV) valve closes at this pressure.

"C" is incorrect but plausible since the Pressurizer Spray valve closes at this pressure.

"D" is incorrect. This is plausible since the Pressurizer Spray valve opens at this pressure but only if it opened due to a MFW pump trip when greater than 80% power. .

This question matches the K/A since the applicant must know when the presurizer spray valve normally opens to be able to monitor automatic operation.

References:

1103.005, Pressurizer Operation

History:

New for 2017 RO Re-exam

QID: 0307 **Rev:** 1 **Rev Date:** 4/9/17 **Source:** Bank **Originator:** J. Cork

TUOI: A1LP-RO-RPS **Objective:** 11 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 012 **System Title:** Reactor Protection System

Description: Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Trip logic circuits

K/A Number: K6.03 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 37 **SRO:**

Given:

* Plant is at 100% power.
* "B" Reactor Protection System channel declared inoperable and cannot be repaired until next month.
* A surveillance test on "D" Reactor Protection System channel is in progress.

What is the Reactor Protection System bistable trip logic under these conditions?

A. Two out-of-two

B. Two out-of-four

C. One out-of-two

D. One out-of-three

Answer:

C. One out-of-two

Notes:

"C" is correct. The reactor protection system logic is normally 2 of 4 required to trip the reactor. The "D" RPS channel is placed in bypass to perform the surveillance test, therefore it cannot trip. Since only one channel can be in bypass at a time, the "B" RPS channel will be tripped due to being inoperable to comply with Tech Specs. The trip logic will therefore be 1 out of 2.

"A" is incorrect but plausible if the applicant does not consider that the "B" channel is already tripped.

"B" is incorrect but plausible if the applicant merely recalls the normal trip logic and does not apply any of the conditions listed.

"D" is incorrect but plausible if the applicant does not consider that the "D" channel is in bypass during a surveillance and can not trip.

Re-ordered answer choices. Revised thrid bullet due to validator comment, used to say "becomes inoperable due to NI-6 failing high", now says simply "declared inoperable yesterday, cannot be repaired until next month". Changed order of bullets so inoperable channel is before surveillance test.

This question matches the K/A since it concerns the RPS and the affect a malfunction has upon the trip logic.

References:

STM-63, Reactor Protection System

History:

Developed for 1999 exam.
Selected for use in 2005 RO exam as replacement.

Selected for 2017 RO Re-exam

QID: 1131 **Rev:** 0 **Rev Date:** 3/6/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-ESAS **Objective:** 6 **Point Value:** 1

Section: 3.2 **Type:** Reactor Coolant System Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation System

Description: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

K/A Number: 2.2.39 **CFR Reference:** 41.7 / 41.10 / 43.2 / 45.13

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.5 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 38 **SRO:**

Given Unit 1 is operating at 100% power.

ESAS Analog 2 RC pressure transmitter fails HIGH.

What action below will be in compliance with Technical Specifications for the above conditions?

A. Initiate administrative controls to document and correct the failure.

B. Initiate plant shutdown to be in Mode 3 within 6 hours.

C. Place the analog instrument channel in trip.

D. Place associated components in their ES positions..

Answer:

C. Place the analog instrument channel in trip.

Notes:

"C" is correct per Tech Spec 3.3.5. If the RCS pressure transmitter fails high, then the low RCS pressure parameter in Table 3.3.5-1 cannot perform it's function (won't trip) so the associated analog channel must be tripped within one hour per 3.3.5 required action A.1.

"A" is incorrect, yet plausible since this would be correct if the transmitter had failed low, this results in an analog channel trip since the channel is already in the required state for the TS required action only administrative controls are necessary. But the RCS pressure transmitter failed high making this incorrect. Reference OP-1105.003, Engineered Safeguards Actuation System

"B" is incorrect, yet plausible since 3.3.5 required action B.1 requires this for more than one inoperable analog channel, but only one channel is inoperable.

"D" is incorrect, yet plausible as this is one hour required action A.1 from 3.3.7 for an inoperable digital ESAS channel.

Modified QID 350 by making it RO level, changed stem to be a one hour action statement. Changed the RCS pressure instrument from failing low to failing high so "A" is no longer correct, modified "B" so that it is more like 3.3.5 required action B.1, changed "C" so that it is required action A.1 for 3.3.5 (now the correct answer), and revised "D" so it is like required action A.1 for LCO 3.3.7.

This matches the K/A since it is about a failed ESAS (ESFAS) channel requiring entry into a one hour Tech Spec action statement.

References:

Technical Specification 3.3.5

History:

Modified QID 350 for 2017 RO Re-exam.

QID: 1153 **Rev:** 0 **Rev Date:** 4/3/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 14 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation

Description: Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate.

K/A Number: A4.01 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 4.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.8 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 39 **SRO:**

Given:
* Unit 1 tripped from 100% power
* ESAS channels 1 though 8 have actuated

CBOT is performing RT-10, Verify Proper ESAS Actuation.

He reports BWST Outlet valve, CV-1408, has failed to open.

What action is required per RT-10?

A. Obtain CRS permission, override ES, open BWST Outlet valve (CV-1408).

B. Override ES, open BWST Outlet valve (CV-1408).

C. Obtain CRS permission, override ES, stop HPI pump (P36B) and LPI pump (P34B).

D. Override ES, stop HPI pump (P36B) and LPI pump (P34B).

Answer:

C. Obtain CRS permission, override ES, stop HPI pump (P36B) and LPI pump (P34B).

Notes:

"C" is correct. RT-10 requires obtaining permission to override ES, then stop HPI and LPI pumps. Permission is required due to taking the pumps to stop which is not their ES position. The pumps are stopped due to CV-1408 has a relatively long stroke time and the pumps should be stopped ASAP to avoid any damage due to loss of suction. A previous step said to verify CV-1408 open and thus the operator should have taken the handswitch to open to attempt to open the valve.

"A" is incorrect, plausible since open is the desired position but there is no requirement to obtain CRS permission to place a component in its' required ES position. Since open is the ES position, there is no reason to override ES. For CV-1407 and CV-1408 one cannot take the valves to close since the ES signal opens series contacts preventing the handswitch from taking the valve to close but the contacts from ES in the open circuit are in parallel to the handswitch so there is nothing to prevent the handswitch from working.

"B" is incorrect, plausible since open is the desired position but again, there is no reason to override ES.

"D" is incorrect, this is plausible since this is the correct action but CRS permission must be obtained prior to placing any component in any position other than it's required ES position.

This question matches the K/A since it involves an ESFAS component which has failed to operate and the actions which must be taken in response to this failure.

References:

1202.012, Repetitive Tasks

History:

New question for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0256 **Rev:** 1 **Rev Date:** 4/4/17 **Source:** Bank **Originator:** D. Slusher

TUOI: A1LP-RO-MSSS **Objective:** 3 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 022 **System Title:** Containment Cooling System

Description: Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system.

K/A Number: K1.01 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 40 **SRO:**

Why are Decay Heat Cooler Outlet Valves SW-22A and SW-22B throttled during normal operation?

A. To maintain adequate service water flow to the Reactor Building Coolers when ES actuates.

B. To raise Service water flow to the Auxiliary Cooling Water System during normal operation.

C. To reduce reactor coolant to service water differential temperature when ES actuates.

D. To maintain the Decay heat coolers full and reduce the chance of water hammer.

Answer:

A. To maintain adequate service water flow to the Reactor Building Coolers when ES actuates.

Notes:

"A" is correct, SW flow to the DH coolers are throttled via SW-22A/B because flow to ES components (specifically the RB coolers) may not be adequate. The valves are marked so that they will not be throttled below the minimum required for the DH coolers. The RB coolers are normally cooled by chilled water and transfer to Service Water upon ESAS actuation (this is the cause-effect relationship between the two systems).

"B" is incorrect, while it is true that ACW demand will be higher during normal operation, this is not the reason for throttling the DH cooler outlets.

"C" is incorrect because while throttling the valves will reduce SW flow to the coolers and thus raise DH outlet temperature, it is not reduce the service water to RCS differential temperature.

"D" is incorrect because while water hammer of service water piping is a concern with SW to the RB coolers, that is not the reason for throttling these valves.

Changed order of answers.

This question matches the K/A since the question requires knowledge of the reason for throttling Service Water (SWS) to the DH coolers, i.e., to ensure adequate flow to the RB coolers (CCS) during an ESAS actuation.

References:

STM 1-09, Reactor Building Ventilation
1309.013, Unit 1 Service Water Flow Test

History:

Used in 1999 exam.
Modified from ExamBank, QID# 1519.
Selected for 2005 RO re-exam. KA 022 A1.04

Selected for 2011 RO Exam.
Seleted for 2017 RO Re-exam

QID: 0705 **Rev:** 1 **Rev Date:** 4/8/17 **Source:** Bank **Originator:** Pullin

TUOI: A1LP-RO-RBS **Objective:** 4 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray System (CSS)

Description: Knowledge of power supplies to the following: MOVs

K/A Number: K2.02 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 41 **SRO:**

Which of the following supply power to Containment Spray Motor Operated Valves
CV-2400 (B RB Spray Block) and CV-2401 (A RB Spray Block)?

A. B6371 and B5371

B. B6171 and B5171

C. B5271 and B6271

D. B6571 and B5571

Answer:

B. B6171 and B5171

Notes:

B is correct as these are the correct power supplies for the spray block valves.

A is incorrect but plausible since these are two other ES powered MCCs.

C is incorrect but plausible since these are two other ES powered MCCs.

D is incorrect but plausible since these are two other ES powered MCCs.

Revised "A" and "C" distractors since the MCCs were non-ES and thus not plausible.

This matches the K/A because it asks what the power supplies are for Containment Spray MOVs.

References:

1107.002, ES Electrical System Operation

History:

New for the 2008 RO Exam.
Selected for 2017 RO Re-exam

QID: 1155 **Rev:** 0 **Rev Date:** 4/4/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-STEAM **Objective:** 12 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 039 **System Title:** Main and Reheat Steam

Description: Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity

K/A Number: K5.08 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 42 **SRO:**

Unit 1 is at 100% power.

A Main Steam leak occurs.

Which of the following summarizes the effects the operators will observe in the control room in the first five (5) minutes?

A. Reactor power will rise, main generator MW will stay the same

B. Reactor power will drop, main generator MW will drop

C. Reactor power will rise, main generator MW will drop

D. Reactor power will drop, main generator MW will stay the same

Answer:

C. Reactor power will rise, main generator MW will drop

Notes:

"C" is correct. With reactor power at 100%, the increase in steam flow will cause primary temperature to decrease. With a negative moderator temperature coefficient, the drop in primary temperature will result in increased moderator density, and a reactor power increase. Main Turbine control system receives pulses from the ICS to control header pressure at setpoint, the steam leak will cause header pressure to drop, the control system will then close the Governor Valves to raise header pressure, and thus Main Generator Megawatts will drop.

"A" and "D" distractors are plausible if the applicant incorrectly believes the Main Turbine control system will function like many control systems and will compensate for the steam leak by opening the governor valves to maintain Main Generator output.

"A" distractor is also plausible since reactor power has the correct trend.

"B" distractor is plausible since the main generator output trend is correct.

This question matches the K/A since it tests the applicant's knowledge of the operational effects of steam removal, including reactor power.

References:

STM 1-65, Integrated Control System

History:

New question for 2017 RO re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0656 **Rev:** 0 **Rev Date:** 12/12/06 **Source:** Bank **Originator:** Possage

TUOI: A1LP-RO-EFIC **Objective:** 4 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 039 **System Title:** Main and Reheat Steam System

Description: Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Automatic isolation of steam line.

K/A Number: K4.05 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 43 **SRO:**

Unless bypassed, the setpoint for Main Steam Line Isolation (MSLI) actuation is at a decreasing SG pressure of:

A. 700 psig

B. 650 psig

C. 600 psig

D. 550 psig

Answer:

C. 600 psig

Notes:

"C" is correct MSLI and EFW actuate at 600 psig decreasing SG pressure.

All of the other choices are simply bracketed around the correct setpoint in 50 psig increments.

This matches the K/A as it requires knowledge of the setpoint for main steam line isolation.

References:

1105.005, Emergency Feedwater Initiation and Control

History:

Selected for 2007 RO Exam. Direct from regular exambank QID# ANO-OPS1-779.
Selected for 2017 RO Re-exam

QID: 0911 **Rev:** 1 **Rev Date:** 3/7/17 **Source:** New **Originator:** Possage/Cork

TUOI: A1LP-RO-EOP01 **Objective:** 10 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 059 **System Title:** Main Feedwater System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overfeeding event.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 44 **SRO:**

Given:

* Reactor tripped from 100% due to a Turbine Trip

* Immediate Actions are complete

* ATC reports ‘A’ S/G level rising rapidly and approaching 410 inches

What action is required by 1202.001, Reactor Trip, and why?

A. Trip ‘A’ MFWP to mitigate overcooling event

B. Trip Both MFWPs to mitigate overcooling event

C. Trip ‘A’ MFWP to prevent feedwater carryover into Main Steam Line

D. Trip Both MFWPs to prevent feedwater carryover into Main Steam Line

Answer:

D. Trip Both MFWPs to prevent feedwater carryover into Main Steam Line

Notes:

"D" is correct per 1202.001 step 11 and the corresponding EOP bases document.

"A" is incorrect but plausible as this action is taken in the Overcooling EOP for overfeed events but if SG level is approaching the high limit both pumps are tripped to prevent putting water in the steam lines.

"B" is incorrect but plausible as this is the correct action but the wrong reason.

"C" is incorrect but plausible as this is the correct reason but both pumps should be tripped.

This question matches the K/A since the conditions describe an overfeeding event and requires the ability to predict what is happening and the knowledge of the proper procedural actions to mitigate the event.

References:

1202.001, Reactor Trip
1202.001 Bases Document

History:

New for 2014 Exam - NOT Used, NRC did not like the "A" & "B" distractors.
New for 2017 RO Re-exam, modified A & B distractors. Cork 3/7/17

QID: 0709 **Rev:** 2 **Rev Date:** 4/4/17 **Source:** Bank **Originator:** Thompson

TUOI: A1LP-RO-EFIC **Objective:** 29 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 061 **System Title:** Auxiliary / Emergency Feedwater System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: AFW flow/motor amps.

K/A Number: A1.05 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 45 **SRO:**

Given:

* Reactor has tripped.

* RCPs have been secured.

* SCM is adequate.

"A" OTSG pressure is 925 psig
"A" OTSG level is 200" and rising
EFW feed rate is 4"/minute

"B" OTSG pressure is 845 psig
"B" OTSG level is 215" and rising
EFW feed rate is 6"/minute

Per RT-5 what OTSG flow rates are required for the above conditions?

A. "A" OTSG filling at 3"/minute,
"B" OTSG filling at 5"/minute.

B. "A" OTSG filling at 5"/minute,
"B" OTSG filling at 3"/minute.

C. "A" OTSG filling at 4"/minute,
"B" OTSG filling at 6"/minute.

D. "A" OTSG filling at 6"/minute,
"B" OTSG filling at 4"/minute.

Answer:

B. "A" OTSG filling at 5"/minute,
"B" OTSG filling at 3"/minute.

Notes:

"B" has the correct fill rates based on Table 1 in RT-5 and knowledge of the proper operation of the fill rate controller. The controller automatically fills the SGs on a linear scale from 2" to 8" based on SG pressure. As SG pressure rises from 800 psig to 1050 psig the fill rate rises linearly to prevent overfill and overcooling. Therefore with A OTSG pressure at 925 psig the fill rate would be 925 - 800 = 125 x .024"/psig = 3 + 2 = 5"/minute. B OTSG fill rate is calculated the same way: 845 - 800 = 45 x 0.024"/psig = 1 + 2 = 3"/minute. The 0.024"/psig is derived by: 1050 - 800 = 250, 8" - 2" = 6", 6"/250 = 0.024"/psig

"A" is incorrect yet plausible as these are the correct fill rates but for the wrong SGs.

"C" is incorrect yet plausible if the candidate believed the given fill rates are correct.

D are incorrect yet plausible if the candidate believed the given fill rates should be swapped.

Revised question by changing B SG pressure from 750 psig to 845 psig, changed all fill rates in answer choices. Revised question to be in compliance with latest format expectations.

This question matches the K/A since it requires candidate to evaluate given condition and monitor/predict the proper EFW flow rates for those conditions.

References:

1105.005, Emergency Feedwater Initiation and Control
1202.012, Repetitive Tasks, RT-5

History:

New for 2008 RO Exam.
Selected for 2017 RO Re-exam.

QID: 1133 **Rev:** 0 **Rev Date:** 3/7/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ELECD **Objective:** 14g **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** AC Electrical Distribution

Description: Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: DC system.

K/A Number: K3.03 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 46 **SRO:**

If Unit 1 experiences a Blackout, what is the minimum time the 125 VDC Vital Batteries (D06 and D07) are designed to supply power to their loads, and what major loads will be secured to extend battery capacity?

A. 1 hour
Main Turbine Emergency Lube Oil Pump (P20)

B. 2 hours
Main Feed Pump Emergency Oil Pumps (P28A & P28B)

C. 1 hour
Main Feed Pump Emergency Oil Pumps (P28A & P28B)

D. 2 hours
Main Turbine Emergency Lube Oil Pump (P20)

Answer:

B. 2 hours
Main Feed Pump Emergency Oil Pumps (P28A & P28B)

Notes:

"B" is the correct answer per the SAR section on the 125 Volt DC System and step 41 of the Blackout EOP. The batteries will last a minimum of two hours and the DC powered oil pumps on the Main FW Pumps are secured to extend battery life.

"A" is incorrect, plausible since the time given is close to the actual time and plausible since the name of the Main Turbine Emergency Lube Oil Pump is similar to the MFW pump emergency lube oil pumps but the P20 pump is powered from B56, not DC.

"C" is incorrect, yet plausible since it has the correct loads to secure but has the incorrect time.

"D" is incorrect, yet plausible as this is the correct time but the incorrect load.

This question matches the K/A since it requires knowledge of DC design (how long the batteries will last and what are DC loads) with respect to a loss of the AC distribution system.

References:

1202.008, Blackout
SAR, 8.3.2

History:

New question for 2017 RO Re-exam

QID: 0336 **Rev:** 1 **Rev Date:** 3/23/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** DC Electrical Distribution

Description: Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.0 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 47 **SRO:**

Given:

Plant is operating at 100% power when the following occurs:

- Turbine Lockout Relay DC Failure (K04-B5),
- D01 Undervoltage (K01-A7),
- D01 Trouble (K01-D7),
- Loss of breaker position indicator lights for plant buses on left side of C10.

Which action, with the correct reason for the action, is required to be performed?

A. Start #1 Diesel Generator from C-10 due to loss of power to undervoltage relays.

B. Transfer D11 to its Emergency Power Supply to energize generator lockout relays.

C. Trip the Generator Output Breakers to prevent the Main Generator from motoring.

D. Line up Battery Charger D03A or D03B to the D01 Bus to restore DC power.

Answer:

B. Transfer D11 to its Emergency Power Supply to energize generator lockout relays.

Notes:

"B" is correct, this action is the use of a design feature: manual transfer of power for 125VDC panel D11 from it's normal power supply of D01 (which has been lost) to it's Emergency Supply (D02). This is the most expedient method of restoring power to D11 and is the action prescribed in 1203.036 for Loss of D01. This action cross-connects both ES trains of DC and is only taken due to the rather significant consequences of losing D01 combined with a loss of generator lockout relays, control power to EDG and ES bus A3, and inadvertent ESAS actuation. Additionally, this action is only taken when D11 is shown NOT to be faulted as evidenced by the presence of the D01 undervoltage alarm so as not to transfer a fault from one train to the other.

"A" is incorrect, although plausible since control power will be lost to #1 EDG and this could be construed as a loss of power to the DG undervoltage relays.

"C" is incorrect, the output breakers may or may not trip on a loss of D01, but plausible since this action is taken within 1203.036 if transfer of D11 to D02 is Unsuccessful.

"D" is incorrect, this is plausible since it might restore power to D01 but it would take too much time and may not work if all of "A" train power is lost.

Swapped answer choices A and B.

This question matches the K/A since it requires knowledge of a DC design feature: manual transfer of DC control power from one train to the other.

References:

1203.036, Loss of 125V DC

History:

Used in 1999 exam. Direct from ExamBank, QID# 1891
Selected for 2005 RO exam, but not used.
Selected for 2007 RO exam.
Selected for 2017 RO Re-exam

QID: 0712 **Rev:** 1 **Rev Date:** 4/9/2008 **Source:** Bank **Originator:** Steve Pullin

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 2 **Type:** Generic Knowledge and Abilities

System Number: 063 **System Title:** DC Electrical Distribution

Description: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

K/A Number: 2.2.42 **CFR Reference:** 41.7 / 41.10 / 43.2 / 43.3 / 45.3

Tier:	2	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 48 **SRO:**

With Unit 1 at 100% power, which of the following conditions requires entry into Technical Specification 3.8.4, DC Systems - Operating?

A. D03A, Battery Charger inoperable and
D04A, Battery Charger inoperable.

B. D03B, Battery Charger inoperable and
D04B, Battery Charger inoperable.

C. D03A, Battery Charger inoperable and
D03B, Battery Charger inoperable.

D. D03B, Battery Charger inoperable and
D04A, Battery Charger inoperable.

Answer:

C. D03A, Battery Charger inoperable and
D03B, Battery Charger inoperable.

Notes:

"C" is correct. Two chargers inoperable on the same train render the train inoperable and require TS entry.

"A" is incorrect. One battery charger inoperable on each train does not require TS entry. Plausible if candidate believes both chargers are on the same train or if any charger being inop requires TS entry per given train.

"B" is incorrect. One battery charger inoperable on each train does not require TS entry. Plausible if candidate believes both chargers are on the same train or if any charger being inop requires TS entry per given train.

"D" is incorrect. One battery charger inoperable on each train does not require TS entry. Plausible if candidate believes both chargers are on the same train or if any charger being inop requires TS entry per given train.

This question matches the K/A since the candidate must know what parts make up a DC electrical system and what would be an entry condition into Tech Specs.

Re-worded distractors A and D to make them more plausible.

References:

Technical Specification 3.8.4

History:

New for 2008 RO Exam
Selected for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1150 **Rev:** 0 **Rev Date:** 3/24/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 10 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generator

Description: Ability to manually operate and/or monitor in the control room: Synchroscope.

K/A Number: A4.03 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 49 **SRO:**

Given:
* DG2 is running during a monthly test per 1104.036, Emergency Diesel Generator Operation.
* The CBOT is ready to parallel DG2 to the grid.

In accordance with 1104.036, Supplement 2, the voltage regulator (Incoming) should be adjusted to ________(1)___________volts of A4 (Running) voltage.
AND
the governor control should be adjusted until frequency is 60 Hz with the synchroscope rotating _________(2)_____________ direction.

A. (1) match or be no greater than 50
(2) slowly in FAST

B. (1) match or be no greater than 50
(2) fast in SLOW

C. (1) match or be no greater than 20
(2) slowly in FAST

D. (1) match or be no greater than 20
(2) fast in SLOW

Answer:

C. (1) match or be no greater than 20
(2) slowly in FAST

Notes:

"C" is correct per steps 2.15.6 and 2.15.7 of 1104.036 Supplement 2 which states that DG2 voltage must match or be no greater than running voltage and the synchroscope should be rotating slowly in the fast direction.

"A" is incorrect, plausible since part (2) is correct but the part (1) voltage direction does not match the direction in 1104.036.

"B" is incorrect, plausible since part (1) is similar to the voltage direction in 1104.036 but the limit of 50 is too high. Part (2) is the opposite of the correct synchroscope requirement.

"D" is incorrect, plausible since the part (1) voltage requirement is correct per 1104.036 but part (2) is the opposite of the correct synchroscope requirement.

This question matches the K/A since the question requires the applicant to know how to monitor for proper synchroscope operation of an Emergency Diesel Generator.

References:

1104.036 Emergency Diesel Generator Operation

History:

New for 2017 RO Re-exam

QID: 0136 **Rev:** 0 **Rev Date:** 06/30/94 **Source:** Bank **Originator:** M. Cooper

TUOI: ANO-1-LP-RO-RMS **Objective:** 2 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring System

Description: Knowledge of the physical connections and/or cause-effect relationships between the PRM system and the following systems: Those systems served by PRMs.

K/A Number: K1.01 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 50 **SRO:**

Given:

- LOCA inside the Reactor Building.
- RE-3814, Service Water Loop I Radiation Monitor alarms.
- SW Loop II indications are normal.

Which of the conditions below when combined with the above condition would make it necessary to isolate the "A" & "B" RB Emergency Coolers?

A. RE-3815, Loop II Service Water monitor is also in alarm.

B. RE-7471, RB ATMOS Gaseous Detector is in alarm.

C. RE-3618, Discharge Flume monitor is also in alarm.

D. RE-8020, RB area monitor is also in alarm.

Answer:

C. RE-3618, Discharge Flume monitor is also in alarm.

Notes:

"C" is the correct answer. Since Loop II is OK, then the confirmation of an actual release via the Discharge Flume monitor necessitates the isolation of Loop I.

"A" would not corroborate a problem with Loop I coolers and indicate a need to isolate the Loop I coolers.

"B" would be expected for a LOCA condition, but does not necessarily impact the SW side of the RB coolers.

"D" is incorrect but plausible since this does seem to confirm a problem with the RB coolers in the Reactor Building but the RB area monitors would be in alarm anyway due to the LOCA.

References:

1203.012I, Annunciator K10 Corrective Action

History:

Taken from Exam Bank QID # 2571
Used in 98 RO Re-exam
Used on 2004 RO/SRO Exam.
Selected for 2017 RO exam

QID: 0978 **Rev:** 0 **Rev Date:** 2/18/13 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-RMS **Objective:** 9 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 51 **SRO:**

During plant operation at 100% power:

- Annunciator alarm K07-A5 ("A" OTSG N-16 TROUBLE) is received with RI-2691 reading 5.5 x 10^3 cpm and stable
- Annunciator alarm K07-A6 ("B" OTSG N-16 TROUBLE) is received with RI-2692 reading 4 x 10^3 cpm and slowly rising

After operators have taken both RI-2691 and RI-2692 from GROSS to ANALYZER per 1203.012F, ANNUNCIATOR K07 CORRECTIVE ACTION, the following readings are observed:

- RI-2691 is reading 5.5 x 10^3 cpm and stable
- RI-2692 is reading 3.5 x 10^3 cpm and slowly rising

Which of the following explains the observations above?

A. RI-2692 has a low voltage condition as confirmed by a drop in count rate following completion of 1203.012F.

B. RI-2691 is operating properly and a steam generator tube leak is confirmed as detected by a steady 5.5 x 10^3 cpm reading.

C. RI-2691 has failed to 5.5 x 10^3 cpm as confirmed by no change in count rate.

D. Both RI-2691 and RI-2692 are operating properly with a confirmed steam generator tube leak present in both steam generators with different leak rates.

Answer:

C. RI-2691 has failed to 5.5 x 10^3 cpm as confirmed by no change in count rate.

Notes:

"C" is correct, 1203.012F directs the rate meter mode to be changed from Gross to Analyzer if the reactor is critical and RI-2691/2692 in Alert or High alarm. This will screen out all activity except N-16 thus the reading would be lower after the change.

"A" is incorrect. A drop in count rate is expected after completion of 1203.012F where the rate meter mode is changed from Gross to Analyzer. A low voltage condition would cause the rad monitor to cease functioning.

"B" is incorrect, but plausible since RI-2692 reading upscale indicates a tube leak is occuring but RI-2691 would read lower after completion of 1203.012F.

"D" is incorrect, but plausible as the indications are that a tube leak is occuring, but RI-2691 would read lower after completion of 1203.012F.

This question matches the K/A since the applicant must know how these monitors detect radiation levels, specifically how they detect N-16 gammas, and requires them to know how the controls (selection switch) changes the reading to arrive at the correct answer: one of the process radiation monitors has failed and this monitor should not be used in determining if design limits have been exceeded.

References:

1203.012F, Annunciator K07 Corrective Action
STM 1-62, Radiation Monitoring

History:

New for 2013 Exam
Selected for 2017 RO Re-exam

QID: 0046 **Rev:** 2 **Rev Date:** 4/9/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 3 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 076 **System Title:** Service Water System

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

K/A Number: A4.02 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 2.6 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 52 **SRO:**

Given:
* Degraded Power
* Both EDGs operating
* ESAS has NOT actuated
* P4C failed to start
* P4B out of service

Which of the following actions are procedurally required for the above conditons?

A. Close ACW Loop Isolation (CV-3643).

B. Close SW Loop II Isolation Valve (SW-10C).

C. Open SW Loop I & II Crossconnects (SW-5 and SW-6).

D. Cross-tie SW Loops at Makeup Pump (SW-14 thru SW-17).

Answer:

A. Close ACW Loop isolation (CV-3643).

Notes:

"A" is correct per the Degraded Power EOP, this action is taken to isolate ACW thereby reducing SW system flow and preventing runout of the sole remaining SW pump. There is no ES actuation so ACW isolation CV-3643 will not close automatically.

"C" is incorrect since this will only isolate the loop flow from P4C with the SW loops still cross-tied at the ICW coolers. This is plausible since it will reduce SW flow but is not required by 1202.007.

"B" is incorrect but plausible since this is a manual isolation which would reduce SW flow by isolating Loop II.

"D" is incorrect but plausible since this would ensure SW flow to the Makeup pumps since only one train of SW is operating but this is not directed by 1202.007.

Re-ordered answer choices due to multiple exam use.

This question matches the K/A since it involves manual operation of a Service Water valve in the control room.

References:

1202.007, Degraded Power

History:

Developed for 1998 RO/SRO Exam.
Revised after 9/98 exam analysis review.
Used in 98 RO Re-exam
Selected for use in 2002 RO exam.
Selected for use on 2007 RO Exam

Selected for 2011 RO Exam.
Selected for 2017 RO Re-exam

QID: 0534 **Rev:** 2 **Rev Date:** 4/9/17 **Source:** Modified **Originator:** NRC

TUOI: A1LP-RO-MSSS **Objective:** 1 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 076 **System Title:** Service Water System

Description: Knowledge of bus power supplies to the following: ESF-actuated MOVs.

K/A Number: K2.08 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 53 **SRO:**

Following a reactor trip, ES channels 1 through 4 have actuated.

While performing RT-10 to verify proper ES actuation, the CBOT notes the following on C16:

* HPI Block to P-32C (CV-1284) has no position indication.
* HPI Block to P-32D (CV-1285) has no position indication.

The CBOT suspects that this is due to a loss of power
to MCC load center _____ on the _____ train.

A. B52 / Red

B. B55 / Red

C. B62 / Green

D. B64 / Green

Answer:

C. B62 / Green

Notes:

"C", this is the correct power supply for both CV-1284 and CV-1285, both are powered from the same MCC, B62. One valve is even numbered and one is odd numbered so applicant must be familiar with power supplies in order to choose correct answer and eliminate distractors.

"A" is incorrect but plausible, this is an ES red train MCC which powers other HPI block valves.

"B" is incorrect but plausible as this is an ES red train MCC.

"D" is incorrect but plausible as this is and ES green train MCC.

Modified this question since the original had two implausible distractors (panel identification eliminated two). Changed valves from Service Water since these valves are in another question, replaced SW valves with HPI block valves. Replaced two distractors which could be eliminated simply by knowing which train they were.

References:

1107.002, ES Electrical System Operation
STM 1-32, Electrical Distribution

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.
Modified for 2005 RO re-exam.
Selected for 2017 RO Re-exam

QID: 0988 **Rev:** 0 **Rev Date:** 2/18/13 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-MSSS **Objective:** 10 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 078 **System Title:** Instrument Air System

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

K/A Number: K1.05 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 54 **SRO:**

What is the design purpose of the backup air accumulators for the MSIVs?

A. To ensure air from the MSIV backup accumulators operate the Atmospheric Dump Valve for 30 minutes due to loss of Instrument air

B. To ensure air from the MSIV backup accumulators will maintain Main Steam Isolation valve open for 30 minutes due to loss of Instrument air

C. To ensure air from the MSIV backup accumulators to operate the Atmospheric Dump Valve for 60 minutes due to loss of Instrument air

D. To ensure air from the MSIV backup accumulators will maintain Main Steam Isolation valve open for 60 minutes due to loss of Instrument air

Answer:

B. To ensure air from the MSIV backup accumulators will maintain Main Steam Isolation valve open for 30 minutes due to loss of Instrument air

Notes:

Answer A is incorrect. The purpose is to ensure the MSIV's have enough air to hold them open for 30 minutes not for ADV control this can be done locally with no Instrument air pressure although the accumulators supply air to the ADV's .

Answer B is correct. The purpose is to ensure the MSIV's have enough air to hold them open for 30 minutes. They are spring to close valves and air to open and maintain open.

Answer C is incorrect. Due to reasons above and the time is too long.

Answer D is incorrect Due to reasons above and the time is too long.

References:

STM1-15, Main Steam System

History:

New for 2013 Exam
Selected for 2017 RO Re-exam

QID: 0867 **Rev:** 1 **Rev Date:** 4/5/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-EOP05 **Objective:** 9 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment

Description: Ability to (a) predict the impacts of the following malfunction or operations on the containment system, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 55 **SRO:**

The reactor has tripped from 100% power with the following conditions:
* RCS pressure, 1750 psig and lowering
* CETs, 500 °F and lowering
* RB pressure is 19.8 psia and rising

Which of the following correctly lists the ESAS channels to verify actuated AND an appropriate corrective action to take based on the conditons given?

A. Channels 1, 2, 3, 4;
Trip all running RCPs

B. Channels 1, 2, 3, 4, 5, 6;
Restore Service Water and ACW per 1104.029 Exhibit B
"Restoring SW to ICW Following ES Actuation"

C. Channels 1, 2, 3, 4;
Restore Service Water and ACW per 1104.029 Exhibit B
"Restoring SW to ICW Following ES Actuation"

D. Channels 1, 2, 3, 4, 5, 6;
Trip all running RCPs

Answer:

D. Channels 1, 2, 3, 4, 5, 6;
Trip all running RCPs

Notes:

"D" is correct, with Reactor Building pressure greater than 18.7 psia, ESAS channels 1 thru 6 will actuate and ESAS will be verified to be operating correctly per RT-10. RT-10 states that if either Channnel 5 or 6 have actuated to secure all running RCPS due to ICW isolation. B&W ESAS channels 1-4 actuate on low RCS pressure (1550 psia) and isolate some RB penetrations, this is the equivalent of the Westinghouse Phase A isolation. If RB pressure exceeds 18.7 psia, then ESAS Channels 1-4 AND channels 5&6 actuate with channels 5&6 being the equivalent of a Westinghouse Phase B isolation. FYI, the conditions given are indicative of a steam line break in the building with RCS pressure low but with plenty of subcooling margin and RB pressure rising.

"A" is incorrect ESAS channels 5 and 6 also should be actuated, but is plausible since Trip all running RCPs is the correct action to take.

"B" is incorrect but plausible since ESAS Channels 1 thru 6 are the correct channels but (per RT-10) Service Water is restored to ICW only if channels 5 nor 6 have NOT actuated.

"C" is incorrect ESAS channels 5 and 6 also should be actuated, but plausible since SW is restored to ICW if only channels 1-4 have actuated.

Revised, this question did not match the K/A originally. Replaced "A LOCA is in progress" with "the reactor

tripped from 100%" and revised condtions to reflect a steam line break; deleted "ESAS has actuated" from conditions; revised A & B as they were implausible since there is no scenario where channels 7 - 10 would actuate alone. Revised all answer choices with actions from RT-10 appropriate to the conditions.

This question matches the K/A since it requires the applicant to predict the impact of rising RB pressure on the Reactor Building (containment) and requires the applicant to know the actions from RT-10 that are taken specifically if ESAS channels 5 or 6 actuate (phase B isolation).

References:

1202.012, Repetitive Tasks

History:

New question written by NRC for 2013 exam
Modified for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1134 **Rev:** 0 **Rev Date:** 3/9/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RPS **Objective:** 4 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive System

Description: Knowledge of the physical connections and/or cause effect relationships between the CRDS and the following systems: CCWS must be cut in before energizing CRDS.

K/A Number: K1.09 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 56 **SRO:**

Unit 1 is recovering from a reactor trip and preparations are underway to re-start.

Which of the follow annunciators ALONE, if in alarm, will prevent closing the CRD AC trip breakers?

A. CRD PUMP PREFILTER DP HI (K08-E1)

B. CRD COOLING RETURN FLOW LO (K08-C1)

C. CRD COOLING RETURN TEMP HI (K08-B1)

D. CRD CLNG SUPP TEMP HI / ROOM FLOOD (K08-A1)

Answer:

B. CRD COOLING RETURN FLOW LO (K08-C1)

Notes:

"B" is correct. FS-2222 causes this annunciator when return flow is ≤127.9 gpm and is an interlock to closing the CRD AC trip breakers. If the alarm is not in, then the AC trip breakers will not close. This is to prevent energizing the CRDM's with inadequate cooling water flow.

"A" is incorrect, although plausible since a clogged filter could be indicative of low flow but the prefilters have bypass valves which open when this alarm comes in to prevent a low flow condition. This alarm has no interlock with the AC trip breakers.

"C" is incorrect, although plausible since a low flow condition could cause this alarm but this alarm could also be caused by inadequate Service Water flow to the Non-Nuclear ICW cooler. This alarm has no interlock with the AC trip breakers.

"D" is incorrect, although plausible since a low flow condition could cause this alarm but this alarm could also be caused by inadequate Service Water flow to the Non-Nuclear ICW cooler. This alarm has no interlock with the AC trip breakers.

This question matches the K/A since it requires knowledge of the cause/effect relationship (interlock) between the CRD system and the ICW (CCWS) system.

References:

1105.009, CRD System Operating Procedure

History:

New question for 2017 RO Re-take exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1135 **Rev:** 0 **Rev Date:** 3/9/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ICS **Objective:** 30 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 002 **System Title:** Reactor Coolant System

Description: Knowledge of the effect that a loss or malfunction of the RCS will have on the following: Fuel.

K/A Number: K3.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	4.2	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.5	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 57 **SRO:**

What feature in the Integrated Control System (ICS) is designed specifically to respond to malfunctions which will have an effect on the nuclear fuel?

A. Feedwater Cross-Limits

B. BTU Limit Alarm

C. Delta Tc Controller

D. Rapid Feedwater Reduction

Answer:

C. Delta Tc Controller

Notes:

"C" is correct. The Delta Tc Controller adjusts FW flow to each SG based on the amount of RCS flow through that SG in order to maintain Delta Tc at minimum. Maintaining Delta Tc at minimum ensures that Quadrant Tilt Limits are not challenged and thus challenges to fuel are minimized.

"A" is incorrect yet plausible since Feedwater Cross-Limits can limit Reactor power based on available Feedwater. The purpose of this however is to ensure heat production and heat removal are balanced so that system transients are minimized and power production can continue.

"B" is incorrect yet plausible since BTU Limits calculations are greatly affected by RCS flow and a reduction in RCS flow will bring in the BTU Limits alarm(s). However, the BTU Limits alarms are to ensure 35°F superheat is available, not nuclear fuel considerations.

"D" is incorrect yet plausible since the RFR rapidly reduces Feedwater following a Reactor Trip which reduces the overcooling of the primary from the secondary. Lowering RCS pressure from overcooling does place the fuel closer to DNB but the primary concern for overcooling the RCS is stresses to the Reactor vessel, not the fuel.

This question matches the K/A since it asks the applicant what ICS feature is designed to keep RCS malfunctions from affecting the nuclear fuel. The applicant must know that a difference in Tcold temperatures caused by a reactor coolant flow malfunction (i.e., a Reactor Coolant Pump trip) will cause reactor core quadrant tilt to become worse and that the purpose of the Delta Tc Controller is to minimize differences in Tcold temperatures between the loops which, in turn, minimizes Quadrant Power Tilts in the reactor core.

References:

STM 1-64, Integrated Control System

History:

New question for 2017 RO Re-exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1136 **Rev:** 0 **Rev Date:** 3/15/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-CRD **Objective:** 14 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 014 **System Title:** Rod Position Indication

Description: Knowledge of the operational implications of the following concepts as they apply to the RPIS: RPIS independent of demand position

K/A Number: K5.02 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 58 **SRO:**

Given:

* Unit 1 is at 100% power

* 1105.009, Supplement 1, Absolute and Relative Position Indication Comparison, is in progress.

* The CBOT reports Control Rod 7-5 is indicating 95% RPI and 100% API via the PI panel.

* All Group 7 Out Limit lights are ON.

Which of the following actions are procedurally required for the above conditions?

A. Declare Control Rod 7-5 inoperable and enter T.S. LCO 3.1.4.

B. Realign Control Rod 7-5 with the rest of Group 7.

C. Adjust Control Rod 7-5 RPI to agree with API at PI panel.

D. Place S-2 toggle switch for Control Rod 7-5 in bypass.

Answer:

C. Adjust Control Rod 7-5 RPI to agree with API at PI panel.

Notes:

"C" is correct, RPI stands for Relative Position Indication and corresponds to rod demand. API stands for Absolute Position Indication and is determined from actual rod position. With all Group 7 Out Limit lights on, a validation of API position being accurate exists since the 100% zone indication comes from the out limit switch, thus RPI is not accurate and should be adjusted per Supplement 1 step 3.1.3.A, third bullet.

"A" is incorrect, yet plausible if applicant cannot recall threshold for operability (6.5%) per TS 3.1.4 and believes API to be invalid.

"B" is incorrect, yet plausible since this is performed for rod misalignments per Supplement 1 step 3.1.3.A (first bullet) but in this case the rod is not misaligned.

"D" is incorrect, yet plausible as this is done when API is determined to be faulty but it is the RPI which is invalid.

This question matches the K/A since it involves the Rod Position Indication and requires applicant to have knowledge of the operational implication (adjust RPI vs. actions for API) of rod position indication independent of demand (RPI).

References:

1105.009, CRD System Operating Procedure
1203.003, Control Rod Drive Malfunction Action

History:

New question for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0464 **Rev:** 1 **Rev Date:** 3/15/17 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-NI **Objective:** 10 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation System

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

K/A Number: A1.02 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 59 **SRO:**

Given:

* Reactor startup is in progress.

* Count rate on NI-1 & NI-2 is 1e3 CPS

During the next rod pull, annunciator HI SUR ROD HOLD (K08-B2) alarms.
Simultaneously annunciator CRD WITHDRAWAL INHIBITED (K08-A2) alarms.

The cause of the alarms is an excessive rod pull causing SUR to reach the setpoint of:

A. 1 DPM on the source range monitors.

B. 2 DPM on the source range monitors.

C. 1 DPM on the intermediate range monitors.

D. 2 DPM on the intermediate range monitors.

Answer:

B. 2 DPM on the source range monitors.

Notes:

"B" is the correct answer, a SUR reaching or exceeding 2 DPM on the source range monitors will cause the HI SUR and CRD withdrawal inhibited alarms.

"A" is an incorrect answer, a SUR of 1 DPM on the source range monitors will not cause the HI SUR alarm to come in. This distractor is plausible since this is the maximum SUR allowed per limit and precaution 6.3 in 1102.008, Approach to Criticality.

"C" is an incorrect answer, a SUR of 1 DPM on the intermediate range monitors will not cause the HI SUR alarm to come in. Setpoint for the IR high SUR rod hold is 3 DPM. This distractor is plausible since this is the maximum SUR allowed per limit and precaution 6.3 in 1102.008, Approach to Criticality.

"D" is an incorrect answer, a SUR of 2 DPM on the intermediate range monitors will not cause the HI SUR alarm to come in. Setpoint for the IR high SUR rod hold is 3 DPM. This is plausible since this is the setpoint for the Hi SUR rod hold from the source range monitors.

This question matches the K/A since it test the ability of the applicant to monitor alarms and NI parameters (SUR) of automatic action which prevent exceeding design limits.

References:

1203.012G, Annunciator K08 Corrective Action

History:

Direct from regular exambank QID 1789.

Selected for use in 2002 RO/SRO exam.
Selected for 2011 RO Exam.
Selected for 2017 RO Re-exam

QID: 1137 **Rev:** 0 **Rev Date:** 3/16/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 5 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation

Description: Ability to monitor automatic operation of the NNIS, including: Automatic selection of NNIS inputs to control systems.

K/A Number: A3.01 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 60 **SRO:**

Given:

* Unit 1 is at 100% power

* Loop A RCS flow 65 e6 lbm/hr

* Loop B RCS flow 70 e6 lbm/hr

* Loop A Tave 580°F

* Loop B Tave 578°F

* Unit Tave 579°F

What causes SASS to select a specific Tave for control (vs. Unit Tave) and which Tave will be selected by the SASS Auto/manual transfer switch?

A. Loop A flow low, Loop B Tave

B. Loop A Tave high, Loop B Tave

C. Loop A Tave high, Loop A Tave

D. Loop A flow low, Loop A Tave

Answer:

A. Loop A flow low, Loop B Tave

Notes:

"A" is correct. SASS will automatically select the Loop Tave for the Loop with the highest RCS flow should either flow drop below 95%. Normally Unit Tave is used for control. Normal RCS loop flow is ~70 E6 lbm/hr, therefore Loop A flow is <95% and SASS will select Loop B for Tave control. This control function protects the core from excessive heat transfer based upon flux to flow.

"B" is incorrect, SASS does not select Tave control based upon which is the highest, it is based on highest RCS flow. This is plausible since most control selections are based upon the highest indication.

"C" is incorrect, SASS does not select Tave control based upon which is the highest, it is based on highest RCS flow. This is plausible since most control selections are based upon the highest indication.

"D" is incorrect, this is the correct cause for Tave control to change from Unit Tave to a specific loop Tave but it will swap to the loop with the highest flow, not the lowest.

This is a modified version of QID 77. All answers were changed to a 2x2 format. The loop with the lower flow was changed to A. The order of the reason and signal selection were reversed in the stem. Added 100% power condition. Changed RCS flow for the lower loop to be closer to the 95% threshold.

This question matches the K/A due to subject is NNI and the knowledge required is what will cause automatic selection of NNIS input (Tave) and which will be selected (monitor).

References:

STM 1-69, Non-Nuclear Instrumentation System

History:

Modified QID 77 for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1138 **Rev:** 0 **Rev Date:** 3/16/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-RBVEN **Objective:** 12 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 029 **System Title:** Containment Purge System

Description: Ability to verify that the alarms are consistent with the plant conditions.

K/A Number: 2.4.46 **CFR Reference:** 41.10 / 43.5 / 45.3 / 45.12

Tier:	2	**RO Imp:**	4.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 61 **SRO:**

Given:

* Unit 1 is in Mode 5.
* Fuel movement is NOT in progress.
* Commencing a Reactor Building Purge per 1104.033, Attachment B.

* CBOT starts RB Purge Supply Fan (VSF-2)
* Annunciator CONT BLDG EXHAUST FAN AUTO TRIP (K15-A3) alarms

Which of the following is a required action per Reactor Building Ventilation(1104.033) and is consistent with the above alarms?

A. Place RB Purge Supply Fan (VSF-2) handswitch in OVERRIDE.

B. Verify RB Purge Exhaust Fan (VEF-15) starts within 10 seconds.

C. Place RB Purge Exhaust Fan (VEF-15) handswitch in MAN.

D. Place RB Purge Exhaust Fan (VEF-15) handswitch in OVERRIDE.

Answer:

C. Place RB Purge Exhaust Fan (VEF-15) handswitch in MAN.

Notes:

"C" is correct, if VEF-15 fails to start when VSF-2 starts, then an attempt is made to start VEF-15 by going to MAN. This alarm could come in if VEF-15 handswitch was not placed in AUTO prior to starting VSF-2.

"A" is incorrect but plausible since placing VSF-2 in "override" will prevent it from tripping and a note in the procedure states this, the procedure does not have a step allowing this action.

"B" is incorrect but plausible since there is a 10 second time delay but that time delay is for tripping VSF-2 within 10 seconds if VEF-15 does not auto-start.

"D" is incorrect but plausible since VSF-2 has an override feature but VEF-15 does not.

This question matches the K/A as it pertains to the containment purge system (RB purge) and requires the applicant to recall which action is necessary in response to the alarm (i.e., the applicant must know how the system works).

References:

1104.033 Reactor Building Ventilation

History:

New question for 2017 RO Re-exam

QID: 0296 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Bank **Originator:** D. Slusher

TUOI: A1LP-RO-ICS **Objective:** 17 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 041 **System Title:** Steam Dump System and Turbine Bypass Control

Description: Ability to manually operate and/or monitor in the control room: Steam dump valves

K/A Number: A4.08 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 62 **SRO:**

Given:

* Startup is in progress.
* Turbine-Generator is in Integrated Control.
* Generator load is 175 megawatts.
* Turbine header pressure is 890 psig.
* Header pressure setpoint is set at 49%

With the above conditions at what pressure will the Turbine Bypass Valves open and close?

A. Turbine Bypass Valves open at 905 psig and close when header pressure is less than 895 psig.

B. Turbine Bypass Valves open at 945 psig and close when header pressure is less than 945 psig.

C. Turbine Bypass Valves open at 995 psig and close when header pressure is less than 985 psig.

D. Turbine Bypass Valves open at 895 psig and close when header pressure is less than 885 psig.

Answer:

B. Turbine Bypass Valves open at 945 psig and close when header pressure is less than 945 psig.

Notes:

"B" is correct, at above 15% MW output (~135MWe) a +50 psig bias is added to the Turbine Bypass Valve setpoint (895 psig, 49% on header pressure setpoint station) to prevent the Turbine Bypass Valves (TBVs) from competing with the turbine governor valves. For the given conditions the Turbine Bypass Valves would open if header pressure reaches 945 psig (setpoint + 50 psig) and close when header pressure falls below 945 psig.

"A" is incorrect, although plausible since this is how the TBVs will operate if generator load dropped to less than 15%, the 50 psig bias would be removed and the TBVs would open at 10 psig above setpoint and close at setpoint. But the given load is above 15%.

"C" is incorrect, although plausible since this is how the TBVs operate following a turbine trip when a 100 psig bias is applied.

"D" is incorrect, although plausible if there was no bias applied to the TBVs and they opened at the normal setpoint of 895 psig.

This question matches the K/A since it concerns the Steam Dump/Turbine Bypass Control system (ICS) and for the applicant to monitor operation in the control room the applicant must know when to expect the TBVs to open.

References:

1105.004, Integrated Control System
STM 1-64, Integrated Control System

History:

Used in 1999 exam, modified from ExamBank, QID# 345.
Selected for 2017 RO Re-exam

QID: 1139 **Rev:** 0 **Rev Date:** 3/17/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 3.9 **Type:** Radioactivity Release

System Number: 068 **System Title:** Liquid Radwaste

Description: 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. (CFR: 41.5 / 43.5 / 45.3 / 45.13)

K/A Number: A2.04 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 63 **SRO:**

Given:

* Unit 1 is at 100% power

* Treated Waste Monitor Tank T-16A release is in progress

* Annunciator PROC MONITOR RADIATION HI (K10-B2) alarms

CBOT reports Liquid Radwaste Process Monitor (RI-4642) was in alarm but is now indicating pre-release value.

CBOT also reports Liquid Waste to Flume valve (CV-4642) is open, CV-4642 was closed using handswitch.

Which of the following actions are required by Liquid Waste Discharge Line High Radiation Alarm (1203.007) in response to the above conditions?

A. Close Treated Waste Discharge to Circ Water Flume (CZ-58)

B. Stop Treated Waste Monitor Pump (P-47A)

C. Reset Liquid Radwaste Process Monitor RI-4642 and re-establish release

D. Obtain independent verification of release path alignment and re-establish release

Answer:

B. Stop Treated Waste Monitor Pump (P-47A)

Notes:

"B" is correct, even though Rad Monitor RI-4642 only experienced a momentary "spike" the applicant should deduce that CV-4642 is inoperable since it did not close when RI-4642 alarmed, and that in addition to verifying CV-4642 goes closed, the Treated Waste Monitor Pump (P-47A) should be stopped per 1203.007. This ensures the release is terminated.

"A" is incorrect, but plausible as this action would be taken if CV-4642 could not be closed. CV-4642 was closed so this step is not applicable.

"C" is incorrect, but plausible since this is allowed due to spikes on RI-4642 but in this instance CV-4642 did not automatically close and the release permit does not contain contingencies for CV-4642 being inoperable.

"D" is incorrect, but plausible since the independent verification is required to be done for an inoperable monitor (RI-4642) but this is not applicable to an inoperable valve (CV-4642).

This question matches the K/A since it involves a liquid radwaste release where automatic isolation failed to occur. The prediction of the impact is the applicant must realize CV-4642 will not perform it's function and

deduce which of the choices given must be performed for CV-4642's failure.

References:

1104.020, Clean Waste System Operation
1203.007, Liquid Waste Discharge Line High Radiation Alarm

History:

New question for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1140 **Rev:** 0 **Rev Date:** 3/17/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-MSSS **Objective:** 5 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 079 **System Title:** Station Air System

Description: Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: Cross-connect with IAS.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 64 **SRO:**

The Service Air System is cross-connected with the Instrument Air System by SA to IA X-Over Valve (SV-5400).

Which of the following correctly describes the interlock between these two air systems?

A. SV-5400 opens when Instrument Air pressure lowers to 60 psig.

B. SV-5400 opens when Instrument Air pressure lowers to 50 psig.

C. SV-5400 closes when Instrument Air pressure lowers to 50 psig.

D. SV-5400 closes when Instrument Air pressure lowers to 60 psig.

Answer:

B. SV-5400 opens when Instrument Air pressure decreases to 50 psig.

Notes:

"B" is correct, to support a lowering Instrument Air header pressure, SV-5400 opens when Inst. Air pressure drops down to 50 psig as a means to prevent Inst. Air from dropping further.

"A" is incorrect, but plausible as this is the correct position for SV-5400 on a dropping Inst. Air header pressure but the setpoint is incorrect. This setpoint is the identical to the setpoint used in 1203.024, Loss of Instrument Air, to determine if the cross-connect between Unit 1 and Unit 2 Inst. Air systems should be closed.

"C" is incorrect, but plausible as this is the correct setpoint but the position is incorrect. This is further plausible if the applicant thinks of the Service Air System as a source of leakage which should be isolated on lowering Inst. Air header pressure but SV-5400 is normally closed.

"D" is incorrect, but plausible if the applicant thinks of the Service Air System as a source of leakage which should be isolated on lowering Inst. Air header pressure but SV-5400 is normally closed. SV-5400 opens automatically to try to hold Instrument Air pressure up. The setpoint in this distractor is the identical to the setpoint used in 1203.024, Loss of Instrument Air, to determine if the cross-connect between Unit 1 and Unit 2 Inst. Air systems should be closed.

This question matches the K/A since it involves the Service Air (Station Air) cross-connect with instrument air and requires applicant knowledge of the interlock between the two.

References:

1104.025, Service Air System

History:

New for 2017 RO Re-exam

QID: 0619 **Rev:** 1 **Rev Date:** 3/17/17 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-RO-FPS **Objective:** 6 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 086 **System Title:** Fire Protection System (FPS)

Description: Knowledge of the effect of a loss or malfunction of the Fire Protection System will have on the following:
Fire, smoke, and heat detectors.

K/A Number: K6.04 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	H

Question: **RO:** 65 **SRO:**

The smoke detector string for the Cable Spreading Room has a trouble alarm on it's respective Zone Indicating Unit (ZIU).

Considering this, if a fire occurs in the Cable Spreading Room which of the following is considered an operable method of actuating the deluge system for this area?

A. Manually by taking the Inhibit switch to "Normal" on C463.

B. Automatic actuation via smoke detector and protectowire detector.

C. Automatic actuation via a protectowire detector.

D. Manually via the Man Trip switch on C463.

Answer:

D. Manually via the Man Trip switch on C463.

Notes:

"D" is correct, a Trouble alarm means a de-energized or open smoke detector string, therefore automatic operation is non-functional for the Cable Spreading Room. The Cable Spreading Room is a cross-zoned system, meaning it takes both a smoke detector and a protectowire detector in alarm to actuate the sprinkler valve. Cross-zoned systems have an Inhibit switch to disable automatic actuation when either detector string has a malfunction, or trouble alarm. Operating the Man Trip switch on C463 bypasses the automatic actuation (and Inhibit) contacts to manually trip the sprinkler valve.

"B" is incorrect, but plausible if the applicant does not understand the significance of the Trouble alarm on the smoke detector string. This is how the system automatically actuates without any malfunction.

"C" is incorrect,but plausible if the applicant cannot recall how a cross-zoned system works and thinks either sting will automatically actuate the system. Automatic actuation is not available on a cross zoned system without both detection strings.

"A" is incorrect, although the Inhibit switch is taken to Inhibit on cross zoned systems when a string is inoperable, taking it out of inhibit will enable automatic actuation but automatic actuation is inoperable due to the inoperable smoke detector string.

This question matches the K/A since it involves a malfunction of a smoke detector string and the applicant must have knowledge of the effect of how this malfunction affects automatic and manual actuation of the sprinkler system.

References:

1203.009, Fire Protection System Annunciator Corrective Action
1104.032, Fire Protection System

History:

New for 2005 RO exam, replacement question.
Selected for 2017 RO Re-exam

QID: 1141 **Rev:** 0 **Rev Date:** 3/20/17 **Source:** New **Originator:** Cork

TUOI: A1QC-RO-QUAL **Objective:** 2.1 **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of shift or short-term relief turnover practices.

K/A Number: 2.1.3 **CFR Reference:** 41.10 / 45.13

Tier:	3	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 66 **SRO:**

The plant is at 100% power.

The CRS, ATC and CBOT are the only licensed operators in the control room.

The ATC needs to go to the computer room to perform a procedure step.

Per EN-OP-115-03, Shift Turnover and Relief, which ONE of the following describes the requirements for this activity?

The ATC:

A. Must be relieved by a licensed operator other than the current CBOT.

B. May leave as long as a currently on-shift RO or SRO monitors the panels until the ATC returns.

C. Must remain in the control room at all times until formally relieved by another licensed operator.

D. May leave the control room, with CRS permission, without being relieved.

Answer:

C. Must remain in the control room at all times until formally relieved by another licensed operator.

Notes:

"C" is correct, a formal relief is required for the ATC to leave the control room.

"A" is incorrect, but plausible since the shift manning requirement is for two licensed operators to be in the control room .

"B" is incorrect, but plausible as this is the requirement during log taking while the ATC goes to the back panels.

"D" is incorrect, the ATC may not leave the control room without being relieved.

This question matches the K/A since it concerns short term relief turnover practices.

References:

EN-OP-115-03, Shift Turnover and Relief
COPD-001, Operations Expectations and Standards

History:

New for 2017 RO-rexam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1142 **Rev:** 0 **Rev Date:** 3/20/17 **Source:** New **Originator:** Cork

TUOI: A1LP-AO-VALVE **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

K/A Number: 2.1.29 **CFR Reference:** 41.10 / 45.1 / 45.12

Tier: 3 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** H

Question: **RO:** 67 **SRO:**

Given:
* Unit 1 is heating up following refueling.
* A valve lineup is in progress but an MOV inside the Reactor Building is leaking by.
* The MOV is to be manually seated.
* This is a non-Q MOV.

What are two of the procedural requirements in Conduct of Operations (1015.001) for manually seating this MOV?

A. Verify MOV breaker open AND manually tighten by hand without using a torque amplifying device.

B. Danger tag MOV breaker open AND manually tighten using a torque wrench.

C. Verify MOV breaker open AND tighten using a torque wrench.

D. Danger tag MOV breaker open AND manually tighten by hand without using a torque amplifying device.

Answer:

A. Verify MOV breaker open AND manually tighten by hand without using a torque amplifying device.

Notes:

"A" is correct. MOVs inside the Reactor Building are not danger tagged due to the inability to leave these tags inside the building when closing out the building prior to heatup, therefore no danger tags should be used during heatup. Additionally, the applicant should deduce since this is a non-Q MOV that torque limits do not apply and therefore the valve is to be hand tightened without the use of a TAD (torque amplifying device).

"B" is incorrect but plausible since MOVs are usually danger tagged if they are to be manually operated, but not if they are in the Reactor Building. Non-Q MOVs additionally do not have to be tightened using a torque wrench.

"C" is incorrect but plausible in this distractor has the correct method of de-energizing the MOV but incorrect method of tightening valve.

"D" is incorrect since MOVs in Reactor Building are not danger tagged but it has the correct method of tightening the valve and is thus plausible.

This question matches the K/A since this is a situation requiring knolwedge of how to perform a valve lineup using an MOV breaker and how to manually close it.

References:

1015.001, Conduct of Operations

History:

New for 2017 RO Re-exam

QID: 1084 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** Repeat **Originator:** Cork

TUOI: ASLP-RO-COPD **Objective:** DD **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

K/A Number: 2.1.37 **CFR Reference:** 41.1 / 43.6 / 45.6

Tier: 3 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.6 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 68 **SRO:**

In accordance with COPD-030, which of the following activities would require a Reactivity Management Brief prior to performance of the activity?

A. Raising the seal injection flow rate to RCP P-32A.

B. Bypassing the E-3/4A Feedwater Heaters.

C. Adding nitrogen to the Makeup Tank T-4.

D. Adjusting the reactive loading on the Main Generator.

Answer:

B. Bypassing the E-3/4A Feedwater Heaters.

Notes:

"B" is correct based on Att. 9.3 in COPD-030. Changing Feedwater flow rate or temperature will affect secondary power and thus will affect reactor power.

"A" is incorrect but plausible since this evolution will increase the amount of fluid going into the RCS, but seal injection is coming from the Makeup Tank so reactivity will not be affected.

"C" is incorrect but plausible since the Makeup Tank is part of Makeup and Purification which makes up to the RCS but changing Makeup Tank pressure will not affect reactivity.

"D" is incorrect but plausible as this contains the word "reactive" but changing reactive load will not change secondary power so there is no reactivity affect.

This question matches the K/A since it requires knowledge of what activity requires a RM brief per ANO's procedure for reactivity management.

References:

COPD-030, ANO Reactivity Management Program

History:

New question for 2016 exam
Repeated for 2017 RO Re-exam

QID: 1082 **Rev:** 0 **Rev Date:** 4/29/16 **Source:** Repeat **Originator:** Cork

TUOI: ASLP-RO-PRCON **Objective:** 1 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for making changes to procedures.

K/A Number: 2.2.6 **CFR Reference:** 41.10 / 43.3 / 45.13

Tier:	3	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 69 **SRO:**

You are assigned to the 4th floor as a procedure writer and are making a procedure change.

Which of the following would be regarded as a change to the INTENT of a procedure?

A. Adding text to clarify the purpose of a procedure step.

B. Deleting a QC hold point in a procedure section for a filter change.

C. Changing the title of a position to correspond to corporate heirarchy.

D. Adding a step to close an open configuration control loop.

Answer:

B. Deleting a QC hold point in a procedure section for a filter change.

Notes:

"B" is the correct answer per 1000.006, definition 4.9.2.

"A", "C", and "D" are common procedure changes and thus plausible, but none of these constitute intent changes per 1000.006.

Changed "C" to "B" to change order of correct answer.

This question matches the K/A since it requires the candidate to recall part of the process of making a procedure change: the definition of an intent change.

References:

1000.006, Procedure Control

History:

New question for 2016 exam
Repeated for 2017 RO Re-exam

QID: 1161 **Rev:** 0 **Rev Date:** 4/6/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-TS **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of limiting conditions for operations and safety limits.

K/A Number: 2.2.22 **CFR Reference:** 41.5 / 43.2 / 45.2

Tier: 3 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.7 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 70 **SRO:**

Which of the following are addressed in the Technical Specification Safety Limits?

A. Quadrant Power Tilt

B. Fuel Centerline Temperature

C. RCS Leakage

D. Reactor Building Pressure

Answer:

B. Fuel Centerline Temperature

Notes:

"B" is correct, fuel pin centerline temperature is Safety Limit 2.1.1.1.

"A" is incorrect, but plausible since there is a Tech Spec LCO for quadrant power tilt and QPT could affect fuel pin centerline temperature but it is not specifically stated in the safety limits.

"C" is incorrect, but plausible since RCS leakage is indicative of a failure of the RCS fission product barrier but is not addressed in safety limits.

"D" is incorrect, but plausible since RB pressure is an important parameter with respect to challenges of a fission product barrier, the Reactor Building, but is not specifically addressed in the safety limits.

References:

Technical Specifications, Section 2.0

History:

New for 2017 RO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0458 **Rev:** 0 **Rev Date:** 5/6/2002 **Source:** Bank **Originator:** S.Pullin

TUOI: A1LP-RO-TS **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to determine Technical Specification Mode of Operation.

K/A Number: 2.2.35 **CFR Reference:** 41.7 / 41.10 / 43.2 / 45.13

Tier:	3	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.5	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 71 **SRO:**

Which one of the following conditions is required by Unit 1 Technical Specifications in order to consider the reactor in Mode 4?

A. The reactor must be subcritical by at least 1.5% Delta k/k.

B. RCS T average must be between 200 °F and 280 °F.

C. The neutron chain reaction is self sustaining and K eff = 1.0.

D. RCS temperature is no more than 200 °F.

Answer:

B. RCS T average must be between 200 °F and 280 °F.

Notes:

"B" is correct, RCS Tavg is >200°F and <280°F for Mode 4.

"A" is incorrect, this choice is plausible since this is the shutdown margin maintained per 1102.002, Plant Startup, during heatup.

"C" is incorrect, this choice is plausible since this is the expected reactivity condition for Mode 2, Startup.

"D" is incorrect, this choioe is plausible since this is the Tavg associated with Mode 5, Refueling.

References:

Technical Specifications

History:

Direct from regular exambank QID 39.
Selected for use in 2002 SRO exam.
Selected for 2011 RO/SRO Exam.
Selected for 2017 RO Re-exam.

QID: 1081 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** Repeat **Originator:** Cork

TUOI: ESLP-GET-RWT01.07 **Objective:** 44 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

K/A Number: 2.3.12 **CFR Reference:** 41.12 / 45.9 / 45.10

Tier:	3	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 72 **SRO:**

You have been directed to perform a task in the Makeup Tank Room, a Locked High Radiation Area (LHRA).

Dose rates in the Makeup Tank Room are 1.1 R/hr.

Which of the following is a requirement per EN-RP-101, Access Control for Radiologically Controlled Areas, SPECIFICALLY for entry into the LHRA?

A. Red trip ticket

B. Continuous RP coverage

C. Approval by on-watch Shift Manager

D. Double PC garments

Answer:

B. Continuous RP coverage

Notes:

"B" is the correct answer per EN-RP-101, continuous RP coverage is required for workers in a field dose rate greater than or equal to 1R/hr which is the definition of an LHRA.

"A" is incorrect but plausible as this is required for HRA (high radiation area) as well as LHRA. The trip ticket with a red border or red background are used by RP but these tickets are not specifically described in a procedure.

"C" is incorrect but plausible as this is required for entry into VHRA (very high radiation area).

"D" is incorrect but plausible, this may be required for highly contaminated areas but is not peculiar to LHRA entry.

This question matches the K/A since it requires the candidate to recall an essential and unique requirement for entry into a locked high radiation area.

Revised C &D, and stem per NRC examiner request. JWC 7/14/16

References:

EN-RP-101, Access Control for Radiologically Controlled Areas

History:

New for 2016 exam
Repeat for 2017 RP Re-exam

QID: 1144 **Rev:** 0 **Rev Date:** 3/21/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP06 **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

K/A Number: 2.3.14 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier: 3 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** F

Question: **RO:** 73 **SRO:**

Given:

* Unit 1 is shutting down due to "A" SG tube leak

* 1203.014, Control of Secondary System Contamination, is in progress

Which Condensate Polishers are preferred to be left in service and why these specific two?

A. E & F, to limit contamination to two polishers

B. C & D, to limit contamination to two polishers

C. E & F, to reduce personnel dose rates

D. C & D, to reduce personnel dose rates

Answer:

D. C & D, to reduce personnel dose rates

Notes:

"D" is correct, two polishers are left in service, C & D are preferred as an ALARA practice since they are in the middle which will increase distance from the polishers to the operator at the polisher panel and increase distance from personnel in the train bay.

"A" is incorrect but plausible since reducing the number of polishers to two is to limit contamination but these are the wrong two and the incorrect reason for the specific two polishers.

"B" is incorrect but plausible since these are the correct two polishers and the reason for reducing the number of polishers to two is to limit contamination but the reason C & D are used is due to their central location.

"C" is incorrect, this is plausible since using E & F as the in-service polishers will reduce dose rates to the operator at the polisher panel but will raise dose rates for personnel in the train bay.

This question matches the K/A since a tube leak is an abnormal condition which introduces radiation hazards, and the question requires the knowledge of why a particular action is taken, i.e., to reduce personnel exposure to a radiation hazard.

References:

1203.014, Control of Secondary System Contamination

History:

New for 2017 RO Re-exam

QID: 0128 **Rev:** 2 **Rev Date:** 8/10/05 **Source:** Bank **Originator:** JCork

TUOI: ASLP-RO-EPLAN **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of the emergency plan.

K/A Number: 2.4.29 **CFR Reference:** 43.5 / 45.11

Tier:	3	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 74 **SRO:**

Which of the following would be classified as a loss, or a potential loss, of a fission product barrier per Emergency Action Level Classification (1903.010)?

A. RCS leakage indicates greater than 50 gpm with letdown isolated.

B. RCS pressure 2450 psig with ERV controlling pressure.

C. CNTMT pressure indicates 17 psia.

D. Engineering Assessment of core damage indicates 0.1% fuel cladding failure.

Answer:

A. RCS leakage indicates greater than 50 gpm with letdown isolated.

Notes:

"A" is correct, leakage is >50 gpm with letdown isolated is a potential loss of the RCS barrier per 1903.010.

"B" is incorrect, this is a challenge to, but not a loss of, the RCS pressure boundary.

"C" is incorrect, CNTMT pressure given is less than ESAS actuation setpoint.

"D" is incorrect, failed fuel must reach 1.0% to be a breach.

References:

1903.010, Emergency Action Level Classification

History:

Developed for 1998 SRO exam.
Revised after 9/98 exam analysis review.
Used in 2001 SRO Exam.
Modified for 2005 RO exam as replacement question.
Selected for 2017 RO Re-exam.

QID: 1143 **Rev:** 0 **Rev Date:** 3/20/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP **Objective:** **Point Value:** 1

Section: 2.0 **Type:** Generic KA's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.

K/A Number: 2.4.16 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	3	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	F

Question: **RO:** 75 **SRO:**

What is the only Emergency Operating Procedure (EOP) which may be directly entered from an Abnormal Operating Procedure (AOP) without first entering 1202.001, Reactor Trip?

A. 1202.002, Loss of Subcooling Margin

B. 1202.006, Tube Rupture

C. 1202.007, Degraded Power

D. 1202.010, ESAS

Answer:

B. 1202.006, Tube Rupture

Notes:

"B" is correct. Per 1015.043, ANO-1 EOP/AOP User Guide, 1202.006 Tube Rupture may be entered directly from AOP 1203.023 Small Generator Tube Leak without first entering 1202.001 so that off-site releases may be limited by performing a controlled shutdown in 1202.006.

"A" is incorrect, yet plausible since this EOP has the highest priority per the EOP User's Guide. Yet it is still entered only after diagnosis is made in 1202.001. It is even entered from 1202.006, Tube Rupture, if problems other than a tube rupture are diagnosed.

"C" is incorrect, yet plausible since this EOP contains several sections designed to mitigate Loss of Subcooling Margin, Overcooling, and Overheating. It's entry conditions are also quite obvious, yet 1202.001 Reactor Trip is still entered first.

"D" is incorrect, yet plausible since this EOP's entry conditions are obvious from the ESAS annunicators, yet 1202.001 Reactor Trip is still entered first.

This question matches the K/A since it requires knowledge of EOP hierarchy and how certain AOPs are used with the EOPs.

References:

1015.043, ANO-1 EOP/AOP User Guide

History:

New question for 2017 RO Re-exam

QID: 1119 **Rev:** 1 **Rev Date:** 3/28/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-MU **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 022 **System Title:** Loss of Reactor Coolant Makeup

Description: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exsts.

K/A Number: AA2.01 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**		**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 76

Given:
* Unit 1 is at 90% power.
* ATC announces that Pressurizer level is 215" and dropping
* Makeup Tank level is 78" and dropping
* Total Seal Injection Flow is 25 gpm and lowering
* MU flow is 30 gpm and lowering
* AREA MONITOR RADIATION HI (K10-B1) alarms due to RE-8011 (Makeup Pump Area)

Which of the following is the correct procedure and mitigating action?

A. Loss of Reactor Coolant Makeup (1203.026);
Stop the running HPI pump.

B. Loss of Reactor Coolant Makeup (1203.026);
Isolate RCP Seal Bleedoff for all RCPs.

C. Reactor Coolant Pump and Motor Emergencies (1203.031);
Stop the running HPI pump.

D. Reactor Coolant Pump and Motor Emergencies (1203.031);
Isolate RCP Seal Bleedoff for all RCPs.

Answer:

A. Loss of Reactor Coolant Makeup (1203.026);
Stop the running HPI pump

Notes:

For the given conditions the event in progress is a line break downstream of the HPI pump but prior to the flow branching off to Seal Injection and Makeup flow as indicated by the lowering Seal Injection and Makeup flow with Pzr level dropping.

"A" is Correct, seal injection flow and makeup flow have lowered due to a leak and the (RCS) leak has caused the area radiation levels to rise enough to cause an alarm. With the leak downstream of the HPI Pump, the procedure directs stopping the pump to allow for isolating the leak.

"B" is wrong but plausible since it is the correct procedure and a lowering flow might indicate a pump issue, if a HPI pump is stopped then isolating Seal Injection is a correct action.

"C" is wrong but plausible since there are RCP issues occurring and making RCP and Motor Emergencies plausible a clogged seal injection filter would cause a lowering seal injection flow and stopping the HPI pump is correct.

"D" is wrong but a plausible procedure as stated in "C". Seal injection isolation is a step found within the RCP and Motor Emergency procedure..

This question matches the K/A since conditions are given for a charging line leak and candidate must

differentiate between RCP seal problem and charging line leak.

References:

1203.026, Loss of Reactor Coolant Makeup
1203.031, Reactor Coolant Pump and Motor Emergencies

History:

New question for 2017 SRO Re-exam.
Rev 1
1. Added location of rad monitor to stem
2. Removed higher flow contradiction/ flow will be lowering due to break location
3, Removed reference to P-32B
4. Total number of bullets lowered and all together
5. Sump level increasing not required but can be added if you like
6. Question changed to which procedure and action required. Applicant must determine that a leak exists to select the appropriate procedure and action.
7. Caps removed
8. Explanation provided.
9. Removed reference to ACAs 10. NO total flow alarm
11. 2x2 format
13. Changed wording

QID: 1165 **Rev:** 0 **Rev Date:** 04/09/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-AOP **Objective:** 2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 027 **System Title:** Pressurizer Pressure Control System Malfunction

Description: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A Number: 2.2.25 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 77

Given:
* Unit 1 trips from 100% power due to a loss of the switchyard from a severe storm
* Pressurizer Spray fails open and the ATC operator was able to close the Spray isolation valve and stop the Reactor Coolant system pressure lowering trend
* Annunciator alarm PZR HEATER GROUND FAULT (K09-E3) comes in.
* RCS pressure response abnormally slow with available Pressurizer heaters energized
* Maintenance is requested to perform Unit 1 Emergency Powered Pressurizer Heater Checkout (1307.009) to determine operability of vital powered pressurizer heaters

Per Technical Specification 3.4.9 bases, If efforts to restore sufficient Emergency Powered Pressurizer Heaters are not successful, then what other means are available to provide an alternate method of maintaining subcooling margin?

A. Use of available HPI pump/s

B. Back feed bus A4 to A2 to energize additional pressurizer heaters

C. Align the Unit 2 Charging Pumps for makeup to the Unit 1 pressurizer

D. Connect the AACG (2K-9) diesel to power to A1 bus to energize additional pressurizer heaters

Answer:

A. Use of available HPI pump/s

Notes:

"A" is correct per TS Bases 3.4.9 which states that although the pressure control provided by the high head pressure
injection pumps is an alternate method of maintaining subcooling.

"B" is the incorrect but plausible since there are non-vital powered PZR heaters on A2, however, in Degraded Power
the CRS would not direct backfeeding from A4 for the sole purpose of restoring additional PZR heaters.

"C" is incorrect but plausible since the installation of the FLEX modifications now provides the necessary connections
and means to align the Unit 2 chargning pumps to supply Unit 1 PZR, but only during an extended Blackout.

"D" is incorrect but plausible since there are non-vital powered PZR heaters on A2, however, in Degraded Power the
CRS would not direct aligning the AACG for the sole purpose of restoring additional PZR heaters.

Matches the KA since the applicant must recall from TS Bases available means to restor

References:

T.S. 3.4.9 Bases
1202.007, Degraded Power
1FSG-001, Long Term RCS Inventory Control
STM 1-03, Reactor Coolant System

History:

New for the SRO 2017 re-exam

QID: 1112 **Rev:** 1 **Rev Date:** 3/28/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP03 **Objective:** 12 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 040 **System Title:** Steam Line Rupture

Description: Ability to determine and interpret the following as they apply to the Steam Line Rupture: Difference between steam line rupture and LOCA

K/A Number: AA2.03 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 78

Given:
* Unit 1 has tripped from 100% power.

CBOT reports the following critical parameters:
* RCS pressure 1620 psig and lowering slowly
* Pressurizer level, 60 inches and lowering slowly
* Thot, 545 °F and lowering slowly
* "A" SG pressure 850 psig and lowering slowly
* "B" SB pressure 1000 psig and steady
* RB pressure 15 psia and rising

Which of the following is the correct procedure and mitigating action for these conditions?

A. Overcooling (1202.003); Isolate Letdown.

B. Overcooling (1202.003); Trip both MFW pumps.

C. Loss of Subcooled Margin (1202.002); Isolate Letdown.

D. Loss of Subcooled Margin (1202.002); Trip both MFW pumps.

Answer:

A. Overcooling (1202.003); Isolate Letdown.

Notes:

The applicant is given conditions of lowering RCS pressure and level and rising RB pressure which can be either an
RCS break or Steam line break. Now they add in the lowering RCS temperature and SG pressures to determine that
a small steam line break has occurred and Overcooling is the correct procedure. Once they identify the correct procedure then determine which if/then step within Overcooling applies.

"A" is correct, Overcooling- tripping. Isolating Letdown is directed when they transition to RT-2.

"B" is wrong but plausible since it is the correct procedure and a contingency action related to RB pressure to stop both MFW pumps can be found within the procedure .

"C" is wrong but plausible because for the given RCS conditions a LOCA could be occurring which can result in a loss of subcooled margin. Isolating Letdown is plausible since this action is located in both procedures.

"D" is wrong but plausible because for the given RCS conditions a LOCA could be occurring which can result in a
loss of subcooled margin. Stopping both MWP is plausible since this action is contained within both procedures.

This question matches the K/A since it requires the applicant to evaluate conditions and choose between a

steam line break procedure (Overcooling) and a LOCA procedure (Loss of Subcooling)

References:

1202.003, Overcooling
1202.002; Loss of Subcooled Margin

History:

New question for 2017 SRO Re-exam
Rev 1
1. Choosing the correct procedure demonstrates LOCA/MSLB
2. Changed temp to T-hot
3. Better explanations provided
4. Referenced this step in support document
5, EOP figure 1 defines adequate SCM (provided)
6. Question has been modified, SG pressure and trend is needed.
7. Reactor is tripped
8. Changed dropping to lowering for entire exam

QID: 0840 **Rev:** 0 **Rev Date:** 5/25/11 **Source:** Bank **Originator:** J. Cork

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 058 **System Title:** Loss of DC Power

Description: Ability to determine the operability and/or availability of safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 79

Unit One has entered Technical Specifications 3.8.4 due to a loss of one of the required DC electrical power subsystems.

In addition to the Battery, Battery Charger and associated cables, which of the following components are considered a part of the DC subsystem per Tech Spec Bases?

1. 125V DC panel
2. Inverter
3. Static Switch

A. 1 only

B. 2 only

C. 1 and 3 only

D. 2 and 3 only

Answer:

A. 1 only

Notes:

"A" is correct per the TS bases for 3.8.4.
The 125 VDC electrical power system consists of two independent redundant safety related subsystems. Each subsystem consists of one 125 VDC battery, the associated battery charger and all the associated control equipment and interconnecting cables

"B" "C" and "D" contain wrong components which are related to the DC system in that the DC system supplies power to the 120v AC vital inverters. Inverters is wrong in A and C. Static switch is wrong in D.

Matches the KA because the applicant must identify the components that must be Operable to support the Operability of DC sub-system

References:

Technical Specifications 3.8.4 Bases

History:

New for 2011 SRO Exam.
Selected for 2017 SRO Re-take exam.
Modified the stem changing from 7 choices to 3 to components to satisfy the Bases.

QID: 1114 **Rev:** 1 **Rev Date:** 2/8/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ALOIA **Objective:** 2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 065 **System Title:** Loss of Instrument Air

Description: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing.

K/A Number: AA2.05 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 80

Given:
* Unit 1 is at 100% power
* INST AIR HEADER PRESS LO (K12-B3) alarms
* Inside AO reports loud hissing sounds on mezzanine in Turbine Building
* Loss of Instrument Air AOP (1203.024) has been entered
* Pressurizer level is 175" and lowering slowly
* CBOT reports Instrument Air pressure is 55 psig and lowering slowly

Which of the following actions and procedures must be performed?

A. Trip the reactor and perform Reactor Trip, (1202.001).

B. Commence plant shutdown at ≥ 10% per minute using Rapid Plant Shutdown, (1203.045).

C. Commence plant shutdown at ≤ 5% per minute using Power Reduction and Plant Shutdown, (1102.016).

D, Take manual control of Pressurizer level using Makeup and Purification System Operation (1104.002)

Answer:

B. Commence plant shutdown at ≥ 10% per minute using Rapid Plant Shutdown, (1203.045).

Notes:

"B" is correct, step 14 of 1203.024 contingency action will direct CRS to step 18 which directs a plant shutdown per Rapid Plant Shutdown AOP at ≥10% per minute. FYI, low instrument air pressure alarm comes in at 75 psig.

"A" is wrong but plausible step 25 directs a Reactor Trip when Instrument Air pressure drops to 35 psig as checked in step 19 with a contingency to transition to step 25.

"C" is wrong but plausible, for most plant shutdowns the normal shutdown procedure is used but if Instrument Air pressure drops to less than 60 psig, then Rapid Plant Shutdown is used.

"D" is wrong but plausible because this procedure refers to pressurizer level and has has multiple contingency actions in regards to level, this requires the applicant to consider this condition and if an action should be taken

This question matches the K/A since conditions are given for a loss of instrument which requires commencement of a plant shutdown.

References:

1203.024, Loss of Instrument Air
1102.016, Power Reduction and Plant Shutdown

History:

New question for 2017 SRO Re-exam
Rev. 1
1. Removed "all previous action completed" statement
2. Changed distractor" so x-connect should not be an issue
3 4 & 5. Provided steps as requested
6. 1102.016 is provided
7. Changed distractor "D"

QID: 1166 **Rev:** 0 **Rev Date:** 4/9/17 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 077 **System Title:** Generator Voltage and Electric Grid Disturbances

Description: Knowledge of abnormal condition procedures.

K/A Number: 2.4.11 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 81

Given:
- Unit 1 is in Mode 1.
- It is August and ambient outside temperature is 103°F.
- CBOT reports that both A3 and A4 bus voltages are ~3750 volts.
- CBOT also reports Startup #1 Transformer voltage is 22.1 KV and SU#2 Transformer is 160KV.
- After being contacted, Dispatcher reports voltage regulators are in-service and working properly but a major capacitor bank is out of service.
- This condition has not improved after several hours.
- Grid disturbances are causing grid voltage and frequency to oscillate.

Procedure 1203.037, Abnormal ES Bus Voltage and Degraded Offsite Power, has been entered.

Which of the following procedure sections should be transitioned to and which procedurally required actions are warranted for the above conditions?

A. In accordance with Section 3, Offsite Voltage Abnormal, start one available DG, parallel the DG to the grid, and separate the associated ES bus from the grid by opening its feeder breaker.

B. In accordance with Section 1, ES Bus Voltage Low, start one available DG, parallel the DG to the grid, and separate the associated ES bus from the grid by opening its feeder breaker.

C. In accordance with Section 3, Offsite Voltage Abnormal, start one available DG, de-energize the associated ES bus by opening its feeder breaker, and verify DG output breaker closes.

D. In accordance with Section 1, ES Bus Voltage Low, start one available DG, de-energize the associated ES bus by opening its feeder breaker, and verify DG output breaker closes.

Answer:

D. In accordance with Section 1, ES Bus Voltage Low, start one available DG, de-energize the associated ES bus by opening its feeder breaker, and verify DG output breaker closes.

Notes:

"D" is correct bus voltage is low but not low enough to autostart the DGs and since grid disturbance are occuring per 1203.037, section 1, the ES bus should be de-energized to allow the DG to automatically re-energize it.

"B" is incorrect, plausible as this would be the correct answer if the grid was stable but grid disturbances are occuring and an EDG should not be paralleled with an unstable grid since this could cause damage to the EDG.

"A" is incorrect since Section 3 does not contain this action, instead major loads are secured to reduce voltage but offsite voltages are not low enough to require this section to be used. Section 4, Offsite Frequency Low, contains this action. "A" contains the correct action but the wrong procedure section.

"C" is incorrect since Section 3 does not contain this action, instead major loads are secured to reduce voltage but offsite voltages are not low enough to require this section to be used. Section 4, Offsite Frequency Low, contains this action. "C" contains the wrong procedure section and the wrong action but completes the 2x2 format.

Modified QID 1047 by changing the last bullet from "No grid disturbances are expected." to grid disturbances are occuring. This changes the correct answer from "C" to "D".

This question matches the K/A since a grid distrubance is ongoing and it requires knowledge of the proper action to take per the proper Abnormal Operating Procedure (1203.037).

References:

1203.037, Abnormal ES Bus Voltage and Degraded Offsite Power

History:

Modified QID 0734 for 2017 SRO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1115 **Rev:** 1 **Rev Date:** 2/9/17 **Source:** New **Originator:** Cork

TUOI: ASCBT-EP-A0081 **Objective:** 2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 036 **System Title:** Fuel Handling Accident

Description: Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Magnitude of potential radioactive release

K/A Number: AA2.03 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 82

*****************REFERENCE PROVIDED***********************************

Given:
* Unit 1 is in a Refueling Outage and core reload is in progress
* Main Fuel Bridge reports a spent fuel assembly has been dropped in the core
* The spent fuel assembly has been confirmed to be damaged
* Area radiation monitor, Fuel Handling Area RE-8017, is in alarm at 0.5 R/hr and rising slowly
* SPING Channel 7, Containment Purge - RX-9820, has been alarming for 17 minutes at 2.5 µCi/cc and rising slowly

The correct EAL classification is _________ and the order in which authorities will be notified is __________.

A. Unusual Event; NRC then State or Local Authorities

B. Alert; NRC then the State or Local Authorities

C. Unusual Event; State and Local Authorities then the NRC

D. Alert; State and Local Authorities then the NRC

Answer:

D. Alert; State and Local Authorities then the NRC

Notes:

"D" is correct, with a damaged fuel assembly and SPING Channel 7 in alarm the EAL is Alert (AA2 in Abnormal Radiological Effluents tab). There are no spurious or input failure related alarms associated with the Rad monitors. With fuel damage and a rising trend on the monitors the SM can determine that the readings are valid. The order is correct per 1903.011-Y.

"A" is wrong but plausible if candidate does not read all of AU2 in the Abnormal Radiological Effluents tab and merely uses the fact that RE-8017 is in alarm as justification for the Unusual Event. The order is the opposite of the correct order in 1903.011Y.

"B" is wrong but plausible since this is the correct EAL classification. The order given is the opposite of the correct order in 1903.011Y.

"C" is wrong but plausible if candidate does not read all of AU2 in the Abnormal Radiological Effluents tab and merely uses the fact that RE-8017 is in alarm as justification for the Unusual Event. The order given is correct.

This question matches the K/A since a fuel handling accident has occurred and the magnitude of the potential radioactive release determines the EAL classification.

References:

1903.010, Emergency Action Level Classification
1903.011-Y, Emergency Class Initial Notification Message

1903.010 EAL Classification (MODIFIED) will be provided as a reference. Information pages 1-18 and 69-73 have been removed.

History:

New question for 2017 SRO Re-exam
Rev 1
1. With fuel damage and a rising trend on the monitors the SM can determine that the readings are valid. The order is correct per 1903.011-Y.
2. LOD and Taxonomy revised
4, Trends provided
5. SPING 7 needs a time element for declaration
7/8. Changed wording as suggested
9. Removed time element since it was common and only requesting correct order of notification

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1116 **Rev:** 0 **Rev Date:** 2/9/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 068 **System Title:** Control Room Evacuation

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

K/A Number: 2.1.23 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**		**RO Select:**	No	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 83

Given:
* Unit 1 is at 100% power.
* Unit 2 has a fire in their Control Room
* Heavy smoke has accumulated in the Unit 1 Control Room

CRS will enter ______________ and direct the CBOT to control SG pressures
950 to 1020 psig using _______________.

A. 1203.029, Remote Shutdown; Turbine Bypass Valves

B. 1203.002, Alternate Shutdown; Turbine Bypass Valves

C. 1203.029, Remote Shutdown; Atmospheric Dump Valves

D. 1203.002, Alternate Shutdown; Atmospheric Dump Valves

Answer:

A. 1203.029, Remote Shutdown; Turbine Bypass Valves

Notes:

When conditions require the Control to be evacuated Remote Shutdown will be entered except in the case of a Fire
in the CR or Cable Spreading Room in which case Alternate Shutdown will be entered. Remote Shutdown controls
SG pressure with Turbine Bypass valves while Alternate Shutdown directs the local manual use of ADVs

"A" is correct since a fire is not forcing evacuation of the Control Room and the Turbine Bypass Valves will be used to
control SG pressures.

"B" is wrong but plausible because Alternate Shutdown is a CR evacuation procedure and correctly identifies the use
Turbine Bypass Valves

"C" is wrong but plausible because it identifies the correct procedure and local manual operation of the ADVs is available.

"D" is wrong but plausible because Alternate Shutdown is a CR evacuation procedure and local manual operation
of the ADVs is available.

Matches the KA because these are CR evacuation procedures and demonstrates the ability to use of plant and system
procedures from Modes 1-3

References:

1203.029, Remote Shutdown
1203.002, Alternate Shutdown

History:

New question for 2017 SRO Re-exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1167 **Rev:** 0 **Rev Date:** 04/10/17 **Source:** New **Originator:** Burton

TUOI: ASLP-RO-EPLAN **Objective:** 6 **Point Value:** 1

Section: 4.3 **Type:** B&W EPEs

System Number: E08 **System Title:** LOCA Cooldown

Description: Knowledge of the emergency action level thresholds and classifications.

K/A Number: 2.4.41 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**	2.3	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 84

*****************REFERENCE PROVIDED***********************************

Given:
* Unit 1 tripped from 100% power 3 hours ago due to a LOCA
* RCS pressure is 50 psig
* HPI flow is 480 gpm, total
* CETs indicate 720 °F and lowering
* Containment High Range monitor 8060 is reading 3700 R/hr
* Containment High Range monitor 8061 is reading 4250 R/hr

Which of the following is the correct classification and barrier status?

A. SAE: Loss of 2 barriers only

B. SAE: Loss of 1 barrier and Potential Loss of 1 barrier

C. GE: Loss of 2 barriers and Potential Loss of 3rd barrier

D. GE: Loss of 3 barriers

Answer:

C. GE: Loss of 2 barriers and Potential Loss of 3rd barrier

Notes:

For the given conditions "C" is the correct answer.
Potential Loss of Containment due to rad levels > 4000R/hr (CNB5)
Loss of RCS due to rad levels > 100R/hr (RCB3)
Loss of Fuel Clad due to rad levels 1000R/hr(FCB4)
Potential Loss of Fuel Clad (CET temp) and HPI flow is for plausibility

"A" is wrong but plausible because 2 barriers are LOST

"B" is wrong but plausible because all 3 barriers have met at least the Potential Loss criteria

"D" is wrong both RCS and Fuel Clad are Lost and Containment is challenged

References:

1903.010, Classification

1903.010 EAL Classification (MODIFIED) will be provided as a reference. Information pages 1-18 and 69-73 have been removed.

History:

Developed for the 2017 SRO Exam.
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QID: 1118 **Rev:** 1 **Rev Date:** 04/01/17 **Source:** New **Originator:** Cork

TUOI: **Objective:** **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: E09 **System Title:** Natural Circulation Cooldown

Description: Ability to determine and interpret the following as they apply to the (Natural Circulation Cooldown):
Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A Number: EA2.1 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**	2.8	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 85

Given:
* A severe thunderstorm caused a loss of offsite power several hours ago
* A stuck open MSSV on "A" SG resulted in this SG being dry and isolated
* Unit 1 is cooling down on "B" SG only
* 1203.013, Natural Circulation Cooldown is in use
* CETs indicate 410 °F and lowering slowly
* Offsite power is NOT expected to return for at least 12 hours
* EDGs are providing power to A3 and A4

NOW

* During RCS de-pressurization the ERV sticks open
* RCS pressure is 390 psig and lowering

Which procedure shall be selected for the above conditions?

A. Stay in 1203.013, Natural Circulation Cooldown

B. Go to 1202.002, Loss of Subcooling Margin

C. Return to 1202.007, Degraded Power

D. Go to 1202.011, HPI Cooldown

Answer:

C. Go to 1202.007, Degraded Power

Notes:

"C" is correct. Section 1, Degraded Power, of 1203.013 would be in use. Step 4 of this section states to go to 1202.007, Degraded Power, if SCM is lost. SCM is > 30 but less than 50 degrees so it is not met.

"A" is wrong. 1203.013 does not contain any mitigating actions for loss of SCM. Other EOPs contain those actions
so this distractor is plausible if the candidate is not familiar with the procedure.

"B" is wrong although 1202.002 would be entered for loss of SCM if the Natural Circulation was being performed
with Section 2, Offsite Power Available. However, Section 1, Degraded Power, is applicable and therefore
1202.007 should be entered.

"D" is wrong although 1202.011, HPI Cooldown, would be in use if there was an unisolable steam leak on BOTH SGs but only only the "A" SG has this problem.

This question matches the K/A since it involves a Natural Circulation Cooldown condtion with conditions

present
that require selection and transition to another procedure.

References:

1203.013, Natural Circulation Cooldown

History:

New question for 2017 SRO Re-exam
Rev 1 removed Loss of SCM statement and provided temp and pressure

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0809 **Rev:** 0 **Rev Date:** 9/23/2009 **Source:** Bank **Originator:** S Pullin

TUOI: A1LP-RO-AOP **Objective:** 6 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 003 **System Title:** Reactor Coolant Pump System (RCPs)

Description: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A Number: 2.1.7 **CFR Reference:** 43.5

Tier:	2	**RO Imp:**		**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 86

Given:
* 100% Power
* "B" RCP seal bleed off temperature 210° F
* "B" RCP motor bearing temperature 172° F
* "B" RCP motor inboard vibration alert alarm
* "B" RCP seal cavity pressure is fluctuating between 650 and 1250 psig

Which of the following is the correct section and action(s) of
1203.031, "Reactor Coolant Pump and Motor Emergency" to perform?

A. Section 2, "Seal Failure", Trip the Reactor and trip the affected RCP.

B. Section 5, "Motor / Bearing Trouble", Trip the Reactor and trip the affected RCP.

C. Section 2, "Seal Failure", Reduce reactor power to within the capacity of unaffected RCP combination and stop the affected RCP per RCP Operations, 1103.006.

D. Section 5, "Motor / Bearing Trouble", Reduce reactor power to within the capacity of unaffected RCP combination and stop the affected RCP per RCP Operations, 1103.006.

Answer:

A. Section 2, "Seal Failure", Trip the Reactor and trip the affected RCP.

Notes:

All of the answers are plausible since they all pertain to the appropriate AOP and require the applicant to access conditions and determine the correct section and action(s) to take.

"A" is correct, a seal bleedoff temperature of greater than 200 F with no change in cooling (seal injection or ICW flow) meets the requirements to trip the RCP due to seal failure section.

"B" is wong. The given conditions do not indicate a bearing problem that warrents stopping the RCP.

"C" is wrong. The given conditions require an abnormal shutdown of an RCP instead of a normal shutdown of an RCP.

"D" is wrong. The given conditions require an abnormal shutdown of an RCP instead of a normal shutdown of an RCP.

Matches the KA because the question requires evaluation and application of the RCP and Motor Emergency procedure and requires operational judgment based on instrument interpretation

References:

1203.031, Reactor Coolant Pump and Motor Emergency

History:

New selected for 2010 SRO exam

Selected for 2017 SRO Re-exam
Bank question with the following changes:
Modified stem question
Affected RCP from "C" to "B"
RCP Motor bearing temperature from 185 to 172" F
Changed order making A correct instead of B.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0594 **Rev:** 1 **Rev Date:** 2/10/17 **Source:** Modified **Originator:** J.Cork

TUOI: A1LP-RO-DH **Objective:** 27 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 005 **System Title:** Residual Heat Removal

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown.

K/A Number: A2.02 **CFR Reference:** 43.5

Tier:	2	**RO Imp:**	3.5	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 87

Given:
* RCS pressure is 200 psig and stable
* RCS temperature is 150 °F and stable
* "A" DH pump is in service

NOW

I&C performs a pressure instrument calibration on "A" channel of RPS.

Which of the following is correct?

A. RCS pressure will drop rapidly, enter Pressurizer Systems Failure (1203.015) and isolate ERV.

B. DH isolation will close, enter Pressurizer Systems Failure (1203.015) and stop "A" DH pump.

C. RCS pressure will drop rapidly, enter Loss of Decay Heat Removal (1203.028) and isolate ERV.

D. DH isolation will close, enter Loss of Decay Heat Removal (1203.028) and stop "A" DH pump.

Answer:

A. RCS pressure will drop rapidly, enter Pressurizer Systems Failure (1203.015) and isolate ERV.

Notes:

"A" is correct, even though the ERV will be in LTOP mode with a setpoint of 400 psig and the pressure input for the low pressure control is wide range RCS pressure from ESAS Analog Channel 1, the high pressure setpoint is still active. The "A" RPS narrow range RCS pressure instrument inputs into high pressure ERV control. The calibration of "A" RPS pressure instrument will lift the ERV and entry into 1203.015 will lead the operators to isolate the ERV.

"B" is wrong but plausible since this is the correct procedure but the Decay Heat pump would not be isolated by this pressure channel calibration, however it would if it was the "C" RPS channel which inputs into the DH suction valve closure logic.

"C" is wrong but plausible since this is the correct action but the wrong procedure. This would be the correct procedure if it was the "C" RPS channel being calibrated.

"D" is wrong this is the wrong action and the wrong procedure. This would be the correct action and procedure if it was the "C" RPS channel being calibrated.

This question matches the K/A since it requires the applicant to "predict" the result of the inappropriate testing of the shutdown pressure protection and then determine the correct procedure and actions to perform.

References:

1203.015, Pressurizer Systems Failure
1203.028, Loss of Decay Heat Removal
STM 1-03, Reactor Coolant System

History:

New for 2005 SRO exam.
Modified this question to be in line with current SRO level questions by converting it to a 2x2 format and adding
actions to the choices. The two procedure choices are the most logical choices and the two actions correspond
to "A" or "C" RPS RCS pressure channel actions.

QID: 1009 **Rev:** 0 **Rev Date:** 2/18/13 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-TS **Objective:** 13 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 006 **System Title:** Emergency Core Cooling System (ECCS)

Description: Knowledge of surveillance procedures.

K/A Number: 2.2.12 **CFR Reference:** 43.2

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 88

Given:
* Unit 1 is operating at 100% power 60 days following a Refueling
* Control Room is informed that a (1) Emergency Core Cooling System (ECCS) valve in an ECCS train was not tested during the outage as required by it's 18 month Surveillance Requirement 3.5.2.3
* This valve was last tested 20 months ago
* This valve is inaccessible with the reactor at power
* A mid-cycle outage is scheduled to occur in 8 months

To comply with Tech Specs the crew __________________.

A. is required to enter TS 3.5.2.A upon notification.

B. is required to enter TS 3.5.2.A when the 25% grace period expires.

C. can delay entering TS 3.5.2.A for a maximum of 24 hours from the time of discovery if a risk assessment is performed and the risk is managed. Entry is then required.

D. can delay entering TS 3.5.2.A until the mid-cycle outage if a risk assessment is performed and the risk is managed.

Answer:

D. can delay entering TS 3.5.2.A until the mid-cycle outage if a risk assessment is performed and the risk is managed.

Notes:

"D" is correct. Per TS SR 3.0.3, if a required surveillance is missed, entry into the applicable TS can be delayed for an additional surveillance frequency (in this case 18 months per SR 3.5.2.3) from the original due date if a risk assessment is done and the risk is managed. Because the mid-cycle outage is scheduled to fall within this window, distractor D is correct.

"A" is wrong based upon the above discussion on SR 3.0.3 but is plausible if the candidate does not recall the nuances of SR 3.0.3.

"B" is wrong based upon the above discussion on SR 3.0.3 but is plausible if the candidate applies SR 3.0.2 which discusses a 25% grace period.

"C" is wrong but plausible based upon the above discussion on SR 3.0.3 but SR3.0.3 states "24 hours or the frequency, whichever is greater" but this is incorrect since 18 months is obviously greater than 24 hours.

This question matches the K/A since a situation requiring the use of Tech Specs is given and the candidate must apply the specifications as well as knowledge of general surveillance requirement applicability (the latter is not part of the handout).

References:

TS 3.5.2, SR 3.0.1, SR 3.0.2, and SR 3.0.3

History:

New for 2013 SRO Exam
Selected for 2017 SRO Re-exam
Changed look of question slightly by changing from 3 valves to a single valve and midcycle from 12 months to 8

QID: 1117 **Rev:** 1 **Rev Date:** 04/05/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ASDCD **Objective:** 12 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation System

Description: Knowledge of EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 89

Given:
* Reactor has tripped due to a LOCA
* ESAS (1202.010) has been entered
* Crew has not initiated a cooldown
* Five (5) hours after a Reactor trip, the following conditions are observed
 -RCS pressure 800 psig and lowering
 -Full HPI flow
 -T-hot indicates 450 °F and lowering
 -RB pressure peaked at 48 psia and is now 30 psia and lowering slowly
 -BWST level is 8 feet and lowering

Based these conditions the crew should _______________ and verify that _________________.

A. remain in ESAS (1202.010);
 Electromatic Relief ERV Isolation (CV 1000) is closed

B. remain in ESAS (1202.010);
 RB Spray flow throttled to maintain 1050 to 1200 gpm per train

C. transition to Small Break LOCA Cooldown (1203.041);
 Electromatic Relief ERV Isolation (CV 1000) is closed

D. transition to Small Break LOCA Cooldown (1203.041);
 RB Spray flow throttled to maintain 1050 to 1200 gpm per train

Answer:

D. transition to Small Break LOCA Cooldown (1203.041);
 RB Spray flow throttled to maintain 1050 to 1200 gpm per train

Notes:

"D" is correct because ESAS directs the transition to SBLOCA if ESAS actuation is NOT corrected and an uncontrolled RCS cooldown is occurring due to HPI/break flow. Action is correct because RB Spray initiated 44 psia and will need to be throttled prior to BWST reaching 6 feet.

"A" is wrong but plausible as entry into this procedure was correct and if the condition had been corrected then the crew would be directed to remain. The action is plausible because both the ESAS and SBLOCA procedures address positioning of this valve and if HPI cooling was not in progress then closed is correct.

"B" is wrong but plausible as entry into this procedure was correct and if the condition had been corrected then the crew would be directed to remain. Throttling RB flow is correct.

"C" is wrong but plausible because ESAS directs the transition to SBLOCA if ESAS actuation is NOT corrected and an uncontrolled RCS cooldown is occurring due to HPI/break flow. The action is plausible because both the

ESAS and
SBLOCA procedures address positioning of this valve and if HPI cooling was not in progress then closed is correct.

Matches the KA because the correct answer addresses both ESFAS actuations (RB Spray) and EOP mitigating strategies

References:

1203.041 Small Break Loca Cooldown
1202.010, ESAS

History:

New question for 2017 SRO Re-exam
Rev 1- Removed reference to SCM, applicaant must determine. Also changed to a 2x2 format as suggested.
Correct answer is now - trtanstition to another procedure

QID: 1152 **Rev:** 0 **Rev Date:** 04/02/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-EOP02 **Objective:** 14 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray System (CSS)

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding or sump level below cutoff (interlock) limit.

K/A Number: A2.07 **CFR Reference:** 43.5

Tier:	2	**RO Imp:**	3.6	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 90

Given:
* Unit 1 tripped from 100% power
* RCS pressure is 100 psig and stable
* RB Spray actuated as required
* CETs indicate 330 °F
* Transfer to RB sump recirculation has been completed.

NOW

* RB Sump level dropped
* RB Flood level is steady
* Both LPI Pump discharge pressures is fluctuating between 100 psig and 160 psig
* Both RB Spray P-35A/B ES Failure annunciators are coming in and out of alarm
* Dose assessment reports there is no offsite release in progress

CRS should mitigate the event using _______________ and direct the crew to Override and Stop _____ RB Spray pump(s) and close the associated RB Spray Block valve(s).

A. 1202.010 (ESAS), only one (1)

B. 1202.010 (ESAS), both (2)

C. 1202.011 (HPI Cooldown), only one (1)

D. 1202.011 (HPI Cooldown), both (2)

Answer:

B. 1202.010 (ESAS), both (2)

Notes:

Conditions require entry into ESAS, RCS pressure and RB pressure are beyond actuation setpoints. HPI Cooldown is
plausible since Primacy to Secondary is not effective under these conditions. HPI cooldown directs use of RT-15 when
BWST level drops to 6 feet. Attachment 1 of ESAS and RT-15 have duplicate steps with regards to LPI, HPI and RB Spray
flow making either procedure plausible. Since there is no breach of Containment stopping both trains of CS is directed.

"B" is correct

"A" is wrong but plausible because it references the correct procedure and a single train would be stopped if containment breach had occurred.

"C" is wrong but plausible as HPI cooldown is plausible and a single train would be stopped if containment breach had occurred.

"D" is wrong but plausible as HPI cooldown is a LOCA based procedure and the actions to stop both trains of CS is correct

Meets the KA since the question involves the CSS and requires the use of a procedure to mitigate the event in progress

References:

1202.010, Chg. 012, ESAS
1202.011, Chg. 008 HPI Cooldown
1202.012, Chg. 017 Repetitive Task 15

History:

Selected for 2017 SRO Exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1154 **Rev:** 0 **Rev Date:** 04/03/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-CRD **Objective:** 12/13 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 014 **System Title:** Rod Position Indication

Description: 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: Dropped rod

K/A Number: A2.03 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 91

Given:
* Unit 1 is operating at 100% power
* Group 5 rods are 100% withdrawn when Grp 5 Rod 5 drops to 0% withdrawn.
* Plant has runback to 40% power.

(1) How would the dropped rod effect the Rod Position Indication System (RPIS)?
(2) If Control Rod alignment and power recovery to 60% is begun 30 hours after the event, then Control Rod Drive Malfunction Action (1203.003) limits power escalation to a maximum of _____ /hr.

A. (1) RPI indicates 0%, API indicates 100%
(2) 3%

B. (1) RPI indicates 0%, API indicates 100%
(2) 30%

C. (1) RPI indicates 100%, API indicates 0%
(2) 3%

D. (1) RPI indicates 100%, API indicates 0%
(2) 30%

Answer:

C. (1) RPI indicates 100%, API indicates 0%
(2) 3%

Notes:

"C" is correct - API position is by use of reed switches so where the rod is what API uses RPI uses demand signal
of the motor therefore in the case of a dropped rod RPI will indicate the last position. Per 1203.033 power excursion is 3%/hr after 24 hours no matter what current power level.
Less than 24 hours the applicant must apply time and power level to determine the correct rate of change for this power change which is 30%/hr.

"A" is wrong but plausible since one set of indicators will read 100% and the other 0% withdrawn. Also 3% is correct

"B" is wrong but plausible since one set of indicators will read 100% and the other 0% withdrawn, 30% is valid.

"D" is wrong but plausible since the indicators are correct and 30% is a valid limit.

Matches the KA because the question requires the applicant to correctly predict the effect on RPIS due to a dropped Control Rod

References:

1203.003, Control Rod Drive Malfunction Action
STM 1-02, Control Rod Drive System

History:

Selected for 2017 SRO Re-exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1041 **Rev:** 0 **Rev Date:** 04/06/14 **Source:** Repeat **Originator:** Possage

TUOI: A1LP-RO-FH **Objective:** 4 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 034 **System Title:** Fuel Handling Equipment

Description: Ability to explain and apply systen limits and precautions

K/A Number: 2.1.32 **CFR Reference:** 43.2

Tier:	2	**RO Imp:**	3.8	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 92

Given:
* Unit 1 is in Mode 6
* SRO in charge of Refueling is verifying that the limits and precautions contained in Unit 1 Refueling (1502.004) are met prior to beginning core off-load.

Which of the following conditions would prevent moving fuel in the Reactor Building?

A. Tornado Watch in effect for Conway County.

B. Reactor was tripped and Mode 3 entered 92 hours ago.

C. One Control Room Emergency Air Conditioning System (CREACS) inoperable for the past 4 days.

D. Reactor Building Radiation monitor RE-8017 inoperable, and portable survey instrument is being monitored on the fuel handling bridge.

Answer:

B. Reactor was tripped and Mode 3 entered 92 hours ago.

Notes:

"B" is correct, the reactor must be subcritical for greater than 100 hours prior to fuel movement

"A" is wrong. Pope, Johnson, Yell and Logan counties in a tornado watch would require stopping fuel movement.
Conway county is immediately east of Pope county.

"C" is wrong. With one CREACS channel inoperable we have 30 days to repair prior to stopping fuel movement.

"D" is wrong. RE-8017 is desired to be operable for monitoring radiation levels on the bridge, however if it becomes
inoperable any portable survey instrument is allowed for monitoring rad levels and continue fuel movement.

Matched the KA because it is knowledge of the limitations and precautions allowing the use of Fuel Handling Equipment
for core off-load.

References:

OP-1502.004 , Control of Unit 1 Refueling
TS 3.7.10, CREACS
TRM 3.9.1, Refueling Operations

History:

Selected for 2014 SRO Exam
Repeated for 2017 SRO Exam

modified an NRC bank question (2014). Modified stem for 2017 exam, (0 hours subcritical to reactor tripped and
mode 3 entered 92 hours ago. Changed correct answer from position "A" to "B"

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1110 **Rev:** 0 **Rev Date:** 2/3/17 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-FPS **Objective:** 10/11 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 086 **System Title:** Fire Protection

Description: Ability to apply Technical Specifications for a system.

K/A Number: 2.2.40 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** H

Question: **RO:** **SRO:** 93

****************REFERENCE PROVIDED***********************

Given:
* Unit 1 is at 100% power
* Annunciator K12-D1 "FIRE PROT SYSTEM TROUBLE" alarms
* CBOT reports the alarm indicated on the C463 panels is a yellow LED on C4-4U "EFW PUMP ROOM SMOKE DET ZONE 38-Y"
* CBOT reports the alarm will not reset.
* No other new alarms are indicated on the C463 panels
* Unit 1 Fire Impairment Database lists no impairments for this zone

Which of the following meets the required action(s) of the Technical Requirements Manual?

A. Establish a one hour roving fire watch for Zone 38-Y.

B. Establish a continuous fire watch for Zone 38-Y.

C. Run fire hoses to Zone 38-Y to establish backup fire suppression.

D. Verify alternate smoke and/or heat detection for Zone 38-Y with control room alarm.

Answer:

B. Establish a continuous fire watch for Zone 38-Y.

Notes:

"B" is correct, with the yellow "trouble" LED in alarm on the P-7A EFW Pump Room, the detection is non-functional
but the sprinkler system also is non-functional, therefore 3.7.9 required action A.1.1, establishing a continuous
fire watch, must be completed within one hour.

"A" is wrong but plausible if the candidate considers only the non-functional smoke detection (3.3.6 action A.1) and not the suppression system whose required actions are more stringent.

"C" iswrong but plausible if the candidate does not recognize that the sprinkler system can still be manually actuated and running fire hoses is unnecessary, and thus is not required.

"D" is incorrect but plausible if the candidate merely reads the 3.7.9 required action A.1.2 statement and does not recognize that there is no other smoke/heat detection in the area.

This question matches the K/A as it requires the candidate to use the provided TRM specifications and apply them to the non-functional fire protection components.

References:

ANO-1 Technical Requirements Manual

TRM TRO's 3.3.6 and 3.7.9 must be in SRO handout!!!!

History:

New

QID: 1109 **Rev:** 1 **Rev Date:** 04/04/17 **Source:** New **Originator:** Cork

TUOI: ASLP-RO-REACT **Objective:** 6 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

K/A Number: 2.1.37 **CFR Reference:** 43.6

Tier:	3	**RO Imp:**		**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.6	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 94

Given:
* Activity or deficiency is determined to have a Reactivity Management Impact

Per COPD-030 (ANO Reactivity Management Program) which of the listed positions determines the Risk Level Classification and Minimum Required Defenses?

A. STA

B. Reactor Engineering

C. Operations SRO

D. Work Week Manager

Answer:

C. Operations SRO

Notes:

"C" is correct per steps 6.1.2 and 6.1.3 of COPD-030.

"A" is wrong but plausible since the STA can be the second individual verifying borations or dilutions are performed
per step 5.5[29] of COPD-030 but they do not have this responsibility.

"B" is wrong but plausible is consulted for power maneuvers, startups, etc., but this is not one of
Rx Engineering's responsibilities.

"D" is wrong but plausible since this position is involved with many parts of work activities but this is
beyond his responsibility in COPD-030. In addition per the On-line Work Management Process (EN_WM_101)
he signs the emergent addition/deletion form which assesses both Risk and Reactivity Impact

This question matches the K/A since it directly asks about responsibilities in ANO's specific Operations Directive for Reactivity Management.

References:

COPD-030, Reactivity Management

History:

New question for SRO Retake exam.
Rev 1 - Did not make any changes then state why Work Week Manager is plausible in justification pages.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1157 **Rev:** 0 **Rev Date:** 04/05/17 **Source:** New **Originator:** Burton

TUOI: A1LP-RO-TS **Objective:** 13 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of primary and secondary plant chemistry limits.

K/A Number: 2.1.34 **CFR Reference:** 43.5

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 3

Group: **SRO Imp:** 3.5 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 95

Given:
* Unit 1 is at 100% power.

(1) Per LCO 3.4.12, RCS Specific Activity, the maximum limit for DOSE EQIVALENT I-131 is ≤ ____ µCi/gm;
(2) Per Bases 3.4.12 the resulting offsite and control room dose will not exceed the applicable 10 CFR 50.67 requirements following a Main Steam Line Break or ______ .

A. (1) 0.1
(2) Loss of Coolant Accident

B. (1) 0.1
(2) Steam Generator Tube Rupture

C. (1) 1.0
(2) Loss of Coolant Accident

D. (1) 1.0
(2) Steam Generator Tube Rupture

.

Answer:

D. (1) 1.0
(2) Steam Generator Tube Rupture

Notes:

"D" is correct because it identifies both the correct I-131 limit for the RCS and the correct accidents that the limit is based.

"A" is wrong but plausible because 0.1 is the secondary limit for I-131and LOCA is a break in one of the 3 barriers
and is identified in numerous bases as a limiting accident.

"B" is wrong but plausible because 0.1 is the secondary limit for I-131 and identifies the correct accident

"C" is wrong but plausible because it correctly identifies the RCS limit for I-131and LOCA is a break in one of the
3 barriers and is identified in numerous bases as a limiting accident.

This question matches the K/A since it requires the applicant to demonstrate knowledge of primary chemistry limits

References:

LCO 3.4.12, RCS Specific Activity and Bases
LCO 3.7.4, Secondary Specific Activity

History:

New question for 2017 SRO Retake exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1158 **Rev:** 2 **Rev Date:** 04/06/17 **Source:** New **Originator:** Burton

TUOI: ASLP-SRO-OPSPR **Objective:** 6 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for conducting special or infrequent tests.

K/A Number:	2.2.7	**CFR Reference:**	43.3				
Tier:	3	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 96

Given:
* Unit is at 100% power
* Crew has been directed to perform a task which requires a new procedure in conjunction with existing procedures
* Potential to cause a significant plant transient exists

Which of the following is correct in regards to required briefs that should be prepared before the task can be allowed to proceed?

A. Pre-Job and Reactivity

B. Refocus and Reactivity

C. Pre-Job and Infrequently Performed Tests or Evolutions

D. Refocus and Infrequently Performed Tests or Evolutions

Answer:

C. Pre-Job and Infrequently Performed Tests or Evolutions

Notes:

"C" is correct per the IPTE and COPD001 procedures both a pre-job and IPTE brief is required because this is a special or infrequent tests, in this case, the applicant must recognize that any first time use procedure with a potential for a plant transient require IPTE controls be established including a specific brief.

"A" is wrong but plausible since both Pre-job is correct and Reactivity briefs are identified in COPD001 also there is a potential for a reactivity excursion due to this task.

"B" is wrong but plausible since both the Refocus and Reactivity briefs are identified in COPD001 and there is a potential for a reactivity excursion due to this task.

"D" is wrong but plausible since IPTE is correct and a Refocus brief is identified in COPD001

This question matches the K/A as it tests the knowledge of performing special or infrequent tests, in this case, the applicant must recognize that any first time use procedure with a potential for a plant transient require IPTE controls be established including a specific brief.

References:

EN-OP-116, Infrequently Performed Test or Evolutions
EN-OP-115-04, Operations Briefs
COPD001, Operations Expectations and Standards

History:

New question for 2017 SRO Retake exam

QID: 1107 **Rev:** 1 **Rev Date:** 04/06/17 **Source:** New **Originator:** Burton

TUOI: A1LP-SRO- **Objective:** **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for controlling equipment configuration or status.

K/A Number: 2.2.14 **CFR Reference:** 43.3

Tier: 3 **RO Imp:** **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** F

Question: **RO:** **SRO:** 97

Given:
* Planned maintenance requires connecting contaminated and non-contaminated systems with a rubber hose
* System Engineer has determined that this is a Temporary Modification

Before the hose is connected between these two systems, who must authorize installation AND what is the MINIMUM number of check valve(s) that must be installed to meet procedural requirements of EN-DC-136, Temporary Modifications?

A. Shift Manager; 1 check valve

B. Shift Manager; 2 check valves in series

C. System Engineering Manager; 1 check valve

D. System Engineering Manager; 2 check valves in series

Answer:

B. Shift Manager; 2 check valves in series

Notes:

"B" is correct, the shift Manager authorizes and the T-mod procedure requires 2 check valves in series

"A" is wrong but plausible since Shift Manager is correct and a single check valve is allowed as a Clearance Boundary and per B3.6.3 can be credited as a Containment Isolation Boundary

"C" is wrong but plausible since the Engineering Manager is identified and has numerous responsibilities within the T-mod process and procedure. Also a single check valve is allowed as a Clearance Boundary and per B3.6.3 can be credited as a Containment Isolation Boundary.

"D" is wrong but plausible since the Engineering Manager is identified and has numerous responsibilities within the T-mod process and procedure. 2 check valves in series is correct.

This question meets the K/A since it requires the applicant to demonstrate knowledge of the T-mod process which is controlling equipment configuration.

References:

EN-DC-136, Temporary Modifications
EN-OP-102, Protective and Caution Tagging
B 3.6.3 Reactor Building Isolation Valves

History:

New question for 2017 SRO Retake exam.
Rev 1 - Changed to a different question we were having trouble with just 1 correct answer that could not be disputed

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1168 **Rev:** 1 **Rev Date:** 04/11/17 **Source:** Modified **Originator:** Burton

TUOI: A1LP-SRO-RAD **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K & A's

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. .

K/A Number: 2.3.12 **CFR Reference:** 43.4

Tier:	3	**RO Imp:**	3.2	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 98

Given:
* Unit 1 has tripped from 100% due to a LOCA
* TSC has been activated
* A Operator has 1.4 Rem accumulative dose for the calendar year
* NRC form 4 is on file
* Access is required to a high radiation area to align a filter
* Expected exposure will be 1.7 Rem

Per Entergy administrative procedures, which of the SPECIFIC recommendations and authorizations listed below is required to extend the workers TEDE exposure limit?

A. Supervisor AND Radiation Protection Manager ONLY.

B. Supervisor, Radiation Protection Manager AND Plant General Manager.

C. Radiation Protection Manager AND EOF Emergency Plant Manager ONLY.

D. Radiation Protection Manager, Plant General Manager AND Site Vice President.

Answer:

B. Supervisor, Radiation Protection Manager, AND Plant General Manager

Notes:

"B" is correct. Dose is > 3R but less than 4R (Supeervisor, RP Manager and Plant GM approves)
All of the remaining choices are plausible as they are found within the procedure table or with a LOCA could be an emergecny dose OP-1903.033

"A" is wrong, this is the authorization required for doses >2 R but <3 R.

"C" is wrong this limit can be authorized, if there is an emergency and an equiment or life saving circumstance existed. (10R/25R)

"D" is wrong this is the authorization for doses >4 R but <4.5 R.

Changed stem to an Operator task and order of the distractor so that correct answer is "D" instead of "C"
Changed the numbers slightly so the add up to 4.1 instead of 4.4

References:

EN-RP-201,Dosimetry Administration
OP-1903.033, PAG for Rescue/Repair & Damage Control Team

History:

Modified QID 931 from the 2014 SRO exam for 2017 SRO Re-take Exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1160 **Rev:** 0 **Rev Date:** 04/06/17 **Source:** New **Originator:** Burton

TUOI: ASCBT-EP-A0081 **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.4 **System Title:** Emergency Procedures / Plan

Description: Knowledge of procedures relating to a security event (non-safeguards information).

K/A Number: 2.4.28 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	3.2	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	H

Question: **RO:** **SRO:** 99

*****************REFERENCE PROVIDED***********************************

Given:
* OAO reports that there is an on-going gun fight with multiple intruders and Security officers by the Intake Structure traveling screen (F-7A)
* Security confirms this report and adds that there has been an explosion in a manhole adjacent to the Intake Structure
* OAO also reports she is exiting the area and there is no visible damage to the Service Water system

Per Emergency Action Level Classification (1903.010) what is the appropriate classification for this event?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer:

C. Site Area Emergency

Notes:

"C" contains the correct classification per 1903.010 since this is a security event involving gun fire \ inside the protected area.

"A" is wrong but plausible because an explosion on site meets the criteria of HU4.

"B" is wrong but plausible because an explosion in the manholes near the Intake Structure is HA4

"D" is wrong but plausible because HG1 would be met if the security event involved a vital area

This matches the K/A since 1903.010 is one of the procedures which would be used during a security event.

References:

1903.010, Emergency Action Level Classification

1903.010 EAL Classification (MODIFIED) will be provided as a reference. Information pages 1-18 and 69-73 have been removed.

History:

New question for 2017 SRO Retake exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1162 **Rev:** 1 **Rev Date:** 04/07/17 **Source:** New **Originator:** Burton

TUOI: ASCBT-EP-A0082 **Objective:** 15 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and abilities

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of emergency plan protective action recommendations.

K/A Number: 2.4.44 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	2.4	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	F

Question: **RO:** **SRO:** 100

Per Emergency Response/Notifications (1903.011)

(1) Protective Action Recommendations (PARs) are required to be provided during ___________.
(2) There has been a change in wind direction, PAR must be reassessed within a MINIMUM of ____________.

A. (1) ONLY General Emergencies
(2) 15 minutes

B. (1) ONLY General Emergencies
(2) 30 minutes

C. (1) General and Site Area Emergencies
(2) 15 minutes

D. (1) General and Site Area Emergencies
(2) 30 minutes

Answer:

A. (1) ONLY General Emergencies
(2) 15 minutes

Notes:

"A" is correct. The Emergency Class Initial Notification Message form 1903.011-Y states that PARs are required for GEs but not for NUE, Alert or SAE classifications. 30 minutes is a plausible distractor because this is the maximum allowed time to complete notifications from the time conditions are available in the Control Room. (15 minutes + 15 to complete notifications)

"B" is wrong. Only General is correct, 30 minutes is wrong but plausible as stated in "A"

"C" is wrong as PARs are not required for SAEs but plausible as plant or localized evacuations could be required for GEs or SAEs based on the plant conditions. 15 minutes is correct.

"D" is wrong as PARs are not required for SAEs as stated in "C". 30 minutes is plausible as stated in "A"

This question matches the K/A since the applicant must have knowledge of when to provide Emergency Plan PARs and how often PARs are re-assessed

References:

1903.010, Emergency Action Level Classification
1903.011, Emergency Response/Notifications

History:

Selected for 2017 SRO Retake exam.