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

QID: 0324 **Rev:** 2 **Rev Date:** 5/17/16 **Source:** Bank **Originator:** J. Cork

TUOI: A1LP-RO-EOP02 **Objective:** 4 **Point Value:** 1

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

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

Description: Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: Actions contained in EOP for PZR vapor space accident/ LOCA.

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

Tier:	1	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	An

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

Given:

- ESAS actuated on low RCS pressure.
- RCS Tave 560 °F and stable
- Pressurizer level 320" and stable
- RCS pressure 1350 psig and rising rapidly
- RB sump level 55% and rising
- Fuel failure of 1% is indicated

Considering the above conditions, which of the following methods, and reason behind the method, will be used to mitigate the RCS pressure transient in accordance with RT-14?

A. Cycle ERV as required, this prevents challenges to the PZR safeties.

B. Raise PZR spray flow, this condenses steam in PZR vapor space.

C. Throttle HPI flow, this reduces input of mass into RCS to match RCS leakage.

D. Raise letdown flow, this lowers RCS mass and thus reduces pressure.

Answer:

A. Cycle ERV as required, this prevents challenges to the PZR safeties.

Notes:

Answer "A" is correct since the conditions given are representative of a steam space leak and the RCS is in a "solid" condition. Using the ERV is the only effective way to reduce RCS pressure with the PZR in a solid condition, and chiefly prevents challenges to the PZR safeties.
Answer "B" is incorrect, but plausible as PZR spray is the normal method of reducing RCS pressure. However, PZR spray is not available since subcooling margin isn't present (RCPs should be off) and since the RCS is solid, it would be ineffective without a steam space to spray into.
Answer "C" is incorrect but plausible since this would reduce pressure, but this is the TMI response to their 1979 vapor space accident, subcooling margin is not present, HPI cannot be throttled.
Answer "D" is incorrect but plausible since this would reduce RCS mass but RT-14 does not allow for Letdown to be re-established with fuel failure indicated.

Revised RCS pressure to make it clear per Figure 3 that SCM is inadequate vs. being "on the line". Revised Pressurizer level from "off scale high" to 320" and stable, this makes question more challenging. Rev.2

This question matches the K/A since it gives the conditions of a PZR steam space leak and directly refers to EOP actions for this event. Reasons for the actions are given which completes the K/A match.

References:

1202.012, Repetitive Tasks, RT-14 "Control RCS Press"

AREVA Technical Document, Vol. 2, V.B-9

History:

Developed for 1999 exam.
Modified for use in 2005 RO exam, replacement question.
Selected for 2016 exam

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

QID: 1089 **Rev:** 1 **Rev Date:** 7/11/16 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 17 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the reasons for the following responses as they apply to the small break LOCA: RCP tripping requirements.

K/A Number: EK3.23 **CFR Reference:** 41.5 / 41.10 / 45.6 / 45.13

Tier: 1 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** K

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

Given:
- Reactor tripped on low RCS pressure
- RB sump is rising
- SCM is inadequate

What is the reason 1202.001, Reactor Trip, directs tripping all RCPs within two minutes following a loss of subcooling margin?

A. To reduce operator burden by tripping them prior to full ESAS actuation.

B. To protect the mechanical seals thus preventing further loss of coolant.

C. To prevent possible core uncovery if the RCPs were tripped later.

D. To prevent overheating of the RCP motors and thus preserve them for later use.

Answer:

C. To prevent possible core uncovery if the RCPs were tripped later.

Notes:

"C" is correct per 1202.001 and AREVA basis document. RCPs are tripped within 2 minutes since for certain size breaks where the void fraction exceeds 70% and then the RCPs were tripped, the phases would then separate and the core would be uncovered.
"A" is incorrect but plausible since if ESAS channels 3 and 4 were to actuate, then the RCPs would need to be tripped.
"B" is incorrect but plausible since damage to the mechanical seals would cause a loss of coolant.
"D" is incorrect but plausible since overheating of the motors could occur in a LOCA environment.

This question is a revision of QID 18. Distracters and correct answer were revised to add reasons to ensure K/A match and in some cases to add plausibility.

This question matches the K/A since conditions are given for a small break LOCA and the question asks for the reason the RCPs are tripped within two minutes of loss of SCM.

Revised stem at suggestion of NRC examiner. Rev.1

References:

1202.001, Reactor Trip
AREVA Technical document 47-1229003, III-A, CT-1

History:

Revised version of QID 18
Selected for 2016 exam

QID: 0684 **Rev:** 2 **Rev Date:** 7/11/16 **Source:** Bank **Originator:** Steve Pullin

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

Section: 4.1 **Type:** EPE

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

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

Group: 1 **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** C

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

Given:

- A Large Break LOCA has occurred.
- Full ESAS actuation has been occurring for 20 minutes.

LPI/HPI flow rates are as follows:

"A" LPI flow--3000 gpm
"B" LPI flow--2950 gpm

"A" HPI total flow--475 gpm
"C" HPI total flow--150 gpm

BWST level is 8 feet

Which of the following action is required per the ESAS EOP for these conditions?

A. Restore full HPI flow on "C" HPI pump.

B. Secure the "C" HPI pump only.

C. Override and secure all HPI pumps.

D. Swap to RB sump recirculation.

Answer:

C. Override and secure all HPI pumps.

Notes:

Answer C is correct. Sufficient LPI flow exists and the procedure directs overriding and securing all HPI.
"A" is incorrect but plausible as this answer would be correct if LPI flow was insufficient (<2800 gpm per pump).
"B" is incorrect but plausible with the degraded HPI flow on "C" pump.
"D" is incorrect but plausible due to low BWST level but this should occur at 6 ft, not 12 ft.

This question matches the K/A due to conditions given are a large break LOCA and the conditions meet the EOP criteria for stopping HPI pumps.

Revised question per NRC examiner comments following initial submittal.

References:

1202.010, ESAS

History:

New question for 2008 RO Exam.
Selected for 2016 exam

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

QID: 0183 **Rev:** 3 **Rev Date:** 7/12/16 **Source:** Bank **Originator:** E. Jacks

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

Section: 4.2 **Type:** Generic AOP

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

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Consequences of thermal shock to RCP seals.

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

Tier: 1 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
- Plant is in Mode 3
- In-service Makeup pump tripped
- PZR level 50"
- RCP seal bleedoff temperatures ~ 190 °F
- Restoration of normal makeup and seal injection is in progress

Which of the following is a required action per 1203.026, Loss of Reactor Coolant Makeup, in order to restore normal makeup and seal injection?

A. Normal makeup is restored before seal injection to raise RCS inventory since inventory has a higher priority.

B. Seal injection control valve (CV-1207) is quickly opened to establish previous flow rate to minimize time without seal injection.

C. Seal injection control valve (CV-1207) is slowly opened to minimize thermal shock to the RCP seals and prevent damage to seals.

D. BWST outlet valve associated with the operating HPI pump must be closed prior to opening seal injection control valve (CV-1207) to prevent borating RCS.

Answer:

C. Seal injection control valve (CV-1207) is slowly opened to minimize thermal shock to the RCP seals and prevent damage to seals.

Notes:

(c) is correct. As stated above, seal injection must be restored slowly to ensure RCP seals are not damaged.
(a) is incorrect. Restoring normal makeup and seal injection has no dependency on the Pressurizer level.
(b) is incorrect. Reestablishing seal injection quickly in any condition has the potential for shocking the RCP seals.
(d) is incorrect. Restoring normal makeup and seal injection has no dependency on the BWST outlet valve position.

This question was originally written for K/A AK3.01. Revised stem and answers to be more focused on K/A AK1.01.

This question matches the K/A since it states that a makeup pump has tripped and asks for the reason and consequence of how to restore seal injection (thermal shock to seals resulting in seal damage).

Revised per NRC examiner suggestion. JWC 7/12/16

References:

1203.026, Loss of Reactor Coolant Makeup

History:

Developed for use in 98 RO Re-exam
Selected for 2005 JG RO re-exam.
Selected for 2008 RO Exam.
Selected for 2016 exam.

QID: 1091 **Rev:** 0 **Rev Date:** 5/18/16 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: LPI pumps.

K/A Number: AA1.03 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier:	1	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	Ap

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

Given:
- Unit 1 is in a refueling outage
- RCS level is 377 ft.
- The "A" Decay Heat pump (P-34A) has tripped due to a breaker malfunction.
- The "B" Decay Heat pump (P-34B) has been placed in service at minimum flow.
- RCS temperatures are beginning to rise.

Which of the following "B" Decay Heat flow values will allow maximum RCS cooling without causing the high DH flow annunciator to alarm (K09-A8 DECAY HEAT FLOW HI/LO)?

A. 1900 gpm

B. 2700 gpm

C. 3500 gpm

D. 3700 gpm

Answer:

C. 3500 gpm

Notes:

Answer "C" is correct, with RCS level >375 feet, the setpoint for the high DH flow alarm is 3550 gpm.
"A" is incorrect but plausible since the high DH flow alarm setpoint is 2000 gpm when RCS level is less than or equal to 375 ft.
"B" is incorrect but plausible, this is just below the low flow alarm setpoint of 2800 gpm when LPI is in service (K11-B5).
"D" is incorrect but plausible, this is just below the high flow alarm setpoint of 3750 gpm when LPI is in service (K11-B5).

This question meets the K/A since the conditions are that a DH pump has been lost and the operator is required to know the high flow alarm setpoint in order to monitor for proper operation of the spare DH pump.

References:

1203.012H, Annunciator K09 Corrective Action

History:

New question for 2016 exam

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

QID: 0008 **Rev:** 1 **Rev Date:** 7/12/16 **Source:** Bank **Originator:** JCork

TUOI: A1LP-AO-ICW **Objective:** 9 **Point Value:** 1

Section: 4.2 **Type:** Generic APE

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

Description: Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: Location of a leak in the CCWS.

K/A Number: AA2.01 **CFR Reference:** 41.6

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 4

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

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

Given:

- Process Radiation Monitor RI-2236, Nuclear ICW, is in alarm.
- Shortly afterwards, reports come in of Nuclear ICW Surge Tank overflowing
- Nuclear ICW flow rate is >3100 gpm

A leak in which of the following components would be capable of causing these conditions?

A. RCP Seal Return Coolers

B. Spent Fuel Coolers

C. Letdown Coolers

D. Pressurizer Sample Cooler

Answer:

C. Letdown Coolers

Notes:

"C" is correct since it is the only component with the piping size and differential pressure to cause the indications given.
All of the other choices are cooled by ICW and are thus plausible but are incorrect because:
"A" RCP seal return cooler differential pressure is only slightly above Makeup Tank pressure and thus leak rate will be too small to cause surge tank overflow;
"B" Spent fuel cooler differential pressure is only slightly greater than ICW pressure and thus leak rate will be too small to cause surge tank overflow;
"D" Pressurizer sample cooler line size is too small to generate enough flow that this leak would go undetected prior to surge tank overflow.

This question matches the K/A since it involves Intermediate Cooling Water (ICW, ANO equivalent of CCW) and question evaluates the candidates ability to recognize which component would be more likely to cause a leak of this size.

Revised per NRC examiner suggestion.

References:

1203.039, Excess RCS Leakage

History:

Developed for 1998 SRO Exam.
Used in 2001 RO/SRO Exam.

Used on 2004 RO/SRO Exam.
Selected for 2016 exam

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

QID: 1101 **Rev:** 0 **Rev Date:** 2/17/16 **Source:** Bank **Originator:** Cork

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

Section: 4.1 **Type:** Generics EPEs

System Number: 007 **System Title:** Reactor Trip

Description: Knowledge of the purpose and function of major system components and controls.

K/A Number: 2.1.28 **CFR Reference:** 41.7

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

Group: 1 **SRO Imp:** 4.1 **SRO Select:** No **Taxonomy:** K

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

The Reactor Protection System has the Module-In-Test/Module-Removal interlock.

Upon removal of a critical module or placing a test module in a position other than "operate", this interlock will ________________ .

A. prevent the associated channel from tripping

B. place the RPS into a 2 out of 3 trip logic

C. lock out the other channels' test switches

D. cause the associated channel to trip

Answer:

D. cause the associated channel to trip

Notes:

"D" is the correct answer. The Module-In-Test/Module-Removal interlock is designed to trip the channel in case the channel is being defeated from tripping by placing a module in test or removing a critical module.
"A" is incorrect but plausible, there are interlocks which prevent a channel from tripping but this is not one.
"B" is incorrect but plausible if the candidate confuses this with placing a channel in bypass which will put RPS in a 2 of 3 logic. Tripping a channel will place RPS in a 1 out of 3 channels to trip logic.
"C" is incorrect but plausible if the candidate believes the purpose of this switch is to prevent placing the other channels in test, like the EFIC system, or confuses this with manual bypass which prevents the other channels from being bypassed when a channel is bypassed.

This question matches the K/A since it involves the RPS which generates a reactor trip signal and requires candidate to have knowledge of the purpose and function of major components and controls, i.e., function of the Module-In-Test/Module-Removal interlock.

References:

1105.001, NI & RPS Operating Procedure

History:

Selected regular exam bank ANO-OPS1-1999 for the 2016 exam

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

QID: 0332 **Rev:** 0 **Rev Date:** 9-6-99 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-EOP06 **Objective:** 3 **Point Value:** 1

Section: 4.1 **Type:** Generic Emergency Plant Evolutions

System Number: 038 **System Title:** Steam Generator Tube Rupture

Description: Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop.

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

Tier: 1 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 2

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

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

Per 1202.006, Tube Rupture, which action below is designed to minimize the rate of leakage into a ruptured steam generator?

A. Controlling reactor coolant system pressure low within the limits of Figure 3.

B. Concurrently performing 1203.014, Control of Secondary System Contamination.

C. Isolation of the "bad" SG with the ruptured tube.

D. Cooling down the reactor coolant system to less than 500 °F.

Answer:

A. Controlling reactor coolant system pressure low within the limits of Figure 3.

Notes:

Reducing the rate of primary to secondary leakage can only be done by reducing the differential pressure between primary and secondary systems.
"A" is correct, controlling RCS pressure low within limits of Figure 3will minimize (but maintain) subcooling margin and thus will decrease primary to secondary differential pressure as low as is reasonable.
"B" is incorrect, yet plausible as this procedure is performed per 1202.006 to reduce the contamination of secondary systemsm but will not reduce leak rate of primary to secondary.
"C" is incorrect, yet plausible since isolation of the OTSG will prevent other systems from receiving fluid from the ruptured OTSG but will do nothing to decrease leakage.
"D is incorrect, yet plausible as this action is designed to place the RCS in a condition which will not lift the MSSV with the lowest setpoint even if the ruptured OTSG is completely filled. This action is therefore designed to reduce offsite releases.

This question matches the K/A since it applies specifically to a SGTR and determines if candidate knows why RCS pressure is maintained low during a SGTR, i.e., the lower the differential pressure the lower the leak rate.

References:

1202.006, Tube Rupture
Bases for 1202.006

History:

Developed for 1999 exam.
Selected for 2007 RO Exam.
Replaced QID 1092 with this question due to NRC examiner comment for 2016 exam.

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

QID: 0686 **Rev:** 4 **Rev Date:** 7/12/16 **Source:** Bank **Originator:** Steve Pullin

TUOI: A1LP-RO-EOP03 **Objective:** 7 **Point Value:** 1

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

System Number: E05 **System Title:** Excessive Heat Transfer

Description: Knowledge of the interrelations between the (excessive heat transfer) 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: EK2.01 **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:**	A

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

The reactor has been tripped due to a Main Steam Line Rupture

The following post-trip conditions exist:

- "A" OTSG pressure = 425 psig
- "B" OTSG pressure = 580 psig

- "A" OTSG EFW flow = 200 gpm
- "B" OTSG EFW flow = 100 gpm

- RCS temperature = 495 degrees F
- RCS pressure = 1500 psig

- MSLI has actuated

Which of the following actions is correct for this event?

A. Verify EFW isolation and control valves to "A" OTSG closed.

B. Verify EFW isolation and control valves to "B" OTSG closed.

C. Verify at least one EFW pump running with flow to both OTSGs.

D. Verify EFW flow rates are ≥ 340 gpm on each SG.

Answer:

A. Verify EFW isolation and control valves to "A" OTSG closed.

Notes:

Answer A is correct. This answer requires an understanding of the automatic features of the Main Steam Line isolation section of the Emergency Feedwater Intiation and Control system and realization that the system is malfunctioning requiring manually completion of the safety function.
Answer B is incorrect as it isolates the good SG.
Answer C is incorrect. This standard post-EFIC-actuation action is incorrect in this situation since flow to the bad SG is caused by a malfunction and is detrimental in this condition.
Answer D is incorrect since EFW flow should be isolated to the A OTSG but plausible since this is the minimum required EFW flow rate if subcooling margin was inadequate.

This question matches the K/A since it involves interrelations between the excessive heat transfer (main steam line rupture) and components (EFW isolation and control valves) and signals (SG pressures) which cause automatic closure of "bad" SG EFW valves.

Revised per NRC examiner suggestion. JWC 7/12/16

References:

1202.012, Repetitive Tasks, RT-6 "Verity Proper MSLI and EFW Actuation and Control."

History:

Exam Bank: OpsUnit1 QuestionID: ANO-OPS1-2856
Selected for the 2008 RO Exam
Selected for 2016 exam

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

QID: 0146 **Rev:** 3 **Rev Date:** 05/20/93 **Source:** Bank **Originator:** E. Wentz

TUOI: A1LP-RO-FW **Objective:** 18 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 054 **System Title:** Loss of Main Feedwater

Description: Ability to recognize abnormal indications for system operating parameters which 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:** 2

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

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

The plant is operating at 40% power, when annunciator K07-C1, REACTOR FEEDWATER LIMITED, alarms.

The following conditions exist:
- RCS pressure and temperature are increasing.
- Both OTSG Operate Range levels = 45% and decreasing.
- Both Main Feedwater flows are decreasing.
- K07-B4, SASS MISMATCH, annunciator is clear.

What procedure contains the required mitigating operator actions for the above conditions?

A. 1203.027, Loss of Steam Generator Feed

B. 1203.001, ICS Abnormal Operation

C. 1203.018, Turbine Trip Below 43% Power

D. 1202.001, Reactor Trip

Answer:

A. 1203.027, Loss of Steam Generator Feed

Notes:

"A" is correct, it contains entry conditions which match the given conditions.
"B" is incorrect but plausible in that it sounds like a logical procedure since an ICS malfunction may be causing the transient, however there should also be an indication of a loss of ICS power to enter this AOP.
"C" is incorrect but plausible as it might be chosen if the candidate associates the symptoms of rising RCS pressure and temperature with a Turbine Trip but this will not cause SG levels and FW flows to change.
"D" the conditions here are similar to some of the conditions of a Rx trip but the annunciator alarm for a Rx trip is not given.

This question matches the K/A because it gives conditions for a loss of Main FW which match the abnormal operating procedure entry conditions.

References:

1203.027, Loss of Steam Generator Feed

History:

Taken from Exam Bank QID # 2800
Used in 98 RO Re-exam
Used on 2004 SRO Exam.
Selected for 2016 exam.

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

QID: 1097 **Rev:** 0 **Rev Date:** 06/06/2016 **Source:** New **Originator:** Coble

TUOI: A1LP-RO-EOP08 **Objective:** E09 **Point Value:** 1

Section: 4.1 **Type:** Generic EPE

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

Description: Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

K/A Number: EA2.03 **CFR Reference:** 41.10

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	No	**Taxonomy:**	C

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

Given the following:

- Both Units have tripped due to a Loss of Offsite Power.
- Startup Transformer #1 primary voltage is 0 KV.
- Startup Transformer #3 primary voltage is 0 KV.
- Unit 2 vital and non-vital buses are aligned to Startup Transformer #2.
- Startup Transformer #2 Voltage is reading 161 KV.
- Both Unit 1 Emergency Diesel Generators failed to start and are locked out.
- Station Blackout EOP recovery procedure has been entered on Unit 1.

Which one of the following would be the correct action to take to initially restore power to Unit 1 for these conditions?

A. Energize 4160v AC buses A1/A3 from Startup #2 Transformer

B. Energize 4160v AC buses A2/A4 from Startup #2 Transformer

C. Energize 4160v AC buses A3 AND A4 from the AAC Diesel Generator

D. Energize either 4160v AC bus A3 or A4 from the AAC Diesel Generator

Answer:

D. Energize either 4160v AC bus A3 or A4 from the AAC Diesel Generator

Notes:

"D" is correct based on the first direction and purpose of the Station Blackout Recovery Procedure 1202.008 Step 3.A. and B which has the unit recover one vital 4160 bus and then exit to the Degraded Power procedure 1202.007 to restore the rest of the busses.
"A" and "B" are incorrect due to the SU#2 Transformer is not available to Unit 1 since Unit 2 is aligned to this source (Refer to the Note on contingency step 8.C. of OP 1202.008) of power. Startup Transformer #2 is not designed to carry loads of both units so only one unit can be aligned to it.
"C" is incorrect because this would cross-tie both ESF buses and is not allowed by procedure, but plausible in that this would re-energize ESF buses.

This question matches the K/A statement in that the candidate must interpret the conditions of both units and apply the knowledge of the Station Blackout Procedure (by recalling actions) to commence restoring power to Unit 1.

References:

1202.008, Station Blackout Steps 3, Contingency Steps 8C/8D and Instruction Steps 34.B./47/48

History:

New question written for 2016 exam

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

QID: 1057 **Rev:** 0 **Rev Date:** 4/7/16 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 056 **System Title:** Loss of Offsite Power

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer.

K/A Number: AK3.01 **CFR Reference:** 41.5, 41.10 / 45.6 / 45.13

Tier: 1 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** K

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

The unit is operating at 100% power when a large break LOCA occurs.
Simultaneously a loss of offsite power occurs.

Which of the following ESF systems will start first and why will they start in this order?

A. RB Cooling Fans will start followed by the RB Spray Pumps due to the difference in Rx Bldg pressure setpoints for their respective ESAS channels.

B. RB Spray Pumps will start followed by the RB Cooling Fans due to the time delay relays which prevent EDG over-loading.

C. RB Cooling Fans will start followed by the RB Spray Pumps due to the time delay relays which prevent EDG over-loading.

D. RB Spray Pumps will start followed by the RB Cooling Fans due to the difference in Rx Bldg pressure setpoints for their respective ESAS channels.

Answer:

B. RB Spray Pumps will start followed by the RB Cooling Fans due to the time delay relays which prevent EDG over-loading.

Notes:

"B" is correct since the time delay relays will sequence on the RB Spray pumps at about 35 seconds followed by the RB Coolers at about 50 seconds.
"A" is incorrect but plausible since the RB pressure rise will cause the ESAS channels 5&6 to actuate first at 4 psig RB pressure and RB spray channels 7&8 will actuate at 30 psig. However, the EDG load sequence overrides this pressure sequence.
"C" is incorrect but plausible since it would be logical that the EDG load sequence would follow the RB pressure setpoints for the ESAS channels but the RB spray pumps start first.
"D is incorrect but necessary to complete the "two by two" order. It is plausible since the Spray pumps will start first but this is according to time delay relays not pressure setpoint differences.

This question matches the K/A since a loss of offsite power condtion is given and question evaluates candidate knowledge of order that ES components will be turned on and the reason why they sequence: don't overload the EDG.

References:

1305.006, Integrated ES System Test

History:

New question for 2016 exam.

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

QID: 1095 **Rev:** 0 **Rev Date:** 06/06/2016 **Source:** New **Originator:** Coble

TUOI: A1LP-RO-ANNI **Objective:** 5 **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: Manual control of components for which automatic control is lost.

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

Tier: 1 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

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

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

Given the following conditions:

- Plant startup in progress
- Plant power at 30%
- NNI Y AC light on C13 goes out

Which one of the following plant components will need to be controlled in manual or locally?

A. Presurizer Level Control Valve CV-1235

B, Pressurizer Heater Banks 3, 4 and 5

C. MFW Pumps P-1A and P-1B

D. RC Pump Seals Total Injection Flow Valve CV-1207

Answer:

C. MFW Pumps P-1A and P-1B

Notes:

C. is the correct answer as this is one of the required actions for only a loss of Vital AC Instrument Bus NNI Y Power with the plant at low power with the startup valves being controlled by dp signals. Main FW transfers from DP control to speed control when the Main Block valves open at ~50% FW demand (50% power).
B. is incorrect but plausible as this is a required action for a Loss of Power to the Vital AC Instrument Bus NNI X Power
A. is incorrect but plausible as this is a required action for a Loss of Power to the Vital AC Instrument Bus NNI X Power
D. is incorrect but plausible as this is a required action for a Loss of Power to the Vital AC Instrument Bus NNI X Power

This question matches the K/A statement in that the candidate must realize that at this power the dp input for feedwater flow control will be lost and must be manually controlled to prevent a feed flow mismatch to the steam generators.

References:

1203.047 Loss of NNI Power Step 8. A.

History:

New question written for 2016 exam

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

QID: 0513 **Rev:** 2 **Rev Date:** 12/8/2003 **Source:** Bank **Originator:** NRC

TUOI: A1LP-RO-EDG **Objective:** 12 **Point Value:** 1

Section: 4.2 **Type:** Generic APE

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:** 41.7

Tier: 1 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** C

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

Given:

- Degraded power event in progress
- K01-D1, "EDG 1 NOT AVAILABLE" is in alarm
- The Inside AO reports that engine DC control power was lost to EDG #1

What is expected effect on EDG #1 following a loss of engine DC control power?

A. EDG #1 will NOT start automatically and can NOT be started manually due to the governor run solenoid loss of power.

B. EDG #1 will start automatically but voltage must be controlled manually.

C. EDG #1 will NOT start automatically but may be started manually by overriding the governor run solenoid.

D. DG #1 will start automatically but can NOT be tied to the A3 bus due to the loss of power causing a lockout on A-308.

Answer:

C. EDG #1 will NOT start automatically but may be started manually by overriding the governor run solenoid.

Notes:

"C" is correct, EDG#1 will NOT start automatically due to loss of DC power to the governor run solenoid. The EDG can be started manually by mechanically overriding the governor run solenoid.
"A" is incorrect, the governor run solenoid can be manually overridden but plausible in that the EDG will not star automatically.
"B" is incorrect the EDG will not start automatically but this is plausible since a loss of DC will result in a loss of automatic voltage control but that is on a separate circuit.
"D" is incorrect, the EDG will not start automatically but plausible since DC control power is removed on A308 during an alternate shutdown situation.

References:

1104.036, Emergency Diesel Generator Operation

History:

Developed by NRC (modified a question from Davis Besse Bank)
Used on 2004 RO/SRO Exam.
Selected for the 2008 RO Exam
Selected for 2016 exam

QID: 1058 **Rev:** 0 **Rev Date:** 4/11/16 **Source:** New **Originator:** Cork

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

Section: **Type:** Generic APEs

System Number: 0662 **System Title:** Loss of Nuclear Service Water

Description: Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Control of flow rates to components cooled by the SWS.

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

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 2

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

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

Plant heat up is in progress with RCS temperature at 210 degrees F.
Service water is lost to the E-35A Decay Heat cooler.

How does the procedure direct you to setup the E-35A DH cooler for re-establishment of SW flow and why?

A. Close SW Inlet (CV-3822) to E-35A and verify SW Outlet (SW-22A) throttled to prevent DH cooler water hammer.

B. Throttle SW Inlet (CV-3822) to E-35A and close SW Outlet (SW-22A) to prevent DH cooler water hammer.

C. Close SW Inlet (CV-3822) to E-35A and verify SW Outlet (SW-22A) throttled to prevent SW pump runout.

D. Throttle SW Inlet (CV-3822) to E-35A and close SW Outlet (SW-22A) to prevent SW pump runout.

Answer:

A. Close SW Inlet (CV-3822) to E-35A and verify SW Outlet (SW-22A) throttled to prevent DH cooler water hammer.

Notes:

"A" is correct per 1203.028 Section 5, Loss of Service Water Flow, the inlet is closed and the outlet verified throttled to prevent water hammer.
"B" is incorrect, the reason is correct but the valve positions are backwards from the procedural requirement.
"C" is incorrect, the valve positions are correct but the reason is incorrect but plausible since the SW outlet is throttled to obtain required SW flow to all SW cooled components.
"D" is incorrect, the valve positions are backwards from the procedural requirement and the reason is incorrect but plausible since the SW outlet is throttled to obtain required SW flow to all SW cooled components.

This question matches the K/A since it involves a loss of Service Water and requires candidate to recall how to setup the DH cooler for service water re-establishment (setup to control flow rate).

References:

1203.028, Loss of Decay Heat Removal

History:

New question for 2016 exam

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

QID: 1102 **Rev:** 1 **Rev Date:** 7/12/16 **Source:** New **Originator:** Possage

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: Components served by instrument air to minimize drain on system.

K/A Number: AA1.02 **CFR Reference:** 41.7 / 45.5 /45.6

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

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

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

Given:
- Instrument Air leak is reported.
- Personnel are using Breathing Air for respiration.
- Instrument Air header pressure low annunciator, K12-B3, alarmed.
- Instrument Air header pressure 70 psig and lowering slowly.
- Breathing Air is being used to supply Instrument Air.
- OP-1203.024, Loss of Instrument Air AOP is in use.

Which of the following is performed SPECIFICALLY to conserve instrument air?

A. Establish SG Pressure control using Atmospheric Dump Isolation valves

B. Isolate Breathing Air from the Instrument Air System.

C. Commence Plant S/D at greater than or equal to 10% per minute.

D. Place P-33B ICW Pump in Pull to Lock.

Answer:

A. Establish SG Pressure control using Atmospheric Dump Isolation valves

Notes:

"A" is correct, SG pressure control is established by throttling ATM Dump Isolation valves closed (these are MOVs) and opening ATM Dump Control valves. This will minimize the amount of air used by this system.
"B" is incorrect, this is plausible however, if personnel are using the BA system for for breathing, low BA pressure can result in over exposure to airborne radiation and/or inadequate air for respiration. In 1203.024 it is not isolated to conserve IA it is isloated to protect personnel.
"C" is incorrect, this is plausible however, if IA pressure continues to degrade, at 60 psig power reduction at the maximum rate is specified in 1203.024 to minimize plant impact from control valves not operating properly, not to conserve IA. The IA header pressure is still greater than 60 psig.
"D" is incorrect, this is plausible since this action is performed in a degraded power situation to conserve IA due to the suction and discharge valves cycling open and closed. However, in 1203.024 it is done to protect the pump from damage when the suction and discharge valves fail closed on a loss of IA.

This question matches the K/A since the correct answer is taken to minimize the load on the IA system.

Revised per NRC examiner suggestion.

References:

1203.024, Loss of Instrument Air

History:

New for 2016 exam

QID: 0626 **Rev:** 2 **Rev Date:** 7/12/16 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-RO-EOP04 **Objective:** 3 **Point Value:** 1

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

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

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

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

Tier: 1 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 2

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

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

Given:

- Loss of all Feedwater
- SPDS "PSHT" screen selector button border is red and flashing
- HPI core cooling started

Per 1202.004, Overheating, which of the following indications confirm adequate HPI core cooling?

A. HPI cooling established for ≥ 120 minutes.

B. CET temperatures stable or dropping.

C. T-hot/T-cold differential temperature dropping.

D. Subcooling margin is adequate

Answer:

B. CET temperatures stable or dropping.

Notes:

"B" is correct since the only criteria for evaluation of adequacy of core cooling via HPI cooling is CET temps stable or dropping.
"A" is incorrect, but plausible since this elapsed time with HPI cooling established is used as a decision point (if CET temps are rising) to try more drastic measures of regaining some form of feedwater.
"C" is incorrect, but plausible since this is an individual indication of adequate primary to secondary heat transfe
"D" is incorrect, but plausible since this is normally an indication that the RCS is adequately cooled but subcooling margin can exist in an overheating condition.

This question matches the K/A as the conditions are for an inadequate heat transfer (Overheating condition for ANO-1), there is an alarm given, a remedial action of HPI cooling is in progress, and the answer choices are indications that would accompany the success of the remedial action.

Revised stem per NRC examiner suggestion.

References:

1202.004, Overheating

History:

This question is a modified version of QID 335 which was used in 1999 and 2004 RO/SRO exam.
Modified for 2005 RO re-exam.
Revised for 2016 exam

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

QID: 0891 **Rev:** 1 **Rev Date:** 9/4/14 **Source:** Bank **Originator:** Cork

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

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

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

Description: Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / generator control.

K/A Number: AK2.07 **CFR Reference:** 41.4, 41.5, 41.7, 41.10 / 45.8

Tier:	1	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	An

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

ANO-1 is at 98% power.
Due to I&C trouble shooting, ICS has been placed in manual per ICS normal operating procedure, 1105.004.
Turbine remains in Integrated Control

Later during the shift, the CBOT reports that Generator MWe load is oscillating by a few megawatts.
The ATC adds that SG pressures have been oscillating as well.

The Dispatcher calls and reports a substation has faulted causing a grid frequency perturbation.

Which of the following actions will stop these oscillations?

A. Place the Generator Automatic Voltage Regulator (AVR) in Manual

B. Place the EHC controls in Turbine Manual

C. Place the S/G Rx Master back in Automatic

D. Place both FW Loop Demands back in Automatic

Answer:

B. Place the EHC controls in Turbine Manual

Notes:

This question comes from ANO specific OE. The speed feedback correction to the Turbine Controls is always there unless the Turbine EHC is taken to Turbine Manual. In the conditions given, the Turbine will be in ICS Auto. Normally, the speed error feedback causes no noticeable changes to the operator since the ICS will adjust for any variation caused by Turbine Control speed correction, and the speed error corrections are very small. However, if the ICS is in Manual, then the Main Turbine acts like a (SG) header pressure controller. If a significant grid disturbance occurs during this mode of operation, then the Main Turbine controls will try to maintain 1800 RPM and will close or open the Governor Valves in an attempt to do so. This will cause SG header pressure to change and the ICS will send a signal to the Main Turbine to position the Governor Valves to correct header pressure, and this signal will be opposite of the speed error correction within the EHC control system. This will cause oscillations until the EHC control is taken to Turbine Manual which removes all feedback corrections, ICS as well as speed. Placing the ICS back in full automatic mode will also correct the oscillations but that is not one of the choices given.
Answer B is correct per the above explanation.
Answer A is incorrect but plausible, an examinee will notice that a grid disturbance is the cause of the problem but changing the generator field voltage will not mitigate the oscillation.
Answer C is incorrect but plausible if the examinee recalls the Turbine signal is downstream of the SG/Rx Master and believes that putting this part of the ICS back in auto will correct the oscillation. However, the speed correction will still be there since it is part of Turbin Controls and not ICS.
Answer D is incorrect but plausible if the examinee believes placing feedwater loop demand control in automatic will allow the ICS to counteract the perturbations.

References:

STM 1-24, Turbine Controls and Auxiliaries
A1LP-RO-TURB, Main Turbine Controls and Auxiliaries

History:

New for 2014 Exam.
Selected for 2016 exam

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

QID: 0320 **Rev:** 0 **Rev Date:** 9-6-99 **Source:** Bank **Originator:** J. Simmons

TUOI: ANO-1-LP-RO-AOP **Objective:** 3 **Point Value:** 1

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: Interaction of ICS control stations as well as purpose, function, and modes of operation of ICS.

K/A Number: AK1.13 **CFR Reference:** CFR: 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** Ap

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

A dropped rod event has occurred (one CRA in Group 7) and the following conditions exist:

- Reactor power = 30% and decreasing.
- Turbine output = 320 MWe and decreasing.
- Annunciator (K07-C3) HIGH LOAD LIMIT is in fast flash.
- Turbine runback is in progress.

What operator action is procedurally required?

A. Allow the runback to terminate normally.

B. Take manual control of the turbine and raise load.

C. Take manual control of SG/RX master.

D. Trip the reactor.

Answer:

C. Take manual control of SG/RX master.

Notes:

"C" is the correct answer as the ICS will runback the plant on a dropped rod to 40% of 902 Mwe. If Rx power is at 30% and still decreasing, then some malfunction must have occurred and the operator is directed to take the SG/RX master to hand per 1203.012F.
"A" is incorrect since the runback should have terminated at ~40%.
"B" is incorrect, this will only raise the turbine generator load and force the rest of the plant to follow it, an undesirable method of plant control and will not correct the ICS malfunction to the Reactor or Feedwater.
"D" is overly conservative, no setpoints have been exceeded and manual control has not been attempted.

This question matches the K/A since a dropped rod is given in the conditions and the candidate must have knowledge of the operational implication of the interaction with ICS (runback should stop at 40%) and the appropriate action to take: take manual control of the SG/Rx master station.

References:

1203.012F, Annunciator K07 Corrective Action
1203.003, Control Rod Drive Malfunction Action, Section 2 - Dropped Rod - Reactor Critical

History:

Used in 1999 exam.
Direct from ExamBank, QID# 2868
Selected for 2016 exam.

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

QID: 0184 **Rev:** 2 **Rev Date:** 7/12/16 **Source:** Bank **Originator:** R. Fuller

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.3 **Point Value:** 1

Section: 4.2 **Type:** Generic AOP

System Number: 032 **System Title:** Loss of Source Range Nuclear Instrumentation

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Guidance contained in EOP for loss of source-range nuclear instrumentation.

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

Tier: 1 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 4

Group: 2 **SRO Imp:** 4.1 **SRO Select:** No **Taxonomy:** An

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

Given:

- Reactor startup in progress.
- Source Range NI-2 and reactor power wide range recorder NR-502 are inoperable.
- Intermediate range NI-3 indicates 5 E-11 amps.
- Intermediate range NI-4 iindicates 7 E-11 amps.

Subsequently Source Range NI-1 fails to 10 E5 cps.

Which of the following is the required procedural action for the above conditions?

A. Continue the startup utilizing NI-3, only one IR channel is required for startup.

B. Immediately initiate a plant shutdown and insert all control rods because both SR and IR channels have failed.

C. Trip the reactor due to no on-scale indication of neutron flux.

D. Hold power constant and restore one SR channel to operable status.

Answer:

C. Trip the reactor due to no on-scale indication of neutron flux.

Notes:

"C" is correct per guidance in 1203.021, if the recorder NR.502 is inoperable AND no SR channel is >10 E5 cps AND no IR channel is > 1 E-10 amps AND 3/4 PR instruments are <10% power, then no on-scale flux indicatior exists and the reactor must be tripped.
"A" is incorrect but plausible since 1203.021 would allow continued operations with both SR channels failed with one IR channel indicating > 10 E-10 amps. However, NI-4 is indicating less than 1 E-10 amps.
"B" is incorrect but plausible since shutting down is conservative and required when both SR channels are failec and both IR channels < 10 E-10 amps, per 1203.021 the reactor must be tripped immediately with no on-scale indication of neutron flux. If some flux indication such as NR-502 were available, then this action would be correct per 1203.021 but the conditions state that NR-502 is inoperable.
"D" is incorrect but plausible as this action sounds like it could rectify this situation, however, it would be contrary to procedural guidance.

This question matches the K/A since the conditions give a loss of Source Range nuclear instrumentation and requires the candidate to recall the correct action in the AOP for this malfunction, and the reason for this action.

Revised at suggestion of NRC examiner.

References:

1203.021, Loss of Neutron Flux Indication

History:

Developed for use in 98 RO Re-exam
Used in 2001 RO/SRO Exam.
Selected for 2002 RO/SRO exam.
Revised for 2016 exam.

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

QID: 1061 **Rev:** 1 **Rev Date:** 7/12/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP06 **Objective:** 9 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 037 **System Title:** Steam Generator Tube Leak

Description: Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Actions to be taken if S/G goes solid and water enters steam lines

K/A Number: AA2.14 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.4 **SRO Select:** No **Taxonomy:** An

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

Given:

- Plant was shutdown due to tube leak in "A" OTSG.
- Emergency cooldown rate was used due to escalation of the tube leak to a tube rupture.
- "A" OTSG level has risen to 415".
- RCS pressure is being maintained by ATC at 1090 psig.
- T Hot has just lowered to 489 degrees F.

Which of the following is a procedurally acceptable RCS pressure band for the above conditions?

A. 950 to 970 psig

B. 1000 to 1020 psig

C. 1030 to 1050 psig

D. 1060 to 1080 psig

Answer:

B. 1000 to 1020 psig

Notes:

"B" is correct since the ruptured SG (A) has risen to above 410", then there is a chance that water could enter the steam lines and the A OTSG should be isolated since Thot is now less than 490°F (less than saturation temperature for 1050 psig, the setpoint of the lowest set MSSV). The Tube Rupture EOP (1202.002) directs maintaining RCS pressure at or below the ADV setpoint of 1020 psig (and the ADV maintained in Auto) to preclude lifting the lowest pressure Main Steam Safety Valve (1050 psig).
"A" is incorrect but plausible since this band is less than 1050 psig but at 1000 psig the Subcooling Margin limit transitions to 50°F from 30°F and this band would cause SCM to become inadequate.
"C" is incorrect but plausible since this band maintains SCM but is higher than the ADV setpoint and the lowest pressure Main Steam Safety Valve (1050 psig).
"D" is incorrect but plausible since this band maintains SCM, is slightly less than the current RCS pressure, but encompassses several MSSV setpoints.

Pressure bands are given in the answers to preclude having more than one correct answer and still have plausible distractors.

This question matches the K/A as the conditions give a SG tube leak and the action taken is due to the possibility of water entering the steam line and lifting the lowest set MSSV.

References:

1202.006, Tube Rupture
Bases for 1202.006

History:

New question for 2016 exam.

QID: 1062 **Rev:** 0 **Rev Date:** 7/13/16 **Source:** Modified **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 051 **System Title:** Loss of Condenser Vacuum

Description: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

K/A Number: 2.4.2 **CFR Reference:** 41.7 / 45.7 / 45.8

Tier:	1	**RO Imp:**	4.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**		**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- K05-B2, CONDENSER VACUUM LO is in alarm
- K05-B3, VACUUM PUMP AUTO START is in alarm
- Power reduction is in progress due to rapidly lowering condenser vacuum.
- Plant is currently at 29% power
- E-11A North Waterbox is OOS for maintenance

Choose the correct procedural requirement:

A. Trip the reactor and turbine when vacuum drops below 26.5 inches Hg.

B. Trip the reactor and turbine when vacuum drops below 24.5 inches Hg.

C. Trip only the turbine when vacuum drops below 26.5 inches Hg.

D. Trip only the turbine when vacuum drops below 24.5 inches Hg.

Answer:

C. Trip only the turbine when vacuum drops below 26.5 inches Hg

Notes:

"C" is correct, since power is below 43% a reactor trip is not required and since power is slightly less than the equivalent of 270 Mwe, then the Westinghouse recommended setpoint manually tripping the turbine at 26.5" Hg lowering condenser vacuum is in effect to preclude stall flutter of the Low Pressure Turbine last stage blading.
"A" is incorrect but plausible, the turbine should be tripped at this vacuum but not the reactor. Power is at 29% which is greater than 43% - the point at which both Reactor and Turbine should be tripped if the Main Turbine is tripped.
"B" is incorrect but plausible, this is the automatic low vacuum trip setpoint for the Main Turbine but at this power level the turbine should be manually tripped earlier due to the stall flutter issue stated above. The reacto does not have to be tripped unless power is less than 43%.
"D" is incorrect but plausible, this s the automatic low vacuum trip setpoint for the Main Turbine but at this powe level the turbine should be manually tripped earlier due to the stall flutter issue stated above. If the operator waits until 24.5" Hg vacuum, then stall flutter could have occurred, caused vibration and cracking of the LP turbine last stage blading, a blade could be ejected possibly causing equipment damage or personnel injury.

Modified QID 10 by lowering plant power from 60% to 29%, this makes choice "C" correct vs. "B". Based on reviewer comment, revised answer choices so that vacuum matches procedure action steps.

This question matches the K/A since it involves a loss of condenser vacuum and requires the candidate to recal AOP setpoints for tripping the turbine.

Revised at suggestion of NRC examiner.

References:

1203.016, Loss of Condenser Vacuum

History:

Modified QID 10 for 2016 exam.

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

QID: 1096 **Rev:** 0 **Rev Date:** 6/6/16 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 076 **System Title:** High Reactor Coolant Activity

Description: Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity : Corrective actions as a result of high fission-product radioactivity level in the RCS

K/A Number: AK3.05 **CFR Reference:** 41.5,41.10 / 45.6 / 45.13

Tier:	1	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	K

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

Given:
- Unit 1 is at 100% power
- "A" SG has a primary-to-secondary leak rate of 5 gpd
- RCS activity has been trending up
- Plant computer R1237 "Failed Fuel Gross" goes into alarm
- Failed Fuel Iodine monitor RI-1237S is out of service

RP reports the dose rate at CA-1 is 100 mR/hr.

Which of the following procedure actions would be taken and why?

A. Secure Zinc Injection to reduce activation of Zinc molecules.

B. Remove all but two Condensate polishers from service to minimize radiation exposure to personnel.

C. Isolate letdown to reduce dose rates in the aux building.

D. Stop continuous trench dump to prevent releasing contaminated effluent to Lake Dardanelle.

Answer:

C. Isolate letdown to reduce dose rates in the aux building.

Notes:

"C" is an action taken per 1203.019, High Activity in Reactor Coolant, Section 1 "High Gross Gamma Activity" and the reason given is to reduce dose rates. The does rate given at CA-1 (RCA exit point) would cause Operations to isolate letdown.
"A", "B", and "D" are all incorrect since these actions are from 1203.014, Control of Secondary System Contamination, which would be performed for a tube rupture event. The condition of "A" SG pri-sec leak rate of 5 gpd gives added plausibility to these distracters but 1203.023, Small Steam Generator Tube Leaks, does not require performance of 1203.014 until "A" SG pri-sec leak rate is up to 10 gpd. The reasons for these actions are correct as well, supporting their plausibility. All of the actions are removing components from service, the same as the correct answer.

This question matches the K/A since the conditions place the operator into the AOP for high activity in the RCS, contains a corrective action from that procedure, and asks for knowledge of the reason for the action.

References:

1203.019, High Activity in Reactor Coolant, Section 1 "High Gross Gamma Activity"

History:

New question for 2016 exam.

QID: 0162 **Rev:** 1 **Rev Date:** 6/10/16 **Source:** Modified **Originator:** J. Cork

TUOI: A1LP-RO-AOP **Objective:** 4.3 **Point Value:** 1

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

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

Description: Ability to operate and/or monitor the following as they apply to the (Plant Runback): Operating behavior characteristics of the facility.

K/A Number: AA1.2 **CFR Reference:** 41.7 / 45.5 / 45.6

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

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

Reactor power is 90% and generated megawatts is 800.

After a loss of one main feedwater pump, the ICS should runback at ____________ and stabilize the plant at _______________.

A. 50%/min, 675 MWe

B. 50%/min, 360 Mwe

C. 30%/min, 675 MWe

D. 30%/min, 360 Mwe

Answer:

B. 50%/min, 360 MWe

Notes:

"B" is correct as this question asks the trainee to recall the ICS runback rate and limit for the loss of one MFW pump.
"A" is incorrect, but plausible as this is the correct runback rate and limit for a loss of one RCP.
"C" is incorrect, but plausbile as this is the runback rate for an asymmetric rod and the limit for a loss of one RCP.
"D" is incorrect, but plausible since 360 Mwe is the correct runback limit value but the rate for an asymmetric rod.

Revised this question due to C and D being implausible distracters. Made question a 2 by 2, adding the runback rate to the stem, and to all four answer choices.

This question matches the K/A since it's focus is on a plant runback and operator must know the operating behavior of the facility by knowing what power the plant will be at and how fast it will get there on a MFW pump trip.

References:

1105.004, Integrated Control System

History:

Taken from Exam Bank QID # 4
Used in 98 RO Re-exam
Selected for use in 2005 RO exam, replacement question. K/A A01 AK2.2
Modified for 2016 exam.

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

QID: 0276 **Rev:** 1 **Rev Date:** 6/10/16 **Source:** Bank **Originator:** D Slusher

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

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

System Number: A05 **System Title:** Emergency Diesel Actuation

Description: Ability to operate and / or monitor the following as they apply to the (Emergency Diesel Actuation): 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:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	K

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

Given:

- A loss of offsite power has occurred.
- Annunciator K01A1, "EDG 1 AUTO START COMMAND", is in alarm.
- Annunciator K01-B1, "EDG 1 BRKR AUTO CLOSE FAILURE", is in alarm.
- No other alarms are in on EDG #1

What action will close EDG #1 output breaker (A-308)?

A. Place EDG #1 output breaker in PULL-TO-LOCK and release.

B. Take EDG #1 lockout handswitch to LOCKOUT and back to NORMAL.

C. Depress reset push-button on local engine control panel.

D. Place EDG #1 output breaker handswitch on C-10 in the CLOSE position.

Answer:

A. Place EDG #1 output breaker in PULL-TO-LOCK and release.

Notes:

"A" is correct, taking HS to PTL will reset anti-pump relays and allow breaker to auto-close.
"B" is incorrect, but plausible. This action will trip the output breaker if cycled while it was closed but will not reset the breaker.
"C" is incorrect, but plausible. This action will reset the K-11 Emergency Trip Relay which will energize the EDG lockout relay but with no other alarms in this could not be the cause. Also, the EDG lockout relay must be reset on A308 to allow the breaker to close if this was the reason.
"D" is incorrect but plausible as this action is direct by the ACA but the breaker cannot be closed manually from C-10 unless the sync switch is ON.

Revised question due to non-plausible distracters. Revised "C" from resetting A1 lockout to using reset PB on local engine control panel. Revised "B" from pressing start pushbutton to cycling lockout HS on C10. Added K01-A1 annunciator since it would be in for this situation.

This question matches the K/A since it involves an EDG actuation and a failure mode as well as the ability to operate handswitches to reset the failure mode (anti-pump relay).

References:

1203.012A, Annunciator K01 Corrective Action

History:

Developed for 1999 exam.
Selected for 2005 exam
Revised for 2016 exam

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

QID: 1064 **Rev:** 0 **Rev Date:** 4/13/16 **Source:** New **Originator:** Cork

TUOI: A1-LP-RO-AOP **Objective:** 5 **Point Value:** 1

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

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

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

K/A Number: AK3.2 **CFR Reference:** 41.5 / 41.10, 45.6, 45.13

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	C

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

Given:
- Heavy rains have caused lake level to rise to 350 ft.
- Lake level is forecast to rise to 355 ft. today.
- All procedural steps for electrical loads have been completed per 1203.025, Natural Emergencies.

How will A3 and A4 4160v buses be powered and why?

A. From their respective EDG's due to elevation of the diesels.

B. From Startup Transformer #1 via A1/A2 due to transformer capacity.

C. From the AAC DG due to flooding concerns with the fuel oil vaults.

D. From Startup Transformer #2 via A1/A2 due to installlation of overhead links.

Answer:

D. From Startup Transformer #2 via A1/A2 due to installlation of overhead links.

Notes:

"D" is correct, per 1203.025, Section 6, Flood, protective trips are defeated and other actions taken so Startup Transformer #2 will be used to power on-site loads and SU#1 de-energized prior to lake level exceeding 354 ft.
"A" is incorrect but plausible in that EDG's are the Class 1E backup to the vital buses.
"B" is incorrect but plausible in that SU #1 does have a greater capacity than SU#2.
"C" is incorrect but plausible in that there have been flooding concerns raised with the EDG fuel oil vaults within the last year, thus this question incorporates site specific OE. A new watertight door has been installed as part of the resolution of these concerns. Condition reports have been initiated on room penetration sealing material but those have yet to be resolved.

This question matches the K/A since it involves flooding and thas the candidate recall actions, and the reasons for the actions, found in the abnormal operating procedure for flooding (1203.025).

References:

1203.025, Natural Emergencies, Section 6, Flood

History:

New question for 2016 exam.

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

QID: 1063 **Rev:** 0 **Rev Date:** 4/13/16 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 10 **Point Value:** 1

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the (LOCA Cooldown): Components, capacity, and function of emergency systems.

K/A Number: EK1.1 **CFR Reference:** 41.8 / 41.10 / 45.3

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

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

Given:
- There are un-isolable steam leaks on both SGs.
- The RCS is solid, procedure 1202.011, HPI Cooldown, is in use.
- RCS pressure = 1700 psig and stable.
- RCS temperature is 560°F and slowly dropping.

The CRS directs the ATC to raise the cooldown rate from 20°F/hr to 25°F/hr.
The ATC raises HPI flow by 5 gpm.

How will RCS pressure respond to the change in HPI flow?

A. Rise initially and then drop to 1650 psig.

B. Remain stable due to PZR spray operation.

C. Drop below the ESAS actuation setpoint.

D. Rise to greater than 2000 psig.

Answer:

D. Rise to greater than 2000 psig.

Notes:

"D" is correct, with an increase in HPI flow without a corresponding increase in letdown flow, RCS pressure will rise signficantly due to solid plant conditions. The increase in pressure will occur first, the resulting rise in cooldown will take longer.
"A" is incorrect, but plausible if the operator believes the change in HPI flow alone will cause the cooldown rate to change enough to override the additional mass input into the RCS and thus will cause RCS pressure to lower .
"B" is incorrect, but plausible since PZR spray is a normal method of pressure control, but ineffective when RCS is solid.
"C" is incorrect, but plausible if candidate believes raising HPI flow will cause a cooldown equivalent to the CRS's direction and uses the thumbrule of 1°F temp change will cause 100 psig pressure change.

QID 471 was modified by adding the un-isolable steam leaks, making the SGs unavailable and the reason for entry into 1202.011.
The pressure and temperature conditions were modified slightly. The CRS direction to raise the cooldown rate was added along with the ATC raising HPI flow (but not letdown) to make "D" the correct answer, vs. "C" in QID 471.

References:

1202.011, HPI Cooldown

History:

Modified QID 471 (last used 2004) for 2016 exam.

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

QID: 0326 **Rev:** 1 **Rev Date:** 7/13/16 **Source:** Direct **Originator:** Stanley

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

Section: 3.4 **Type:** Heat Removal From Reactor Core

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

Description: Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply.

K/A Number: K6.02 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** C

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

Reactor Coolant Pump (P32A) has a 2.6 gallon seal bleedoff flow.

What should happen to seal bleedoff temperature if seal injection is subsequently lost?

A. Rise to potentially seal damaging temperature >200 °F due to bleedoff in excess of seal cooler capacity.

B. Rise to potentially seal damaging temperature >200 °F due to loss of flow to the seal cooler.

C. Rise to ~170 °F due to seal bleedoff within cooler capacity, no seal damage expected.

D. Remain the same due to seal recirc flow impeller circulation.

Answer:

A. Rise to potentially seal damaging temperature >200 °F due to bleedoff in excess of seal cooler capacity.

Notes:

"A" is correct. The RCP seal cooler is rated at 2.5 gpm, seal leakage plus bleedoff. If seal injection is lost, RCP seal bleedoff temperatures will rise above 170°F. Therefore if the bleedoff flow is >2.5 gpm, seal bleedoff temperatures will rise.
"B" is incorrect but plausible since the seal cooler is the issue but ICW supplies cooling water to the seal cooler.
"C" is incorrect but plausible if candidate cannot recall seal cooler capacity
"D" is incorrect but plausible because the recirc impeller provides seal cooling but the capacity of the seal cooler is exceeded in this situation.

Revised using NRC examiner suggestions.

References:

1203.031, Reactor Coolant Pump and Motor Emergency

History:

Used in 1999 exam Direct from ExamBank, QID# 3266 KA 003 A4.06
Selected for 2014 Exam.
Selected for 2016 exam.

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

QID: 0258 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-MU **Objective:** 8 **Point Value:** 1

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the CVCS: Relationship between VCT pressure and NPSH for charging pumps

K/A Number: K5.26 **CFR Reference:** CFR: 41.5/45.7

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

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

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

Given:

- "A" HPI pump is operating.
- Makeup tank level is 80 inches.
- Makeup tank pressure is 12 psig.
- RCS sampling is in progress.

With no operator action, what will occur if the Makeup Tank Inlet Valve (MU-12) was accidentally closed by chemistry personnel?

A. "A" HPI pump will be damaged due to loss of suction.

B. The Makeup Tank vent valve CV-1257 will open on low pressure.

C. The RCP seals will be damaged due to low seal injection flow.

D. The BWST Outlet Valve CV-1407 receives an open signal.

Answer:

a. "A" HPI pump will be damaged due to loss of suction.

Notes:

"A" is correct, with the loss of letdown into the Makeup Tank the level will continue to lower until low level and low pressure cause a loss of NPSH for the "A" HPI pump followed by pump damage. This question is based on ANO-1 specific OE when a chemist meant to isolate an RCS sample and closed the MUT inlet instead.
"B" is incorrect, but plausible since the Makeup Tank vent valve opens on low level of 18" but not low pressure.
"C"" is incorrect, but plausible since seal injection flow will cease to exist but as long as seal cooling is still provided by ICW, then the seals should be OK.
"D" is incorrect, but plausible as the BWST outlet valve does automatically open but on ESAS signal, not MUT level.

References:

1104.002, Makeup and Purification System

History:

Developed for 1999 exam.
Selected as replacement question for QID1069 based on NRC examiner comment on 2016 exam

QID: 0654 **Rev:** 0 **Rev Date:** 12/8/06 **Source:** Bank **Originator:** Cork/Possage

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

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

System Number: 004 **System Title:** Chemical and Volume Control System (CVCS)

Description: Ability to monitor automatic operation of the CVCS, including: Letdown isolation.

K/A Number: A3.02 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

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

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

The makeup and purification system is in operation with 70 gpm letdown flow, when the following indications are observed.

Letdown flow-- 0 gpm
Letdown pressure-- 200 psig
Makeup tank level-- 76" decreasing

Which of the following transients caused the above indications?

A. Loss of power to the letdown demineralizer inlet valves.

B. Loss of Inst. Air to the letdown block orifice inlet and bypass valves.

C. Letdown isolation due to high temperature.

D. Inadvertent closure of the Makeup Tank Outlet Isolation.

Answer:

A. Loss of power to the letdown demineralizer inlet valves.

Notes:

"A" is correct, a loss of power or air to the letdown DI inlets will cause them to go closed. Their closure will cause pressure to increase rapidly lifting the letdown relief valve at 200 psig.
"B" is incorrect, but plausible since these valves are in the flowpath, this will not cause isolation of letdown, the bypass closes but the block fails as-is on loss of instrument air.
"C" is incorrect, but plausible since letdown flow is zero and this will isolate letdown but pressure will not be high since the isolation occurs upstream of the relief valve.
Answer "D" is incorrrect, but plausible since this is in the suction path of the makeup pumps as is letdown, but this will not isolate letdown, flow will not be at zero.

This question matches the K/A since it involves the ANO-1 equivalent of CVCS (Makeup & Purification) and the question involves the condtions which would be seen if letdown was isolated by closure of the letdown DI inlet valves.

References:

1203.036, Loss of 125V DC

History:

Selected for 2007 RO Exam. Direct from regular exambank QID# ANO-OPS1-3211
Selected for 2016 exam.

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

QID: 1068 **Rev:** 0 **Rev Date:** 4/18/16 **Source:** New **Originator:** Cork

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

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

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

Description: Knowledge of the physical connections and/or cause-effect relationships between the RHRS system and the following systems: RCS.

K/A Number: K1.09 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** C

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

Given:
- Plant Shutdown and cooldown is in progress.
- Reactor Coolant Pumps P-32C and P-32D are running.
- RCS pressure is 240 psig.
- Procedure preparations are in progress to place the first Decay Heat Removal pump, P-34A, in service.
- DH Cooler Outlet valve CV-1428 is closed.
- E-35A Cooler Bypass valve CV-1433 is 50% open.
- Decay Heat Block valve CV-1401 is open.

Which of the following will occur when P-34A Decay Heat pump is started?

A. BWST level will rise

B. Pressurizer level will drop

C. DH pump will be damaged

D. RCS cooldown rate will be exceeded

Answer:

B. Pressurizer level will drop

Notes:

"B" is the correct answer per 1104.004, the Cooler Bypass valve should be throttled to 74-80% open and past experience has shown that even at this position if RCS pressure is greater than 220 psig (DH pump normally develops ~150 psig), the discharge relief could lift (see step 7.4.8.A.2). With the Cooler Bypass valve incorrectly throttled to 50% open, the discharge relief will definitely lift and Pressurizer level will lower.
"A" is incorrect but plausible if candidate does not recall that pressure surge path is isolated prior to opening DH suction from RCS CV-1404. Pressure surge path was installed due to injection line backleakage and concerns with voids due to gases coming out of solution.
"C" is incorrect but plausible if candidate does not recall system dynamics correctly and believes the pressure could cause CV-1050 or CV-1410 to auto close.
"D" is incorrect but plausible if candidate does not recall system startup configuration and believes the bypass is too far open and flow will be too high resulting in a cooldown from relatively cool stagnant water in A Decay Hea loop.

This question matches the K/A since it involves Decay Heat system operation (RHRS), involves physical connections between RCS and DH, and determines if the candidate understands the cause-effect on both DH and RCS when starting a DH pump without the Cooler Bypass valve open suffficiently.

References:

1104.004, Decay Heat Removal Operating Procedure

History:

New question for 2016 exam.

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

QID: 0611 **Rev:** 1 **Rev Date:** 6/1/16 **Source:** Bank **Originator:** Cork/Pullin

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

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

System Number: 005 **System Title:** Residual Heat Removal System (RHRS)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Detection of and response to presence of water in RHR emergency sump.

K/A Number: A1.05 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 3

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

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

Given:

- Plant is in Mode 5.
- "A" Decay Heat Removal system is in service.
- Annunciators K09-C7 "TRAIN A RCS LEVEL LO" and K09-D7 "TRAIN B RCS LEVEL LO" alarm.
- RB sump level 40% and rising. This is a step change of 2%.

Which of the following actions are procedurally required to be performed FIRST?

A. Start P-34B Decay Heat pump and secure P-34A Decay Heat pump.

B. Stop P-34A Decay Heat pump and close CV-1404, Decay Heat Suction.

C. Stop P-34A Decay Heat pump and close CV-1434, P-34A Suction from RCS.

D. Fill RCS by starting P-34B Decay Heat pump using LPI flowpath.

Answer:

B. Stop P-34A Decay Heat pump and close CV-1404, Decay Heat Suction.

Notes:

Answer "B" is the correct response, an RCS leak of >20 gpm is indicated (2% step change in RB sump level = ~90 gallons), the pump should be secured and the suction from the RCS isolated.
Answer "A" is incorrect but plausible, this is the response to other problems with P-34A, this will not mitigate the low level.
Answer "C" is incorrect but plausible, closing CV-1434 will isolate the suction to P-34A from the RCS (this is required in Section 1 of 1203.028 for a leak <20 gpm) but isolating the entire DH system from the RCS is required.
Answer "D" is incorrect, although taking suction from the BWST and injecting to RCS would raise level (this is the LPI flowpath) and is plausible since it is an option to makeup for lost inventory later in the procedure, it is no an action to take "first" to mitigate the level loss.

References:

1203.028, Loss of Decay Heat Removal
1203.012H, Annunciator K09 Corrective Action

History:

New for 2005 RO exam, replacement question.
Selected for 2013 RO exam
Selected fof 2016 exam

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

QID: 1090 **Rev:** 0 **Rev Date:** 6/2/16 **Source:** Mod **Originator:** Cork

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

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

System Number: 006 **System Title:** Emergency Core Cooling System (ECCS)

Description: Knowledge of bus power supplies to the following: ESFAS-operated valves.

K/A Number: K2.04 **CFR Reference:** 41.7

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

Group: 1 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

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

Which of the following provides motor power to CV-1408, BWST Outlet Valve?

A. B-8

B. B-6

C. B-4

D. B-2

Answer:

B. B-6

Notes:

"B" is correct, B-6 supplies B-61 which supplies power to CV-1408.
"A", "C", and "D" are incorrect but plausible since they are also even numbered green train load centers.

This is a modified version of QID 903, it asks for power for "green" train BWST outlet CV-1408 (vs. 1407) which required changing all of the answers to green train load centers.

References:

1107.002, ES Electrical System Operation

History:

Modified QID 903 for 2016 exam

QID: 0561 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** Bank **Originator:** S.Pullin

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

Section: 3.5 **Type:** Containment Integrity

System Number: 007 **System Title:** Pressurizer Relief Tank/Quench Tank System

Description: Knowledge of the operational implications of the following concepts as they apply to the PRTS: Method of forming a steam bubble in the PZR.

K/A Number: K5.02 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** Ap

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

A plant startup is in progress with a steam bubble being drawn in the Pressurizer.
- Initial Quench Tank pressure is 3 psig.
- RCS pressure 75 psig.
- Pressurizer temperature 320°F.

Which of the following assures that venting and steam bubble formation is complete in the Pressurizer?

A. Quench Tank pressure 7.6 psig after a 3 minute blow of the ERV.

B. Quench Tank pressure 6.2 psig after a 3 minute blow of the ERV.

C. Quench Tank pressure 4.8 psig after a 3 minute blow of the ERV.

D. Quench Tank pressure 3.5 psig after a 3 minute blow of the ERV.

Answer:

D. Quench Tank pressure 3.5 psig after a 3 minute blow of the ERV.

Notes:

"D" is correct with Quench Tank pressure rise less than or equal to 1 psig.
All other choices contain greater than 1 psig pressure rise which indicates nitrogen is still being vented to the Quench Tank. They are all plausible if the candidate cannot recall the indications of bubble formation from 1103.005.

This question matches the K/A since it involves the Quench Tanks as it relates to forming a steam bubble.

References:

1103.005, Pressurizer Operation

History:

New for 2005 RO exam, later modified for replacement.
Selected for 2010 RO/SRO exam.
Selected for 2016 exam.

QID: 0627 **Rev:** 0 **Rev Date:** 11/7/05 **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: Ability to monitor automatic operation of the CCWS, including: Requirements on and for the CCWS for different conditions of the power plant. .

K/A Number: A3.04 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
- Plant is at 100% power.
- ICW pump P-33B is in service on Nuclear ICW.

What would be the effect on the ICW system if the Non-Nuclear ICW pump tripped?

A. ICW pump P-33A would auto-start, P-33B would be unchanged.

B. ICW pump P-33C would auto-start, P-33B would be unchanged.

C. ICW pump P-33B would shift to Non-Nuclear loop, P-33C would auto-start.

D. ICW pump P-33B would shift to Non-Nuclear loop, P-33A would auto-start.

Answer:

C. ICW pump P-33B would shift to Non-Nuclear loop, P-33C would auto-start.

Notes:

"C" is correct, P-33B will shift to loop with lowest pressure (non-nuclear) and the non-swing nuclear pump P-33C would auto-start.
"A" is incorrect, although plausible since it is one of the other two ICW pumps, P-33A is the non-nuclear ICW pump.
"B" is incorrect, although P-33C will auto-start, P-33B is the swing pump and will re-align to the non-nuclear loo
"D" is incorrect, but plausible, however P-33A is the non-nuclear ICW pump.

This question matches the K/A since it involves ANO-1 equivalent of CCWS (Intermediate Cooling Water - ICW) and it requires the candidate to recall knowledge of the auto-start sequence of the ICW pumps.

References:

STM 1-43, Intermediate Cooling Water

History:

New for 2005 RO re-exam.
Selected for 2016 exam.

QID: 1070 **Rev:** 1 **Rev Date:** 7/20/16 **Source:** Modified **Originator:** Cork

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

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

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

Description: Knowledge of annunciator alarms, indications, or response procedures.

K/A Number: 2.4.31 **CFR Reference:** 41.10 / 45.3

Tier:	2	**RO Imp:**	4.2	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	An

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

Given:
- Plant is at 100% power.
- ICW Booster pump P-114A is in service.

Simultaneously the following alarms come in:
- K12-C4 BOOSTER PUMP DISCH PRESS LO
- K12-D4 ICW PUMP DISCH PRESS LO
- K08-E7 RCP SEAL COOLING FLOW LO

The ATC announces that BOTH P-33C ICW pump AND P-114A ICW Booster pump have tripped.
The CBOT states that ALL RCP Seal Cooling Flow Low lamps are lit on C13.

Which of the following will occur FIRST in response to the above conditions?

A. ICW pump P-33B starts immediately.

B. RCP Seal Cooling Pump Bypass CV-2287 opens.

C. ICW Booster pump P-114B starts.

D. ICW Nuclear Loop Inlet Isolation CV-2233 closes.

Answer:

B. RCP Seal Cooling Pump Bypass CV-2287 will open

Notes:

"B" is correct, with the ICW pump discharge pressure low alarm at 35 psig, the standby pump P-33B will start but only after a 10 second time delay. The standby ICW Booster Pump P-114B will not start on low discarge pressure due to it's suction pressure less than 45 psig. Therefore, with both Booster Pumps off the RCP Seal Cooling Bypass valve CV-2287 will open to try to maintain some ICW flow to the RCP seal coolers.
"A" is incorrect but plausible since ICW pump P-33B will start on low discharge pressure of P-33C but only after a 10 second time delay.
"C" is incorrect but plausible since the ICW Booster pump P-114B will start on low discharge pressure of P-114A but only if it's suction pressure is greater than 45 psig. P-114B suction pressure can't be greater than 45 psig however since P-33C discharge pressure is <35 psig.
"D" is incorrect but plausible since this valve (CV-2233) will isolate the ICW flow to these pumps but this only occurs on ESAS actuation.

QID 94 was modified extensively by giving initial conditions, adding the alarms, adding the operator announcements, changing the stem, and replacing one distracter.

This question matches the K/A since it concerns ICW (CCW) and requires the candidate to have knowledge of alarm setpoints and alarm response procedure corrective actions (verify bypass valve opens).

Replaced D distracter per NRC examiner comment.

References:

1203.012K, Annunciator K12 Corrective Action

History:

Modified QID 94 for 2016 exam.

QID: 1071 **Rev:** 0 **Rev Date:** 4/20/16 **Source:** New **Originator:** Cork

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

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

System Number: 010 **System Title:** Pressurizer Pressure Control System (PZR PCS)

Description: Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR sprays and heaters.

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

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

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

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

Given:
- The unit is at 55% power following a feedwater transient.
- RCS pressure lowered to 2115 psig during the transient.
- RCS pressure is currently 2135 psig and slowly rising.

Which of the following is indicative of a Pressurizer Pressure Control System malfunction and should be controlled in manual?

A. Proportional heaters ON

B. Heater Bank 3 ON

C. Heater Bank 4 OFF

D. Heater Bank 5 OFF

Answer:

C. Heater Bank 4 OFF

Notes:

"C" is correct, Heater Bank 4 should be OFF at 2140 psig rising so it is malfunctioning.
"A" is incorrect but plausible if the candidate cannot recall that the proportional heaters are full ON at 2135 psig (lowering) and stay on until pressure is 2155 psig.
"B" is incorrect but plausible if the candidate cannot recall that Heater Bank 3 is ON at 2135 psig (lowering) and stays on until pressure is 2155 psig.
"D" is incorrect but plausible if the candidate cannot recall that Heater Bank 5 is ON at 2105 psig (lowering) and turns off at a pressure of 2125 psig.

This question matches the K/A since it requires the candidate to determine that a malfunction of the PZR heaters has occurred.

References:

1103.005, Pressurizer Operation

History:

New question for 2016 exam.

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

QID: 0085 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Bank **Originator:** JCork

TUOI: ANO-1-LP-RO-RPS **Objective:** 6.4 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

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

Description: Knowledge of bus power supplies to the following: RPS channels, components, and interconnections.

K/A Number: K2.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

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

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

Which of the following power supplies is the normal source for RPS channel D?

A. Inverter Y22 from B65

B. Inverter Y22 from D02

C. Inverter Y24 from B61

D. Inverter Y24 from D02

Answer:

D. Inverter Y24 from D02

Notes:

"D" is correct. D RPS is powered from RS-4. RS-4 is normally supplied by Y-24 (can be supplied by Y-25). The normal power source for the inverter Y24 is DC bus D02. B-61 supplies alternate AC power to Y-24. "A", "B", and "C" are all plausible since they are green train vital AC instrument power alignments. However, they are all incorrect either to being the supply for "C" RPS or the alternate AC for Y24.

Question was revised due to having implausible distracters and being incorrect.

This question is a direct K/A match since it requires the candidate to know normal power supply arrangement for an RPS channel.

References:

1107.003, Inverter and 120V Vital AC Distribution

History:

Used in 1998 SRO exam
Used in NRC developed RO exam no. 45, 2/28/94
Selected for 2002 RO/SRO exam.
Revised for 2016 exam

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

QID: 1093 **Rev:** 0 **Rev Date:** 6/3/16 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-RPS **Objective:** 18 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

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

Description: Ability to manually operate and/or monitor in the control room: Bistable, trips, reset and test switches.

K/A Number: A4.04 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	An

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

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During a plant startup the following indications are observed:

- Rx power is 12%

- "A" MFP is operating.

- "B" MFP is tripped.

- All RPS alarms on K08 are clear.

- In the "A" RPS cabinet, the upper red light on both "A" and "B" MFP contact buffers are ON.

- In the "B", "C", and "D" RPS cabinets, the upper red lights on the "A" MFP contact buffers are ON, while the upper red lights on the "B" MFP contact buffers are OFF.

With the above conditions, what is the RPS coincidence logic for MFP trip?

A. 1 out of 3

B. 2 out of 3

C. 1 out of 4

D. 2 out of 4

Answer:

B. 2 out of 3

Notes:

"B" is correct, with the upper red lights ON in "A" RPS cabinet for both MFPs, this means the Anticipatory Reactor Trip System (ARTS) has NOT been reset in "A" RPS and thus will NOT trip on a loss of the A MFP. B/C/D RPS cabinets have been reset, so it takes two out of three channels to open the reactor trip breakers on a loss of the A MFP.
"A" is incorrect but plausible if the student believes that the indications given are that the "A" RPS channel is already tripped and only one more channel needs to trip for a reactor tip on loss of MFP.
"C" is incorrect but plausible if the student believes that the indications given are that the "A" RPS channel is already tripped and uses the standard coincidence logic (2 out of 4) for a reactor trip.
"D" is incorrect but plausible if the student correctly concludes the "A" RPS channel will not trip but mistakenly uses the standard coincidence logic (2 out of 4) for a reactor trip.

This question matches the K/A since it requires the candidate to have knowledge of correct and incorrect RPS indications for bistables and thus the monitoring aspect is addressed.

References:

1102.002, Plant Startup

History:

Selected regular exam bank ANO-OPS1-5329 for 2016 exam

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

QID: 1073 **Rev:** 1 **Rev Date:** 7/13/16 **Source:** New **Originator:** Cork

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

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

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

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

K/A Number: K3.03 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 2

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

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

Given:
- Plant is at 100% power.
- Unit 1 is in a Tech Spec LCO time clock due to P-35B RB Spray Pump out of service.

Subsequently a large break LOCA occurs.
ESAS Channel 9 fails to actuate.

Which of the following would be challenged by the above conditions?

A. Containment isolation would be challenged

B. Containment design pressure would be exceeded

C. Containment atmosphere iodine concentration would be higher.

D. Hydrogen production would be greater than design

Answer:

C. Containment atmposphere iodine concentration would be higher.

Notes:

"C" is correct, P-35B is OOS and ESAS Channel 9 failure means that no sodium hyroxide will be injected into P 35A's RB spray so Iodine removal will be diminished due to sump pH no longer being adjusted.
"A" is incorrect but plausible if candidate believes that ESAS Channel 9 contains some means of containment isolation like channels 1-6.
"B" is incorrect but plausible if candidate believes that failure of ESAS Channel 9 means that P-35A Spray pum won't start. However, four RB Coolers are available in this scenario.
"D" is incorrect, but plausible if candidate believes purpose of sodimum hydroxide pH adjustment was for reducing corrosion and thus hydrogen production.

This question matches the K/A since it involves a failure of the ESFAS (sodium hydroxide injection) and requires the candidate to know what effect that would have on containment and the resulting effect on post accident doses.

Revised question based on NRC examiner suggestion.

References:

SAR, chapter 6

History:

New question for 2016 exam.

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

QID: 1103 **Rev:** 0 **Rev Date:** 6/18/16 **Source:** New **Originator:** Cork

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

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

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

Description: Knowledge of the purpose and function of major system components and controls.

K/A Number: 2.1.28 **CFR Reference:** 41.7

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

Group: 1 **SRO Imp:** 4.1 **SRO Select:** No **Taxonomy:** K

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

In the ESAS system there are many different, but important, modules.

What is the primary purpose of bistables?

A. Provide for conversion of analog signals to digital output signals

B. Provide signals for computer and annunciator alarms

C. Provide communication links between analog and digital subsystems

D. Provide electrical isolation for signals outside of the system

Answer:

A. Provide for conversion of analog signals to digital output signals

Notes:

"A" is correct, this is the purpose of a bistable.
"B" is incorrect, this is plausible since it is a major ESAS component, but this is the purpose of auxiliary relays.
"C" is incorrect, this is plausible since it is a major ESAS component, but this is the purpose of logic buffers.
"D" is incorrect, tthis is plausible since it is a major ESAS component, but this is the purpose of contact buffers.

This question matches the K/A since it requires the candidate to know the purpose of major ESAS components, i.e., logic buffers.

References:

STM 1-65, Engineered Safeguards Actuation System

History:

New question for 2016 exam

QID: 0909 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** Bank **Originator:** Possage

TUOI: A1LP-RO-EOP10 **Objective:** 2 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 022 **System Title:** Containment Cooling System (CCS)

Description: Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Automatic containment isolation.

K/A Number: K4.03 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 42 **SRO:**

Complete the following statement:

ESAS Channel ________ will automatically isolate ________ to the Reactor Building.

A. 3 & 4
CRD Cooling, Chilled Water, RCP Motor Cooling

B. 3 & 4
Reactor Building Leak Detector, Fire Water, Letdown

C. 5 & 6
CRD Cooling, Chilled Water, RCP Motor Cooling

D. 5 & 6
Reactor Building Leak Detector, Fire Water, Letdown

Answer:

C. 5 & 6
CRD Cooling, Chilled Water, RCP Motor Cooling

Notes:

C is correct as it is the only answer with the correct ESAS channels and systems isolated.
A is incorrect but plausible as these systems are isolated by ESAS but by channels 5&6, not 3&4.
B is incorrect but plausible as two of these systems are isolated by ESAS 3&4 but Letdown is isolated by 1&2.
D s incorrect but plausible as these systems are isolated by ESAS but the first two by 3&4 and Letdown is isolated by 1&2.

This question matches the K/A since it requires the candidate to have knowledge of autmatic containment isolation of chilled water to the containment coolers.

References:

STM 1-65, Engineered Safeguards Actuation System

History:

Modified 139 for 2014 Exam
Selected for 2016 exam

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

QID: 1075 **Rev:** 0 **Rev Date:** 4/21/16 **Source:** New **Originator:** Cork

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment sump level.

K/A Number: A1.03 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 4

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

Question: **RO:** 43 **SRO:**

Given:
- Large Break LOCA has occurred.
- All ECCS components are operating as designed.
- There is no evidence of a Containment breach.
- BWST level 5.5 ft.
- RB Sump Outlet valves have been opened
- BWST Outlet valves have been closed

Which of the following parameters would indicate the RB Spray pumps are required to be secured per 1202.010, ESAS, Attachment 1?

A. Reactor Building pressure less than 4 psig

B. Both LPI pump flows greater than 2800 gpm

C. NaOH Tank level less than 16 ft

D. Reactor Building sump level dropping

Answer:

D. Reactor Building sump level dropping

Notes:

"D" is correct per 1202.010, Attachment 1. If RB sump levels drop, then this could be indicative of blockage, LPI pump flows will be throttled back to minimum and RB spray pumps secured if there are no indications of a CNTMT breach.
"A" is incorrect but plausible if the candidate believes that RB Spray pumps can be secured if the RB pressure is less than the ESAS setpoint for Channels 1-6.
"B" is incorrect but plausible, the LPI pump flow rate given is the minimum flow value.
"C" is incorrect but plausible as this NaOH level is the value indicating an adequate amount of sodium hydroxide has been injected and the NaOH valves may be closed.

This question matches the K/A since it requires the candidate to predict changes to RB sump levels as they relate to operating the RB spray pumps.

References:

1202.010, ESAS

History:

New question for 2016 exam.

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

QID: 1074 **Rev:** 1 **Rev Date:** 7/13/16 **Source:** Modified **Originator:** Cork

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

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

System Number: 039 **System Title:** Main and Reheat Steam System (MRSS)

Description: Knowledge of the operational implications of the following concepts as they apply to the MRSS: Bases for RCS cooldown limits.

K/A Number: K5.05 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 44 **SRO:**

Given:
- RCS Temperature 490 °F
- Turbine Bypass Valves being used to control cooldown
- Plant cooldown in progress due to SG tube leak.
- Transition has been made to 1202.006, Tube Rupture.
- Emergency cooldown is NOT required.

Per the 1202.006, Tube Rupture, what is the MAXIMUM cooldown rate and what is it based on?

A. 50 °F/hr, minimize stresses on bowed tie rods in S/G

B. 50 °F/hr, prevent brittle fracture of the Rx Vessel due to neutron embrittlement

C. 100 °F/hr, minimize stresses on bowed tie rods in S/G

D. 100 °F/hr, prevent brittle fracture of the Rx Vessel due to neutron embrittlement

Answer:

D. 100 °F/hr, prevent brittle fracture of the Rx Vessel due to neutron embrittlement

Notes:

"D" is correct per EOP technical bases document and Tech Spec bases.
"A" is incorrect but plausible since this would be the correct answer per 1102.010 and guidance from Framatome.
"B" is incorrect but plausible as this has the cooldown rate per 1102.010 and the correct bases for Tech Spec 3.4.3 limits.
"C" is incorrect but plausible as it has the correct cooldown rate per Tech Spec 3.4.3 but with the bases for the Framatome cooldown rate guidance in 1102.010.

Modified QID 910 by changing "1102.010, Plant Shutdown and Cooldown" to "EOP". This made "D" correct (vs. "A").

Changed stem to state "per 1202.006" due to NRC examiner comment.

References:

1202.006, Tube Rupture, EOP Technical Guide
Areva Technical Document, Vol. 3, III.E-17
Techncial Specifications, 3.4.3 and B3.4.3

History:

Modified QID 910 (2014) for 2016 exam.

QID: 0565 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** Bank **Originator:** J.Cork

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

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

System Number: 059 **System Title:** Main Feedwater (MFW) System

Description: Ability to manually operate and monitor in the control room: ICS.

K/A Number: A4.10 **CFR Reference:** 41.7 / 45.5 to 45.8

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

Group: 1 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** An

Question: **RO:** 45 **SRO:**

Given:

- 100% power
- ICS in full automatic

The CBOR places the ICS Delta T-Cold Hand Auto Station meter selection switch in "POS" (position). The meter reads 46%.

What does this mean in terms of ICS control of Main Feedwater?

A. The average of feedwater loop A and feedwater loop B demand is 46%.

B. Feedwater loop B demand is greater than feedwater loop A demand.

C. The feedwater loop A demand is being boosted by a 4 °F Delta T-Cold error.

D. Feedwater loop A demand is greater than feedwater loop B demand.

Answer:

B. Feedwater loop B demand is greater than feedwater loop A demand.

Notes:

"B" is correct, with the Delta Tc H/A station meter reading <50% in POS (position), this indicates that loop B demand is > loop A demand.
"A" is incorrect but plausible as the value 46% is stated but the meter does not indicate average demand,
"C" applies to looking at the MV (measured variable) reading (for which it would still be incorrect) but it still appears to be plausible answer.
"D" is incorrect but plausible, this is the opposite of the correct answer.

This question matches the K/A since it pertains to the Main Feedwater system and requires candidate to have the ability to monitor the relationship between MFW and the ICS Delta Tc controller indications.

References:

STM 1-64, Integrated Control System

History:

Developed for the 1998 RO/SRO Exam.
Selected for use in 2002 RO/SRO exam.
QID #63 used on 2004 RO/SRO Exam.
Modified for 2005 RO exam.
Selected for 2016 exam.

QID: 0269 **Rev:** 1 **Rev Date:** 4/22/16 **Source:** Bank **Originator:** Cork

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

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 061 **System Title:** Auxiliary/ Emergency Feedwater System

Description: Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: Emergency water source

K/A Number: K1.07 **CFR Reference:** 41.2to 41.9 / 45.7 to 45.8

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

Group: 1 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 46 **SRO:**

Which of the following is the assured water source for the Emergency Feedwater System?

A. Condensate Storage Tank T-41

B. EFW Condensate Storage Tank T-41B

C. Service Water System Loops I and II

D. ECP via FLEX transfer pump

Answer:

C. Service Water System Loops I and II

Notes:

"C" is correct, Service Water is the assured source of water to the Emergency Feedwater System.
"A","B", and "D" are alternate sources of water for the EFW system therefore they are all plausible, but incorrect

Revised question to eliminate two implausible distractors.

This question matches the K/A since it involves the emergency feedwater system and requires the candidate to recall the emergency water source for the EFW system.

References:

1106.006, Emergency Feedwater Pump Operation

History:

Used in 1999 exam, direct from ExamBank, QID# 91
Revised question for 2016 exam.

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

QID: 1076 **Rev:** 0 **Rev Date:** 4/22/16 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-EFIC **Objective:** 9 **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: S/G level.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** Ap

Question: **RO:** 47 **SRO:**

Given:
- Reactor is tripped with the plant in degraded power.
- Primary and secondary parameters are stable post trip conditions for degraded power.

Which of the following would be the proper OTSG fill rate by EFIC for the EFW system as it feeds to the required level?

A. 4 to 4.5 "/min

B. 6 to 6.5 "/min

C. 7 to 7.5 "/min

D. 7.6 to 8 "/min

Answer:

C. 7 to 7.5 "/min

Notes:

OTSG fill rate is adjusted to prevent overcooling, so the OTSG levels rise at 2 inches/minute at an OTSG pressure of 800 psig and 8 inches/minute at an OTSG pressure of 1050 psig. That equates to 0.024" per psig. At the ADV control pressure of 1020 psig (degraded power means no condenser vacuum so ADVs will be controlling) the OTSG fill rate will be 7.3 inches/minute. "C" is the correct answer.
"A" is incorrect but plausible as this would be the fill rate if the TBVs were controlling at 895 psig (normal setpoint).
"B" is incorrect but plausible as this would be the fill rate if the TBVs were controlling post-trip with a 100 psig bias.
"D" is incorrect but plausible as this would be the fill rate if SG pressure were floating on the lowest MSSV of 1050 psig.

Modified QID 270 by removing the purpose of the fill rate, removed the OTSG pressure (it was wrong anyway given the condition of degraded power) - just stated plant was in degraded power, and added bands. This changed the correct answer to ~7" per min, the previous correct answer was ~4" per min.

This question matches the K/A since it requires the candidate to have the ability to monitor the fill rate from EFW as it raises SG level.

References:

1105.005, Emergency Feedwater Initiation and Control

History:

Modified QID 270 for 2016 exam

QID: 1077 **Rev:** 1 **Rev Date:** 7/13/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ELECD **Objective:** 11j **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** A.C. Electrical Distribution

Description: Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers.

K/A Number: K4.03 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 48 **SRO:**

Given:
- Unit 1 is at 100% power.
- A spurious Reactor trip occurs.

Which of the following MUST be actuated to directly cause a "fast" transfer of the 4160v/6900v buses from the Unit Aux Transformer to the Startup #1 Transformer?

A. Startup #2 Transformer Lockout

B. Main Turbine Lockout

C. Main Generator Lockout

D. Main Generator Backup Reverse Power

Answer:

C. Main Generator Lockout

Notes:

"C" is correct, for any automatic transfer to occur a Main Generator Lockout must be present.
"A" is incorrect but plausible since a Startup #1 lockout will cause a transfer to Startup #2.
"B" is incorrect but plausbile since a Main Turbine Lockout combined with a Main Generator Reverse Power relay actuation will generate a Main Generator Lockout but a Reactor Trip will initiate the transfer without this.
"D" is incorrect but plausible in that a Main Generator Backup Reverse Power will cause a Main Generator Lockout but it is the Main Generator Lockout signal which is essential for auto transfers.

This question matches the K/A since it requires the candidate to have knowledge of the AC electrical system interlocks for automatic bus transfers, they have to know that a fast bus transfer requires a main generator lockout.

Revised stem per NRC examiner suggestion.

References:

STM 1-32, Electrical Distribution

History:

New question for 2016 exam

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

QID: 0140 **Rev:** 1 **Rev Date:** 8/1/05 **Source:** Bank **Originator:** J. Haynes

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

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** A. C. Electrical Distribution

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the AC Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Types of loads that, if de-energized, would degrade or hinder plant operation.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 49 **SRO:**

Initial conditions:

- 100% power
- P-36C is the operating makeup pump
- ICW pumps P-33A and P-33C in service

Subsequently, annunciator K01-B7 "A4 L.O. RELAY TRIP" alarms.

What RCP support system would be affected by a loss of bus A4 and which procedural actions are used to mitigate the loss of this support system?

A. Loss of Seal Injection, verify seal cooling is maintained.

B. Loss of RCP Motor Cooling, trip reactor and trip all RCPs.

C. Normal Seal Bleedoff flowpath isolated, open alternate bleedoff path to Quench Tank.

D. Loss of AC Oil Lift pumps, verify emergency DC lift pumps are available.

Answer:

A. Loss of Seal Injection, verify seal cooling is maintained

Notes:

"A" is correct. Loss of A4 results in a loss of the running HPI pump and seal cooling (via ICW) must be maintained. ICW is not lost since the pumps are powered from A1 or A2.
"B" is incorrect, but plausible, however P-33A will remain in service since it is powered from B12 (via A1) which provides motor cooling.
"C" is incorrect but plausible since a loss of A4 causes a loss of B6 which supplies power to a lot of valves, but Seal bleedoff isolation is an MOV, it won't change position, and thus seal bleedoff will not be affected by the loss of A4.
"D" is incorrect, although it sounds plausible but RCP lift oil pumps are non-vital powered.

This question matches the K/A since it involves a malfunction of the AC distribution system and determines if the candidate can evaluate which important load was lost and what action to take for the loss of this load. A loss of RCP seal injection could certainly hinder or degrade plant operation.

References:

1203.026, Loss of Reactor Coolant Makeup

History:

Taken from Exam Bank QID # 3714
Used in 98 RO Re-exam
Selected for use in 2005 RO exam.
Selected for 2016 exam.

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

QID: 1078 **Rev:** 0 **Rev Date:** 4/26/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-ELECD **Objective:** 14h **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** DC Electrical Distribution

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

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:** 3.1 **SRO Select:** No **Taxonomy:** An

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Question: **RO:** 50 **SRO:**

A malfunction of the red train Vital 125V DC electrical system has occurred.

Using the attached photograph, determine which of the following local alarms would accompany the indications shown:

A. Local annunciator for D01, "BLOWN FUSE"

B. Local annunciator for Charger D03A, "DC OUTPUT BREAKER OPEN"

C. Local annunciator for D01, "BATTERY DISCONNECT OPEN"

D. Local annunciator for Charger D03A, "HIGH DC FLOAT VOLTAGE"

Answer:

B. Local annunciator for Charger D03A, "DC OUTPUT BREAKER OPEN"

Notes:

"B" is correct. The photo shows that D-01 Amps are approximately 100. Normally, the amps on the battery are "zero" indicating that the battery charger is carrying the load and the battery is in standby. The amp meter is indicative of the battery automatically picking up the load due to failure of the battery charger which would be indicated by the local alarm "DC Output Breaker Open".
"A" is incorrect but plausible if the candidate does not know what this alarm means and believes the photo is indicative of the battery charger carrying the load vs. the battery. The blown fuse this alarm refers to is the one between the D07 battery and bus D01.
"C" is incorrect but plausible if the candidate believes the photo is indicative of the battery charger carrying the load vs. the battery which would be the case if the battery disconnect were open.
"D" is incorrect but plausible if the candidate believes that the voltage indicated is abnormal. The system is called 125 volt but with the battery charger connected it is normally 130 volts.

References:

E-17, Red Train Vital AC and 125V DC Single Line and Distribution
1203.012, Annunciator K01 Corrective Action

PHOTO (separate file) MUST FOLLOW THIS QUESTION ON THE EXAM!!!!!!

History:

New question for 2016 exam

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

QID: 0792 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** Bank **Originator:** S. Pullin

TUOI: A1LP-RO-EDG **Objective:** 19 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators (ED/G)

Description: Knowledge of the physical connections and / or cause-effect relationships between the ED/G system and the following systems: Starting air system.

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:**	3
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 51 **SRO:**

Given:

- Plant at 100%
- CBOT is performing #1 EDG monthly surveillance per 1104.036 Supplement 1.

The CBOT presses the start pushbutton on C10.
A short time later annunciator K01-B2 "EDG 1 OVERCRANK" alarms

What is the cause of the alarm and how long did the starting air system attempt to start the engine?

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

B. #1 EDG did not exceed 300 rpm in 8 seconds and air start motors engaged for 2.5 seconds.

C. #1 EDG did not exceed 30 rpm in 45 seconds and air start motors engaged for 8 seconds.

D. #1 EDG did not exceed 30 rpm in 8 seconds and air start motors engaged for 2.5 seconds.

Answer:

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

Notes:

A is correct, following a start signal one bank of the air start system will engage and crank the engine. If the EDG does not reach 30 rpm in 2.5 seconds, then that air start system is disengaged and the other engaged (it will crank for 2.5 seconds). If the EDG does not achieve 30 rpm after 8 seconds, then all cranking is stopped. A timer will cause the Overcrank alarm if the EDG does not achieve 300 rpm in 45 seconds. This 45 seconds allows time for the EDG to continue attempting to start in case it was sputtering and trying to start.
"B" is incorrect but plausible since this is a combination of the correct RPM for the overcrank, the 8 second time limit for cranking, and the time limit for one bank.
"C" is incorrect but plausible since this is a combination of the RPM for the air start timer, the correct time for the overcrank alarm, and the time limit for cranking.
"D" is incorrect but plausible since this is a combination of the RPM for the air start timer, the 8 second time limi for cranking, and the time limit for one bank.

This question matches the K/A as it requires the candidate to recall the relationship between the EDG and the starting air system (times and sequence of air start motors and banks).

References:

STM 1-31, Emergency Diesel Generators
1203.012A, Annunciator K01 Corrective Action

History:

New 2010 RO/SRO exam
Selected for 2016 exam

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

QID: 1065 **Rev:** 1 **Rev Date:** 7/13/16 **Source:** New **Originator:** Cork

TUOI: A1-LP-RO-AOP **Objective:** 5 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring

Description: Knowledge of the operational implications of the following concepts as they apply to the PRM system: Radiation theory, including sources, types, units, and effects.

K/A Number: K5.01 **CFR Reference:** 41.5 / 45.7

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

Group: 1 **SRO Imp:** 3.0 **SRO Select:** No **Taxonomy:** C

Question: RO: 52 SRO:

Given:
- Plant heatup is in progress per 1102.002, Plant Startup.
- RCS Tcold is ~480°F.
- RCS pressure 2150 psig.
- Fourth RCP was started an hour ago.

The Process Monitor Radiation High annunciator alarms.
The plant computer indicates the failed fuel ratio has dropped from 21.49 to 12.72.

What is the cause of this alarm and what operational implication does this have?

A. Crud burst from starting the fourth RCP is releasing activated iron and nickel isotopes, letdown flow must be raised to increase RCS filtration.

B. Iodine portion of the failed fuel detector is failing low, a mode change is not allowed.

C. In-service letdown demineralizer is exhausted, and must be swapped.

D. RCS activity due to release of fission products is rising, a reactor startup may not commence.

Answer:

D. RCS activity due to release of fission products is rising, a reactor startup may not commence.

Notes:

"D" is correct, a marked drop in gross/iodine ratio (failed fuel ratio) indicates a rise in fission products in the RCS. As the iodine portion of the failed fuel monitor's output rises, this is compared with gross activity (all activity) and the gross to iodine ratio gets smaller (both rise but the Iodine rises by a larger percentage). This indicates the amount of fission fragments such as I-131 is rising in the RCS, an indicator of failed fuel.
"A" is incorrect but plausible in that starting RCPs often produces crud bursts but this will result in the gross activity rising due to activated corrosion products and this will cause the failed fuel ratio to rise, not lower.
"B" is incorrect but plausible as this will result in the failed fuel ratio changing but again this will cause the failed fuel ration to rise, not lower.
"C" is incorrect but plausible as letdown demineralizers are often changed based on activity but this would be ar activity differential across the demineralizer, not the RCS as a whole.

This question matches the K/A since the failed fuel ratio comes from the Failed Fuel Monitor, a process rad monitor on letdown. The question asks for an operational implication and the implication is that failed fuel is present due to the type of activity being seen on the failed fuel monitor.

Revised based upon NRC examiner suggestion.

References:

1203.019, High Activity in Reactor Coolant

History:

New question for 2016 exam.

QID: 1079 **Rev:** 0 **Rev Date:** 4/26/16 **Source:** Bank **Originator:** Cork

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

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

System Number: 076 **System Title:** Service Water System (SWS)

Description: Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Automatic start features associated with SWS pump controls.

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

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 53 **SRO:**

The plant is operating at 100% with the following conditions:
- P-4A and P-4C SW pumps running
- P-4B SW pump is aligned to A3 but is tagged out for bay maintenance.
- SW valve alignment is normal otherwise.
- B55/56 is aligned to B5.

Subsequently, ESAS actuates on low RCS pressure with a concurrent Loss of Offsite Power.

#2 EDG fails to start.

What will the service water pump alignment be?

A. P-4A to P-4B crosstie valves CV-3644 & CV-3646 CLOSED;
P-4C to P-4B crosstie valves CV-3640 & CV-3642 OPEN;
ACW isolation CV-3643 CLOSED.

B. P-4A to P-4B crosstie valves CV-3644 & CV-3646 OPEN;
P-4C to P-4B crosstie valves CV-3640 & CV-3642 CLOSED;
ACW isolation CV-3643 OPEN.

C. P-4A to P-4B crosstie valves CV-3644 & CV-3646 OPEN;
P-4C to P-4B crosstie valves CV-3640 CLOSED & CV-3642 OPEN;
ACW isolation CV-3643 OPEN.

D. P-4A to P-4B crosstie valves CV-3644 OPEN & CV-3646 CLOSED;
P-4C to P-4B crosstie valves CV-3640 CLOSED & CV-3642 OPEN;
ACW isolation CV-3643 CLOSED.

Answer:

D. P-4A to P-4B crosstie valves CV-3644 OPEN & CV-3646 CLOSED;
P-4C to P-4B crosstie valves CV-3640 CLOSED & CV-3642 OPEN;
ACW isolation CV-3643 CLOSED.

Notes:

"D" is correct, with ESAS, no offsite power, and failure of #2 EDG the green train crosstie valves will remain open (all of the valves mentioned are open at the beginning of the event) and the red train crosstie valves will close to ensure train separation (because they have power). All of the crosstie valves re-position (or not) based upon which SW pump is running. B55/56 is aligned to B5 (red train) so the ACW isolation will close on ESAS as designed to ensure SW flow goes to ESF components and the sole SW pump P-4A is not operating in a runout condition.
"A", "B", and "C" are all alternate combinations which are plausible since they would be correct for different combinations of SW pumps running and different alignment of B55/56, but they are incorrect for this set of conditions.

This question matches the K/A due to the components are all SW components and the question asks if the candidate understands how the crosstie valves and ACW isolation will atuomatically align based upon which SW pump auto-starts (auto start features).

References:

1104.029, Service Water & Auxiliary Cooling Water
STM 1-42, Service & Auxiliary Cooling Water

History:

Selected regular exam bank question QID ANO-OPS1-05891a for 2016 exam

QID: 0227 **Rev:** 2 **Rev Date:** 7/13/16 **Source:** Bank **Originator:** Cork

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

Section: 3.8 **Type:** Plant Services Systems

System Number: 078 **System Title:** Instrument Air System (IAS)

Description: Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Cross-tied units.

K/A Number: K3.03 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 54 **SRO:**

Given:
- Both units are at 100% power.
- Unit 2 2C28A Instrument Air Compressor is out of service.
- Instrument Air pressure has dropped to 68 psig.
- Field operators can not find an Inst. Air leak on Unit One.
- Instrument Air pressure is now at 58 psig.

Which of the following is the procedurally required response per 1203.024, Loss of Instrument Air, to restore or conserve Instrument Air pressure?

A. Dispatch operator to take manual control of Pzr level control valve CV-1235.

B. Close Unit 1 to Unit 2 Instrument Air cross-connect.

C. Trip Reactor, actuate EFW and MSLI on both SGs.

D. Isolate Seal Injection by closing CV-1206.

Answer:

B. Close Unit 1 to Unit 2 Instrument Air cross-connect.

Notes:

"B" is correct, per 1203.024, the U1 to U2 cross connect should be closed if instrument air pressure drops below 60 psig.
"A" is incorrect, but plausible as it is an action in 1203.024 however this does not occur until pressure is less than 35 psig.
"C" is incorrect, but plausible as it is an action in 1203.024 however this would not be done unless pressure was less than 35 psig.
"D" is incorrect, but plausible as it is an action in 1203.024 however this would not be done unless necessary to maintain PZR level <290".

Revised "A" distracter as it is no longer in procedure.

This question matches the K/A since it involves the Instrument Air system and it requires the candidate to exhibit knowledge of when the cross-tie between the units is closed.

Revised question per NRC examiner suggestion.

References:

1203.024, Loss of Instrument Air

History:

Developed for 1998 RO/SRO Exam QID 0102.
Modified for 98 RO Re-exam
Modified for 2005 RO exam.
Selected for 2011 RO Exam.
Selected for 2016 exam.

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

QID: 0104 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** GGiles

TUOI: A1LP-RO-EOP10 **Objective:** 15.5 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment System

Description: Ability to monitor automatic operation of the containment system, including: Containment isolation.

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

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

Group: 1 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 55 **SRO:**

Following an ESAS actuation the CBOT is directed to perform RT-10 to verify proper actuation.
The RT instructs you to verify each component properly actuated on C16, C18, and C26.

How is this accomplished for containment isolation valves (assume no components have been overridden)?

A. Verify all containment isolation valve "closed" indication lights are illuminated.

B. Verify containment isolation valve positions are in positions marked with green dots.

C. Verify containment isolation valve positions are in position marked by green or red background.

D. Verify containment isolation valves are in position marked with black tape background.

Answer:

D. Verify containment isolation valves are in position marked with black tape background.

Notes:

"D" is the correct response. A black tape background identifies the proper actuation position of ES components.
"A" is incorrect but plausible since the verification is for containment isolation valves, however not all containment penetration valves will be closed.
"B" is incorrect but plausible since Reg Guide 1.97 instrumentation is identified in this manner.
"C" is incorrect but plausible since this is the method used by Unit 2 for ESFAS actuated components.

This question is a direct match for the K/A as it requires the candidate to know how to properly monitor for automatic containment isolation.

Revised due tor NRC examiner comments.

References:

1015.018 , Plant Labeling
1202.012, Repetitive Tasks, RT-10 "Verify Proper ES Actuation"

History:

Developed for 1998 RO Exam.
Selected for use in 2002 RO/SRO exam.
Used on 2004 RO/SRO Exam.
Selected for 2016 exam

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

QID: 0674 **Rev:** 0 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** Possage

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

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive System

Description: Ability to (a) predict the impacts of the following malfunction or operations on the CRDS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rod-misalignment alarm.

K/A Number: A2.17 **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:**	3.8	**SRO Select:**	No	**Taxonomy:**	Ap

Question: **RO:** 56 **SRO:**

Given:
- Approach to criticality is in progress.
- Groups 1 - 5 are fully withdrawn.
- Group 6 is at 30% withdrawn.
- Group 6 Rod 4 drops into the core.
- K08-C2, "CONTROL ROD ASYMMETRIC" in alarm.

Which of the following procedural actions is required for the given conditions?

A. Relatch Group 6 Rod 4 and withdraw to 30% in increments of <25%.

B. Insert Group 6 rods and verify reactor remains subcritical.

C. Insert Group 5 and Group 6 rods in sequence.

D. Trip the reactor and perform 1202.001.

Answer:

C. Insert Group 5 and Group 6 rods in sequence.

Notes:

"C" is correct. If a dropped rod exists and NI power is <2%, then per Section 5 of 1203.003 the starup is terminated by inserting regulating rods in sequence.
"A" is incorrect but plausible since recovery of a single dropped rod is allowed in Mode 1, however recovery of a dropped rod from a subcritical condition can result in uncontrolled criticality and unanalyzed control rod configurations.
"B" is incorrect, but plausible since the rod is in Group 6 and the reactor is not yet critical, however procedure direction is to insert ALL regulating control rods.
"D" is incorrect, but plausible since a Reactor trip is required for two dropped rods if NI power is greater than or equal to 2%.

This question matches the K/A since it involves the Control Rod Drive System and requires candidate to predict the impacts of a malfunction of the CRD (dropped rod generating the rod misalignment alarm) by analyzing the given conditions and to recall procedure requirements for these conditions. This is acceptable per ES-401, D.2.a, 2nd paragraph.

References:

1203.003, Control Rod Drive Malfunction Action

History:

New for 2007 RO Exam. KA 2.4.50
Selected for 2016 exam

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

QID: 0193 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Modified **Originator:** Cork

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

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

System Number: 002 **System Title:** Reactor Coolant

Description: Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship between reactor power and RCS differential temperature.

K/A Number: K5.10 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.1 **SRO Select:** No **Taxonomy:** Ap

Question: **RO:** 57 **SRO:**

As a reactor operator it is important to ensure indicated reactor power is accurate.

Which of the following sets of parameters would correspond to 85% power range NI power?

A. Thot 592 degrees, Tcold 565 degrees

B. Thot 594 degrees, Tcold 559 degrees

C. Thot 596 degrees, Tcold 557 degrees

D. Thot 598 degrees, Tcold 556 degrees

Answer:

C. Thot 596 degrees, Tcold 557 degrees

Notes:

Normal Thot is 602 and normal Tcold is 556 at 100%. This equates to a delta T of 46 degrees. 46 x 0.85 = 39
"C" is correct. Delta T is 39 degrees.
"A" is incorrect, this is a delta T of 27 degrees. This is the previous correct answer and is thus plausible.
"B" is incorrect. Delta T is 35 degrees, indicative of ~76% power.
"D" is incorrect. Delta T is 42 degrees and indicates ~91% power.

Revised question due to implausible distracter and to make correct answer "more" correct. Eliminated Tave in answers as it did not add anything.
Revised question at suggestion of NRC examiner to make power in stem 85% and changed "C" parameters to make it correct and changed "B" and "D" distracters to bound correct answer. Question thus meets definition of modified. JWC 7/14/16

This question matches the K/A since it involves RCS parameters (Thot and Tcold) and requires candidate to exhibit knowledge of the relationship between these parameters and reactor power.

References:

1102.004, Power Operation

History:

Developed for use in 98 RO Re-exam
Used in 2001 RO/SRO Exam.
Selected for 2005 RO re-exam.
Modified for 2016 exam.

QID: 1066 **Rev:** 0 **Rev Date:** 4/15/16 **Source:** New **Originator:** Cork

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

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation

Description: Knowledge of bus power supplies to the following: NIS channels, components, and interconnections.

K/A Number: K2.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 58 **SRO:**

Which of the following supplies power to Power Range channel NI-7?

A. RS-1

B. RS-2

C. RS-3

D. RS-4

Answer:

C. RS-3

Notes:

"C" is the correct power supply for Power Range NI channel 7.
"A", "B", and "D" are the other possible choices and are thus plausible (but incorrect) if one does not know the Power Range NI channel arrangement.

This question matches the K/A since it requires candidate to exhibit knowledge of NIS channels, specifically NI-7.

References:

1105.001, NI & RPS Operating Procedure
1107.003, Inverter and 120V Vital AC Distribution

History:

New question for 2016 exam.

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

QID: 1067 **Rev:** 1 **Rev Date:** 7/20/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP05 **Objective:** 11 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-Core Temperature Monitor

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations:
Core damage.

K/A Number: A2.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 59 **SRO:**

Given:
- An overheating event has been in progress.
- 1202.005, Inadequate Core Cooling, is in use.
- Core Exit Thermocouples = 1460 degrees F (average) and rising.
- RCS pressure = 2350 psig
- All actions have been performed for the current Region.

Critical parameters have been updated by the ATC:
- Core Exit Thermocouples = 1520 degrees F (average) and rising.
- RCS pressure = 2400 psig

CBOT reports multiple CETs are alarming on the plant computer with the status "INVALID" or "FAIL_LO".

What is occurring and what procedural action is required for the above conditions?

A. CETs are experiencing thermionic emission, trip all running RCPs.

B. CETS are failing due to short circuits, trip all running RCPs.

C. CETs are experiencing thermionic emission, use ADVs to reduce SG T-sat to ~100°F below current value.

D. CETS are failing due to short circuits, use ADVs to reduce SG T-sat to ~100°F below current value.

Answer:

B. CETS are failing due to short circuits, trip all running RCPs.

Notes:

Answer "B" is correct, Region 4 of Inadequate Core Cooling has been entered, all running RCPs should be tripped per step 12. CETs fail low when they have open circuits which could occur due to the extremely high temps experienced in Region 4.
"A" is incorrect, although plausible since thermionic emission affects the other type of Incore instrument - Incore Self Powered Neutron Detectors (SPNDs), this phenomenon does not affect CETs. The action given is correct.
"C" is incorrect, although plausible since thermionic emission affects the other type of Incore instrument - Incore Self Powered Neutron Detectors (SPNDs), this phenomenon does not affect CETs. The action is incorrect, this action is performed after entering Region 3.
"D" is incorrect, but plausible since this is the failure mechanism for the CETs. The action is incorrect however, this action is performed after entering Region 3.

This question matches the K/A since it involves the In-Core Temperature Monitor (CETs in the ICCMDS at ANO 1), requires the candidate to predict the impact of core damage on the CETs, and to recall the specific procedural direction required for this region in the ICC procedure.

Per NRC examiner suggestion added Region 3 conditions to question. JWC 7/20/16

References:

1202.005, Inadequate Core Cooling
1202.013, EOP Figures, Figure 4
1105.008, Inadequate Core Cooling Monitor and Display

History:

New question for 2016 exam.

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

QID: 0200 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** B. Short

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

Section: 3.8 **Type:** Plant Services Systems

System Number: 033 **System Title:** Spent Fuel Pool Cooling System

Description: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Maintenance of spent fuel level.

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:**	K

Question: **RO:** 60 **SRO:**

A break has occurred on the discharge line downstream of the discharge valve of the in service Spent Fuel Cooling Pump (P-40A).
The pump is stopped and the discharge valve (SF-5A) is closed.

What should happen with Spent Fuel Pool level?

A. The SFP will drain to 2' above fuel assemblies due to elevation of bottom of tilt pit gate.

B. The SFP will drain to point of uncovering the spent fuel assemblies.

C. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

D. The SFP level will drop ~3 feet to the bottom of the suction pipe.

Answer:

C. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

Notes:

"C" is correct. The discharge pipe has the siphon break holes located at the normal pool level.
"A" is incorrect but plausible if the candidate doesn't know the answer and believes the design will drain quite a bit of the pool but not uncover the assemblies due to bottom of tilt pit gate elevation, which is ~2' above top of fuel racks.
"B" is incorrect, but plausible since one discharge line goes all the way to the bottom of the pool. However with no operator action at all, the lowest the level would go is ~3 feet to the bottom of the suction pipe. This is still ~ 20 feet above the fuel.
"D" is incorrect but plausible, the suction pipe bottom is at ~3 feet, however, with the discharge valve closed the pool will stop draining out the break at the normal pool level due to the siphon holes on the discharge pipe.

This question matches the K/A since it is about knowledg of the spent fuel cooling system and requires the candidate to recall the design feature which maintains spent fuel level despite a break in the return line.

Revised question by re-wording stem, distractor B, and adding discharge valve designator per request of NRC examiner.

References:

STM 1-07, Spent Fuel Cooling System

History:

Selected for 2011 RO Exam.
Selected for 2016 exam.

QID: 1094 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP03 **Objective:** 15 **Point Value:** 1

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

System Number: 035 **System Title:** Steam Generator

Description: Ability to manually operate and/or monitor in the control room: Fill of dry S/G.

K/A Number: A4.02 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 2.8 **SRO Select:** No **Taxonomy:** An

Question: **RO:** 61 **SRO:**

Given:
- Recovery from an Overcooling condition is in progress.
- Reactor Coolant Pumps are not runnning.
- Auxiliary Feedwater Pump, P-75 is the only available source of water.

- "A" SG level is 10 inches and stable
- "B" SG level is 21 inches and lowering
- "A" SG shell temperature is 485 °F
- "B" SG shell temperature is 445 °F
- CET Temperatures are ~425 °F
- RCS Pressure is 1950 psig

Steam leak has been isolated locally.
RT-16, Feeding Intact SG, is in use.

Which of the following is the procedure action required by RT-16 for the above conditions?

A. Feed only "B" SG, reduce flow to ≤ 450 gpm due to Tube-to-Shell DT limit has been exceeded.

B. Feed both SGs, reduce flow to ≤ 200 gpm due to Tube-to-Shell DT limit has been exceeded.

C. Feed both SGs as necessary to establish 300 to 340" level while maintaining Tube-to-Shell DT limits.

D. Feed only "B" SG as necessary to establish 300 to 340" level while maintaining Tube-to-Shell DT limits.

Answer:

C. Feed both SGs as necessary to establish 300 to 340" level while maintaining Tube-to-Shell DT limits.

Notes:

"C" is correct, subcooling margin is adequate, no tube-to-shell DT limits have been exceeded, thus AFW should be fed until 300 to 340" is established in each SG while maintaining tube-to-shell DT limits. ANO-1 has been analyzed for AFW flow to a dry S/G so no specific requirements (other than the EOP limits) are in effect.
"A" is incorrect but plausible as this action would be correct if tube-to-shell DT had been exceeded. Candidate has to recall tube-to-shell DT limits of 100°F tubes colder and 60°F tubes hotter. Indications given show that tube-to-shell DT is 60°F tubes colder. Also both SGs can be fed, not just "B". Flow limit of ≤ 450 gpm is for feeding with EFW (not AFW) with any RCP running and primary to secondary heat transfer not yet established.
"B" is incorrect but plausible as this action would be correct if tube-to-shell DT had been exceeded. Candidate has to recall tube-to-shell DT limits of 100°F tubes colder and 60°F tubes hotter. Indications given show that tube-to-shell DT is 60°F tubes colder. Flow limit of ≤ 200 gpm is for feeding with EFW (not AFW) with all RCP's off and primary to secondary heat transfer not yet established.
"D" is incorrect but plausible as this action would be correct if this answer stated to feed both SGs.

This question is based upon QID 662 which assumes an Overheating condition and RT-16 in use with AFW, this new question assumes an Overcooling condition with RT-16 in use with AFW.

This question matches the K/A as it involves operating SG feedwater controls within procedural limits with indications of a dry S/G (A).

Revised question based upon NRC examiner comments. JWC 7/14/16

References:

1202.003, Overcooling
1202.012, Repetitive Tasks, RT-16 Feeding Intact SG

History:

New question for 2016 exam

QID: 1080 **Rev:** 0 **Rev Date:** 4/27/16 **Source:** Modified **Originator:** Cork

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 predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure.

K/A Number: A1.02 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 62 **SRO:**

Given:
- Plant startup in progress
- Turbine control is in ICS Auto
- Generated MW was 140 but has lowered to 130 Mwe
- Turbine Bypass Valves (TBVs) are closed

If turbine header pressure setpoint is 895 psig, then the TBVs would be expected to begin opening at _________ psig.

A. 895

B. 905

C. 945

D. 995

Answer:

B. 905

Notes:

"B" is correct, once 15% (135 MW) power is reached, a 50 psig bias is applied to the TBVs. However, if power lowers to less than 15%, then the bias is removed and the TBVs open when the setpoint is exceeded by 10 psig
"A" is incorrect but plausible since the TBVs will control at setpoint prior to initially reaching 15% power.
"C" is incorrect but plausible since this is the header pressure setpoint plus the 50 psig bias.
"D" is incorrect but plausible as this is the header pressure setpoint plus the 100 psig bias applied following a reactor trip (to limit cooldown).

This question was modified from QID ANO-OPS1-345 by adding that generated MW had gone above 15% but had lowered to less than 15%. Changed header pressure setpoint to 895 psig which necessitated modifying ALL of the answer choices.

This question matches the K/A since it requires the candidate to recognize (monitor) when the Turbine Bypass Valves will open (to control steam pressure) if a transient were to occur.

References:

STM 1-64, Integrated Control System

History:

Modified regular exam bank question (ANO-OPS1-345) for 2016 exam

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

QID: 0470 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Modified **Originator:** J.Cork

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

Section: 3.9 **Type:** Radioactivity Release

System Number: 071 **System Title:** Waste Gas Disposal System (WGDS)

Description: K4.04 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Isolatior of waste gas release tanks.

K/A Number: K4.04 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 63 **SRO:**

Given:
- Waste Gas Decay Tank T-18A is in progress.
- Shortly afterwards RE-4830 Gaseous Waste Discharge Process Monitor goes into high alarm.

Which of the following will occur as a result of this actuation?

1. C-9A and C-9B Waste Gas Compressors stop.

2. Gaseous Waste Discharge Isolation valve (CV-4830) closes.

3. Aux. Building Vent Header (CV-4806) diverts to the in-service Waste Gas Decay Tank.

4. Gaseous Radwaste Discharge Header flow control valve (CV-4820) closes.

A. 1 and 4

B. 2 and 3

C. 3 and 4

D. 2 and 4

Answer:

D. 2 and 4

Notes:

Answer "D" is correct, on high radiation signal from RI-4830 the discharge isolation CV-4830 closes, flow contro CV-4820 closes, and CV-4806 opens to allow flow to divert to the Waste Gas Surge Tank.
"A" is incorrect but plausible since CV-4820 does close and the compressors do have auto stop functions but not on high radiation.
"B" is incorrect but plausible since CV-4830 does close and CV-4806 diverts but a check valve arrangement prevents it from diverting to the Waste Gas Decay Tanks.
"C" is incorrect but plausible since CV-4820 does close and CV-4806 diverts but a check valve arrangement prevents it from diverting to the Waste Gas Decay Tanks.

Modified question by revising the stem into an operational context. Revised choices 1-4 so that they all had component IDs. Added choice 2 and removed "old" 2 so that "D" is now the correct answer. Changed A and D answers.

This question matches the K/A since it reqiures knowledge that the gaseous waste rad monitor has a design feature which will isolated the waste gas decay tanks on a high alarm.

References:

1104.002, Gaseous Radwaste System

History:

Direct from regular ExamBank QID ANO-OPS1-1399.
Selected for use on 2007 RO Exam.
Modified for use in 2016 exam.

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

QID: 0379 **Rev:** 2 **Rev Date:** 4/27/16 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-WCO-ARMS **Objective:** 7 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 072 **System Title:** Area Radiation Monitoring (ARM) System

Description: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: radiation levels.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 64 **SRO:**

During performance of 1305.001, Supplement 6, Area Radiation Monitor Monthly Alarm Check, you discover Relay Room Area Monitor, RI-8002, high alarm setpoint is greater than the maximum allowable value.

What are the required actions?

A. Record the value found, and document set-point drift in Section 3.0 of the surveillence test.

B. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

C. Record the value found, then have I&C make the required adjustment under a "blanket" Work Order.

D. Adjust the setpoint to twice the background reading, then record the As-Left Setpoint.

Answer:

B. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

Notes:

Answer [B] is correct per the procedure supplement as it maintains the system alarms within the design criteria of the system.
Answer [A] is incorrect but would be correct for discrepancies not governed by a procedural response.
Answer [C] is incorrect but this is how it was handled in the past.
Answer [D] is incorrect but this is how adjustments are made for process rad monitors in Supplement 5.

Revised choice D to make it plausible.

This question matches the K/A since it requires the candidate to know how to adjust alarm setpoints on area radiation monitors, setting the high alarm setpoint to less than or equal to the procedural max high alarm setpoint ensures the area monitor will alarm when radiation levels change slightly.

References:

1305.001, Radiation Monitoring System Check and Test

History:

Modified regular exambank QID #2645
Selected for the 2008 RO Exam.
Revised for the 2016 exam

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

QID: 0309 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** Possage

TUOI: A1LP-RO-ICS **Objective:** 40 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation System

Description: Knowledge of the effect that a loss or malfunctions of the NNIS will have on the following: MFW system.

K/A Number: K3.04 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.7	**SRO Select:**	No	**Taxonomy:**	An

Question: **RO:** 65 **SRO:**

Given:

- The plant is operating at 100% power.
- Loop "A" T-cold Narrow Range Temperature instrument fails HIGH.

If this instrument was hard selected by the SASS selector switch, what ICS HAND/AUTO stations should be placed in HAND per 1203.001, ICS Abnormal Operation?

A. Both Feedwater Loop Demands, Reactor Demand and Diamond Panel.

B. SG/Rx Master, Loop Delta Tc and Reactor Demand

C. Both Feedwater Loop Demands, SG/Rx Master and Loop Delta Tc.

D. Both MFW Pumps, Loop Delta Tc and Turbine (EHC).

Answer:

A. Both Feedwater Loop Demands, Reactor Demand and Diamond Panel.

Notes:

A cold leg temperature instrument failure causes the reactor demand signal to drive rods inward due to a high indicated Tave. Feedwater flows are changed to balance loop cold leg temperatures.
"A" is correct. Both feedwater loop demand stations reactor demand and diamond panel must be taken to manual.
"B" is incorrect because feedwater is affected downstream of the SG/Rx Master, but plausible because it includes reactor demand.
"C" is incorrect because reactor demand is affected downstream of the SG/Rx Master, but plausible because it includes both FW loop demands
"D" is incorrect because the turbine is not affected but plausible since it includes loop delta Tc and a failure of a Tc instrument is given.

This matches the K/A since it involves a failure of an NNI instrument (A Tc) and evaluates knowledge of ICS handstations to take to "hand" to mitigate this failure on feedwater and the reactor.

Added AOP to stem per suggestion from NRC examiner. JWC 7/14/16

References:

1203.001, ICS Abnormal Operation

History:

2007 RO Exam.
Selected for 2016 exam

QID: 1083 **Rev:** 0 **Rev Date:** 5/2/16 **Source:** New **Originator:** Cork

TUOI: ASLP-RO-OPSPR **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of operator responsibilities during all modes of plant operation.

K/A Number: 2.1.2 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.4 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 66 **SRO:**

The Annunciator Control Periodic Review is performed to verify the continued need for each annunciator that has been removed from service or modified.

What two positions are responsible for completing this review in accordance with 1015.001, Conduct of Operations?

A. Control Board Operator or I&C Superintendent

B. System Engineer or I&C Superintendent

C. Control Board Operator or STA

D. STA or System Engineer

Answer:

C. Control Board Operator or STA

Notes:

"C" is correct per 1015.001, Conduct of Operations, form 1015.001C. The form states this review shall be performed by a licensed operator or STA.
"A" is incorrect since the I&C Superintendent does not have this responsibiilty per 1015.001C but this is plausible due to the position and department, the System Engineer is notified whenever an annunciator is removed from service.
"B" is incorrect since neither of these positions are responsible for the review but plausible in that a System Engineer is notified whenever an annunciator is removed from service and I&C Supt. is plausible due to the position and department.
"D" is incorrect since the System Engineer (SE) does not have this responsibiilty per 1015.001C but plausible since the SE is notified whenever an annunciator is removed from service and the STA is one of the responsible positions.

This question matches the K/A since it involves a licensed operator responsibility which may be performed during any mode of plant operation.

References:

COPD001, Operations Expectations and Standards

History:

New question for 2016 exam

QID: 0838 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** Bank **Originator:** J. Cork

TUOI: ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledge and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

K/A Number: 2.1.4 **CFR Reference:** 41.10 / 43.2

Tier: 3 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 67 **SRO:**

For the purpose of maintaining an NRC operator's license, which of the following should be reported to the NRC?

A. A change in marital status.

B. A traffic citation for speeding.

C. A new diagnosis for high blood pressure.

D. An audit by the IRS of previous year's tax return.

Answer:

C. A new diagnosis for high blood pressure.

Notes:

Only "C" is requred to be reported per EN-NS-112 and 1063.008.
The others are plausible situations which can occur in life that are not required to be reported as part of an operator's license.

This question matches the K/A since it relates to an individual licensed operator responsibility to maintain an active license.

References:

1063.008, Operations Training Sequence

History:

New for 2011 RO Exam.
Selected for 2016 exam.

QID: 1084 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** New **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:** K

Question: **RO:** 68 **SRO:**

As a licensed operator you are responsible for compliance with COPD-030, ANO Reactivity Management Program.

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.

Replaced A distractor at request of NRC examiner. JWC 7/14/16

References:

COPD-030, ANO Reactivity Management Program

History:

New question for 2016 exam

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

QID: 0231 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** J.Cork

TUOI: FLP-OPS-ESOMS **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of tagging and clearance procedures.

K/A Number: 2.2.13 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

Group: G **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 69 **SRO:**

Which of the following conditions is correct with regard to preparation and installation authorization of a common unit tagout?

A. Installation may be authorized by either the Unit 1 or the Unit 2 Operations Supervisor.

B. Preparer and reviewer must include a licensed operator from each unit.

C. Preparer and reviewer may be non-licensed if installation is authorized by a Unit Operations Supervisor from each unit.

D. Preparer may be non-licensed as long as the opposite unit reviewer is licensed.

Answer:

B. Preparer and reviewer must include a licensed operator from each unit.

Notes:

"B" is correct because the procedure requires both the preparer and the reviewer preparing the tagout to be licensed on their respective unit one on Unit 1 and one on Unit 2.
"A" is incorrect but plausible because a common unit tagout requires both Unit's Operations Supervisors to approve it not just one.
"C" is incorrect, but plausible because both Unit Ops Supervisors must approve but the preparation & review must be done by licensed operators even though the non-licensed operator is qualified to do so.
"D" is incorrect but plausuble because the preparation & review must be done by licensed operators on their respective units even though the non-licensed operator is qualified to do so.

This question matches the K&A because to answer this question correctly requires knowledge of the tagging and clearance procedure.

Revised B and C choices at request of NRC examiner. JWC 7/14/16

References:

EN-OP-102, Attachement 9.5 Sheet 1.

History:

Developed for use in 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.
Selected for use on 2007 RO Exam.
Selected for use on 2016 RO Exam.

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

QID: 1082 **Rev:** 0 **Rev Date:** 4/29/16 **Source:** New **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:** K

Question: **RO:** 70 **SRO:**

Being a part of Operations might require you to make 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. Changing the title of a position to correspond to corporate heirarchy.

C. Deleting a QC hold point in a procedure section for a filter change.

D. Adding a step to close an open configuration control loop.

Answer:

C. Deleting a QC hold point in a procedure section for a filter change.

Notes:

"C" is the correct answer per 1000.006, definition 4.9.2.
"A", "B", and "C" are common procedure changes and thus plausible, but do not constitute intent changes per 1000.006.

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

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

QID: 1081 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** New **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:** K

Question: **RO:** 71 **SRO:**

You have been directed to perform a task in the Makeup Tank Room, a Locked High Radiation Area (LHRA).

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.
"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

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

QID: 0751 **Rev:** 1 **Rev Date:** 7/11/2008 **Source:** Bank **Originator:** Spullin

TUOI: ASLP-RO-RADPRO **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledge's and Abilities

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier: 3 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3

Group: **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 72 **SRO:**

Which of the following exposure limits are
Entergy's Routine Annual Administrative Guidelines?

A. TEDE 2000 mrem per year; SDE, WB= 40 rem; and LDE= 12 rem

B. TEDE 5000 mrem per year; SDE, WB= 40 rem; LDE= 12 rem

C. TEDE 5000 mrem per year; SDE, WB= 50 rem; LDE= 15 rem

D. TEDE 2000 mrem per year; SDE, WB= 50 rem; LDE= 15 rem

Answer:

A. TEDE 2000 mrem per year; SDE, WB= 40 rem; and LDE= 12 rem

Notes:

A. is correct, the limits are Entergy Routine Annual Administrative Guidelines
B. is incorrect but plausible as these are the Maximum Annual Administrative guidelines for Entergy
C. is incorrect, but plausible as these are the Annual Regulatory limits
D. is incorrect, but plausible as they are a mix of different limits

This question matches the K/A since it requires the candidate to recall one of the normal radiation exposure limits.

References:

EN-RP-201, Dosimetry Administration

History:

New for the 2008 RO exam
Selected for 2016 exam

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

QID: 0242 **Rev:** 0 **Rev Date:** 9-1-99 **Source:** Bank **Originator:** D. Slusher

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

Section: 2 **Type:** Generic

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Ability to identify post-accident instrumentation.

K/A Number: 2.4.3 **CFR Reference:** CFR: 41.6/45.4

Tier:	3	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 73 **SRO:**

What instruments are marked with a green dot?

A. Instruments designated for use during an alternate shutdown.

B. Instruments that should be reliable during accident conditions.

C. Instruments the Shift Engineer uses after a reactor trip.

D. Instruments designated as important to the Emergency Plan.

Answer:

B. Instruments that should be reliable during accident conditions.

Notes:

"B" is correct because instruments which are reliable and to be used for accident conditions are marked by a green dot as required by Reg Guide 1.97 and IAW OP 1305.028 Section 3.0.
"A" is incorrect but plausible because SPDS instrumentation is designated for the alternate shutdown AOP OP-1203.002 Attachment 10.
"C" is incorrect but plausible because System Engineering instruments used after a reactor trip are designated by the Post Transient Procedure Admin Procedure OP-1015.037 Attachment I.
"D" is incorrect but plalusible since equipment important to the Emergency Plan is identified in 1903.069 but there are no specific instrument markings for control room instrumentation.

This Question matches the K&A because the candidate must have the ability to identify which instruments he can use post accident and the ones he will use have the green dots on the indicator.

References:

1305.028, Reg Guide 1.97 Instrumentation Verification

History:

Developed for 1999 exam.
Selected for the 2010 RO/SRO Exam
Selected for the 2016 RO/SRO Exam

QID: 0051 **Rev:** 2 **Rev Date:** 7/14/2016 **Source:** Bank **Originator:** GGiles

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 general operating crew responsibilities during emergency operations:

K/A Number: 2.4.12 **CFR Reference:** 41.10 / 45.12

Tier: 3 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 74 **SRO:**

A General Emergency has been declared on Unit 1.

The Shift Manager has announced a plant evacuation.

Which of the following actions should be performed?

A. All Operations personnel on watch should report to the Control Room.

B. All on watch Operations personnel, with the exception of the control room staff, should report to the Operations Support Center (OSC).

C. All non-watchstanding Operations personnel (training/support) should report to the Technical Support Center (TSC).

D. All Operations personnel, with the exception of the on watch Operations personnel, should evacuate the plant.

Answer:

A. All Operations personnel on watch should report to the Control Room.

Notes:

"A" is correct IAW OP-1903.030 Step 6.2.2.B.
"B" is incorrect but plausible because on shift crew members should report to the control room and all other operation personnel should report to the OSC IAW OP-1903.030 Step 6.2.2.D.
"C" is incorrect but plausible because all other shift operations personnel should report to the OSC not the TSC.
"D" is incorrect but plausible because Non-essential personnel will be evacuating the plant.

This question matches the K&A because it describes the general responsibilities of an operating crew during a plant emergency evacuation.

Changed "on duty" to "on watch" or to "non-watchstanding" where applicable, per NRC examiner suggestion. JWC 7/14/16

References:

1903.030, Evacuation, Sections 6.2.2.B, & C

History:

Developed for 1998 SRO Exam.
Used on the 2016 NRC Exam.

QID: 0848 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Modified **Originator:** J. Cork

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

Section: 2 **Type:** Generic K/A's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of fire protection procedures.

K/A Number: 2.4.25 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 3 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

Group: G **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 75 **SRO:**

You are standing watch and it is now 1300 on 08/26/2016.

Which of the following would be considered a fire system impairment in accordance with 1000.120, ANO Fire Impairment Program?

A. P-6A Electric Fire Pump non-functional due to on-going surveillance since 0400.

B. A smoke detector string in Corridor 98 defeated for PMs since 0900.

C. Fire hose station in Aux Bldg 335 elevation isolated since 1030 for hose replacement.

D. Control Room Halon System #3 defeated for corrective maintenance since 0830.

Answer:

A. P-6A Electric Fire Pump non-functional due to on-going surveillance at 0600.

Notes:

A is correct since per 1000.120 this non-functionality has not been corrected prior to the end of the night shift and has carried over to day shift.
B, C, and D are incorrect but plausible since they are non-functional but any systems out of service for less than one shift for surveillances, corrective maintenance, or PMs are not considered an impairment per 1003.002.

Revised conditions and times to make "A" correct, this is a modified question.

This question matches the K/A as it applies to a Tier 3 topic: it requires the candidate to recall portions of a fire protection procedure (1000.120) and to ascertain which condition requires the performance of the administrative task of reporting a fire impairment.

Changed A time to 0400 and all "at" before times to "since" per request of NRC examiner. JWC 7/14/16

References:

1000.120, ANO Fire Impairment Program

History:

New question for 2011 exam.
Modified for 2016 exam.

QID: 1100 **Rev:** 0 **Rev Date:** 6/17/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to determine or interpret the following as they apply to a small break LOCA: Adequate core cooling.

K/A Number: EA2.39 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 4.3 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 76

Given:
- A small break LOCA has occurred.
- ESAS actuated on low RCS pressure.
- Subcooling margin (SCM) was lost and all RCP's were tripped.
- 1202.002, Loss of Subcooling Margin is in use.

The break has been isolated.

Current plant conditions are:
- RCS pressure1800 psig and slowly rising
- CET average 552 °F (Thot temps ~ the same) and lowering
- Thot/Tcold delta temperature dropping
- SG pressures are being controlled by ADVs and ATC operator
- Tcold tracking SG Tsat

Which procedure should be transitioned to given the above conditions?

A. Small Break LOCA Cooldown, 1203.041

B. Reactor Trip, 1202.001

C. ESAS, 1202.007

D. Natural Circulation Cooldown, 1203.013

Answer:

B. Reactor Trip, 1202.001

Notes:

"B" is correct. The condtions given show the loss of adequate SCM has been corrected, SCM is adequate based on RCS pressure and CETs, and primary to secondary heat transfer is in progress. In accordance with step 19 of 1202.002, Loss of Subcooling Margin, the CRS will transition to 1202.001, Reactor Trip, to complete analysis of plant status.
"A" is incorrect, but plausible since step 19 contingency action of 1202.002 would require a transition to 1203.041 if an uncontrolled RCS cooldown was occurring due to HPI break flow.
"C" is incorrect but plausible since ESAS had actuated.
"D" is incorrect but plausible since step 19 contingency action of 1202.002 would require a transition to 1203.013 if RCS leak were unisolable, RCS press was > 150 psig, and RCPs were not available.

This question is SRO only since it relates to 10CFR55.43(b)(5), assessment of facility condtions and selection of procedures. The candidate is presented with facility conditions and must be able to select the appropriate procedure to transition to.

This question matches the K/A as it requires the candidate to have the ability to determine if core cooling is adequate following a small break LOCA.

References:

1202.002, Loss of Subcooling Margin

History:

New question for 2016 SRO exam

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

QID: 0639 **Rev:** 1 **Rev Date:** 5/4/16 **Source:** Modified **Originator:** Cork

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

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

System Number: 025 **System Title:** Loss of Residual Heat Removal System (RHRS)

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

K/A Number: AA2.02 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 77

Given:
- Plant is in Mode 5, cooldown in progress for refueling outage
- Both Decay Heat Removal trains are in service
- Both Decay Heat Removal flows are steady at ~1900 gpm.

Then the following occurs:
- K10-B2 "PROCESS MONITOR RADIATION HIGH" alarms
- CBOT reports RI-3809, Loop A DH Process Rad Monitor is in alarm

For the above conditions which of the following actions are required and which section of 1203.028, Loss of Decay Heat Removal, should be in use?

A. Stop P-34A DH pump and close P-34A suction valve from RCS (CV-1434) per Section 3, DH Removal System Leak >20 GPM

B. Stop both P-34A and P-34B DH pumps and close DH suction valve (CV-1050) per Section 9, Loss of Both DH Systems - RCS Pressure Boundary Intact

C. Stop both P-34A and P-34B DH pumps and close DH suction valve (CV-1050) per Section 3, DH Removal System Leak >20 GPM

D. Stop P-34A DH pump and close P-34A suction valve from RCS (CV-1434) per Section 9, Loss of Both DH Systems - RCS Pressure Boundary Intact

Answer:

A. Stop P-34A DH pump and close P-34A suction valve from RCS (CV-1434) per Section 3, DH Removal System Leak >20 GPM

Notes:

Answer "A" is correct, the alarm response procedure directs a transition to 1203.028 and Section 3 of the AOP contains actions for a DH cooler leak into Service Water at step 14 and only the affected pump should be stopped and its associated suction valve closed.
"B" is incorrect, but plausible since stopping both pumps is appropriate IF the operator closes common suction valve CV-1050. This section would be used for a condition affecting both pumps but should not be used for a condition only affecting the A pump.
"C" is incorrect, but plausible since this is the correct action but Section 9 will have the operator close a common suction isolation such as CV-1050.
"D" is incorrect, but plausible since this is the correct action but the wrong procedure section.

This question is SRO only since it meets 10CFR55.43(b)(5): the question requires the candidate to evaluate the conditions given and to select the appropriate procedure and action within that procedure which would assist in mitigating the event. It cannot be answered solely by knowing the major mitigative strategy nor can it be answered solely by knowing entry conditions

This question meets the K/A since it requires the candidate to assess the malfunction of the DH system, determine that it is DH cooler leakage and select the procedure section and appropriate action to mitigate the malfunction.

References:

1203.028, Loss of Decay Heat Removal
1203.012I, Annunciator K10 Corrective Action

History:

New, created for 2007 SRO exam.
Selected for 2011 SRO Exam.
Modified extensively for 2016 SRO exam due to previous version no longer meeting SRO Only question standards.

QID: 1085 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** Cork

TUOI: ASCBT-EP-A0081 **Objective:** 5 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 029 **System Title:** Anticipated Transient Without Scram (ATWS)

Description: Ability to determine or interpret the following as they apply to a ATWS: Reactor nuclear instrumentation

K/A Number: EA2.01 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 4.4 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 78

Given:
- Plant startup is in progress.
- Reactor power is 46%.

While going through the Main FW Block valves a MFW transient occurs:
- Total Main FW flow lowers to 1.4 x e 6 lbm/hr,
- Generated Mwe goes to zero,
- EFW actuated on both trains,
- RCS pressure rising rapidly,
- Reactor power 5% and dropping rapidly,
- All Safety Groups full out,
- All Regulating Groups fully inserted.

Subsequent actions taken by the ATC successfully trip all CRDs, Power Range channels indicate 0%.

Which of the following Emergency Action Level classifications should be declared?

A. General Emergency

B. Site Area Emergency

C. Alert

D. Unusual Event

Answer:

C. Alert

Notes:

"C" is correct per 1903.010, an automatic trip failed to shutdown the reactor and manual actions successfully shutdown the reactor as indicated by reactor power <5%. Indications are that AMSAC actuated on low MFW flow tripping the turbine, then DSS tripped the CRD regulating groups but not the safety groups, indicating a failure of RPS to trip the reactor.
"A" is incorrect but plausible as a failure of an automatic trip could require declaration of a GE but only if manua actions were unsuccessful and FW flow rates were less than 430 gpm or CETs approaching 1200°F.
"B" is incorrect but plausible as a failure of an automatic trip could require declaration of an SAE but only if manual actions taken in the control room at C03 are unsuccessful.
"D" is incorrect but plausible if the candidate cannot deduce was has occurred or cannot recall which EAL is me when RPS fails to trip.

This question is SRO level because it meets 10CFR55.43(b)(5) assessment of facility condtions and selection of appropriate procedures. Classification of emergencies is a specific SRO only responsibility.

This question meets the K/A since the candidate must assess the conditions given, determine that an ATWS has occurred, and then using the nuclear instrumentation parameters given must determine the EAL classification that is appropriate.

Lowered Main FW flow to 1.4 x e 6 at suggestion of NRC examiner. JWC 7/15/16

References:

1903.010, Emergency Action Level Classification
1203.012G, Annunciator K08 Corrective Action
1202.001, Reactor Trip

History:

New question for 2016 SRO exam.

QID: 0584 **Rev:** 1 **Rev Date:** 5/9/16 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-EOP03 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of symptom based EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 43.2

Tier:	1	**RO Imp:**	3.7	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 79

A steam line rupture has occurred in the Reactor Building with the following conditions now present:
- ESAS actuated on channels 1 thru 6.
- All RCPs secured per RT-10.
- RB pressure 3 psig and dropping.
- RCS pressure is 1050 psig.
- T-hot is 390°F.
- HPI has been throttled.
- EOP actions have terminated the overcooling.

The STA recommends to the CRS to restore normal operating pressure per RT-14 in order to reset ESAS and re-start RCPs.

As CRS, does this recommendation follow the EOP mitigation strategies, and why or why not?

A. Yes, the overcooling event has been terminated.

B. No, this action could overstress reactor vessel.

C. Yes, adequate SCM exists so this is allowable.

D. No, RB pressure is not within normal limits.

Answer:

B. No, this could overstress reactor vessel.

Notes:

"B" is correct, trainee must recognize that with RCPs secured and HPI having been initiated that Pressurized Thermal Shock (PTS) limits apply until an evaluation is performed prior to returning to normal pressure. PTS limits prevent overstressing reactor vessel.
"A" is incorrect, yes this is plausible as the overcooling has been terminated but normal operating pressure would violate procedure.
"C" is incorrect, but plausible as adequate subcooling margin is present but normal operating pressure would violate procedure.
"D" is incorrect, but plausible: RB pressure is a concern and is outside normal limits, but the overriding concern is with PTS.

Revised the T-hot value given from 490°F to 390°F so that raising pressure from 1050 to 2155 would definitely violate the NDTT limit. Revised the "C" distractor by simply stating SCM exists since it was implausible that SCM would have been lost. Removed "due to existence of adequate SCM" since the candidate should evaluate RCS pressure-temperature conditions independently.

This question is SRO level, it meets 10CFR55.43(b)(2) since it requires knowledge of the Technical Specification bases.

This question matches the K/A since it has direct ties to the EOP mitigation strategy followin an RCS cooldown caused by a steam line rupture to limit RCS pressure low within limits of Figure 3 if PTS limits apply.

References:

1202.012, Repetitive Tasks, RT-14
Technical Specification Bases, 3.4.3
1202.013, EOP Figures, Figure 3

History:

New for 2005 SRO exam.
Selected for the 2010 SRO exam
Selected for the 2016 SRO exam

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

QID: 0586 **Rev:** 1 **Rev Date:** 5/9/16 **Source:** Modified **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 056 **System Title:** Loss of Offsite Power

Description: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

K/A Number: 2.2.36 **CFR Reference:** 43.2

Tier: 1 **RO Imp:** 3.1 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 80

REFERENCE PROVIDED

Given:
- Plant is at 100% power with no failed equipment.
- A loss of the 161 KV ring bus occurs (de-energized).
- Autotransformer is energized from the 500 KV ring bus.

Entergy Arkansas states that maintenance to repair the 161 KV ring bus will take 1 to 5 days.

Which of the following is the maximum time allowed before the plant is required to be in Mode 3?

A. 30 hours

B. 72 hours

C. 78 hours

D. 84 hours

Answer:

C. 78 hours

Notes:

Answer "C" is correct, knowledge of the switchyard layout is required to know that the auto transformer supplies SU Transformer #1 (for Unit 1) and the 161KV ring bus supplies SU Transformer #2. With the 161KV ring bus de-energized so will SU Transformer #2 be de-energized , thus 1 of the 2 required offsite power sources is inoperable. The time for Required Action A.3 must be added to Required Action F.1 to arrive at the correct time limit.
"A" is incorrect, but plausible since 24 hours is the completion time for Required Action C.2 added to Required Action F.1.
"B" is incorrect but plausible as 72 hours is the completion time for Required Action A.3 alone.
"D" is incorrect but plausible as this is the completion time for Required Action A.3 added to the tjime for Required Action F.2.

This question is considered modified since the Tech Spec it is based on has changed since the last time the question was used. This changed the correct answer and one distracter.

This question is SRO level because it meets 10CFR55.43(b)(2), facility operating limitations in the Technical Specifications, specifically 3.8.1.

The question meets the K/A since it presents the candidate with a degraded offsite power source requiring maintenance, and requires the candidate to apply the Technical Specifications to these conditions and arrive at a correct answer.

References:

Technical Specification 3.8.1

This reference must be included in the student's exam handout!!!

History:

Selected for 2011 SRO Exam.
Modified for 2016 SRO exam

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

QID: 1050 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-RO-EOP04 **Objective:** 10 **Point Value:** 1

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

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

Description: Ability to determine and interpret the following as they apply to the (Inadequate Heat Transfer): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

K/A Number: EA2.2 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 81

Given:
- Reactor tripped due to loss of both MFW Pumps at time 1020.
- P-75 AFW pump is tagged out for maintenance.
- Bus A3 is locked out.
- P-7A EFW pump trips.
- P-32B and P-32D RCPs have been tripped, P-32A and P-32C are running.
- RCS Thot temperatures are 620°F and rising.
- RCS pressure is 2320 psig and rising.
- SG Tube-to-Shell Delta-T is 65°F, tubes hotter.

It is now 1021 and based on the above current conditions, the procedurally required action is to _______.

A. Trip P-32A and P-32C RCPs in accordance with Loss of Subcooling Margin (1202.002).

B. Initiate HPI Cooling per RT-4 in accordance with Overheating (1202.004).

C. Trip P-32A and P-32C RCPs in accordance with Overheating (1202.004).

D. Initiate Full HPI per RT-3 in accordance with Loss of Subcooling Margin (1202.002).

Answer:

C. Trip P-32A and P-32C RCPs in accordance with Overheating (1202.004).

Notes:

C" is correct, in accordance with the Overheating EOP the running RCPs should be tripped if tube-to-shell DT exceeds 60°F tubes hotter.
"A" is incorrect since the Overheating EOP would be entered first and not exited to Loss of SCM, it is plausible since the conditions indicate extremely hot conditions close to but not inadequate SCM.
"B" is incorrect in accordance with step 1 and step 5 of 1202.004, Overheating. This distracter is incorrect since conditions are close to the ERV lifting but it has not reached the auto open setpoint, it pressure had reached 2450 psig, then a transition to RT-4 is made to initiate HPI Cooling.
"D" is incorrect, but plausible since the conditions indicate extremely hot conditions (close to but not inadequate SCM) and the Loss of SCM EOP does initiate full HPI early in the procedure.

Added times to question based on discussion with Chief Examiner. Rev. 1 - moved time to loss of both MFW pumps per NRC examiner request. JWC 7/15/16

This question is SRO only since per 10CFR55.43(b)(5) it requires the candidate to evaluate the conditions giver and to select the appropriate procedure and action within that procedure which would mitigate the conditions with the highest priority.

This question meets the intent of the K/A as the candidate must be able to determine which procedure is applicable and adhere to the guidance within that procedure to mitigate the Inadequate Heat Transfer.

References:

1202.004, Overheating

History:

New SRO question for 2016 exam.

QID: 0347 **Rev:** 1 **Rev Date:** 3/28/16 **Source:** Bank **Originator:** G. Alden

TUOI: A1LP-RO-FH **Objective:** 1.4 **Point Value:** 1

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

System Number: A08 **System Title:** Refueling Canal Level Decrease

Description: Ability to determine and interpret the following as they apply to the (Refueling Canal Level Decrease) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A Number: AA2.2 **CFR Reference:** 43.7

Tier: 1 **RO Imp:** 3.8 **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 82

The main fuel bridge has a grappled spent fuel assembly and is indexed over the core when an NI seal plate cover failure occurs.

Water level in the canal is falling at two inches per minute.

As SRO in Charge of Fuel Handling you should direct the main fuel bridge operator to:

A. Continue to the upender and place the assembly in the upender.

B. Leave the fuel assembly in the mast and evacuate the area.

C. Place the assembly in the fuel rack in the deep end of the canal.

D. Return the assembly to an available location in the reactor vessel.

Answer:

D. Return the assembly to an available location in the reactor vessel.

Notes:

In this scenario the fuel transfer canal level is dropping and the fuel assembly must be placed in an area that wi remain covered with water after the canal is drained.
Therefore, "D" is the correct answer in accordance with 1203.042, Section 2, step 3.A. With Refueling Canal level at minimum (400') there is 23.5 feet to the top of the reactor vessel (bottom of refueling canal) so at 2"/mir there are 282" of water which gives 141 minutes to take action before reaching the top of the vessel. At one ft/min there would only be 23.5 minutes to take action. The fuel mast can move at 20 ft/min in fast speed down to ~12" above the core then shifts to slow speed (5 ft/min) to the bottom of the core, so it would take ~4 minutes to place the assembly in the core. A seal plate failure cannot drain the RCS level below the top of the vessel. There are still ~9.5 ft from the top of the vessel to the top of the fuel assemblies.
"A" is incorrect but plausible if the assembly could not be returned to the reactor vessel.
"B" is incorrect but plausible per the Note before step 3 in Section 2 but conditions are not given to indicate that dose levels in the area are hazardous.
"C" is incorrect but plausible if the SRO in Charge of Fuel Handling determined that level was dropping too fast to transfer the assembly back to the vessel, but level is dropping slowly enough to allow returning the assembly to the vessel.

This question is SRO level since it meets 10CFR55.43(b)(7) and concerns fuel handling procedures.

This question meets the K/A since the conditions given meet the entry condtions for a Refueling Canal Level Decrease section in ANO-1's Refueling Abnormal Operations procedure and requires the candidate to know which is the proper course of action given the conditions.

References:

1203.042, Refueling Abnormal Operations

History:

Last used 2011 SRO exam.
Selected for 2016 SRO Exam.

QID: 1086 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** Cork

TUOI: **Objective:** **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 059 **System Title:** Accidental Liquid Radwaste Release

Description: Ability to control radiation releases.

K/A Number: 2.3.11 **CFR Reference:** 43.4

Tier:	1	**RO Imp:**	3.8	**RO Select:**	No	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.3	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 83

Given:
- Unit 1 is at 100% power.
- Chemistry reports the "A" OTSG secondary leak rate is 25 gpd.
- A report from the Inside AO reveals a Turbine Building Trench continuous release is in progress.
- Discharge Flume (RI-3618) subsequently goes into high alarm.

Which of the following actions is required for the above conditions?

A. Obtain a grab sample and perform gamma and I-131 analysis per ODCM L2.3.1.

B. Suspend the release and initiate a special report per Technical Specifications 3.7.4, Secondary Activity.

C. Obtain a grab sample and perform gamma and I-131 analysis per Technical Specifications 3.7.4, Secondary Activity.

D. Suspend the release and initiate a condition report per ODCM L2.3.1.

Answer:

D. Stop the release and initiate a condition report per ODCM L2.3.1.

Notes:

"D" is correct per ODCM L2.3.1 B.1 and B.2. This action is also consistent with guidance in 1104.044, Turbine Building Draining System, and 1203.014, Control of Secondary System Contamination when a primary to secondary tube leak is in progress (also refer to 1203.023, Small Steam Generator Tube Leaks). During tube leaks only batch releases are allowed. The ODCM spec for continuous releases is not applicable since the release should be terminated immediately. Only batch releases are allowed and the sampling requirements are not met for a batch release so the release should be suspended and a condition report initiated.
"A" is incorrect but plausible as this is the correct action (and correct licensing basis document) for exceeding I-131 limits for continuous releases of secondary coolant but continuous releases should be isolated per 1203.014, Control of Secondary System Contamination when a primary to secondary tube leak >15 gpd is in progress.
"B" is incorrect but plausible as this is the correct action but the incorrect licensing basis document. Technical Specifications adds to the plausibility since during a SG tube leak TS LCO 3.4.13 for primary-to-secondary leakage and 3.7.4 for secondary system activity could apply.
"C" is incorrect but plausible correct action for exceeding I-131 limits for continuous releases of secondary coolant but this distracter gives the wrong licensing basis document.

This question is SRO level because it meets 10CFR55.43(b)(4) as it addresses radiation hazards that may arise during normal and abnormal situations.

This question meets the K/A since it concerns accidental liquid radwaste releases (an alarming radiation monitor and tube leak constitute an accidental liquid release since the sampling and analysis requirements are not met) and the question concerns the ability to control radiation releases.

Revised question per NRC examiner suggestions. JWC 7/15/16

References:

Offsite Dose Calculations Manual, L2.3.1
1203.023, Small Steam Generator Tube Leaks
1203.014, Control of Secondary System Contamination

History:

New question for 2016 SRO exam

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

QID: 1045 **Rev:** 0 **Rev Date:** 3/17/16 **Source:** New **Originator:** J. Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 067 **System Title:** Plant Fire On Site

Description: Knowledge of fire protection procedures.

K/A Number: 2.4.25 **CFR Reference:** 43.2

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

Question: **RO:** **SRO:** 84

REFERENCE PROVIDED

Given:
- Unit 1 is at 100% power.
- Annunciator K12-D1 "FIRE PROT SYSTEM TROUBLE" goes into alarm.
- ATC observes "C463 PANEL TROUBLE" LED illuminated on K125 on C19.
- CBOT investigates and reports yellow trouble LED illuminated for Upper North Electrical Penetration Room (UNEPR) smoke detector and that no other alarms are in.
- Inside AO reports that one smoke detector in UNEPR has red LED lit on base. Smoke detector will not reset.
- STA reports that his review of e-prints show there are six smoke detectors in the UNEPR detector string.

Which of the following actions are required to comply with plant procedures and regulatory requirements?

A. Submit condition report on the smoke detector, the detection string is functional.

B. Establish a one hour roving fire watch within one hour for the UNEPR.

C. Determine any limitations for continued operation of the plant within 24 hours.

D. Establish a continuous fire watch within one hour for the UNEPR.

Answer:

D. Establish a continuous fire watch within one hour for the UNEPR.

Notes:

"D" is the correct answer, the examinee must recognize that a trouble LED on the UNEPR smoke detector string renders the entire detector string inoperable since the trouble alarm will stay in and another will not be received. The 1203.009 fire alarm response procedure directs declaring the detection non-functional. Since the smoke detector string actuates the sprinkler system for the UNEPR, the sprinkler system is thus non-functional and TRM 3.7.9 applies. One of the actions for this specification is to establish a continuous fire watch for the UNEPR within one hour (TRM 3.7.9.A.2).
"A" is incorrect but plausible since the conditions state that there is a problem with only 1 detector and 6 detectors are in the string, thus it might appear that 50% of the detection is available. Per the explanation for "D" above, only one trouble alarm will be received thus the entire string must be declared inoperable.
"B is the other required action for 3.3.6.A (inoperable detection and thus plausible) but a one hour roving fire watch is not an adequate compensatory measure for an inoperable sprinkler system as TRM 3.7.9 requires a continuous fire watch.
"C" is the standard action whenever required actions and completion times are not met (thus plausible), it is present in TRM 3.7.9 but it does not apply since the conditions do not indicate it should be applicable.

This is an SRO level question as it meets 10 CFR 55.43(b)(1), conditions and limitations in the facility license.

The administration of fire protection program requirements is a specific example of an SRO level question in ES 401, Att. 2, page 17. ANO does not expect Ros to be familiar with TRM requirements NOR does it require SROs to commit TRM fire protection specifications to memory despite the one hour time requirements. The TRM fire protection specifications are too numerous and too complex.

This question matches the K/A since it involves fire protection procedures which are used to ensure fire protection systems are functional to detect a plant fire on site. This question requires the candidate to know the alarm response procedure requires declaring the detection string to be non-functional when a trouble alarm occurs.

References:

TRM 3.7.9 - THIS REFERENCE ALONG WITH 3.3.6 MUST BE IN EXAMINEE'S HANDOUT.
1203.009, Fire Protection System Annunciator Corrective Action

History:

New SRO question for 2016 exam.

QID: 0737 **Rev:** 2 **Rev Date:** 6/2/2008 **Source:** Bank **Originator:** Steve Pullin

TUOI: A1LP-RO-EOP **Objective:** 06 **Point Value:** 1

Section: 2 **Type:** Generic K&A

System Number: E09 **System Title:** Natural Circulation Cooldown

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

K/A Number: 2.2.44 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 4.2 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 85

Given:
- Reactor tripped due to Degraded Power condition.
- "B" OTSG has been isolated due to leaking MSSV.

- Plant is cooling down on "A" OTSG.
- Tube to Shell delta T is 110 °F tubes colder.
- Subcooling Margin is adequate.

Which procedural action is correct for this condition?

A. Reduce cooldown rate per 1202.007, Degraded Power.

B. Establish 40-60 °F primary to secondary delta T per 1203.013, Natural Circulation Cooldown.

C. Reduce cooldown rate per 1203.013, Natural Circulation Cooldown.

D. Establish 40-60 °F primary to secondary delta T per 1202.007, Degraded Power.

Answer:

C. Reduce cooldown rate per 1203.013, Natural Circulation Cooldown.

Notes:

"C" is correct, a transition is made to Natural Circulation Cooldown from Degraded Power, and 1203.013 states that when cooling down with one dry SG and tube to shell delta temperature exceeds 100°F, then reduce cooldown rate.
"A" is incorrect but plausible since Degraded Power conditions are stated, however Degraded Power sends one to Natural Circulation Cooldown.
"B" is incorrect but plausible since the delta T reduction is an action for overheating but this action is not in 1203.013, it would be in Degraded Power if one returned to that EOP from Natural Circulation Cooldown due to inadequate SCM.
"D" is incorrect but plausible since Degraded Power conditions are given but the delta T reduction is an action for overheating, not excessive tube to shell delta T.

Revised question by changing order of correct answer, was A, now C. Removed "on Natural Circulation" to given conditions by stating plant is cooling down to remove cueing. Student should recognize plant is in Degraded Power.

This question is SRO level because it meets 10CFR55.43(b)(5) assessment of facility condtions and selection of appropriate procedures. This question cannot be answered solely by knowing a major mitigative strategy nor solely by knowing entry conditions to EOP/AOP. Candidate must know that one transitions to 1203.013 from 1202.007 with the given conditions.

This question meets the K/A since the candidate must interpret the control room conditions given to ascertain the status of the steam generators, and know the correct actions to direct to affect plant conditions positively, al during a natural circulation cooldown.

References:

1203.013, Natural Circulation Cooldown

History:

New for 2008 SRO Exam.
Selected for 2016 SRO exam

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

QID: 0638 **Rev:** 2 **Rev Date:** 6/13/16 **Source:** Modified **Originator:** Possage/Cork

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off.

K/A Number: A2.01 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 86

Given:
- Rx power is 100%
- HPI PUMP TRIP, K10-A6, in alarms
- RCP BLEEDOFF FLOW HI, K08-B7, in alarm

The CBO reports that RCP P-32B Seal Bleedoff Flow is 2.8 gpm.

Based on the above indications an RCP seal _____________ is occurring and as CRS you would direct ____________________ (actions) in response to the above conditions.

A. Degradation;
Trip reactor, trip RCP, and go to 1202.001, Reactor Trip.

B. Failure;
Trip reactor, trip RCP, and go to 1202.001, Reactor Trip.

C. Degradation;
Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP.

D. Failure;
Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP.

Answer:

C. Degradation;
Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP.

Notes:

Answer "C" is correct, a transition is made from the alarm response procedure to the RCP AOP 1203.031. Since seal bleedoff is >2.5 gpm with a loss of seal injection, the action is to reduce power (another transition to a separate procedure) and stop the RCP as required by Section 1 of 1203.031 due to seal degradation vs. seal failure.
Section 2 of 1203.031 defines seal failure as:
>10 gpm rise in RCS leak AND change in seal cavity pressure behavior
RCP seal bleedoff or seal stage temp 200F AND no change in SI or ICW
DP across a single stage = RCS press, with seal BO established.

Answer "A" is incorrect but plausible, since the symptoms are those of seal degradation, however the actions to trip the reactor and then trip the RCP are incorrect for the given conditions.
Answer "B" is incorrect but plausible, since the symptoms for a seal failure are similar to those of seal degradation. The symptoms for a seal failure are not present and the RCP should be stopped, not tripped.
Answer "D" is incorrect but plausible, although the actions to reduce power and stop the RCP is correct, the symptoms are those of seal degradation vs. failure.

Modified original question since it no longer met SRO Only criteria. Revised all stem and all answers.

This question is SRO level since it meets 10CFR55.43(b)(5) due to assessment of condtions in the stem and selection of procedures.
The candidate must know which section of 1203.031 applies and what the subsequent procedural actions to direct are.

This question meets the K/A since it involves predicting the impact of a Reactor Coolant Pump malfunction (RCP seal degradation) and has the candidate select the procedure actions to use to mitigate the high RCP seal leakoff rate.

References:

1203.031, Reactor Coolant Pump and Motor Emergency

History:

New, created for 2007 SRO exam.
Modified for 2016 SRO exam.

QID: 1052 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** Cork

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

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

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

Description: Ability to (a) predict the impacts of the following malfunction or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc control power.

K/A Number: A2.05 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 87

A reactor trip has occurred from 100% power with the following indications:

- Breaker position lights on the RIGHT side of C10 are off.
- Both OTSGs pressures are ~890 psig and slowly trending down.
- PZR level 75" and slowly trending down.
- Attempts to transfer 125V DC panel D21 to its emergency supply are unsuccessful.

Which of the following is the required action and which procedure should be implemented?

A. Manually actuate MSLI and EFW for both OTSGs per 1203.036, Loss of 125V DC.

B. Isolate Letdown by closing Letdown Cooler outlet valves CV-1214 & CV-1216 per 1203.036, Loss of 125V DC.

C. Manually actuate MSLI and EFW for both OTSGs per 1202.003, Overcooling.

D. Isolate Letdown by closing Letdown Cooler outlet valves CV-1214 & CV-1216 per 1202.003, Overcooling.

Answer:

B. Isolate Letdown by closing Letdown Cooler outlet valves CV-1214 & CV-1216 per 1203.036, Loss of 125V DC.

Notes:

"B" is the correct answer per Section 2, Loss of D02, in 1203.036. Loss of DC AOP should be performed in conjunction with entry into the Reactor Trip EOP 1202.001, not exactly a transition but it meets the intent of a transition. Loss of Y02 panel (due to loss of D02) causes closure of Letdown Demineralizer inlet valves, causing Letdown relief valve to lift, therefore letdown must be isolated.
"A" is incorrect, this action is from Section 1, Loss of D01, in 1203.036. It is plausible since OTSG pressures are below normal.
"C" is incorrect, this action is in the Overcooling EOP which should be entered due to low OTSG pressure following a Rx trip but this action is not taken in 1202.003 unless SG pressures are less than 600 psig.
"D" is incorrect, this action is in the Overcooling EOP which should be entered due to low OTSG pressure following a Rx trip but is not taken in 1202.003 unless PZR level is less than 55 inches.

This question is SRO only since it meets 10CFR55.43(b)(5): the question requires the candidate to assess the facility conditions given and to select the appropriate procedure, and action within that procedure, which would assist in mitigating the event. The action is not part of the major mitigating strategy, it is simply an action within 1203.036 for a loss of D02.

This question meets the K/A as it requires the candidate to predict the impact of the malfunction of ESFAS (a loss of DC control power - D02), and it requires the candidate to know which action (in the appropriate procedure) will mitigate the loss of D02.

Revised stem per NRC examiner commennt. JWC 7/15/16

References:

1203.036, Loss of 125V DC

History:

New question for 2016 SRO exam.

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

QID: 1099 **Rev:** 0 **Rev Date:** 6/14/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EFIC **Objective:** 43 **Point Value:** 1

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

System Number: 061 **System Title:** Auxiliary/Emergency Feedwater

Description: Ability to determine operability and/or availability of safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 43.2

Tier: 2 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 88

REFERENCE PROVIDED

Given:
- Unit is at 10% power
- Surveillance has just been performed on P-7A EFW pump.
- CBOT notes that indication is lost on SG A steam admission valve to K3 EFW pump turbine, CV-2613.
- Investigation shows that the breaker for CV-2613 (D-2512) is open and will not reset.

Which of the following actions will comply with all Technical Specifications?

A. Declare P-7A inoperable and restore to operable status within 72 hours.

B. Declare CV-2613 inoperable, de-energize in the closed position, and restore to operable status within 48 hours.

C. Declare CV-2613 inoperable and restore to operable status within 7 days.

D. De-energize CV-2613 in the closed position to maintain operability of P-7A.

Answer:

A. Declare P-7A inoperable and restore to operable status within 72 hours.

Notes:

"A" is correct, P-7A is the "green" train of EFW and the green DC powered steam admission valve CV-2613 is required for operability of P-7A, so TS 3.7.5.B must be entered and P-7A declared inoperable.
"B" is incorrect but plausible if the candidate confuses steam admission valve CV-2613 with steam supply valve CV-2617 and applies LCO 3.6.3 since CV-2617's duty as a reactor building isolation valve.
"C" is incorrect but plausible if the candidate confuses steam admission valve CV-2613 with steam supply valve CV-2617 and applies LCO 3.7.5.A.
"D" is incorrect but plausible sounding as CV-2613 one of two steam admission valves so with CV-2613 closed CV-2663 is still available but CV-2663 is an "enhancement" and will not maintain operability per 1106.006. De-energizing CV-2663 in the closed position will maintain operabiilty of P-7A but the opposite is not true.

This question is SRO only because it requires the candidate to apply LCO action statements to the conditions given and determine the applicable action and completion time. This particular LCO requires knowledge of TS bases to recognize which action is applicable.

This question matches the K/A since it pertains to Emergency Feedwater and requires the candidate to determine operability of P-7A, a safety related EFW pump.

References:

Technical Specifications 3.7.5 and 3.6.3 (both must be in SRO handout)
1106.006, Emergency Feedwater Pump Operation.

History:

New for 2016 SRO exam

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

QID: 1046 **Rev:** 1 **Rev Date:** 7/21/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-RO-EOP07 **Objective:** 12.3 **Point Value:** 1

Section: 3.6 **Type:** Plant Systems: Electrical

System Number: 062 **System Title:** AC Electrical Distribution System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Keeping the safeguards buses electrically separate.

K/A Number: A2.06 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 89

Given:

- Plant has lost all offsite power.
- 1202.007, Degraded Power, EOP is in use.
- #2 EDG is supplying A4.
- #1 EDG has tripped due to low lube oil pressure.
- AAC Generator is OOS for maintenance.

- P-7A EFW pump tripped and could not be reset.
- RCS pressure 2325 psig.
- CET average indicates 612°F.
- A3 and A4 buses were cross-tied and P-7B EFW pump started.
- CBOT reports SU1 voltage is now 22.5 KV.

Which procedure should you transition to and which is the procedurally required action for the above conditions'

A. Go to 1202.004, Overheating, while continuing with bus restoration section of 1202.007.

B. Go to 1202.005, Inadequate Core Cooling, while continuing with bus restoration section of 1202.007.

C. Go to 1202.011, HPI Cooldown, and dispatch an operator to perform Att. 2, "Recovery from Degraded Power Breaker Alignment and UV Relay Defeat".

D. Go to 1107.002, ES Electrical System Operation, and restore buses to normal using "Returning Paralleled Buses A3 and A4 to Normal" section, while continuing with 1202.007.

Answer:

D. Go to 1107.002, ES Electrical System Operation, and restore buses to normal using "Returning Paralleled Buses A3 and A4 to Normal" section, while continuing with 1202.007.

Notes:

"D" is correct, per step 57.D.2 of 1202.007, Degraded Power, once off-site power becomes available, then buses should be restored to normal using 1107.002 (transition). This will maintain separation of ES buses.
"A" is incorrect but plausible. The RCS temperature given indicates entry conditions are met for the Overheating EOP but an SRO candidate should know to stay in the Degraded Power EOP since it has a sectior for mitigating an overheating condition.
"B" is incorrect but plausible. The 1202.007 EOP does direct entry into the Inadequate Core Cooling EOP (step 37) and while the RCS temperature given is quite high, the RCS is not in an ICC condition. If a candidate misreads the EOP figures, then this distracter is quite plausible.
"C" is incorrect but plausible. The 1202.007 EOP does direct entry into the HPI Cooldown EOP (step 47) and Attachment 2 is directed to be performed if A2 is energized from A4 in step 115 but the two are not performed

together.

This is an SRO level question as it meets 10 CFR 55.43(b)(5), assessment of conditions and selection of procedures. It is not RO level since it requires in-depth knowledge of AOPs and EOPs.

This question meets the K/A as it requires the candidate to assess the conditions given and predict the impact, i.e., the ES buses A3 and A4 are cross-tied and now that SU1 is available as indicated by 22.5 KV, then the A3 and A4 buses should be electrically separated to protect them from common faults. The candidate also needs to know the procedure heirarchy, i.e., the EOP user's guide and for ANO that means staying in the Degraded Power EOP despite the indications of heat transfer upsets.

Modified D per NRC examiner suggestion. JWC 7/21/16

References:

1202.007, Degraded Power
1202.013, EOP Figures

History:

New SRO question for 2016 exam.

QID: 1047 **Rev:** 0 **Rev Date:** 3/21/16 **Source:** New **Originator:** J. Cork

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

Section: 3.6 **Type:** Plant Systems: Electrical

System Number: 064 **System Title:** Emergency Diesel Generator

Description: Knowledge of abnormal condition procedures.

K/A Number: 2.4.11 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 4.0 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 90

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 ~3700 volts.
- CBOT also reports Startup #1 Transformer 22 Kvvoltage is 22.1 KV and SU#2 Transformer 161 KV 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.
- No grid disturbances are occurring.

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 2, 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 withSection 2, 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:

B. In accordance with Section 2, ES Bus Voltage Low, start one available DG, parallel the DG to the gid, and separate the associated ES bus from the grid by opening its feeder breaker.

Notes:

"B" is correct, bus voltage is low but not low enough to autostart the DGs, no grid disturbance is expected, so per 1203.037, section 1, step 6.A one DG should be started, paralleled to the grid, and the associated ES bus separated from the grid.
"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.
"D" is incorrect since no grid disturbance is expected, therefore the ES bus should not be de-energized to allow the DG to automatically re-energize it, but plausible since it refers to the correct procedure.

This is an SRO level question as it meets 10 CFR 55.43(b)(5), assessment of conditions and selection of procedures. It is not RO level since it requires in-depth knowledge of AOPs and has an SRO specific learning objective.

This question matches the K/A since it concerns both Emergency Diesel Generators and abnormal condition procedures.

References:

1203.037, Abnormal ES Bus Voltage and Degraded Offsite Power

History:

New SRO question for 2016 exam.

QID: 1056 **Rev:** 1 **Rev Date:** 7/21/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 35 **Point Value:** 1

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

System Number: 011 **System Title:** Pressurizer Level Control

Description: Ability to apply technical specifications for a system.

K/A Number: 2.2.40 **CFR Reference:** 43.2

Tier: 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 91

It is 0900 on Sunday, July 31, 2016.
You are assuming the watch after being on assignment to Training.
The plant is operating at 100% power.
While reviewing the logs you notice PZR level transmitter LT-1001 failed LOW at 2100 on Thursday, June 30.

Technical Specification LCO 3.3.15 Action B.1 states to initiate action to prepare and submit aa special report to the NRC.

When is the special report to the NRC due?

A. 8 hours

B. 24 hours

C. 30 days

D. 60 days

Answer:

C. 30 days

Notes:

"C" is correct, LCO action 3.3.15.B.1 is required since LT-1001 has been inoperable for greater than 30 days. The bases for this LCO action states the report is due within 30 days.
"A" is incorrect but plausible if the candidate incorrectly concludes that the report due time is similar to some LCO action time requirements.
"B" is incorrect but plausible if the candidate incorrectly concludes that the report due time is similar to some LCO action time requirements.
"D" is incorrect but plausible if the candidate incorrectly concludes that the report due time is twice that of the actual time.

This is SRO level since it involves 10CFR55.43(b)(2), application of Technical Specifications.

This question meet the K/A since it involves a Pressurizer Level channel (LT-1001) used in the Pzr Level Control system and it involves applicatlion of Tehcnical Specifications.

References:

Technical Specifications 3.3.15 and bases

History:

New question for 2016 SRO exam

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

QID: 1048 **Rev:** 0 **Rev Date:** 3/22/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-FUEL-FHPRO **Objective:** 1 **Point Value:** 1

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

System Number: 034 **System Title:** Fuel Handling Equipment

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Mispositioned fuel element.

K/A Number: A2.03 **CFR Reference:** 43.7

Tier:	2	**RO Imp:**	3.3	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** **SRO:** 92

Given:
- Unit 1 is in Mode 6 with core reload in progress
- You are the SRO In Charge of Fuel Handling
- You just reported to control room communicator a fuel assembly is inserted into core location G6.
- ATC reports to control room communicator that Source Range count rate is increasing continuously.

Which of the following actions are procedurally required for the above conditions?

A. Commence Attachment K "Setting Containment Closure" of 1015.002, Decay Heat Removal and LTOP System Control.

B. Notify Radiation Protection to monitor Main Fuel Bridge area with beta sensitive monitoring equipment.

C. Direct fuel handlers to remove the last assembly inserted into the core.

D. Make announcement over plant PA system to evacuate the Rx Bldg and activate RB evacuation alarm.

Answer:

C. Direct fuel handlers to remove the last assembly inserted into the core.

Notes:

"C" is correct per 1502.004, step 8.15.8, the last assembly inserted into the core or replacing the last removed control rod when source range count rate rises unexpectedly.
"A" is incorrect, this action is not required for this situation but is required for a damaged fuel assembly per 1203.042, Refueling Abnormal Operations, Section 1, Fuel Handling Accident.
"B" is incorrect, this action is not required for this situation but is required for a damaged fuel assembly per 1203.042, Refueling Abnormal Operations, Section 1, Fuel Handling Accident.
"D" is incorrect, this action is not required for this situation but is required for a damaged fuel assembly per 1203.042, Refueling Abnormal Operations, Section 1, Fuel Handling Accident.

This is an SRO level question as it meets 10 CFR 55.43(b)(7), fuel handling facilities and procedures. It is not RO level since this specific action is an SRO In Charge of Fuel Handling responsibilty iin 1502.004.

This question meets the K/A as it has the candidate assess the impact of a mispositioned fuel assembly and must choose one of the choices, all of which are procedural steps. The specific procedure references are NOT included in these choices since three of the four come from the same procedure and would reduce their plausibility.

References:

1502.004, Control of Unit 1 Refueling

History:

New SRO question for 2016 exam.

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

QID: 1053 **Rev:** 3 **Rev Date:** 7/15/16 **Source:** Modified **Originator:** Cork

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

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

System Number: 086 **System Title:** Fire Protection

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure to actuate the FPS when required, resulting in fire damage.

K/A Number: A2.04 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 93

The plant is at 100% power when the "FIRE" alarm comes in.

The CBOT checks the C463 panels and reports a fire alarm is indicated in the Lower South Electrical Penetration Room (LSEPR), Zone 105-T.

The investigating Inside AO reports the LSEPR deluge valve did NOT actuate and can NOT be manually actuated.

The Inside AO also reports the fire as severe.

Which of the following procedures should be transitioned to and will contain actions that will allow the control room staff to quickly mitigate the specific consequences of components damaged by fire in this area?

A. 1203.009, Fire Protection System Annunciator Corrective Action

B. 1203.049, Fires in Areas Affecting Safe Shutdown

C. 2203.034, Fire or Explosion

D. ANO Pre-Fire Plan for Zone 105-T

Answer:

B. 1203.049, Fires in Areas Affecting Safe Shutdown

Notes:

"B" is correct, starting with the annunciator corrective action (1203.009), the CRS will transition to 2203.034 after the fire is confirmed, and then transition to 1203.049 from section 2 of 2203.034. The LSEPR (105-T) will be listed in 1203.049 and contains specific actions for a fire in this area.
"A" is incorrect, this distracter is plausible in that it will be used to respond to the annunciator and will contain direction to actuate the deluge but it does not contain specific actions for affected components in this area.
"C" iis incorrect, this distracter is plausible in that it will be used to dispatch the fire brigade but it will direct the user to go to 1203.049.
"D" is incorrect, this distracter is plausible in that it will be used by the fire brigade to respond to the fire but it does not contain specific actions for affected components in this area since it is a 1203.049 area.

This question is SRO only since it meets 10CFR55.43(b)(5): the question requires the candidate to evaluate the conditions given and to select the appropriate procedure and action within that procedure which would assist in mitigating the event.

This question meets the K/A since it requires the candidate to assess the malfunction of the LSER deluge valve and select the procedure containing actions which will assist in mitigating the malfunction.

Added "specific" prior to "consequences" in stem at request of NRC examiner. JWC 7/15/16

References:

1203.049, Fires in Areas Affecting Safe Shutdown

History:

This is a modification of QID 014, last used on 2007 SRO exam.
Selected for 2016 SRO exam.

QID: 1055 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** Bank **Originator:** Cork

TUOI: ASLP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

K/A Number: 2.1.5 **CFR Reference:** 43.5

Tier: 3 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 94

The plant is at 100% power on New Year's Eve night shift.
The on-duty CRS has a heart attack and must be transported to St. Mary's at 0530.

What is the LATEST time at which a replacement CRS must be in the Control Room to preclude a violation of Technical Specifications?

A. 0830

B. 0730

C. 0630

D. 0530

Answer:

B. 0730

Notes:

Answer "B" is the correct answer since the maximum time the shift can be below the minimum complement is two hours.
Answer "A" is incorrect but plausible if the candidate doesn't recall the time requirement and believes the replacement must be in the control room within 3 hours.
Answer "C" is incorrect but plausible if the candidate doesn't recall the time requirement and believes the replacement must be in the control room within 1 hour.
Answer "D" is incorrect but plausible if the candidate doesn't recall the time requirement and believes the replacement must be immediate.

This is an SRO level question as it meets 10CFR55.43(b)(2), it requires knowledge of Technical Specification staffing requirements not expected of Ros and has an SRO specific learning objective.

This question matches the K/A as the candidate must be able to recall shift staffing requirements.

References:

Technical Specification 5.2.c

History:

Revised QID 885 for 2016 SRO exam (still bank).
Revised question QID 885 (last used in the 2014 SRO exam) by changing time in stem from 0430 to 0530 thereby making 0730 the correct answer (vs. 0600 in QID 885"A"). Revised all answer choices. Made choices highest to lowest vs. lowest to highest.

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

QID: 0846 **Rev:** 0 **Rev Date:** 5/26/11 **Source:** Modified **Originator:** D. Thompson

TUOI: A1LP-SRO-FH **Objective:** 6 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledge and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of the fuel-handling responsibilities of the SRO.

K/A Number: 2.1.35 **CFR Reference:** 43.7

Tier: 3 **RO Imp:** 2.2 **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** **SRO:** 95

Given:
- Unit 1 refueling is in progress.
- Due to difficulties inserting a fuel assembly into the core, the Bridge Operator requests an alteration in the fuel load sequence.

In order to change the fuel shuffle sequence the SRO in Charge of Fuel Handling and ____________ must approve the change per 1502.004, Control of Unit 1 Refueling.

A. Control Room Supervisor

B. Shift Manager

C. Reactor Engineer

D. Refueling Project Manager

Answer:

C. Reactor Engineer

Notes:

"C" is correct per 1502.004, step 8.11.
"A" is incorrect but plausible as the CRS does have an SRO license and is directly in charge of control room operations.
"B" is incorrect but plausible as the Shift Manager is the the person responsible for shift operations.
"D" is incorrect but plausible as this person is the the most senior individual in the Refueling Project.

This questions is SRO level because it involves fuel handling facilities and procedures, i.e., meets 10CFR55.43(b)(7).

This question matches the K/A since it requires knowledge of the fuel handling responsibility of who is requires to alter the fuel load sequence.

References:

1502.004, Control of Unit 1 Refueling

History:

Modified QID 250 for 2011 SRO Exam.
Selected for 2016 SRO exam.

QID: 0486 **Rev:** 1 **Rev Date:** 5/11/16 **Source:** Modified **Originator:** Cork

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:** C

Question: **RO:** **SRO:** 96

The system engineer for the Makeup and Purification System gives you a test procedure he wants interim approval of.
The test involves entering the Makeup Tank pressure curve's Restricted Operating Region of Exhibit A of 1104.002.
The system engineer believes the curve is too conservative.

Which of the following options are required to ensure the test procedure is in compliance with the licensing basis?

A. Get Asst. Ops Manager approval and implement the test procedure per EN-OP-112, Night and Standing Orders.

B. Approve the procedure after it has a Technical Review per 1000.006, Procedure Control.

C. Use the Procedure Modification process in 1000.006, Procedure Control, to implement the test procedure.

D. Ensure a PAD review per EN-LI-100, Process Applicability Determination, is completed to support the test procedure.

Answer:

D. Ensure a PAD review per EN-LI-100, Process Applicability Determination, is completed to support the test procedure.

Notes:

"D" is correct, any test or experiment that is not described in the SAR is required to have a PAD review which will then intitiate a 50.59 evaluation which in turn will determine if the test procedure conflicts with our licensing basis.
"A" is incorrect, a night order may not be used for a test procedure but plausible since night orders may be used for additional instructions which do NOT conflict with the SAR or existing procedural guidance.
"B" is incorrect, but plausible since all procedure changes require a technical review but this is insufficient scrutiny for a test procedure.
"C" is incorrect, but plausible since the procedure modification process has replaced the old procedure "deviation" section allowing operations to continue when the existing procedure guidance won't allow it but this process may not be used for something like a test procedure.

This question is SRO level since it tests the knowledge identified in 10CFR55.43(b)(3): what is needed to make operating changes to the facility. This question is based upon OE.

This question matches the K/A as it tests the knowledge of performing special or infrequent tests, in this case, the SRO must recognize that any new test must have a PAD review.

References:

1000.006, Procedure Control

History:

New question for 2004 SRO exam.
Used on 2004 SRO Exam.
This question was heavily modified to reflect current procedural guidance for the 2016 SRO exam. The initial conditions are the sam but ALL answer choices were revised. The question stem was revised to be more specific.

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

QID: 0879 **Rev:** 1 **Rev Date:** 5/16/16 **Source:** Bank **Originator:** NRC Exam Bank

TUOI: ASLP-SRO-MNTC **Objective:** 2 **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 managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

K/A Number: 2.2.17 **CFR Reference:** 43.5

Tier: 3 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 3

Group: **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 97

In accordance with EN-WM-100, Work Request (WR) Generation, Screening and Classification, an approved Work Order Package ______ required for Emergency maintenance. Prior to performing work, authorization to begin the work must be approved at a MINIMUM by the ______ .

A. Is
Shift Manager

B. Is NOT
Shift Manager

C. Is
Work Week Manager

D. Is NOT
Work Week Manager

Answer:

B. is NOT
Shift Manager

Notes:

Answer B is correct per EN-WM-100. Emergency maintenance can be approved by the Shift Manager and a work order is used to document the work performed as soon as practical afterwards.
Answer A is incorrect, this answer is plausible in that it has the proper authority but a work order package is not required prior to the work, however this is the normal (non-emergency) sequence.
Answer C is incorrect, this answer is plausible in that it has the correct sequence for work order preparation but the incorrect approval authority although the Work Week Manager is the ultimate authority for executing work per WN-WM-101 for non-emergency situations.
Answer D is incorrect, this answer has the incorrect authority (although plausible as in the explanation for C) and the incorrect sequence (although plausible in the explanation for A).

Revised question by removing "Priority 1" to avoid the possibility of having two correct answers. Also made the stem two separate sentences.

This is SRO level since it involves work package approval and authorization for plant maintenance, this is part of 10CR55.43(b)(5). Training is NOT given to ROs for this administrative duty.

This meets the K/A since it involves knowledge of the managing maintenance activities, i.e., emergency maintenance, and work prioritization.

References:

EN-WM-100, Work Request Generation, Screening and Classification

History:

Selected for 2014 SRO Exam. (Direct from Crystal River Exam 2011 SRO Question #21, slightly changed to align with ANO)
Selected for 2016 SRO exam

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

QID: 1049 **Rev:** 1 **Rev Date:** 7/8/16 **Source:** Modified **Originator:** J. Cork

TUOI: A1QC-SRO-QUAL **Objective:** 3.23 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 43.4

Tier: 3 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 98

Given:

- A Site Area Emergency has been declared on Unit 1.
- An Emergency Medical Team must enter a 500 REM/hr area to rescue a critically injured employee (they are directed, i.e., not volunteers).

Which of the following is the MAXIMUM time each individual team member can stay in this area and who can authorize a team member to extend this maximum time if they volunteer to do so?

A. 1 minute
Shift Manager

B. 3 minutes
Shift Manager

C. 1 minute
OSC Manager

D. 3 minutes
OSC Manager

Answer:

B. 3 minutes
Shift Manager

Notes:

"B" is correct since the exposure limit to save a life is 25 Rem in 1903.033 and in a 500 REM/hr area this equates to 8.33 REM/minute so the total stay time would be 3 minutes. Authorization to exceed 10CFR20 limits is given by Shift Manager or Emergency Director or Emergency Plant Manger per 1903.033.
"A" is incorrect, but plausible in case the examinee uses the limit of 10 Rem to save valuable equipment, and the responsible person is correct.
"C" is incorrect, but plausible in case the examinee uses the limit of 10 Rem to save valuable equipment, and the responsible person is incorrect.
"D" is incorrect, but plausible since the stay time is correct but the responsible person is incorrect.

This is an SRO level question as it meets 10 CFR 55.43(b)(4), radiation hazards. It is not RO level since this specific knowledge is not part of the initial RO curriculum and has an SRO specific objective.

This question meets the K/A since it is specifically about emergency radiation exposure limits.

The dose rate for the area was changed from 100 R/hr to 500 R/hr which changes the correct answer from 15 minutes to 3 minutes. Changing a condition to make a different choice correct meets the critieria of a modified question per ES-401, D.2.f. Modified other distracters to make them more plausible.

Revised per recommendation of NRC examiner. JWC 7/21/16

References:

1903.033, Protective Action Guidelines for Rescue/Repair & Damage Control Teams

History:

This is a Modified version of QID 120 for use in the 2016 SRO exam.

QID: 1088 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** Cork

TUOI: **Objective:** **Point Value:** 1

Section: 2.0 **Type:** Generic Kas

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

Tier:	3	**RO Imp:**	3.1	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** **SRO:** 99

Which of the following must be performed if a General Emergency is declared?

A. Offsite evacuation

B. Localized evacuation

C. TSC/OSC evacuation

D. Exclusion area evacuation

Answer:

D. Exclusion area evacuation

Notes:

"D" is correct, per step 6.1.3.A of 1903.030 an exclusion area evacuation shall be initiated if a GE is declared.
"A" is incorrect but plausible since, in most cases, an offiste evacuation will be recommended by the Emergency Director but there are situations where sheltering vs. evacuation will be recommended.
"B" is incorrect but plausible since a localized evacuation is quite possible during a GE event but is not always required.
"C" is incorrect but plausible since a GE could cause an TSC/OSC evacuation but it is not a certainty, IAW 1903.030 these are not normally evacuated in the event of a plant evacuation. They are only evacuated if the raditiation levels exceed exceed 2.5 mR/hr or airborne activity >9E-10 µCi/cc.

Knowledge of evacuations during a GE is taught exclusively to the SRO candidates, this is therefore an SRO level question that is linked to 10CFR55.43(b)(5).

This question matches the K/A since it requires knowledge of the emergency plan.

Changed "C" to TSC/OSC per request of NRC examiner. JWC 7/15/16

References:

1903.030, Evacuation

History:

New question for 2016 SRO exam.

QID: 0411 **Rev:** 1 **Rev Date:** 5/16/2016 **Source:** Bank **Originator:** E-Plan

TUOI: ASLP-RO EPLAN **Objective:** 7 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

K/A Number: 2.4.30 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	2.7	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 100

Unit 1 is shutdown for a refueling outage.
A fire was reported at 0844 in the Reactor Building.
It is now 0920 and the fire is still burning.

Based on the above conditions what is the time requirement per 1903.011, Emergency Response/Notifications for notification to the NRC?

A. Notification to the NRC is required within 15 minutes of the declaration of an emergency class and notify the Arkansas Department of Health within 1 hour.

B. Notification to the NRC is required immediately following notification of the Arkansas Department of Health and within 1 hour of the declaration of an emergency class.

C. Notification to the NRC is required immediately following declaration of an emergency class and notify the Arkansas Department of Health within 1 hour.

D. Notification to the NRC is required within 30 minutes of the declaration of an emergency class, after notifying the Arkansas Department of Health.

Answer:

B. Notification to the NRC is required immediately following notification of the Arkansas Department of Health and within 1 hour of the declaration of an emergency class.

Notes:

"B" is correct since this is the procedural requirement.
"A" is incorrect but plausible as it is the reverse of the correct requirement.
"C" is incorrect but plausible since the NRC is notified immediately and it has a one hour requirement, but the sequence is incorrect.
"D" is incorrect but plausible since the sequence is correct and there are notifications which must be made within 4 hours, but this is not in accordance with 1903.011.

References:

1903.011Y, Emergency Class Initial Notification Message

History:

Modified E-Plan exam bank QID#61 for use in 2001 SRO Exam.
Selected for use in 2002 SRO exam.
Selected for 2010 SRO exam
Repeated for 2011 SRO Exam.