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

QID: 0214 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** E. Jacks

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to identify post-accident instrumentation.

K/A Number: 2.4.3 **CFR Reference:** 41.6 / 45.4

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

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

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

Which of the following Control Room control panels does NOT have post accident components that are verified per Repetitive Task (RT-10) following an ESAS actuation?

a. Control Panel C26

b. Control Panel C16

c. Control Panel C10

d. Control Panel C13

Answer:

d. Control Panel C13

Notes:

All of the listed panels with the exception of C-13 have components that need to be verified following an ESAS actuation per RT-10.

References:

1202.012, Chg. 004-03-0

History:

Developed for use in sensitive 98 RO Re-exam
Selected for use in sensitive RO re-exam.

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

QID: 0027 **Rev:** 1 **Rev Date:** 3/16/05 **Source:** Modified **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 008 **System Title:** Pressurizer (PZR) Vapor Space Accident

Description: Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves.

K/A Number: AK2.01 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

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

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

The following plant conditions exist:

- Pressurizer temperature is 588 °F
- Pressurizer level is 285 inches and rising
- RCS Pressure is 1400 psig and lowering
- Quench Tank pressure is 0 psig and stable
- The ERV acoustic monitor indicates flow noise

What would be the expected temperature as indicated on the ERV PSV-1000 Outlet Temp on the Safety Parameter Display System (SPDS)?

A. Approximately 212 °F

B. Approximately 260 °F

C. Approximately 280 °F

D. Approximately 588 °F

Answer:

B. Approximately 260 °F

Notes:

Candidates should be provided with steam tables. The temperature elements on the ERV tailpipe would indicate the temperature for the Quench Tank pressure from the Mollier diagram, therefore, answer (b) is correct since the isenthalpic throttling process will produce a superheated value. The disclaimers are incorrect because: (a) is the saturation temperature for 14.7 psia, (c) is temperature on the saturation line coming straight across the Mollier diagram for 1400 psia and (d) is the saturation temperature of the pressurizer.

References:

ASME Steam Tables

History:

Developed for 1998 RO Exam.
Selected for 2002 RO exam under 008 AK1.01.
Modified for 2005 RO exam but not used.
Selected for 2005 sensitive RO re-exam.

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

QID: 0029 **Rev:** 0 **Rev Date:** 7/8/98 **Source:** Direct **Originator:** GGiles

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: Natural circulation and cooling, including reflux boiling.

K/A Number: EK1.01 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 3

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

Question: RO: 3 SRO:

Given the following plant conditions:

- Reactor trip from full power
- Full ES actuation
- ICCMDS Display Subcooling Margin indicates 0 °F
- ICCMDS CET temperatures are alternating between superheated and saturated conditions.

Which of the following best describes the mode of RCS cooling for these conditions?

a. Reflux Boiling

b. Forced Convection

c. Natural Circulation

d. Natural Conduction

Answer:

a. Reflux Boiling

Notes:

Answer (a) is correct since the conditions listed would indicate an ICC event in which boiler condenser cooling would occur (commonly referred to as "reflux boiling" at ANO). (b) is incorrect since RCPs are OFF, (c) is incorrect since SCM is lost, (d) is incorrect because conductive heat transfer would provide minimal heat removal.

References:

1202.010, Chg. 005-03-0

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0183 **Rev:** 0 **Rev Date:** 11/21/98 **Source:** Direct **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 reasons for the following responses as they apply to Loss of Rx Coolant Makeup: Adjustment of RCP seal backpressure regulator valve to obtain normal flow.

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

Tier:	1	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

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

During restoration of normal makeup and seal injection, which of the following is true?

a. If PZR level is <55", normal makeup is restored before seal injection.

b. If RCP seal bleedoff temperatures are >180 degrees, seal injection control valve (CV-1207) is quickly opened to establish previous flow rate.

c. If RCP seal bleedoff temperatures are >180 degrees, CV-1207 is slowly opened to minimize thermal shock to the RCP seals.

d. BWST outlet valve associated with the operating HPI pump must be closed prior to opening CV-1207.

Answer:

c. If RCP seal bleedoff temperatures are >180 degrees, CV-1207 is slowly opened to minimize thermal shock to the RCP seals.

Notes:

(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.
(c) is correct. As stated above, seal injection must be restored slowly to ensure RCP seals are not damaged.
(d) is incorrect. Restoring normal makeup and seal injection has no dependency on the BWST outlet valve position.

References:

1203.026, Chg. 009-05-0

History:

Developed for use in sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0327 **Rev:** 0 **Rev Date:** 9-06-99 **Source:** Direct **Originator:** D. Fowler

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

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

System Number: 025 **System Title:** Loss of Residual Heat Removal System

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: Isolation of RHR low pressure piping prior to pressure increase above specified level.

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

Tier: 1 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

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

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

The RCS pressure setpoints at which the Decay Heat suction isolation valves close are __________ for CV-1050 and __________ for CV-1410.

a. 290 psig, 320 psig

b. 340 psig, 400 psig

c. 320 psig, 385 psig

d. 340 psig, 385 psig

Answer:

c. 320 psig, 385 psig

Notes:

"c" lists the correct setpoints, the others are close but incorrect.

References:

1104.004, Chg. 072-01-0

History:

Used in 1999 exam.
Direct from ExamBank, QID# 1773
Selected for 2005 sensitive RO re-exam.

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

QID: 0013 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-AO-ICW **Objective:** 6 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 026 **System Title:** Loss of Component Cooling Water (CCW)

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:** 43.5 / 45.13

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

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

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

During the winter months, which of the following would indicate a tube leak in Nuclear ICW Cooler, E28C?

A. Rise in SW system activity

b. Rise in ICW Surge Tank level

c. Rise in SW header pressure

d. Drop in Nuclear ICW header flow

Answer:

b. Increase in ICW Surge Tank level

Notes:

SW pressure is higher than ICW pressure therefore a tube leak in an ICW cooler would result in flow from SW to ICW, therefore "B" is correct. The other answers would be indicative of flow from ICW to SW.

References:

1104.028, Chg. 023-07-0

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0622 **Rev:** 0 **Rev Date:** 10/31/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

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

K/A Number: AK2.03 **CFR Reference:** 41.5, 41.10 / 45.6 / 45.13

Tier:	1	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.8	**SRO Select:**	No	**Taxonomy:**	C

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

Given:
- RCS pressure has dropped to 2115 psig.

Which of the following would indicate a malfunction in the response of the Pressurizer heater controller in response to the above parameter?

A. Heater banks 1 and 2 full ON.

B. Heater banks 1, 2, and 3 full ON.

C. Heater banks 1, 2, 3, and 4 full ON.

D. Heater banks 1, 2, 3, 4, and 5 full ON.

Answer:

D. Heater banks 1, 2, 3, 4, and 5 full ON.

Notes:

"D" is correct since the bank 5 heaters are not full ON until pressure drops to 2105 psig.
The other choices are incorrect since the other heater banks are all full ON at 2115 psig (bank 4 is full ON at 2120 psig lowering).

References:

1203.015, Chg. 011-01-0

History:

New for 2005 sensitive exam.

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

QID: 0035 **Rev:** 0 **Rev Date:** 7/9/98 **Source:** Direct **Originator:** GGiles

TUOI: A1LP-RO-EOP01 **Objective:** 13 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 029 **System Title:** Anticipated Transient Without Scram (ATWS)

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

K/A Number: EK2.06 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	K

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

The plant is at 100% power when "B" MFW pump trips.

- FW pumps discharge cross-tie valve, CV-2827, fails to open due to mechanical binding.
- The CBOR depresses the manual reactor trip pushbutton.
- All rods remain out.

Which one of the following actions should be the number one priority?

a. Manually trip the turbine and verify GVs/TVs closed.

b. Open BWST outlet valve to running Makeup pump.

c. Depress CRD power supply breaker trip PBs on C03.

d. Manually insert rods and send operator to locally trip CRD breakers.

Answer:

c. Depress CRD power supply breaker trip PBs on C03.

Notes:

Candidate should know that "C" is the first action which should be attempted, this is the most expedient action to insert negative reactivity into the core to shutdown the reactor. A reactor trip is required due to loss of MFW to "B" OTSG.
"A" and "D" are also followup actions, "D" should only be done if "C" is unsuccessful. "A" is a followup action to verify turbine has tripped but is a lower priority to reactivity.
"B" is a method of emergency borating to insert negative reactivity but is slow and thus a lower priority to step "C".

References:

1202.001, Chg. 028-03-0

History:

Developed for 1998 SRO Exam.
Selected for 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0364 **Rev:** 0 **Rev Date:** 11/8/00 **Source:** Direct **Originator:** J.Cork

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

Section: 4.1 **Type:** Generic EPEs

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

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

K/A Number: 2.4.48 **CFR Reference:** 43.5 / 45.12 / 41.10

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	An

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

After a reactor trip, the following indications are observed:

- Makeup Tank level has lost 5 inches in the last 5 minutes
- RB and Aux. Bldg. Sump levels are stable
- "A" EFIC level is 35 rising and "A" MFW Flow is 0.1 mlb/hr
- "B" EFIC level is 31 stable and "B" MFW Flow is 0.3 mlb/hr

Which of the following actions would be required to minimize the threat of a potential radioactive release to the public?

a. Initiate HPI per RT-2

b. Cooldown and isolate the "B" SG

c. Cooldown and isolate the "A" SG

d. Commence a rapid RCS cooldown at 240 °F/hr

Answer:

c. Cooldown and isolate the "A" SG

Notes:

Answer [c] is correct, the SG level parameters indicate a rupture on the "A" SG and a cooldown should be commenced to reduce RCS temperature to <500 F to minimize the possibility of lifting a secondary safety on the "A" SG.
[a] is incorrect, the leak size is 30 gpm, this is within the capacity of normal makeup.
[b] is incorrect, a cooldown and isolation is required but not on this SG.
[d] is incorrect, a rapid cooldown at this rate is not required until overfilling of ruptured SG is imminent.

References:

1202.006, Chg. 007-04-0

History:

Created for 2001 RO/SRO Exam.
Selected for 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0285 **Rev:** 0 **Rev Date:** 9-3-99 **Source:** Direct **Originator:** D. Slusher

TUOI: A1LP-RO-EOP03 **Objective:** 4 **Point Value:** 1

Section: 4.3 **Type:** Generic Abnormal Plant Evolutions

System Number: 040 **System Title:** Steam Line Rupture - Excessive Heat Transfer

Description: Knowledge of symptom based EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	K

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

Following a turbine and reactor trip, an overcooling transient is occurring due to a stuck open safety.

The affected SG pressure is 825 psig and falling.

What actions are taken to seat the MSSV per the Overcooling procedure?

a. Actuate Main Steam Line Isolation for the SG with the lowest pressure.

b. Quickly reduce the lowest SG pressure to 700 psig or the MSSV open alarm is clear.

c. Trip both Main Feedwater pumps, actuate EFW, and perform RT-5.

d. Shut the Main Feedwater Isolation Valve for the affected Steam Generator.

Answer:

b. Quickly reduce the lowest SG pressure to 700 psig or the MSSV open alarm is clear.

Notes:

"a", "c", and "d" are incorrect because they are actions to isolate the OTSG not actions to seat the MSSV.

References:

1202.003, Chg. 005-01-0

History:

Used in 1999 exam.
Direct from ExamBank, QID# 5274 used in class exam
Selected for use in 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0623 **Rev:** 0 **Rev Date:** 10/31/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

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

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

What has occurred during this transient?

A. Loss of Instrument Air.

B. All RCPs have tripped.

C. Reactor tripped.

D. Turbine tripped.

Answer:

C. Reactor tripped.

Notes:

"C" is correct, the ICS Rapid Feedwater Reduction (RFR) circuit will close the Startup and Low Load control valves.
"A" is incorrect, the Low Load Control Valves fail "As-Is" on a loss of Inst. Air.
"B" is incorrect, this would have closed the block valves but not the control valves.
"D" is incorrect, this would have had no direct effect on the FW system.

References:

STM 1-64, Rev. 9

History:

New for 2005 sensitive RO re-exam.

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

QID: 0496 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Repeat **Originator:** NRC

TUOI: ELP-NLO-ELEC1 **Objective:** 29 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to operate and monitor the following as they apply to a Station Blackout: Battery, when approaching fully discharged.

K/A Number: EA1.05 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

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

Unit 1 has been in a station black-out for 1.5 hours with battery bank D06 supplying bus D02 with power without a battery charger online for this entire time.

If the equipment on bus D02 does NOT change, which one of the following statements describes the battery's discharge rate (in amps) as the battery is expended?

A. The discharge rate will be fairly constant until the design battery capacity (1500 amp-hrs) is exhausted.

B. The discharge rate will decrease steadily until the design battery capacity is exhausted.

C. The discharge rate will increase steadily until the design battery capacity is exhausted.

D. The discharge rate will be fairly constant until the design battery capacity (amp-hours) is exhausted and then will rapidly decrease.

Answer:

C. The discharge rate will increase steadily until the design battery capacity is exhausted.

Notes:

P=IE; As the battery discharges under a constant load, battery voltage will drop and discharge current will rise.

References:

ELP-NLO-ELEC1

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0366 **Rev:** 0 **Rev Date:** 1/8/00 **Source:** Direct **Originator:** J.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:** 2

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

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

An electrical storm has caused a Degraded Power situation with a spurious ES actuation of the even channels.

In which order will the following ES components be started automatically?

a. SW pump, HPI pump, LPI pump, RB Spray pump

b. HPI pump, SW pump, LPI pump, RB Spray pump

c. SW pump, HPI pump, RB Spray pump, LPI pump

d. HPI pump, LPI pump, SW pump, RB Spray pump

Answer:

d. HPI pump, LPI pump, SW pump, RB Spray pump

Notes:

Answer [d] lists the correct order of load sequence with loss of offsite power and ES actuation.
The others are incorrect sequences of the correct components.

References:

1305.006, Chg. 021-01-0

History:

Created for 2001 RO/SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0624 **Rev:** 0 **Rev Date:** 11/2/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: S/G pressure and level meters.

K/A Number: AA2.05 **CFR Reference:** 43.5 / 45.13

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

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

Question: RO: 14 SRO:

What would be the effect on the SG pressure and level instruments if a loss of the RS-1 bus occurred?

A. Instrument power would automatically be transferred to YO-2 by the ABT.

B. The NNI-X S1 and S2 switches would open and SASS would transfer to NNI-Y.

C. Both NNI-X SG pressure indicators would fail mid-scale and ICS could not generate a BTU limit alarm.

D. Instrument power would automatically be transferred to YO-1 by the ABT.

Answer:

A. Instrument power would automatically be transferred to YO-2 by the ABT.

Notes:

"A" is correct, a loss of RS-1 would simply cause NNI-X to be powered from YO-2, -24vDC logic power is auctioneered and instrument power would transfer by the ABT within 0.5 seconds.
"B" is incorrect, it would take a loss of both RS-1 and YO-2 to cause the S1 and S2 switches to open.
"C" is incorrect, although SG pressure does input to the BTU limit alarm, the NNI-X SG pressure inidcators would not be failed due to the power transfer to YO-2.
"D" is incorrect, the alternate power to NNI-Y is from YO-1.

References:

STM 1-69, Rev. 8

History:

New for 2005 sensitive RO re-exam.

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

QID: 0187 **Rev:** 1 **Rev Date:** 4/25/2002 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-AOP **Objective:** 4.5 **Point Value:** 1

Section: 4.2 **Type:** Generic APE

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

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation.

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:** 3.1 **SRO Select:** No **Taxonomy:** C

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

Given the following indications at 100% power:

- Annunciator D02 UNDERVOLTAGE (K01-A8) in alarm.
- Annunciator D02 TROUBLE (K01-D8) in alarm.
- Annunciator D02 CHARGER TROUBLE (K01-E8) in alarm.
- The reactor has tripped.
- The turbine trip solenoid light is on.
- Breaker position lights on the RIGHT side of C10 are off.

What are the actions required of the CBOT?

a. Trip the main generator output breakers.

b. Transfer D11 to emergency supply D02.

c. Trip all RCPs.

d. Transfer D21 to emergency supply D01.

Answer:

d. Transfer D21 to emergency supply D01.

Notes:

[d] is correct per 1203.036 as the conditions are indicative of a loss of D02.
[a] or [b] might be chosen if candidate mistakenly concludes that D01 has been lost.
[c] might be chosen if candidate believes all RCP oil pumps have been lost.

References:

1203.036, Chg. 005-04-0

History:

Developed for use in sensitive 98 RO Re-exam
Selected for use in 2002 RO/SRO exam, revised slightly.
Selected for 2005 sensitive RO re-exam.

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

QID: 0625 **Rev:** 0 **Rev Date:** 11/2/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Nuclear service water temperature indications.

K/A Number: AA1.01 **CFR Reference:** 41.7/ 45.5 / 45.6

Tier:	1	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

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

Given:
- ESAS actuation has occurred on channels 1 through 6
- SW Pump P-4A does not start, P-4B is aligned to the other SW loop

Which components' temperatures should be closely monitored until P-4B can be re-aligned?

A. Circ Water Pumps P-3A & P-3B, HPI pump P-36B, Nuclear ICW components

B. Condensate Pumps, Circ Water Pumps P-3C & P-3D, HPI pump P-36A, Nuclear ICW components

C. Circ Water Pumps P-3A & P-3B, HPI pump P-36A, Nuclear ICW components

D. Circ Water Pumps P-3C & P-3D, HPI pump P-36C, Non-Nuclear ICW components

Answer:

C. Circ Water Pumps P-3A & P-3B, HPI pump P-36A, Nuclear ICW components

Notes:

"C" is correct, P-4A supplies cooling water to Loop I which serves the components listed.
"A" is incorrect, Loop I does not cool P-36B.
"B" is incorrect, Loop I does not cool Circ Water pumps P-3C & P-3D.
"D" is incorrect, this is a list of Loop II cooled components.

References:

1203.030, Chg. 013-01-0
STM 1-42, Rev. 10

History:

New for 2005 sensitive RO re-exam.

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

QID: 0103 **Rev:** 1 **Rev Date:** 11/2/05 **Source:** Modified **Originator:** JCork

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

Section: 4.2 **Type:** Generic APEs

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

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:** C

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

Given:
- Unit One Instrument Air pressure has been degrading for approximately 30 minutes.
- Unit Two Instrument Air pressure is steady.
- Inst. Air is being used for respirable air.

Suddenly, Unit One Inst. Air pressure drops to 70 psig.

Which of the following actions are required at this pressure?

A. Inform RP and isolate Inst. Air from respirable air.

B. Isolate Unit 2 Inst. Air from Unit 1.

C. Place VSF-9 Outside Air damper in RESERVE position.

D. Place RCP Seal Injection Block, CV-1207, in OVERRIDE.

Answer:

A. Inform RP and isolate Inst. Air from respirable air.

Notes:

"A" is correct, respirable aiir should be isolated.
"B" is incorrect, this is done only if the leak is originating from Unit Two but their pressure is given as steady.
"C" is incorrect, this is done only if CR isolation occurs.
"D" is incorrect, this isn't done until IA pressure is <60 psig.

References:

1203.024, Chg. 010-08-0

History:

Developed for 1998 SRO exam
Modified for 2005 sensitive RO re-exam.

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

QID: 0626 **Rev:** 0 **Rev Date:** 11/3/05 **Source:** Modified **Originator:** J.Cork

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

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

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

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

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

Tier:	1	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	K

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

Given:

- Loss of all Feedwater
- HPI core cooling started
- Only one HPI pump is running

Which of the following indicates 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. One °F RCS temp change causes a 100 psig pressure change.

Answer:

B. CET temperatures stable or dropping.

Notes:

"B" is correct since the only criteria for evaluation of adequacy of core cooling via HPI is CET temps stable or dropping.
"A" is incorrect, this is used as a point where if CET temps are rising to try more drastic measures of regaining some form of feedwater.
"C" is incorrect, this is an individual indication of adequate primary to secondary heat transfer.
"D" is incorrect, this is an effect from the RCS being solid.

References:

1202.004, Chg. 004-02-0
EOP Technical Bases Document, 074-1152414, Rev. 8

History:

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

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

QID: 0369 **Rev:** 0 **Rev Date:** 11/13/00 **Source:** Direct **Originator:** R.Soukup

TUOI: A1LP-RO-CRD **Objective:** 15 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of annunciators alarms and indications, and use of the response instructions.

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

Tier: 1 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** Ap

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

Given:

- Plant is at 38% power.
- ICS is in full automatic.
- Rod 7 in Group 6 has dropped.
- Annunciator K08-C2 "CONTROL ROD ASYMMETRIC" is in alarm.
- All actions in response to the dropped rod have been completed.

Which of the following actions must be performed FIRST to recover the dropped rod?

a. Depress FAULT RESET on the Diamond panel.

b. Transfer dropped rod to its normal power supply.

c. Latch the dropped rod using auxiliary power supply.

d. Pull the dropped rod's motor power fuses.

Answer:

c. Latch the dropped rod using auxiliary power supply.

Notes:

Answer [c] is correct since an individual rod must be recovered using the auxiliary power supply and relatched.
Answer [a] is incorrect, this is not done until the rod is realigned with its group.
Answer [b] is incorrect, the rod's group is left on the normal supply.
Answer [d] is incorrect, this is only done for a rod with a high stator temperature condition.

References:

1105.009, Chg. 019-02-0

History:

Created for 2001 RO/SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0329 **Rev:** 1 **Rev Date:** 11/7/00 **Source:** Direct **Originator:** D. Slusher

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

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

System Number: 033 **System Title:** Loss of Intermediate Range Nuclear Instrumentation

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

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

Tier: 1	**RO Imp:** 3.6	**RO Select:** Yes	**Difficulty:** 4
Group: 2	**SRO Imp:** 3.9	**SRO Select:** No	**Taxonomy:** Ap

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

Given:

- Plant startup in progress
- NI501 at 9 x E4 cps
- NI502 at 1 x E5 cps
- NR502 is operable and at 5 x E-2% power
- NI3 at 2 x E-10 amps
- NI4 at 5 x E-10 amps
- NI5 thru 8 at 0%

Subsequently NI3 fails low.

What action should be taken by control room operators?

A. Maintain flux level in the source range

b. Trip the reactor

c. Continue with startup

d. Stablize and maintain power at 1 x E-8 amps

Answer:

c. Continue with startup

Notes:

The Intermediate Range NI values are on scale and represent one decade of overlap with the Source Range indications. Therefore with IR NI3 failed, the startup may continue as in answer © in accordance with 1203.021.
(a) is incorrect, this action would only be taken if less than one decade of overlap existed.
(b) is incorrect, although this action would be taken IAW 1203.021 if no on scale flux indication existed.
(d) is incorrect, power can continue up through 10-8 amps.

References:

1203.021, Chg. 008-01-0

History:

Used in 1999 exam
Direct from ExamBank, QID# 3099 used in class exam
Modified for use in 2001 RO/SRO Exam.

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

QID: 0426 **Rev:** 0 **Rev Date:** 4/25/2002 **Source:** Direct **Originator:** S.Pullin

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: When to isolate one or more S/Gs.

K/A Number: AA2.11 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	A

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.
- T Hot 485 degrees F.
- RCS pressure 1050 psig.

Which of the following would require isolating the "A" OTSG?

a. BWST level drops to 23.5 ft

b. Offsite dose rates meet the NUE criteria

c. "A" OTSG level rises to 415"

d. "A" OTSG tube-to-shell delta T rises to 50°F (tubes hotter)

Answer:

c. "A" OTSG level rises to 415"

Notes:

"C" is correct, if bad SG level rises to >410", then the SG is isolated.
"A" is incorrect, BWST level must drop to < 23 ft. to isolate SG.
"B" is incorrect, offsite dose rate must meet Alert critieria to isolate SG.
"D" is incorrect, tube to shell delta T is used as critieria to secure RCPs in SGTR.

References:

1202.006, Chg. 007-04-0

History:

New for 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0043 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** JCork

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

Section: 4.2 **Type:** Generic APEs

System Number: 060 **System Title:** Accidental Gaseous Radwaste Release

Description: Knowledge of the interrelationships between the Accidental Gaseous Radwaste Release and the following: ARM system, including the normal radiation-level indications and the operability status.

K/A Number: AK2.01 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	C

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

A radioactive gas release of WGDT T-18B has been terminated by Gaseous Radwaste Monitor, RE-4830, in high alarm. Examination of the recorder paper shows RE-4830 had been trending steadily at ~2000 cpm when it jumped to 30,000 cpm and then dropped back to ~2000 cpm.

What action should be taken?

a. Submit Condition Report and check system alignment.

b. Notify Nuclear Chemistry to perform offsite dose projections via RDACS.

c. Isolate T-18B and then re-submit sampling and release permit.

d. Reset RE-4830 and re-establish T-18B gaseous release.

Answer:

d. Reset RE-4830 and re-establish T-18B gaseous release.

Notes:

The indications given are representative of an instantaneous spike, therefore "d" is the correct answer. The other three are actions to be performed if RE-4830 did NOT "spike."

References:

1203.006, Chg. 010-02-0
1203.012I, Chg. 040-06-0

History:

Developed for 1998 RO Exam.
Selected for 2005 $\boxed{\text{sensitive}}$ RO re-exam.

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

QID: 0634 **Rev:** 0 **Rev Date:** 11/8/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 061 **System Title:** Area Radiation Monitoring (ARM) System Alarms

Description: Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring (ARM) System Alarms: Guidance contained in alarm response for ARM system.

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

Tier: 1 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 4

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

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

Given:
- AREA MONITOR RADIATION HI (K10-B1) in alarm
- RADIATION MONITOR TROUBLE (K10-C1) in alarm

In accordance with the alarm response procedure, the area monitors on C25 Bay 3 must be inspected.

What indication(s) would you expect to find on the alarming monitor drawer with both of the above annunciators in alarm?

A. WARNING and POWER ON lights on

B. POWER ON light off

C. HIGH ALARM light on and POWER ON light off

D. FAILURE light on

Answer:

B. POWER ON light off

Notes:

"B" is correct, a loss of power will cause both the Hi Radiation and Trouble annunciators to come in.
"A" is incorrect, this would cause the Hi Radiation but not the Trouble annunciator.
"C" is incorrect, the POWER ON light off will cause both annunicators but the HIGH ALARM light will not be on with a loss of power.
"D" is incorrect, this will cause the Trouble annunciator but not the Hi Radiation annunciator.

References:

1203.012I, Chg. 040-06-0
STM 1-62, Rev. 9

History:

New for 2005 sensitive RO re-exam.

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

QID: 0633 **Rev:** 0 **Rev Date:** 11/8/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 069 **System Title:** Loss of Containment Integrity

Description: Knowledge of the operational implications of the following concepts as they apply to Loss of Containment Integrity: Effect of pressure on leak rate.

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

Tier: 1 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** K

Question: RO: 24 SRO:

What is the key concern with a DBA occurring while the Reactor Building (RB) is inoperable per Technical Specifications?

A. A large release of radioactivity would cause embrittlement of the steel liner plate.

B. High temperatures would cause unalyzed stresses on piping within the RB.

C. High internal pressure would cause greater leakage of radioactivity to atmosphere.

D. Humidity inside RB would cause RB instrumentation accuracy to be unacceptable.

Answer:

C. High internal pressure would cause greater leakage of radioactivity to atmosphere.

Notes:

"C" is correct, RB inoperability is based on leakage and leakage in excess of Tech Specs could cause an unacceptable amount of radioactivity to be released due to a larger leak rate from the high internal pressures assumed during a DBA.
"A" is incorrect, although radiation does embrittle metal, this is not as significant as leakage.
"B" is incorrect, although high temperatures will raise stress levels on piping, this is not as significant as leakage.
"D" is incorrect, although humidity can cause instrumentation inaccuracies, specific RB instrumentation is qualified for this environment.

References:

T.S. 3.6.1 bases

History:

New for 2005 sensitive RO re-exam.

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

QID: 0289 **Rev:** 0 **Rev Date:** 9/4/99 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-EOP05 **Objective:** 6 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 074 **System Title:** Inadequate Core Cooling

Description: Knowledge of the operational implications of the following concepts as they apply to the Inadequate Core Cooling:
Processes for removing decay heat from the core.

K/A Number: EK1.03 **CFR Reference:** 41.8 / 41.10 / 45.3

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

Group: 1 **SRO Imp:** 4.9 **SRO Select:** No **Taxonomy:** C

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

For reflux boiling to be effective, the primary steam bubble must extend:

a. below the high point of the Tcold.

b. above the secondary side water level.

c. below the secondary side water level.

d. above the upper tube sheet.

Answer:

c. below the secondary side water level.

Notes:

In order to cool and condense the steam in the reactor coolant system the primary steam bubble must be below the secondary side water level. Therefore, "a", "b", and "d" are incorrect.

References:

EOP Technical Bases Document, Vol. 3, IV.C

History:

Used in 1999 exam.
Direct from ExamBank, QID# 536 used in class exam
Selected for 2005 sensitve RO re-exam.

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

QID: 0635 **Rev:** 0 **Rev Date:** 11/8/05 **Source:** New **Originator:** J.Cork

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

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

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

Description: Ability to operate and/or monitor the following as they apply to the (Plant Runback): Desired operating results during abnormal and emergency situations.

K/A Number: AA1.3 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** Ap

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

Given:
- Plant is at 100% power.
- "A" RCP suddenly trips.

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

A. T hot 593°F, T cold 564°F, MFW flow 3.3 x e6 lbm/hr each loop

B. T hot 588°F, T cold 569°F, MFW flow 2.2 x e6 lbm/hr each loop

C. T hot 597°F, T cold 561°F, MFW flow 4.1 x e6 lbm/hr each loop

D. T hot 592°F, T cold 565°F, MFW flow 3.0 x e6 lbm/hr each loop

Answer:

C. T hot 597°F, T cold 561°F, MFW flow 4.1 x e6 lbm/hr each loop

Notes:

"C" is correct, ICS will runback the plant to 75% power, the indications given are for that level. 100% power was assumed to be 11 x e6 lbm/hr total.
"A" is incorrect, these are the indications for 60% power. There is no runback to this level but there are only three different runback levels.
"B" is incorrect, these are the indications for 40% power. There is a runback to this power for a loss of 1 MFWP or asymmetric rod.
"D" is incorrect, these are the indications for 55% power. There is a runback to this power for a loss of 1 RCP in each loop.

References:

1105.004, Chg. 016-03-0
1102.004, Chg. 042-00-0

History:

New for 2005 sensitive RO re-exam.

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

QID: 0363 **Rev:** 0 **Rev Date:** 11/6/00 **Source:** Direct **Originator:** S.Pullin

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

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

System Number: E09 **System Title:** Natural Circulation Cooldown

Description: Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Cooldown): Normal, abnormal and emergency operating procedures associated with (Natural Circulation Cooldown).

K/A Number: EK1.2 **CFR Reference:** 41.8 / 41.10, 45.3

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- Natural circulation cooldown in progress
- CETs at 550°F
- Reactor vessel head temperatures at 614°F
- Pressurizer level = 150 inches, then makes step change to 180 inches
- RCS pressure at 1700 psig and slowly dropping
- "A" OTSG pressure = 945 psig
- "B" OTSG pressure = 950 psig

The required operator action for the above conditions is to pressurize the RCS slightly and reduce cooldown rate per 1203.013, Natural Circulation Cooldown.

What is the reason for this action?

a. Reduce the thermal stresses on the Reactor Vessel.

b. Restore adequate Subcooling Margin.

c. Collapse a steam void in the Rx Vessel head.

d. Comply with SG tube to shell delta-T limits.

Answer:

c. Collapse a steam void in the Rx Vessel head.

Notes:

Answer (c) is correct since this action is due to a steam void in the upper head as evidenced by the sudden change in Pzr level with no RCS pressure increase.
Answer (a) would be the proper response for PTS concerns.
Answer (b) is not applicable, subcooling margin is adequate with the values given.
Answer (d) only applies when the OTSGs are not being used to cool the RCS.

References:

1203.013, Chg. 017-00-0

History:

Developed for 2001 RO/SRO Exam
Selected for use in 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0053 **Rev:** 0 **Rev Date:** 7/8/98 **Source:** Direct **Originator:** JCork

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 (RCPS)

Description: Knowledge of the RCPS design feature(s) and/or interlock(s) which provide for the following: Adequate cooling of RCP motor and seals.

K/A Number: K4.04 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

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

Without operator action, which of the following incidents would have the most detrimental effect on RCP operation?

a. Loss of nuclear ICW to RCP

b. Main steam line rupture inside RB

c. Loss of RCP seal injection

d. RCP Bleedoff Normal Return, CV-1274, fails closed

Answer:

b. Main steam line rupture inside RB

Notes:

"b" is correct due to isolation of non-nuclear ICW to all RCP motors from ESAS.
"a" is incorrect since a loss of nuclear ICW will not harm RCPs unless seal injection is also lost.
"c" is incorrect since a loss of seal injection will not harm RCPs unless nuclear ICW is also lost.
"d" is incorrect, isolation of seal return will only cause seal staging pressures to change.

References:

1202.012, Chg. 004-03-0

History:

Used in 1998 initial RO exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0637 **Rev:** 0 **Rev Date:** 11/8/05 **Source:** Modified **Originator:** J.Cork

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

Section: 3.1 **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 CT pressure and NPSH for charging pumps.

K/A Number: K5.26 **CFR Reference:** 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:**	Ap

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

Initial Conditions: 100% power, steady state.

Makeup tank level is 70 inches.

What Makeup tank pressure would raise the risk the HPI pumps could lose suction during ESAS?

A. 10 psig

B. 25 psig

C. 30 psig

D. 35 psig

Answer:

D. 35 psig

Notes:

"D" is correct. If ESAS initiates HPI and the Makeup Tank Vent valve fails to open on low level of 18", then there is a 20 minute allowance for operator action to isolate the Makeup Tank. Going above the curve on Exhibit A reduces the time margin and raises the risk of introducing hydrogen to the suction of the HPI pumps.
"A" is incorrect, this limit is the low pressure limit and is still within Exhibit A.
"B" and "C" are incorrect, they are within the curve on Exhibit A.

References:

1104.002, Chg. 058-02-0 (Exhibit A must be in candidate's handout!)

History:

Modified regular exambank QID#55 for 2005 sensitive RO re-exam.

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

QID: 0630 **Rev:** 0 **Rev Date:** 11/7/05 **Source:** New **Originator:** J.Cork

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

Section: 3.1 **Type:** Reactivity Control

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

Description: Knowledge of the effect of a loss or malfunction on the following CVCS components: Heat exchangers and condensers.

K/A Number: K6.07 **CFR Reference:** 41.7 / 45.7

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

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

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

Which of the following would occur if the interlock between the Letdown Coolers Inlet valves and ICW Supply valves (to coolers) malfunctioned so the Letdown Cooler Inlet opened before the ICW Supply?

A. Letdown relief would pass steam, causing damage to relief.

B. Letdown cooler flow capacity would be exceeded.

C. Exceed thermal limits on N16 expansion tank.

D. Letdown isolation valve would close.

Answer:

D. Letdown isolation valve would close.

Notes:

"D" is correct. Without the interlock, hot RCS would cause letdown to isolated on high temperature to protect the DI resin.
"A" is incorrect, flashing on the outlet of a relief normally occurs.
"B" is incorrect, although there is a limit on cooler flow, this interlock does not prevent exceeding it.
"C" is incorrect, although the N16 tank will get hot, this will be within its design.

References:

STM 1-04, Rev. 7

History:

New for 2005 sensitive RO re-exam.

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

QID: 0293 **Rev:** 1 **Rev Date:** 11/7/05 **Source:** Direct **Originator:** D Slusher

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

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

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

Description: Knowledge of bus power supplies to the following: RHR pumps.

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

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2.5

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

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

Given:

- Plant is in Cold Shutdown
- "B" Decay Heat pump is running

Which of the following would cause a loss of Decay Heat Removal?

A. A-1 voltage of 2650 volts

b. A-2 voltage of 2650 volts

c. B-5 voltage of 415 volts

d. B-6 voltage of 415 volts

Answer:

d. B-6 voltage of 415 volts

Notes:

"B" Decay Heat Removal Pump is powered from A-4. An undervoltage on the A buses or B buses will trip A-409 (A4 feeder breaker). The undervoltage setpoint for A-4 is 2450 volts. The undervoltage setpoint for B-6 is 429 volts. Therefore, "a", "b" and "c" are incorrect.

References:

1107.002, Chg. 020-00-0

History:

Developed for 1999 exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0070 **Rev:** 0 **Rev Date:** 7/8/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-MU **Objective:** 3.2 **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 the effect that a loss or malfunction of the ECCS will have on the following: Fuel.

K/A Number: K3.02 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	An

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

Given:
- LOCA in progress with Degraded Power conditions
- EDG #2 did NOT start
- RCS pressure stabilized at 1750 psig
- RCS leakage calculated to be 210 gpm

With these accident conditions, which of the following would pose the greatest challenge to core cooling?

a. RCS leakage increases to 1000 gpm

b. Both CFT pressures at 560 psig

c. Lockout relay trips on 4160v A3 bus

d. Initial BWST level at 37 ft.

Answer:

c. Lockout relay trips on 4160v A3 bus

Notes:

Due to the high system pressure, the HPI system is the only system available to keep the core cool. Also, with only one train of ECCS available, if A3 were lost (due to lockout from high current) there wouldn't be any core cooling until the RCS depressurized to CFT pressure and LPI would not be available to cool the core at lower pressures. All of this makes [c] the greatest threat to core cooling.
[a] is incorrect due to ability of CFT's and LPI to cool core. With the larger break the RCS will depressurize, CFT's will dump and LPI will provide ~3000 gpm cooling flow.
[d] is incorrect since a BWST level slightly lower than T.S. only means that the ECCS will go on RB sump recirc earlier than if the level was at the minimum T.S. level of 38.4 ft. In addition with only one train of ECCS is operation, the demand on the BWST is lessened.
[b] is incorrect, even if both CFT's do not discharge all of their volume into the RCS, a sizable quantity will be dumped, and LPI is still available to cool core as the RCS is cooled down.

References:

SAR, Chapter 6.1

History:

Developed for the 1998 RO/SRO Exam.
Used in sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0208 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** J. Haynes

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal pressure in the PRT.

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

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- All Pressurizer heaters ON.
- RCS pressure is 2000 psig and going down.
- Quench Tank (T-42) pressure rising
- Pressurizer Level Control Valve, CV-1235, full closed.

What actions should be performed by the control room operators?

a. Close the Spray Line Isolation Valve (CV-1009).

b. Close the ERV Isolation Valve (CV-1000).

c. Take manual control of CV-1235 and open.

d. Maximize letdown flow.

Answer:

b. Close the ERV Isolation Valve (CV-1000).

Notes:

(a.) is incorrect. This would be the correct action if the spray valve was leaking but with the conditions given this would not accomplish anything
(b.) is correct. The RCS pressure is below the setpoint that the ERV should have reseated and the Quench Tank pressure is rising, so the ERV should be isolated.
(c.) is incorrect. A pressurizer steam space leak will cause level to swell, the level control valve is responding properly to the event.
(d.) is incorrect. This action would make the condition worse instead of helping to correct the condition.

References:

1203.015, Chg. 011-01-0

History:

Developed for use in sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0627 **Rev:** 0 **Rev Date:** 11/7/05 **Source:** New **Originator:** J.Cork

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

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

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

Description: Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: The standby feature for the CCW pumps.

K/A Number: K4.09 **CFR Reference:** 41.7

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

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

Question: **RO:** 34 **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, 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 loop.
"D" is incorrect, P-33A is the non-nuclear ICW pump.

References:

STM 1-43, Rev.8

History:

New for 2005 sensitive RO re-exam.

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

QID: 0628 **Rev:** 0 **Rev Date:** 11/7/05 **Source:** New **Originator:** J.Cork

TUOI: A1LP-RO-NNI **Objective:** 13 **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 physical connections and/or cause-effect relationships between the PZR PCS and the following systems: RPS.

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

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

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

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

Given:
- Plant is shutdown and cooldown is in progress.
- RCS pressure 215 psig.
- RCS temperature 200 °F.
- DH is in service.

Which of the following would resulf if the "A" RPS narrow range pressure transmitter (PT-1021) slowly failed high?

A. DH would be lost.

B. SASS would select "C" RPS (PT-1038).

C. ERV would open.

D. All Pressurizer heaters would turn off.

Answer:

D. All Pressurizer heaters would turn off.

Notes:

"D" is correct, PT-1021 from "A" RPS is the NNI-X input to RCS pressure control.
"A" is incorrect, the inputs for valve closure come from PT-1020 and PT-1041.
"B" is incorrect, a slow failure will not trigger SASS to swap channels.
"C" is incorrect, the low range input to the ERV comes from WR PT-1020 (ESAS).

References:

STM 1-69, Rev. 8

History:

New for 2005 sensitive RO re-exam.

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

QID: 0629 **Rev:** 0 **Rev Date:** 11/7/05 **Source:** New **Originator:** J.Cork

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

Section: 3.7 **Type:** Instrumentation

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

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

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

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

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

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

I&C is performing a surveillance on the "A" RPS channel.

During this surveillance, Power Range channel NI-7 fails high.

Which of the following will occur?

A. Nothing, SASS will transfer to NNI-Y.

B. Control rods will start to insert.

C. Control rods will start to withdraw.

D. Feedwater flows will go down.

Answer:

B. Control rods will start to insert.

Notes:

"B" is correct. "A" RPS will be in channel bypass and the NI selector switch will be in the NNI-Y position since the auctioneer circuit delivers the highest of 5 and 6 via NNI-X to ICS.
"A" is incorrect, SASS would do this but the NI selector switch is already in the NNI-Y position and NI-7 is one of the auctioneered inputs to NNI-Y.
"C" is incorrect, rods will insert, not withdraw.
"D" is incorrect, feedwater will go up, not down.

References:

STM 1-67, Rev. 9
STM 1-64, Rev. 9

History:

New for 2005 sensitive RO re-exam.

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

QID: 0563 **Rev:** 0 **Rev Date:** 4/5/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-RPS **Objective:** 16 **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:**	2
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	K

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

Upon removal of a critical module or placing a test module in other than the operate position, the Module-In-Test/Module-Removal interlock will:

A. Prevent that channel from tripping.

B. Place the RPS into a two-out-of-three trip logic.

C. Lock out the other channels' test switches.

D. De-energize the associated channel's trip relay.

Answer:

D. De-energize the associated channel's trip relay.

Notes:

"D" is correct.
"A" is incorrect, channel bypass performs this function.
"B" is incorrect, the logic would be one out of three since this trips the channel.
"C" is incorrect, channel bypass will lock out the other channels.

References:

STM 1-63, Rev. 6

History:

Direct from regular exam bank QID#1999
Selected for 2005 RO exam, but not used.
Selected for 2005 sensitive RO re-exam.

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

QID: 0057 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** GGiles

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

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Safety system logic and reliability.

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

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	C

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

Which of the following would result in all ten (10) ESAS Digital Channels receiving an Analog Channel Trip signal?

a. Removal of the low RCS Pressure Bistable from Analog Channel 1.

b. Removal of the RB Pressure Buffer Amplifier from Analog Channel 1.

c. Removal of the 30 psig RB Pressure Bistable from Analog Channel 2.

d. Removal of the RCS Pressure Buffer Amplifier from Analog Channel 2.

Answer:

b. Removal of the RB Pressure Buffer Amplifier from Analog Channel 1.

Notes:

(b) is correct. (a) and (d) will result in an analog channel trip signal to digital channels 1,2,3 and 4. (c) will result in an analog channel trip signal to digital channels 7 and 8 or 9 and 10 (depending on which 30 psig bistable was removed).

References:

STM 1-65, Rev. 3

History:

Developed for the 1998 RO/SRO exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0256 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Cooling water flow.

K/A Number: A1.04 **CFR Reference:** 41.5 / 45.5

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

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

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

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

a. Service water flow to the Auxiliary Cooling Water System is raised during normal operation.

b. Maintains adequate service water flow to the Reactor Building Coolers when ES actuates.

c. Reactor coolant to service water differential temperature is reduced when ES actuates.

d. Decay heat coolers are maintained full and reduces the chance of water hammer.

Answer:

b. Maintains adequate service water flow to the Reactor Building Coolers when ES actuates.

Notes:

"a" is incorrect, while it is true that ACW demand will be higher during normal operation, this is not the reason for throttling the DH cooler outlets.
"b" is correct, SW flow to the DH coolers are throttle because flow to ES components may not be adequate (in particular RB Coolers are EDG coolers). The valves are marked so that they will not be throttled below the minimum required for the DH coolers.
"c" is incorrect because throttling the valves will not reduce the service water to RCS differential temperature.
"d" is incorrect because water hammer of service water piping of the DH coolers is not a concern.

References:

STM1-05, Rev 13
1309.013, Chg. 012-01-0

History:

Used in 1999 exam.
Modified from ExamBank, QID# 1519.
Selected for 2005 sensitive RO re-exam.

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

QID: 0078 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: ECCS.

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

Tier:	2	**RO Imp:**	4.2	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	K

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

If an ESAS occurs simultaneously with a Loss of Offsite Power, the start of RB Spray pumps is delayed by 35 sec. Why?

A. To allow the EDGs to come up to speed.

B. To allow SW pumps to start for spray pump cooling.

C. To prevent overload of the EDGs.

D. To prevent water hammer of the spray headers.

Answer:

C. To prevent overload of the EDGs.

Notes:

With an ES signal present, ES loads will sequence on to the EDG to prevent overload, therefore "C" is correct. (a), (b) and (d) are reasons for other aspects of RB spray operation but are not applicable to the basis for the time delay.

References:

1107.002, Chg. 020-00-0

History:

Developed for 1998 RO/SRO Exam.
Used in sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0631 **Rev:** 0 **Rev Date:** 11/7/05 **Source:** New **Originator:** J.Cork

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

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

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

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

K/A Number: A3.02 **CFR Reference:** 41.5 / 45.5

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

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

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

Given:

- RCS pressure is 1800 psig and dropping,
- RCS temperature is 545 degrees F and dropping,
- "A" OTSG pressure is 650 psig and dropping,
- "B" OTSG pressure is 970 psig and dropping slightly,
- Reactor Building Pressure is 2 psig and steady.

What automatic actuation (not currently actuated) will occur shortly to stop this transient?

A. RB Spray actuation

B. MSIV closure

C. EFW actuation

D. PZR heater actuation

Answer:

B. MSIV closure

Notes:

"B" is correct, MSLI is imminent at 600 psig.
"A" is incorrect, although RCS pressure and temperature are dropping, RB pressure is steady.
"C" is incorrect, although EFW actuation is imminent, it will not stop this transient.
"D" is incorrect, although RCS pressure is dropping, heaters should be full on at this point and will not be sufficient to maintain pressure.

References:

1202.003, Chg. 005-01-0
1202.012, Chg. 004-03-0

History:

New for 2005 sensitive RO re-exam.

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

QID: 0445 **Rev:** 0 **Rev Date:** 5/1/2002 **Source:** Direct **Originator:** J.Cork

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

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

System Number: 039 **System Title:** Main Steam and Reheat Steam System

Description: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

K/A Number: 2.1.20 **CFR Reference:** 45.2

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

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

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

Which button should be depressed on the MSR (Moisture Seperator Reheater) Control Panel on C11 to close the 2" Main Steam valves to the MSR's ?

A. Reset

B. Manual valve position

C. Hot Start

D. Ramp

Answer:

A. Reset

Notes:

"A" is correct, this is the button depressed prior to startup of the MSRs.
"B", "C", and "D" are other buttons on C11 but none of these will close the 2" Main Steam valves.

References:

1106.009, Chg. 031-17-0

History:

Direct from regular exambank QID 2273.
Selected for use in 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0537 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Repeat **Originator:** NRC

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

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

System Number: 059 **System Title:** Main Feedwater System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves.

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

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

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

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

ANO Unit 1 has just completed a refueling outage and has commenced plant start-up in accordance with 1102.002 and 1102.004.

The second Main feed pump should be placed in service:

A. Prior to exceeding 50% power using the Gamma Metrics Linear Power instrument.

B. Prior to exceeding 36% setting (~360 Mwe) on the Unit Load Demand (ULD) HI-Load Limit.

C. Prior to reaching 50% open on MFW pump Low Load Control Valve demand.

D. Prior to exceeding 45% setting (~450 Mwe) on the Unit Load Demand (ULD) HI-Load Limit.

Answer:

B. Prior to exceeding 36% setting (~360 Mwe) on the Unit Load Demand (ULD) HI-Load Limit.

Notes:

"B", Prior to exceeding 36% setting (~360 Mwe) on the Unit Load Demand (ULD) HI-Load Limit, is the correct answer. "A" is incorrect because the actual procedure statement is "Verify MFW flow is >0.90 x 106 lbm/hr prior to Gamma Metrics Linear Power raising above 45% power. "C" is incorrect because section 7.9 states that "WHEN ~350 Mwe is reached OR prior to reaching 90% open on Low Load Control Valve demand, THEN perform the following: 7.9.1 Place second MFWP (P-1A or P-1B) in service. "D" is incorrect because the value is too high at 45% power.

References:

1102.004, Chg. 042-00-0

History:

Developed by NRC.
Used on 2004 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0107 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-ICS **Objective:** 22 **Point Value:** 1

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

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

Description: Ability to monitor automatic operation of the MFW, including: ICS.

K/A Number: A3.07 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	Ap

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

The plant is operating at 60% power with Delta Tc and SG/RX Master stations in Hand.
All other ICS stations are in Auto.

If one RCP has to be tripped due to high vibration, how will the ICS respond?
(Assume no operator action other than tripping the RCP.)

A. The ICS will runback the plant to 45% load at 50%/min.

B. No change to FW will occur since the SG/RX Master is in Hand.

C. Demand is less than the RCP runback limit, no changes occur to FW.

D. The RC flow difference will re-ratio the FW flow demand.

Answer:

D. The RC flow difference will re-ratio the FW flow demand.

Notes:

Following an RCP trip Delta Tc will re-ratio feedwater demands, therefore answer (d) is correct. Answer (a) is incorrect since the plant is operating below the runback setpoint, while (b) and (c) are incorrect because they state that ICS will not re-ratio feedwater demands.

References:

1203.012F, Chg. 026-06-0
STM 1-64, Rev.9

History:

Modified QID 4408 for use on 1998 RO/SRO Exam.
Modified for use in 2001 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0435 **Rev:** 1 **Rev Date:** 11/7/05 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-EOP04 **Objective:** 8 **Point Value:** 1

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the AFW: Relationship between AFW flow and RCS heat transfer.

K/A Number: K5.01 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	An

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

Given:

- Reactor tripped due to loss of all offsite power.
- RCS T cold is 545°F and dropping.
- RCS pressure 1800 psig and dropping.
- OTSG pressures are ~930 psig and dropping.
- "A" OTSG level is 210" and rising.
- "B" OTSG level is 195" and rising.
- "A" EFW flow is 370 gpm.
- "B" EFW flow is 350 gpm.

Which of the following is an appropriate response to the above conditions in accordance with RT-5, Verify Proper EFW Actuation and Control?

A. Maintain 280 gpm to each SG in HAND.

B. Throttle EFW to prevent overcooling.

C. Select Reflux Boiling setpoint.

D. Actuate MSLI on both OTSGs.

Answer:

B. Throttle EFW to prevent overcooling.

Notes:

"B" is correct since overcooling is evident by RCS pressure and T cold dropping. SCM is adequate and both SGs have >340 gpm flow so EFW should be throttled per RT-5.
"A" is incorrect, this is the "old" minimum EFW flow value.
"C" is incorrect, although no RCPs are running, SCM is adequate, therefore the REFLUX BOILING setpoint would be inapproriate.
"D" is incorrect, although overcooling is evident, MSLI is too drastic a measure as SGs are much greater than 600 psig, EFW should be throttled first.

References:

1202.012, Chg. 004-03-0

History:

Created for 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0270 **Rev:** 1 **Rev Date:** 11/8/05 **Source:** Modified **Originator:** D. Slusher

TUOI: A1LP-RO-EFIC **Objective:** 29 **Point Value:** 1

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

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

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

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

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 2.5

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

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

The EFIC automatic fill rate is designed to prevent overcooling.
Given a SG pressure of 885 psig, determine the proper OTSG fill rate by EFIC for the EFW system:

a. ~3"/min

b. ~4"/min

c. ~5"/min

d. ~6"/min

Answer:

b. ~4"/min

Notes:

OTSG fill rate is adjusted so that OTSG levels raise at 2 inches/minute at OTSG pressure of 800 psig and 8 inches/minute at OTSG pressure of 1050 psig. This limits the overcooling effects of feeding OTSGs with EFW. At 885 psig OTSG fill rate will be 4 inches/minute. "b" is the correct answer.

References:

1105.005, Chg. 028-01-0

History:

Used in 1999 exam.
Direct from ExamBank, QID# 92 used in class exam
Modified for 2005 sensitive RO re-exam.

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

QID: 0632 **Rev:** 0 **Rev Date:** 11/7/05 **Source:** Modified **Originator:** J.Cork

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

Section: 3.6 **Type:** 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: Effect on plant of de-energizing a bus.

K/A Number: A2.04 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

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

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

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

Given:
- A loss of offsite power occurred.

If #2 EDG failed to start, which of the following actions are procedurally required to restore power to buses A4 and B6 following restoration of offsite power?

A. Open Test Switch to defeat bus A4 undervoltage.

B. Open Test Switch to defeat bus B6 undervoltage.

C. Take A409 handswitch to Pull-to-Lock and release.

D. Hold A409 handswitch in closed position until bus energized.

Answer:

D. Hold A409 handswitch in closed position until bus energized.

Notes:

"D" is correct, this will maintain breaker closed until undervoltage relays are no longer tripped.
"A" and "B" are incorrect, this may work but is not proceduralized.
"C" incorrect, this is the method to re-energize A1 and A2 buses.

References:

1202.008, Chg. 007-01-0

History:

Modified regular exambank QID#5147 for 2005 sensitive RO re-exam.

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

QID: 0636 **Rev:** 0 **Rev Date:** 11/8/05 **Source:** Direct **Originator:** J.Cork

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

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** AC Electrical Dist

Description: - Knowledge of limiting conditions for operations and safety limits.

K/A Number: 2.2.22 **CFR Reference:** 43.2 / 45.2

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 4

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

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

Given:

- 100% power

- A fire has caused a lockout trip on the switchyard auto transformer

- All in plant equipment operable.

What actions should be taken according to Technical Specifications?

A. Place the plant in Mode 3 within 84 hours if the auto transformer cannot be returned to service.

B. Place the plant in Mode 3 within 72 hours if the auto transformer cannot be returned to service.

C. Place the plant in Mode 3 within 36 hours if the auto transformer cannot be returned to service.

D. Place the plant in Mode 3 within 12 hours if the auto transformer cannot be returned to service.

Answer:

A. Place the plant in Mode 3 within 84 hours if the auto transformer cannot be returned to service.

Notes:

"A" is correct, per 3.8.1 action A.3 72 hours are allowed to return the offsite source to operable status with an additional 12 hours (F.1) to be in Mode 3 if it can't be restored for a total of 84 hours.
"B" is incorrect, if candidate only considers 3.8.1 A.3 and not F.1.
"C" is incorrect, if candidate choose action F.2 instead of F.1.
"D" is incorrect, if candidate only considers F.1 and not A.3.

References:

T.S. 3.8.1 (this ref must be in candidate's handout!!)

History:

Direct from regular exambank QID#ANO-OPS-3057 for 2005 sensitive RO re-exam.

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

QID: 0317 **Rev:** 0 **Rev Date:** 9/5/99 **Source:** Direct **Originator:** J Cork

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

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** D.C. Electrical Distribution

Description: Knowledge of bus power supplies to the following: Major DC loads.

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

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

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

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

Unit One is at 100% power and experiences a loss of 125V DC Bus D02.

Which of the following D02 loads will cause the reactor to trip?

a. MCC D25

b. Panel RA2

c. Inverter Y22

d. Inverter Y28

Answer:

b. Panel RA2

Notes:

All of the loads listed are from D02, however, RA2 supplies power to the RCP B&D pump monitors. A loss of two RCP's at 100% power results in an immediate Rx trip.
The other answers might cause minor transients and bring up T.S. issues but no trip will result.

References:

1107.004, Chg. 013-00-0

History:

Developed for 1999 exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0529 **Rev:** 1 **Rev Date:** 11/7/05 **Source:** Repeat **Originator:** NRC

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

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** DC Electrical Distribution System

Description: Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses.

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

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 4

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

Question: **RO:** 50 **SRO:**

Given:

- An electrical fault has made the normal supply to D11 inoperable.
- D11 should be transferred to it's emergency supply.

Which of the following is a concern in this infrequently performed procedure?

A. The discharge rate of the battery cannot be monitored in this configuration.

B. The static guard circuit is disabled and could cause inadvertent grounds on the bus.

C. A single fault may disable both trains of safety equipment such as EDG's and ES pumps.

D. It will prevent EDG #2 from starting, should it be required in an emergency situation.

Answer:

C. A single fault may disable both trains of safety equipment such as EDG's and ES pumps.

Notes:

"C" is correct. All others are incorrect distracters for this procedure.

References:

1107.004, Chg. 013-00-0

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0204 **Rev:** 2 **Rev Date:** 11/7/05 **Source:** Direct **Originator:** R. Walters

TUOI: A1LP-RO-EDG **Objective:** 16.3 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators

Description: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks.

K/A Number: K6.08 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 51 **SRO:**

The #2 EDG monthly surveillance is in progress with the EDG on-line and fully loaded.

- EDG 2 Non-critical Trouble (K01-D4) is in alarm.
- EDG 2 Critical Trouble (K01-C4) comes into alarm.
- The Outside AO reports that EDG 2 Fuel Oil Transfer Pump P-16B is tripped and will NOT start.
- The Inside AO reports that the T-30B day tank level is ~180 gallons and going down slowly.
- T-57B level is >168 inches.

Which of the following is correct with regard to #2 EDG status?

A. EDG 2 is operable because fuel oil can be supplied from EDG 1 transfer pump P-16A.

B. EDG 2 is inoperable because the level in T-30B is <190 gallons.

C. EDG 2 is operable because Emergency Fuel Storage Tank T-57B is full.

D. EDG 2 is inoperable because EDG 2 Fuel Oil Transfer Pump P-16B is inoperable.

Answer:

D. EDG 2 is inoperable because EDG 2 Fuel Oil Transfer Pump P-16B is inoperable.

Notes:

(a.) is incorrect. The system is can be configured as stated, however, the diesel is considered inoperable.
(b.) is incorrect. Tech Specs requires a separate day tank containing a minimum of 160 gallons.
(c.) is incorrect. Even though the storage tank is above the limit a fuel transfer pump is required to pump the fuel oil to the day tank.
(d.) is correct. The #2 EDG must have an operable transfer pump to be considered operable.

References:

Tech Spec 3.8.1

History:

Developed for use in 1998 RO Re-exam
Modified for use in 1999 exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0271 **Rev:** 1 **Rev Date:** 11/7/05 **Source:** Modified **Originator:** D. Slusher

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

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring System (PRM)

Description: Knowledge of the effect of a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases.

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

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2.5

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

Question: RO: 52 SRO:

Which of the following must be performed to release TWMT T-16A contents with the Liquid Radwaste Process Monitor (RI-4642) inoperable?

A. Chemistry personnel must have indepent sample and anaylsis results as well as indepently verified computer input data.

B. A Waste Control Operator must independently verify release path alignment prior to release.

C. The release flow rate must be estimated at least once every four hours during the release.

D. Discharge Flume process monitor RI-3618 must be checked for operability.

Answer:

A. Chemistry personnel must have indepent sample and anaylsis results as well as indepently verified computer input data.

Notes:

The requirements for release when the Liquid Radwaste Process Monitor is inoperable are
a. An independent sample and analysis of the tank contents
b. Computer input data independently verified.
"A" is therfore the correct answer.
"B" is incorrect, this was once a requrement but is no longer.
"C" is incorrect, this is done if the flow recorder is inoperable.
"D" is incorrect, although this detector also monitors the flume, it is not required.

References:

1104.020, Chg. 043-02-0

History:

Used in 1999 exam.
Direct from ExamBank, QID# 2765
Used in 2001 RO/SRO Exam.
Modified for 2005 sensitive RO re-exam.

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

QID: 0534 **Rev:** 1 **Rev Date:** 11/7/05 **Source:** Modified **Originator:** NRC

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

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

System Number: 076 **System Title:** Service Water System

Description: Knowledge of bus power supplies to the following: ESF-actuated MOVs.

K/A Number: K2.08 **CFR Reference:** 41.7

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

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

Question: **RO:** 53 **SRO:**

Following a reactor trip, ES channels 1 through 4 have actuated.

While performing RT-10 to verify proper ES actuation, the CBOT notes the following at C16:

- P4A to P4B Crosstie (CV-3644) has no position indication.
- P4B to P4C Crosstie (CV-3642) has no position indication.

The CBOT suspects that this is due to a loss of power to MCC load center _____ on the _____ train.

A. B62 / Green

B. B52 / Green

C. B62 / Red

D. B52 / Red

Answer:

A. B62 / Green

Notes:

"A", B62 on Green train is correct. Others are incorrect combinations and choices.

References:

STM 1-42, Rev. 10

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.
Modified for 2005 sensitive RO re-exam.

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

QID: 0535 **Rev:** 1 **Rev Date:** 10/13/200 **Source:** Modified **Originator:** J.Cork

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

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

System Number: 078 **System Title:** Instrument Air System

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

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

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

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

Question: **RO:** 54 **SRO:**

Instrument Air pressure has dropped to 48 psig.

Which of the following manual or automatic actions should be performed or will occur in response to the low Instrument Air pressure?

Note: All actions for higher pressures have been completed at the required pressure and answer the question considering only the action for the current pressure.

A. Service Air to Instrument Air cross-connect automatically opens.

B. Open Unit 1 to Unit 2 Instrument Air cross-connect.

C. Trip Reactor, actuate EFW and MSLI on both SGs.

D. Close Letdown Cooler outlet to isolate Letdown.

Answer:

A. *Service Air to Instrument Air cross-connect automatically opens.*

Notes:

"B" is incorrect, this was done when pressure dropped to 75 psig.
"A" is correct, this automatically occurs when pressure drops to 50 psig.
"C" is incorrect, this would not be done until pressure was less than 35 psig.
"D" is incorrect, this would not be done until pressure was less than 35 psig.

References:

1104.025, Chg. 012-05-0

History:

Developed for 1998 RO exam (similar to QID 102)
Modified question for sensitive 98 RO Re-exam
Modified for sensitive 2005 re-exam.

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

QID: 0139 **Rev:** 1 **Rev Date:** 04/07/94 **Source:** Direct **Originator:** K. Canitz

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

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment System

Description: Knowledge of physical connections and/or cause-effect relationships between the containment system and the following systems: Containment isolation/containment integrity.

K/A Number: K1.02 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

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

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

Question: **RO:** 55 **SRO:**

Which of the following systems are isolated to the Reactor Building by ESAS?

A. RCP motor cooling, chill water, RB service water cooling.

B. Seal injection, CRD cooling, RB leak detectors.

C. Seal injection, RCP motor cooling, RB service water cooling.

D. CRD cooling, chill water, RB leak detector.

Answer:

D. CRD cooling, chill water, RB leak detector.

Notes:

[d] is the only answer in which all items are isolated by ESAS.
[a] & [c], Service Water is not isolated by ESAS, the other two are.
[b] seal injection is not isolated by ESAS, the other two are.

References:

STM 1-09, Rev. 6
STM 1-43, Rev. 8
STM 1-62, Rev. 9

History:

Taken from Exam Bank QID # 3608
Used in sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0145 **Rev:** 1 **Rev Date:** 11/16/94 **Source:** Direct **Originator:** G. Alden

TUOI: A1LP-RO-CRD **Objective:** 16 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive System

Description: Knowledge of CRDS design feature(s) and/or interlocks which provide for the following: Operation of latching controls for groups and individual rods.

K/A Number: K4.15 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.0	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 56 **SRO:**

The purpose of the IN-LIMIT (LATCH) BYPASS switch on the Diamond panel is to:

a. Apply power to the CRD motor which will engage the latching mechanism.

b. Reset a fault condition provided the fault has cleared.

c. Reset the AC breakers, DC breakers, and programmer controls.

d. Allow driving in Groups 1 thru 7 to engage roller nuts with lead screws.

Answer:

d. Allow driving in Groups 1 thru 7 to engage roller nuts with lead screws.

Notes:

(a) is incorrect. The CRD motor has power applied at all times.
(b) is incorrect. This is accomplished with the fault reset pushbutton.
(c) is incorrect. This is accomplished with the Trip/reset pushbutton.
(d) is correct. The Latch pushbutton must be depressed to allow the in-limits to be bypassed when engaging lead screws on groups 1-7 after they have been deenergized.

References:

STM 1-02, Rev. 7

History:

Taken from Exam Bank QID # 2684
Used in sensitive 98 RO Re-exam
Used in 2001 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0193 **Rev:** 1 **Rev Date:** 11/4/05 **Source:** Direct **Originator:** R. Fuller

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:**	An

Question: **RO:** 57 **SRO:**

Several plant parameters can be monitored to ensure accurate indications of reactor power are available.

Which of the following sets of parameters would be indicative of 60% reactor power?

A. Tave 579 degrees, Thot 593 degrees, Tcold 564 degrees, total FW flow 6.5 million lbm/hr.

B. Tave 580 degrees, Thot 599 degrees, Tcold 560 degrees, total FW flow 8.4 million lbm/hr.

C. Tave 579 degrees, Thot 588 degrees, Tcold 570 degrees, total FW flow 4.5 million lbm/hr.

d. Tave 581 degrees, Thot 590 degrees, Tcold 565 degrees, total FW flow 9.8 million lbm/hr.

Answer:

a. Tave 579 degrees, Thot 593 degrees, Tcold 564 degrees, total FW flow 6.5 million lbm/hr.

Notes:

(a.) is correct.
(b.) is incorrect. Parameters are indicative of >70% power.
(c.) is incorrect. Delta T is indicative of ~40% power.
(d.) is incorrect. Delta T is indicative of ~56% power, however, FW flow is indicative of ~90% power.

References:

1102.004, Chg. 042-00-0

History:

Developed for use in sensitive 98 RO Re-exam
Used in 2001 RO/SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0220 **Rev:** 0 **Rev Date:** 11/12/98 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-NNI **Objective:** 19 **Point Value:** 1

Section: 3,2 **Type:** Reactor Coolant System Inventory Control

System Number: 011 **System Title:** Pressurizer Level Control System (PZR LCS)

Description: Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: CVCS.

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

Tier: 2 **RO Imp:** 3.2* **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** An

Question: **RO:** 58 **SRO:**

Given:

- Reactor power steady at 80%.
- PZR Level Control selected to LT-1001 on C04
- A break in LT-1001's reference leg causes it to drain completely.

How will Makeup Tank (T-4) level be affected by this failure?

A. Makeup flow drops, Makeup Tank level will rise.

B. No effect, SASS will auto select LT-1002.

C. Makeup flow rises, Makeup Tank level will drop.

D. Letdown flow drops, Makeup tank level also drops.

Answer:

A. Makeup flow drops, Makeup Tank level will rise.

Notes:

With LT-1001's reference leg empty, the delta pressure transmitter will sense a minimum DP and max level. SASS has no effect on PZR level instruments, (b is thus incorrect) therefore the makeup valve, CV-1235, will close due to the indicated high level. Letdown flow will remain constant (making distracter "d" incorrect) and therefore Makeup Tank level will increase as in answer [a].
If the trainee determines indicated PZR level will decrease, then he will probably choose [c].

References:

1203.015, Chg. 011-01-0
1105.006, Chg. 009-03-0

History:

Developed for sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0431 **Rev:** 2 **Rev Date:** 8/21/2002 **Source:** Modified **Originator:** S.Pullin

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

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier:	2	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 59 **SRO:**

Given:

- 100% power
- NI channel 5 failed full upscale (to 125% power)
- SASS failed to automatically transfer

Which operator actions are required per 1203.021, Loss of Neutron Flux Indication?

A. Place SG/RX Master and Feedwater Loop Demands stations in manual and stabilize reactor power less than 100% .

B. Place Diamond and Reactor Demand stations in manual and stabilize reactor power less than 100% .

C. Place Unit Load Demand and Turbine in manual and stabilize reactor power less than 100% .

D. Place Unit Load Demand and SG/RX Master stations in manual and stabilize reactor power less than 100%.

Answer:

B. Place Diamond and Reactor Demand stations in manual and stabilize reactor power less than 100%

Notes:

(b.) is the correct action per 1203.021, Loss of Neutron Flux Indication. These actions will stop the transient since both the Diamond (rod control station) and Reactor Demand (ICS station) are taken to manual.

a. c. & d. are incorrect since these actions do not stop the transient since none of these take both the Diamond and Reactor Demand to manual.

References:

1203.021, Chg. 008-01-0

History:

Direct from exambank QID 1793.
Selected for use in 2002 RO/SRO exam.
This question has been modified after NRC review to match K&A (Rev. 1)
Changed NI channel 7 to channnel 5 following exam administration. Jcork 8/21/02
Selected for 2005 sensitive RO re-exam.

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

QID: 0403 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Direct **Originator:** D.Slusher

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

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-nuclear Instrumentation

Description: Ability to manually operate and/or monitor in the control room: NNI channel select controls.

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

Tier:	2	**RO Imp:**	2.9*	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	2.8*	**SRO Select:**	No	**Taxonomy:**	An

Question: **RO:** 60 **SRO:**

Initial Conditions:

- SASS Mismatch Annunciator
- Rx is Feedwater Limited Annunciator
- Feedwater is Rx Limited Annunciator
- Unit Tave TI-1032 is 583 degrees F
- Loop A Tave TI-1020 is 588 degrees F
- Loop B Tave TI-1043 is 578 degrees F

Which of the following actions would clear the cross limits and return temperature indications to normal?

a. Select the NNI-Y signal for RCS loop A hot leg temperature.

b. Place the Controlling Tave Selector Switch in the Loop B position.

c. Place Loop A Feedwater Demand in hand and raise Loop A feedwater flow.

d. Select the NNI-Y signal for RCS loop B cold leg temperature.

Answer:

a. Select the NNI-Y signal for RCS loop A hot leg temperature.

Notes:

Answer [a] is correct, because cross limits indicate a problem with a false high temperature feeding into the unit Tave.
Answer [b] is incorrect, due to instrument failure, the controlling Tave would not be selected until ICS components were placed in Manual.
Answer [c] is incorrect, ICS would be trying to raise A FW flow due to the cross limits.
Answer [d] is incorrect, "B" T cold is not the problem, the loop with the higher temperature has the failed instrument.

References:

STM 1-64, Rev. 9

History:

Direct from exambank 4574, used in 2001 SRO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0375 **Rev:** 0 **Rev Date:** 11/14/00 **Source:** Direct **Originator:** J.Cork

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

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-Core Temperature Monitor System

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

K/A Number: K1.02 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 61 **SRO:**

With the plant operating at 100%, which condition would cause an "ICC Event Train A" or "B" alarm?

a. Core exit thermocouple (one) fails high.

b. Wide range RCS pressure instrument fails high.

c. Hot leg level transmitter fails high.

d. CET subcooling margin < 25 degrees F.

Answer:

d. CET subcooling margin < 25 degrees F.

Notes:

Answer [d] is correct, if CET subcooling margin was less than the minimum, then an ICC alarm would be generated.
Answer [a] is incorrect, while this seemingly would generate the alarm, the alarm uses an average CET vs. any CET.
Answer [c] is incorrect, while the hot leg level transmitters have an input to this alarm, a level transmitter failing high would be indicative of no voids and thus the ICC alarm would not be in alarm.
Answer [b] is incorrect, while wide range RCS pressure is compared to avg. CET temp to calculate SCM, a pressure transmitter failing high would provide more SCM, not less.

References:

1203.012J, Chg. 035-02-0

History:

Modified regular exambank QID #3685 for use in 2001 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0200 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **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: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Spent Fuel Pool Cooling System controls including: Spent fuel pool water level.

K/A Number: A1.01 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 4

Group: 2 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 62 **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 is closed.

Which of the following statements is correct concerning the Spent Fuel Pool inventory?

a. The SFP will drain to ~ 2 feet above the spent fuel assemblies.

b. Emergency makeup from service water will be needed to prevent the SFP level from reaching 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 pipe.

Answer:

c. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

Notes:

(a.) & (b.) are incorrect. 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.
(c.) is correct. The discharge pipe has the siphon break holes located at normal pool level.
(d.) is incorrect. 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.

References:

STM 1-07, Rev. 4

History:

Developed for use in sensitive 98 RO Re-exam
Selected for use in RO/SRO exam. 2002
Selected for 2005 sensitive RO re-exam.

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

QID: 0160 **Rev:** 1 **Rev Date:** 08/05/94 **Source:** Direct **Originator:** E. Wentz

TUOI: A1LP-RO-EOP09 **Objective:** 1 **Point Value:** 1

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

System Number: 041 **System Title:** Steam Dump System (SDS) and Turbine Bypass Control

Description: Knowledge of bus power supplies to the following: ICS, normal and alternate power supply.

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

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 63 **SRO:**

During a Rx trip transient, all nine of the + or - 24v DC NNI/ICS power supply status lights go out.

What would cause this condition?

a. Loss of Offsite Power

b. Loss of DC Bus D11

c. Loss of Y02

d. Loss of DC Bus D21

Answer:

b. Loss of DC Bus D11

Notes:

The first step of EOP 1202.009 checks these indicating lights and a note gives the power supply as breaker 25 on D-11.
Therefore, [b] is the correct answer as long as no other abnormal indications are present.
[a] is incorrect, D11 is battery backed.
[c] is incorrect, although Y02 supplies power to ICS and NNI, neither of these power the indicating lights.
[d] is incorrect, the lights are only powered from D11, although the two are very similar in other functions.

References:

1203.047, Chg. 000-01-0

History:

Taken from Exam Bank QID # 3202
Used in sensitive 98 RO Re-exam
Used in 2001 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0536 **Rev:** 1 **Rev Date:** 11/8/05 **Source:** Modified **Originator:** NRC

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

Section: 2 **Type:** Generic

System Number: 056 **System Title:** Condensate System

Description: Ability to locate and operate components, including local controls.

K/A Number: 2.1.30 **CFR Reference:** 41.7/45.7

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

Group: 2 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 64 **SRO:**

Initial conditions:

- Unit 1 is starting up after a refueling outage
- Startup of the Condensate System is in progress

The procedure directs that the Gland Steam Desuperheater Bypass (CS-58-3) be closed.

Where must the AO be dispatched to verify the position of CS-58-3?

A. Inside HP turbine doghouse, east side.

B. West side of EH oil skid deck.

C. Northwest side of the condenser on the mezzanine.

D. North end of FW lube oil deck by Gland Steam condenser.

Answer:

A. Inside HP turbine doghouse, east side.

Notes:

"A" is correct answer, other answers are locations of other condenser connecting systems.

References:

1106.013, Chg. 014-07-0
1106.016, Chg. 042-00-0

History:

Developed by NRC.
Used on 2004 RO Exam.
Modified for 2005 sensitive RO re-exam.

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

QID: 0153 **Rev:** 2 **Rev Date:** 11/4/05 **Source:** Modified **Originator:** J. Cork

TUOI: ASLP-RO-CRVNT **Objective:** 6 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 072 **System Title:** Area Radiation Monitoring System

Description: Ability to monitor automatic operation of the ARM system, including: Changes in ventilation alignment.

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

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 65 **SRO:**

A high rad alarm on RE-8001, Control Room area radiation monitor, will cause all CR normal ventilation isolation dampers to close and:

A. Both emergency Recirc Fans (VSF-9 and 2VSF-9) start, normal supply fans (VSF-8A/B) stop.

B. Emergency Recirc Fan (VSF-9) starts, normal supply fans (VSF-8A/B) stop.

C. Emergency Recirc Fan (VSF-9) starts, all normal supply fans (VSF-8A&B, 2VSF-8A/B) stop, normal exhaust fans (2VEF-43A/B) stop.

D. Emergency Recirc Fan (2VSF-9) starts, all normal supply fans (VSF-8A&B, 2VSF-8A/B) stop, normal exhaust fans (2VEF-43A/B) stop.

Answer:

B. Emergency Recirc Fan (VSF-9) starts, normal supply fans (VSF-8A/B) stop.

Notes:

"B" is correct, RE-8001 and 2RITS-8001A/B stop Unit 1 supply fans and starts Unit 1 emerg. Recirc. Fan only.
"A" is incorrect, it would take an actuation of RE-8001 and 2RITS-8750-1A or 1B to cause this.
"C" is incorrect, RE-8001 does start VSF-9 but does not shut down Unit Two fans.
"D" is incorrect, these are the acuations that would occur for 2RITS-8750-1A or 1B.

References:

2104.007, Chg. 027-02-0

History:

Taken from Exam Bank QID # 3971 (stem modified slightly)
for use in sensitive 98 RO Re-exam.
Modified for 2005 sensitive RO re-exam.

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

QID: 0038 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** GGiles

TUOI: A1LP-RO-EOP06 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 038 **System Title:** Steam Generator Tube Rupture (SGTR)

Description: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

K/A Number: 2.1.7 **CFR Reference:** 43.5 / 45.12 / 45.13

Tier:	3	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 66 **SRO:**

The plant is shutdown and a cooldown is in progress due to a 200 gpm tube leak in the A OTSG.

- RCS temperature is 520 °F and lowering
- BWST level is at 35 ft and lowering
- A OTSG level is 290 inches and rising
- Dose rates at site boundary are normal

Which of the following RCS cooldown limits apply?

a. Less than or equal to 50 °F/hour.

b. Less than or equal to 100 °F/hour.

c. Less than or equal to 240 °F/hour.

d. Less than or equal to 520 °F/hour.

Answer:

b. Less than or equal to 100 °F/hour.

Notes:

(b) is correct in accordance with 1202.006, Tube Rupture. (a) is incorrect, it is the limit if RCS temp is between 300 and 170 °F. (c) is incorrect, it is the emergency cooldown limit which only applies if the affected steam generator level is at 410 inches or off site dose rates reach alert criteria, (d) is incorrect, it is the pressurizer cooldown limit.

References:

1202.006, Chg. 007-04-0

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam, but not used.
Selected for 2005 sensitive RO re-exam.

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

QID: 0016 **Rev:** 0 **Rev Date:** 6/30/98 **Source:** Direct **Originator:** GGiles

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to coordinate personnel activities outside the control room.

K/A Number: 2.1.8 **CFR Reference:** 45.5 / 45.12 / 45.13

Tier:	3	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 67 **SRO:**

To coordinate plant activities during a remote shutdown, what is the primary means of communications?

a. Dial 111 on the plant telephone system to access the party line.

b. Portable radios available in the alternate shutdown cabinet.

c. The plant Gaitronics phones located throughout the plant.

d. Cellular phones available in the control room extension.

Answer:

b. Portable radios available in the alternate shutdown cabinet.

Notes:

The remote shutdown procedure specifies portable hand-held radios as the primary means of communications, therefore (b) is the correct response. (a), (c) and (d) are other means of communications.

References:

1203.029, Chg. 006-03-0

History:

Developed for 1998 SRO Exam.
Selected for 2005 sensitive RO re-exam.

QID: 0143 **Rev:** 3 **Rev Date:** 11/3/05 **Source:** Direct **Originator:** D.Walls

TUOI: ASLP-AO-DUTYS **Objective:** 14 **Point Value:** 1

Section: 2 **Type:** Generic K&As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of how to conduct and verify valve lineups.

K/A Number: 2.1.29 **CFR Reference:** 41.10 / 45.1 / 45.12

Tier:	3	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	Ap

Question: **RO:** 68 **SRO:**

Given:

- The plant is shut down for Refueling.
- A Core Flood system valve alignment is in progress inside Controlled Access.
- The primary sample room has become a high radiation area due to hydrogen peroxide cleanup.
- The first check was made on CF-2, Core Flood Combined Sample Isolation, but the Shift Manager decided to waive the second check to reduce the exposure to high radiation.

Which one of the following statements most accurately describes why the Shift Manager's decision is acceptable or unacceptable?

A. Acceptable, independent verifications for manual valves can be waived for valve alignments inside High Radiation Areas.

B. Unacceptable, independent verification should not be waived if remote valve position indication is provided.

C. Acceptable, independent verification can be waived for at any time with the Shift Manager's approval.

D. Unacceptable, independent verifications cannot be waived for valve alignments without the approval of the Manager of Plant Operations.

Answer:

A. Acceptable, independent verifications for manual valves can be waived for valve alignments inside High Radiation Areas.

Notes:

"A" is correct, this meets the guidance of 1015.001.
"B" is untrue, independent verification of a manual valve in a high rad area should be waived, but only if no other means are available to verify it's position.
"C" is untrue, the Shift Manager can only waive verification in specific situations.
"D" is untrue, it lists the wrong approval authority.

References:

1015.001, Chg. 056-08-0

History:

Taken from Exam Bank QID # 3273
Used in sensitive 98 RO Re-exam
Modified for use in 2001 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0118 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of surveillance procedures.

K/A Number: 2.2.12 **CFR Reference:** 41.10 / 45.13

Tier:	3	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 69 **SRO:**

Which of the following describes a Channel Check as defined by Tech Specs?

a. A test of logic elements in a protection channel to verify their associated trip action.

b. Adjustment of a channel such that it responds accurately to known values.

c. Verification of acceptable channel performance by observation of other available indications.

d. Injection of a simulated signal into the channel to verify proper response.

Answer:

c. Verification of acceptable channel performance by observation of other available indications.

Notes:

"C" is the correct definition of channel check.
"A" defines Trip Test.
"B" is an Instrument Test Channel Calibration.
"D" is a Channel Test.

References:

Technical Specifications, section 1.1

History:

Used in 1998 RO exam
Used in 8/24/92 NRC developed ANO-1 RO exam, no. 75
Used in sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0362 **Rev:** 0 **Rev Date:** 11/29/99 **Source:** Repeat **Originator:** J. Cork

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

Section: 2 **Type:** Generic Knowledges and abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

K/A Number: 2.2.30 **CFR Reference:** 45.12

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

Group: G **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 70 **SRO:**

When is the Aux. Building Equipment Hatch on 404' NOT allowed to be opened?

A. Operators are moving irradiated fuel in the SF area.

B. The spent fuel Crane is being used to move a pump in the SF area.

C. Operators are moving new fuel in the new fuel storage pit.

D. Spent Fuel pool level is being raised.

Answer:

A. Operators are moving Irradiated Fuel in the SF area.

Notes:

"A" is correct because movement of irradiated fuel necessitates closure of equipment hatch in case of fuel handling accident.
"B" and "D" are incorrect, these do not pose a risk of a fuel handling accident.
"C" is incorrect since it involves new fuel and is in the new fuel pit vs. in or over the SFP.

References:

1502.010, Chg. 008-07-0

History:

Direct from ExamBank, QID# 3236
Used in 1999 exam.
Used on 2004 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0121 **Rev:** 1 **Rev Date:** 11/4/05 **Source:** Direct **Originator:** S. Pullin

TUOI: ASLP-RO-RADP **Objective:** 15 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of 10CFR20 and related facility radiation control requirements.

K/A Number: 2.3.1 **CFR Reference:** 41.12 / 43.4 / 45.9, 45.10

Tier:	3	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.0	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 71 **SRO:**

What is the federal occupational exposure limit to the SKIN (SDE, Shallow Dose Equivalent) in accordance with 10CFR20?

A. 5.0 rems/calendar year

B. 15.0 rems/calendar year

C. 50.0 rems/calendar year

D. 100.0 mrems/calendar year

Answer:

C. 50.0 rems/calendar year

Notes:

"C" is the correct answer.
"A", "B", and "D" are incorrect values for TEDE, LDE, and general public TEDE.

References:

EN-S-RP-201, Rev. 3

History:

New question developed for 2001 RO/SRO NRC Exam.
Selected for use in 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0235 **Rev:** 2 **Rev Date:** 11/4/05 **Source:** Direct **Originator:** B. Short

TUOI: ASLP-RO-RADP **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

K/A Number: 2.3.4 **CFR Reference:** 43.4 / 45.10

Tier: 3 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3

Group: G **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** Ap

Question: **RO:** 72 **SRO:**

A reactor building entry is required at power to investigate a potential leak on the RCP Seal Cooling Pump (P-114).
You have been assigned to perform this task.
The dose rate in area of P-114 is 1.8 rem/hr.
Your yearly exposure to date is 1.6 rem.

Which one of the following is the LONGEST stay time allowable WITHOUT exceeding Administrative Dose Control Limits (TEDE)?

A. 5 minutes

B. 10 minutes

C. 15 minutes

D. 20 minutes

Answer:

B. 10 minutes

Notes:

The Administrative Dose Control Level is 2.0 Rem TEDE for the calendar year. With a dose rate of 1.8 R/hr a stay time of 10 minutes would result in a dose of 0.3 Rem. The WCO's total accumulated dose would be 1.9 Rem, therefore "B" is correct.

References:

EN-S-RP-201, Rev. 3

History:

Developed for 1998 RO/SRO Exam QID 0122.
Modified for sensitive 98 RO Re-exam
Selected for 2005 sensitive RO re-exam.

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

QID: 0234 **Rev:** 0 **Rev Date:** 12/3/98 **Source:** Direct **Originator:** J. Cork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

K/A Number: 2.3.10 **CFR Reference:** 43.4 / 45.10 / 41.13

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

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

Question: **RO:** 73 **SRO:**

During a SGTR, which of the following actions is performed specifically to reduce plant personnel exposure?

A. Maintaining RCS pressure low within limits of Fig. 3.

B. Steaming bad SG to maintain tube-to-shell DT <150°F.

C. Aligning HPI to provide PZR Aux Spray.

D. Removing all but C & D condensate polishers from service.

Answer:

d. Removing all but C & D condensate polishers from service.

Notes:

(a.) is incorrect. This is performed to curtail primary system losses.
(b.) is incorrect. This is done to alleviate tube stresses and to prevent failing more tubes.
(c.) is incorrect. Aligning aux spray during SGTR is done for pressure control without RCPs and to reduce primary system losses.
(d.) is correct. This task is performed to clean up the secondary while using centrally located polishers to maintain doses ALARA to the train bay and polisher panel.

References:

1203.014, Chg. 012-05-0

History:

Developed for use on sensitive 98 RO Re-exam
Selected for use in 2002 RO/SRO exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0039 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** GGiles

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of annunciator response procedures.

K/A Number: 2.4.10 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 3 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 74 **SRO:**

ICS is in full automatic and the CBOR is verifying proper plant response to a trip of the "A" main feedwater pump from 100% power.

Which of the following should the CBOR expect to occur?

a. The operating main feedwater pump demand will have a bias of 30% added.

b. The pressurizer spray valve will come open when RCS pressure reaches 2205 psig.

c. The control rods will start inserting immediately due to a crosslimit from feedwater.

d. The main feedwater block valves will close in fast speed once power goes below 80%.

Answer:

d. The main feedwater block valves will close in fast speed once power goes below 80%.

Notes:

Answer (d) is correct since the main feedwater block valve closure is inhibited above 80%, once <80% they will close in fast speed. Answers (a) is incorrect, no bias is applied to demand although it will be maximized due to the high power level and the MFP trip.
Answer (b) is incorrect, a bias of 125 psig is subtracted from the normal setpoint of 2205, the spray valves should open at 2080 psig.
Answer (c) is incorrect, control rods will start to insert but not due to a crosslimit from feedwater, rather the reactor system will be crosslimiting feedwater. A bias subtraction from the reactor demand signal will cause the rods to insert.

References:

1203.012F, Chg. 026-06-0

History:

Developed for 1998 RO Exam.
Used in 2001 RO Exam.
Selected for 2005 sensitive RO re-exam.

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

QID: 0392 **Rev:** 0 **Rev Date:** 11/20/00 **Source:** Direct **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 / 45.13

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

Group: G **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 75 **SRO:**

Who is the final authority on evaluating the fire loading of a safety related area for transient and insitu combustibles?

a. Safety Coordinator

b. Licensing Safety Engineer

c. Fire Barrier Watch Supervisor

d. Fire Protection Engineer

Answer:

d. Fire Protection Engineer

Notes:

Answer [d] is correct per 1000.152, all others have assigned responsibilities with fire protection or safety but not this one.

References:

1000.152, Chg. 004-00-0

History:

New question created for 2001 RO Exam.
Selected for 2005 sensitive RO re-exam.