RO

Tier 1

Group 1

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

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

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

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

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

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

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

Group: 1 **SRO Imp:** 4.6 **SRO Select:** No **Taxonomy:** An

Question: RO: 1 SRO:

QUESTION #1 DELETED PER POST-EXAM COMMENTS

Given:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

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

References:

1202.012, Repetitive Tasks, RT-14 "Control RCS Press"
AREVA Technical Document, Vol. 2, V.B-9

History:

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

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

QID: 0324 **Rev:** 1 **Rev Date:** 8/8/05 **Source:** Bank **Originator:** J. Cork

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

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

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

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

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

Tier: 1 **RO Imp:** 4.1 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.6 **SRO Select:** No **Taxonomy:** An

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

Given:

Prior to Revision

- ESAS actuated on low RCS pressure.
- RCS Tave 560 °F and stable
- Pressurizer level off-scale high
- RCS pressure 1400 psig and rising rapidly
- RB sump level 55% and rising
- Fuel failure of 1% is indicated

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

A. Cycle ERV as required to quickly and effectively control the pressure rise.

B. Raise PZR spray to condense steam in PZR vapor space.

C. Throttle HPI flow to reduce input of mass into RCS and match RCS leakage.

D. Raise letdown flow to lower RCS mass and reduce pressure.

Answer:

A. Cycle ERV as required to quickly and effectively control the pressure rise.

Notes:

Answer "A" is correct since the conditions given are representative of a steam space leak and the RCS is in a "solid" condition.
Answer "B" is incorrect, PZR spray is not available since subcooling margin isn't present and there is not vapor space to spray into anyway with the RCS solid.
Answer "C" is incorrect, this is the TMI response to their 1979 accident, subcooling margin is not present, HPI cannot be throttled.
Answer "D" is incorrect, RT-14 does not allow for Letdown to be re-established with fuel failure indicated.

References:

1202.012, Chg. 004-03-0

History:

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

CONTROL RCS PRESS

NOTE

- PTS limits apply if any of the following has occurred:
  - HPI on with all RCPs off
  - RCS C/D rate > 100°F/hr with Tcold < 355°F
  - RCS C/D rate > 50°F/hr with Tcold < 300°F
- Once invoked, PTS limits apply until an evaluation is performed to allow normal press control.
- When PTS limits are invoked OR SGTR is in progress, PZR cooldown rate limits do not apply.
- PZR cooldown rate <100°F/hr.

1. **IF PTS limits apply or RCS leak exists,**
   THEN maintain RCS press low within limits of Figure 3.

2. **IF RCS press is controlled AND will be reduced below 1650 psig,**
   THEN bypass ESAS as RCS press drops below 1700 psig.

3. **IF PZR steam space leak exists,**
   THEN limit RCS press as PZR goes solid by one or more of the following:

   A. Throttle makeup flow.

   B. IF SCM is adequate, ← SCM is inadequate
      THEN throttle HPI flow by performing the following:

      1) Verify both HPI Recirc Blocks open:
         - CV-1300
         - CV-1301

      2) Throttle HPI.

   C. Raise Letdown flow.

      1) IF ESAS has actuated, ← Failed fuel or 1% indicated
         THEN unless fuel damage or RCS to ICW leak is suspected, restore Letdown per RT-13.

   D. Verify Electromatic Relief ERV Isolation open (CV-1000)
      AND cycle Electromatic Relief ERV (PSV-1000).

3.3 **IF the RCS will be taken solid, THEN limit RCS pressure increase by one or more of the following, as applicable:**

3.3.1 **Throttle MU/HPI (Rule 2.0).**

3.3.2 **Increase letdown flow.**

3.3.3 **Place pressurizer heaters in OFF.**

3.3.4 **Cycle the PORV or pressurizer vent as necessary.**

Indicators and Controls

Indicators:
- RCS pressure
- RCS temperature (incore thermocouple)
- MU/HPI flow
- Letdown flow
- PORV and pressurizer vent indication
- Pressurizer heaters status

Controls:
- MU/HPI valve controls
- Letdown flow controls
- PORV and pressurizer vent controls
- Pressurizer heater controls

Purpose of Step

The purpose of this step is to ensure RCS pressure control to prevent large pressure swings and possible lift of the PSVs.

Bases

Primary pressure control is more sensitive during solid plant operation to small changes in inventory and temperature. It is especially desirable to prevent challenges to the PSVs to preclude passing water.

Sequence

There is no specific sequence requirement.

TBD Volume 3 References

III.G.3.4 and V.2.0

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

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

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

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

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

Question: RO: 2 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

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

Revised stem at suggestion of NRC examiner. Rev.1

References:

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

History:

Revised version of QID 18

Selected for 2016 exam

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

QID: 0018 **Rev:** 0 **Rev Date:** 7/6/98 **Source:** Direct **Originator:** GGiles

TUOI: AA51003-007 **Objective:** 7.2 **Point Value:** 1

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

System Number: E03 **System Title:** Inadequate Subcooling Margin

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

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

Tier:	1	**RO Imp:**	3.8	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	C

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

Following a reactor trip the plant experiences a loss of subcooling margin.

Why is it desirable to secure all Reactor Coolant Pumps within 2 minutes following a loss of subcooling margin?

a. To allow the void coefficient of reactivity to add negative reactivity to the core.

b. To protect the mechanical seals on the Reactor Coolant Pumps.

c. To prevent exceeding a 70% void fraction and possible core uncovery.

d. To reduce operator burden by securing the RCP's prior to overheating of motors.

Prior to revision

Answer:

c. To prevent exceeding a 70% void fraction and possible core uncovery.

Notes:

A note above step 1 of 1202.002, Loss of Subcooling Margin, states: "Tripping all RCPs >2 minutes after a loss of adequate subcooling margin could cause Rx core to become uncovered". The B&W technical bases document discusses analyses which have showed that continued RCP operation could allow the RCS to evolve to a void fraction of 70% or greater if a certain range of break sizes were present. If RCPs were tripped when the void fraction was 70% or greater, core uncovery would occur.

References:

1202.002 (Rev 3, PC-3), Loss of Subcooling Margin,

History:

Developed for 1998 RO/SRO Exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
3. **Check adequate SCM.**	3. Check elapsed time since loss of adequate SCM **AND** perform the following: A. IF ≤ 2 minutes have elapsed, THEN trip all RCPs: • P32A • P32C • P32B • P32D B. IF > 2 minutes have elapsed, THEN leave currently running RCPs on. C. Advise Shift Manager to implement Emergency Action Level Classification (1903.010). D. Perform the following: 1) IF 4160V bus A1 or A2 is energized, THEN **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure. 2) IF only EDG power is supplying 4160V buses, THEN **GO TO 1202.007, "DEGRADED POWER"** procedure.
4. **Perform the following:** • **Advise Shift Manager to implement Emergency Action Level Classification (1903.010).** • **Direct Control Board Operators to monitor floating steps** 5. **Verify Orifice Bypass (CV-1223) demand adjusted to zero.** 6. **Open BWST T3 Outlet (CV-1407 or CV-1408) to operating HPI pump.** 7. **IF Emergency Boration is not in progress, THEN adjust Pressurizer Level Control setpoint to 100".**	

III.A DESCRIPTION OF CTs BASED ON: ADDING/MAINTAINING APPROPRIATE RCS WATER MASS

CT-1: TRIP ALL RCPs (Rule 1.0)[1]

Fulfillment of this CT requires the following:
Trip all RCPs

1.0 PLANT CONDITIONS

The GEOG prescribes performance of this CT anytime adequate subcooling margin (SCM) is lost. It is intended that tripping of all RCPs be accomplished immediately following a loss of SCM (and verification that the reactor is shutdown), and no later than [1or 2] minutes of loss of SCM, depending on plant-specific ECCS capability. If this is not accomplished then it is intended that the RCPs not be tripped, i.e., the RCPs should remain running if they cannot be tripped within [1 or 2] minutes of the loss of SCM.

2.0 ASSOCIATED GEOG BASES

SBLOCA analyses were performed using conservative Appendix K assumptions with the objective of meeting 10CFR50.46 criteria. These analyses predicted that continued RCP operation, during certain SBLOCAs, could lead to RCS void fractions of 70% if RCPs continued to operate longer than [1 or 2] minutes following initiation of the SBLOCA. The analyses predicted that if RCPs were tripped after these high void fractions occurred, the core would not be adequately covered and fuel clad failure would occur.

For more realistic assumptions (e.g., full flow from 2 HPI pumps, 1.0 times decay heat, etc.) the time period to reach these high RCS void fractions was > 10 minutes. However, the GEOG maintained the [1 or 2] minute time period for the following reasons:

[1] Rule 1.0 provides the following guidance relative to this CT:
- If RCPs not tripped within [1 or 2] minutes after a loss of SCM, then RCP operation (existing RCP combination) must be maintained until SCM is restored or until minimum LPI flow is established.
- If SCM is lost, immediately following RCP restart, then the RCPs do not need to be tripped immediately but must be tripped if SCM is not restored within 2 minutes.

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

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

Section: 4.1 **Type:** EPE

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

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

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

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

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

Question: RO: 3 SRO:

Given:

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

LPI/HPI flow rates are as follows:

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

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

BWST level is 8 feet.

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

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

B. Secure the "C" HPI pump only.

C. Override and secure all HPI pumps.

D. Swap to RB sump recirculation.

Answer:

C. Override and secure all HPI pumps.

Notes:

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

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

Revised question per NRC examiner comments following initial submittal.

References:

1202.010, ESAS

History:

New question for 2008 RO Exam.
Selected for 2016 exam

NOTE

Aligning Pressurizer AUX Spray to LPI system before going on sump recirc reduces personnel exposure should the lineup be required for boron precipitation mitigation at a later time. Transfer to RB Sump suction must commence when BWST level reaches 6', even if this alignment is not complete.

13. Dispatch an operator to perform Decay Heat Removal Operating Procedure (1104.004), "DH System Aux Spray Alignment Prior to RB Sump Recirc" section.

A. IF BWST level reaches 6' before alignment is complete,
THEN notify dispatched operator to exit the Aux Bldg, regardless of alignment status, until transfer to RB sump suction is complete and radiation levels can be determined.

14. Check LPI flow meets the following criteria:

2 LPI pumps	1 LPI pump
≥ 2800 gpm/pump	≥ 3050 gpm

14. **GO TO step 15.**

A. Override all HPI pumps:

- P36A
- P36B (C18)
- P36B (C16)
- P36C

B. Perform the following to secure HPI:

1) Start AUX Lube Oil pumps for running HPI pumps:

P36A	P36B	P36C
P64A	P64B	P64C

(14. CONTINUED ON NEXT PAGE)

14. (Continued).

2) Stop running HPI pumps:

- P36A
- P36B
- P36C

3) Override **AND** close all HPI Block valves.

P36A/B	P36B/C
• CV-1219	• CV-1227
• CV-1220	• CV-1228
• CV-1278	• CV-1284
• CV-1279	• CV-1285

4) Verify RCP Seal INJ Block (CV-1206) closed.

C. Dispatch an operator to isolate CFTs as follows:

1) Remove Danger Tag, unlock **AND** close Core Flood Tank Outlet supply breakers:

- B5661
- B5545

2) Close Core Flood Tank Outlet valves:

- CV-2415
- CV-2419

3) Open **AND** lock Core Flood Tank Outlet supply breakers:

- B5661
- B5545

(14. CONTINUED ON NEXT PAGE)

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

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

Section: 4.2 **Type:** Generic AOP

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

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

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

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

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

Question: RO: 4 SRO:

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

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

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

B. Seal injection control valve (CV-1207) is slowly opened to minimize thermal shock to the RCP seals and prevent damage to seals, independent of normal makeup restoration.

C. Seal injection control valve (CV-1207) is quickly opened to establish previous flow rate to minimize time without seal injection, independent of normal makeup restoration.

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

Answer:

B. Seal injection control valve (CV-1207) is slowly opened to minimize thermal shock to the RCP seals and prevent damage to seals, independent of normal makeup restoration.

Notes:

"B" 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 Pressurizer level.
"C" is incorrect. Reestablishing seal injection quickly in any condition has the potential for shocking the RCP seals.
"A" is incorrect. Restoring normal makeup and seal injection has no dependency on the BWST outlet valve position.

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

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

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

References:

1203.026, Loss of Reactor Coolant Makeup

History:

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

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

QID: 0183 **Rev:** 1 **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:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

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

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

A. If PZR level is <55", normal makeup is restored before seal injection to raise RCS inventory.

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, seal injection control valve (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 seal injection control valve (CV-1207) to prevent borating RCS.

Prior to Revision

Answer:

C. If RCP seal bleedoff temperatures are >180 degrees, seal injection control valve (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. 010-00-0

History:

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

SECTION 1 -- LOSS OF HPI PUMP (continued)

CAUTION

With RCP seal bleed off temperature >180°F rapid establishment of seal injection, <30 min, will result in seal damage.

I. **IF** RCP Seal Bleed off temperatures are ≤180°F,
THEN slowly open CV-1207 until RCP Seals Total INJ Flow is 30 to 40 gpm.

1) Place CV-1207 in AUTO.

J. **IF** RCP Seal Bleed off temperatures are >180°F,
THEN slowly open CV-1207 until RCP Seals Total INJ Flow is 8 to 12 gpm.

1) Record current time _______.

2) Maintain 8 to 12 gpm total flow for >30 minutes.

3) After 30 minutes, slowly open CV-1207 until 30 to 40 gpm total flow is reached.

a) Place CV-1207 in AUTO.

K. **WHEN** RCP seals total injection flow is above setpoint of ~22 gpm (CV-1206 Flow light on),
THEN return CV-1206 OVRD pushbutton to normal (OVRD light off).

L. Slowly open CV-1235 until makeup flow indication is on-scale.

M. Adjust CV-1235 setpoint to desired value.

N. Place CV-1235 in AUTO.

9. Restore letdown per Repetitive Tasks (1202.012), Restore Letdown (RT-13).

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

Tier: 1 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

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

Question: RO: 5 SRO:

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

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

A. 1900 gpm

B. 2700 gpm

C. 3500 gpm

D. 3700 gpm

Answer:

C. 3500 gpm

Notes:

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

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

References:

1203.012H, Annunciator K09 Corrective Action

History:

New question for 2016 exam

Location: C14

Device and Setpoint: see next page

DECAY HEAT
FLOW
HI/LO

Alarm: K09-A8

1.0 OPERATOR ACTIONS

NOTE
SPDS Safety System Diagnostic Instrumentation Display may be helpful in monitoring DH Pump (P-34A or P-34B).

1. IF flow is low,
   THEN attempt to raise flow >1000 gpm and within limits of Attachment B of Decay Heat Removal Operating Procedure (1104.004).

   A. IF loop "A",
      THEN use E-35A Cooler Bypass (CV-1433) or Decay Heat Cooler E-35A Outlet (CV-1428).

   B. IF loop "B",
      THEN use E-35B Cooler Bypass (CV-1432) or Decay Heat Cooler E-35B Outlet (CV-1429).

2. IF low flow persists,
   THEN GO TO Loss of Decay Heat Removal (1203.028).

3. IF flow is high,
   THEN lower flow to within limits of Attachment B of Decay Heat Removal Operating Procedure (1104.004).

   A. IF loop "A",
      THEN use E-35A Cooler Bypass (CV-1433) or Decay Heat Cooler E-35A Outlet (CV-1428).

   B. IF loop "B",
      THEN use E-35B Cooler Bypass (CV-1432) or Decay Heat Cooler E-35B Outlet (CV-1429).

4. IF high flow alarm is due to idle loop surveillance testing while drained below 375',
   THEN monitor the operating loop decay heat flow within the limits of Attachment B of Decay Heat Removal Operating Procedure (1104.004).

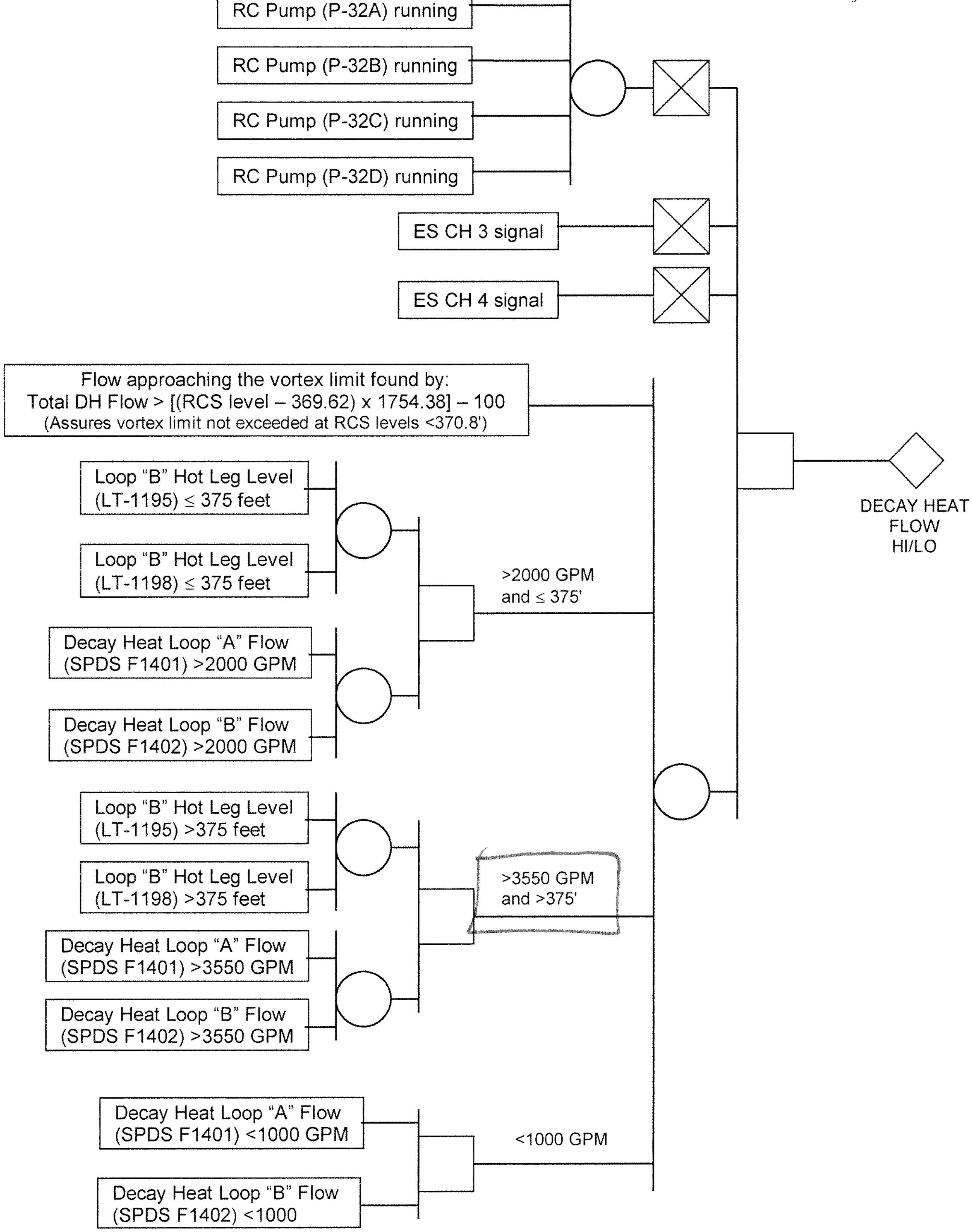
2.0 PROBABLE CAUSES

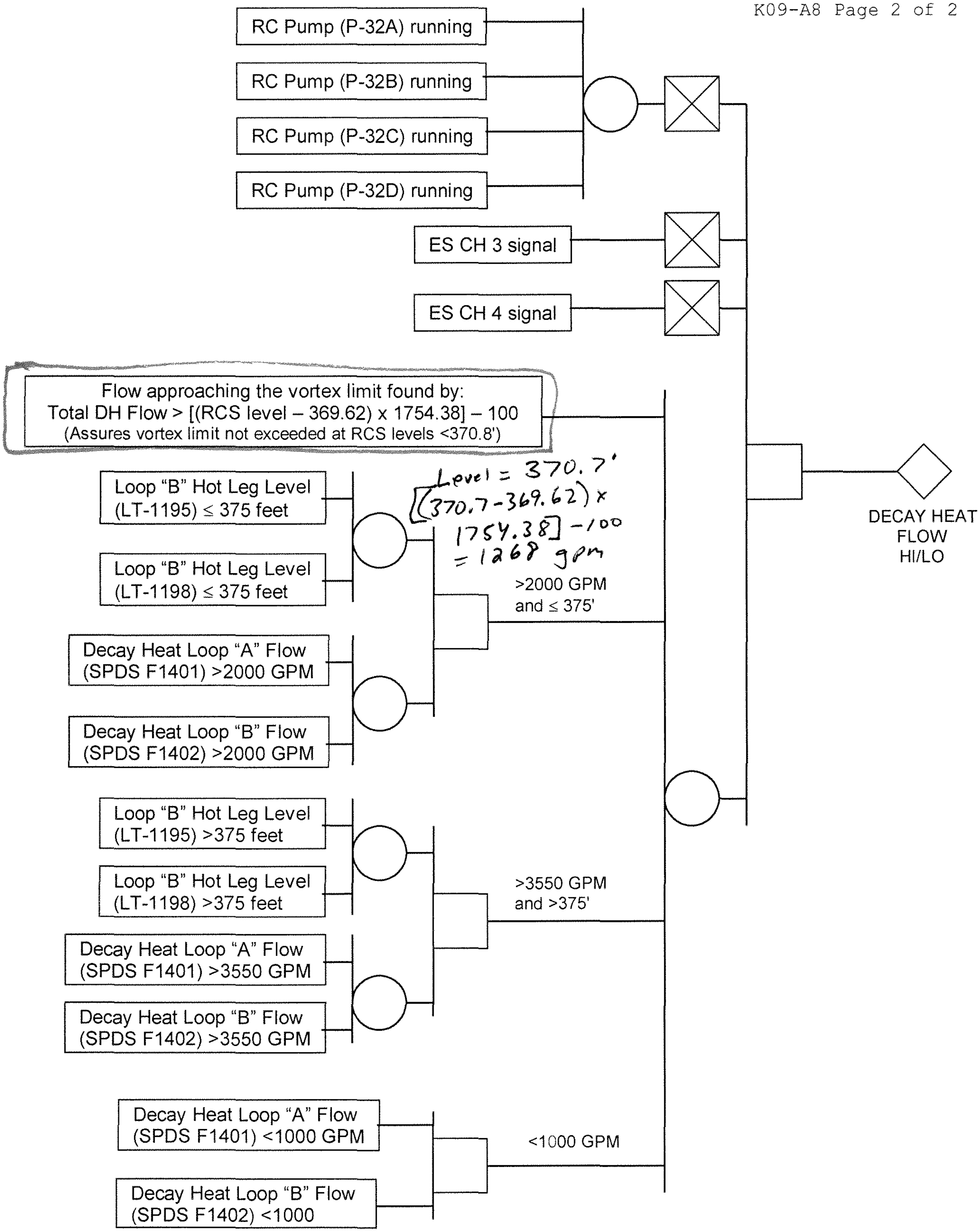
- For high flow high: surveillance testing in progress in idle loop
- Loop CVs out of adjustment

3.0 REFERENCES

Window Arrangement Annunciator K09 (E-459, sheets 1-4)

K09-A8 Page 2 of 2





RC Pump (P-32A) running
RC Pump (P-32B) running
RC Pump (P-32C) running
RC Pump (P-32D) running
ES CH 3 signal
ES CH 4 signal
Flow approaching the vortex limit found by:
Total DH Flow > [(RCS level – 369.62) x 1754.38] – 100
(Assures vortex limit not exceeded at RCS levels <370.8')
Level = 370.7'
[(370.7-369.62) x 1754.38] -100
= 1268 gpm
Loop "B" Hot Leg Level (LT-1195) ≤ 375 feet
Loop "B" Hot Leg Level (LT-1198) ≤ 375 feet
Decay Heat Loop "A" Flow (SPDS F1401) >2000 GPM
Decay Heat Loop "B" Flow (SPDS F1402) >2000 GPM
>2000 GPM and ≤ 375'
Loop "B" Hot Leg Level (LT-1195) >375 feet
Loop "B" Hot Leg Level (LT-1198) >375 feet
Decay Heat Loop "A" Flow (SPDS F1401) >3550 GPM
Decay Heat Loop "B" Flow (SPDS F1402) >3550 GPM
>3550 GPM and >375'
Decay Heat Loop "A" Flow (SPDS F1401) <1000 GPM
Decay Heat Loop "B" Flow (SPDS F1402) <1000
<1000 GPM
DECAY HEAT FLOW HI/LO

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

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

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

Section: 4.2 **Type:** Generic APE

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

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

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

Tier:	1	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	Ap

Question: RO: 6 SRO:

Given:

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

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

A. RCP Seal Return Coolers

B. Spent Fuel Coolers

C. Letdown Coolers

D. Pressurizer Sample Cooler

Answer:

C. Letdown Coolers

Notes:

"C" is correct since it is the only component with the piping size and differential pressure (2155 to ~100 psig) to cause the indications given.
All of the other choices are cooled by ICW and are thus plausible but are incorrect because:
"A" RCP seal return cooler pressure is only slightly above Makeup Tank pressure (10 to 45 psig) and thus leak rate will be too small (or would be from ICW to seal return cooler) to cause surge tank overflow;
"B" Spent fuel cooler pressure (~95 psig) is less than ICW pressure (~100 psig) and thus leak rate will be from ICW to SFP cooler.
"D" Pressurizer sample cooler has a large DP (2155 to 100 psig) but the reactor coolant goes through a small line which wraps around the shell (cooled by Nuc ICW) so a tube failure would not necessarily cause a typical tube-to-shell leak. OE at ANO does not show any failure of the primary sample cooler but ANO has experienced a Letdown cooler leak.

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

Revised per NRC examiner suggestion.

References:

1203.039, Excess RCS Leakage

History:

Developed for 1998 SRO Exam.
Used in 2001 RO/SRO Exam.
Used on 2004 RO/SRO Exam.
Selected for 2016 exam

NOTE

The RB Sump contains 45.4 gal/percent.

5. Monitor RB parameters:

- Humidity (PMS/PDS M6278, M6278RTD, M6279, M6279RTD)
- RB temperature
- RB pressure
- RB Sump level

A. **IF** leakage into RB Sump is indicated,
THEN perform the following:

1) Consider performing Repetitive Tasks (1202.012), Maximize RB Cooling (RT-9).

2) Determine RCS Leakrate (Exhibit 1).

3) **GO TO step 16.**

6. Check any of the following for indications of RCS leakage into ICW system:

- Nuclear Loop ICW activity rising
- Indication of Letdown Cooler RCS leak into ICW:
  - ~~Letdown Cooler ICW Outlet temp rising on PMS~~:
    - 8P ICW trend
    - T2214 for E29A
    - T2215 for E29B
- Indication of RCP Seal Cooler RCS leak into ICW:
  - RCP Seal Temp rising
  - RCP Seal Bleedoff Temp rising
  - Skewed RCP Seal Injection Flows

None of the other components in Q#6 are listed.

NOTE

ICW Surge Tank T-37B Level (PDIS 2229) 0.5 to 2.7 psid (1 psid = 333 gallons)

A. Dispatch an operator to determine Nuclear Loop ICW Surge Tank (T37B) level trend.

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

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

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

Section: 4.1 **Type:** Generics EPEs

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

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

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

Tier:	1	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 7 SRO:

The Reactor Protection System includes a Module-In-Test/Module-Removal interlock.

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

A. prevent the associated channel from tripping

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

C. lock out the other channels' test switches

D. cause the associated channel to trip

Answer:

D. cause the associated channel to trip

Notes:

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

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

References:

1105.001, NI & RPS Operating Procedure

History:

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

If a protective channel module is removed, or if a module test switch is placed in a position other than OPERATE, its respective channel will trip.

The RPS sends signals to EFIC for actuation of EFW on the following:

- Loss of all RCPs.
- Loss of both MFWPs ≥9% full power.

Bypassing the loss of both MFW pump RPS trip also bypasses the EFIC actuation of EFW on loss of both MFW Pumps.

RPS inputs the following signals to other systems:

- A narrow range pressure input to non-nuclear instrumentation for RCS pressure controls via SASS automatic signal selector.
- RC flow input to ICS for feedwater ratioing via SASS automatic signal selector.

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

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

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

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

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

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

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

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

Question: RO: 8 SRO:

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

1202.006, Tube Rupture
Bases for 1202.006

History:

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

INSTRUCTIONS	CONTINGENCY ACTIONS
18. IF only DG power is available, THEN GO TO 1202.007, "DEGRADED POWER" procedure unless entry was from that procedure.	
19. Check Main Generator and Exciter Field breakers open: • **5114** • **5118** • **Exciter Field breaker**	19. Perform the following: A. IF 125 V DC Bus D01 is energized, THEN manually trip Main Generator and Exciter Field breakers: • 5114 • 5118 • Exciter Field breaker B. IF 125 V DC Bus D01 is de-energized, THEN leave Main Generator and Exciter Field breakers closed.

NOTE

PZR cooldown rate limits do not apply during SGTR.

INSTRUCTIONS	CONTINGENCY ACTIONS
20. Operate Pressurizer Heaters AND Pressurizer Spray valve (CV-1008) to maintain RCS press low within limits of Figure 3 (RT-14).	20. Verify Electromatic Relief ERV Isolation (CV-1000) open AND cycle ERV to maintain RCS press low within limits of Figure 3 (RT-14).
A. IF RCS press drops below 1700 psig AND SCM is adequate AND RCS press is controlled, THEN bypass ESAS.	

ANO1 EOP Step No.	B&W TBD Step No.	Explanation or Basis for Difference
20.	GEOG III.E 6.0, 7.0	This step ensures RCS pressure is maintained low within acceptable limits to minimize RCS to SG ΔP. During the cooldown it is desirable to maintain RCS pressure and temperature close to, but above, the minimum SCM. This minimizes the differential pressure between the RCS and the affected SG, thus minimizing the tube leak flow rate. If normal PZR spray is not available, then the ERV is used to lower RCS pressure. When depressurizing the RCS during mitigation of a tube rupture, PZR cooldown rate limits may be exceeded.
		This step directs bypass of ESAS when RCS press <1700 psig as long as SCM is adequate and RCS press is controlled.
21.	GEOG III.E 1.0, 7.0	This step stabilizes PZR level to prevent repressurization due to refill with HPI when initial post-trip cooldown is over.
		This step provides instructions to manually adjust HPI flow, as necessary, to maintain PZR level >55" and RCS Pressure low within Figure 3. This complies with the PTS rule.
22.	N/A	This step ensures N-16 detectors will monitor gross activity now that N-16 production is insignificant. For information on deviation determination, see "Deviations" section.
23.	GEOG III.E 2.0	This step identifies the leaking SG. The affected SG must be identified early since subsequent actions depend on this information. Since an automatic trip could have occurred prior to positively identifying the leaking SG, this step is necessary following the reactor trip.
24.	GEOG III.E 2.1	This step ensures secondary system contamination is controlled as much as possible. This step also isolates non-essential steam loads to reduce potential release paths to the atmosphere. Since an automatic trip could have occurred prior to performance of this step earlier in the procedure, this step is necessary following the reactor trip.
25.	GEOG III.E 2.1	This step prevents unmonitored release via P7A if EFW actuates. Since the valve could have previously been closed in this procedure, it is only required to be verified closed at this point.

QID: 0686 **Rev:** 4 **Rev Date:** 7/12/16 **Source:** Bank **Originator:** Steve Pullin
TUOI: A1LP-RO-EOP03 **Objective:** 7 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE
System Number: E05 **System Title:** Excessive Heat Transfer
Description: Knowledge of the interrelations between the (excessive heat transfer) and the following: components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.
K/A Number: EK2.01 **CFR Reference:** 41.7/45.7

Tier:	1	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	A

Question: RO: 9 SRO:

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

The following post-trip conditions exist:

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

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

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

- MSLI has actuated

Which of the following actions is correct for this event?

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

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

C. Verify EFW flow rates are ≥ 570 gpm on each OTSG.

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

Answer:

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

Notes:

Answer A is correct. This answer requires an understanding of the automatic features of the Main Steam Line isolation section of the Emergency Feedwater Intiation and Control system and realization that the system is malfunctioning requiring manually completion of the safety function.
Answer B is incorrect as it isolates the good SG.
Answer C is incorrect. This standard post-EFIC-actuation action is incorrect in this situation since both SGs are less than 600 psig and the A SG is greater than 150 psig less than the B SG so flow should NOT go the A SG per step 6 of RT-6. This flow rate is plausible since this is the minimum required EFW flow rate if subcooling margin was inadequate and only one SG was available. Also, flow would be going to the A SG in this situation IF the DP was less than 150 psid.
Answer D is incorrect since EFW flow should be isolated to the A OTSG but plausible since this is the minimum required EFW flow rate if subcooling margin was inadequate.

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

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

References:

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

History:

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

VERIFY PROPER MSLI AND EFW ACTUATION AND CONTROL

5. IF bad SG press is ≤ 600 psig and other SG press is > 600 psig
OR
ΔP between SGs is > 150 psig and both SGs < 600 psig,
THEN verify EFW ISOL and EFW CNTRL valves to bad SG closed:

Correct answer

SG A			SG B	
CV-2627	CV-2670	**ISOL**	CV-2620	CV-2626
CV-2645	CV-2646	**CNTRL**	CV-2647	CV-2648

NOTE
Table 1 contains EFW fill rate and level bands for various plant conditions.

6. Verify at least one EFW pump (P7A or P7B) running with flow to good SG(s)
OR both SGs if both are ≤ 600 psig and ΔP is ≤ 150 psig
through applicable EFW CNTRL valves:

SG A		SG B
CV-2645	**P7A**	CV-2647
CV-2646	**P7B**	CV-2648

7. IF SCM is not adequate,
THEN perform the following:

A. Select Reflux Boiling setpoint for the following:

- Train A
- Train B

NOTE
Table 2 contains examples of less than adequate/excessive EFW flow.

B. Verify EFW CNTRL valves operate to establish and maintain good SG level(s) 370 to 410".

(7. CONTINUED ON NEXT PAGE)

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

Question: RO: 10 SRO:

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

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

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

A. 1203.027, Loss of Steam Generator Feed

B. 1203.001, ICS Abnormal Operation

C. 1203.018, Turbine Trip Below 43% Power

D. 1202.001, Reactor Trip

Answer:

A. 1203.027, Loss of Steam Generator Feed

Notes:

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

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

References:

1203.027, Loss of Steam Generator Feed

History:

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

ENTRY CONDITIONS

One or more of the following:

- Drop in feedwater flow
- Drop in SG level
- Rise in RC pressure and temperature
- Annunciator alarms on any of the following:
  - REACTOR IS FEEDWATER LIMITED (K07-C1)
  - A MFP TURBINE TRIP (K07-A7)
  - B MFP TURBINE TRIP (K07-A8)
  - EFW ACTUATION SIGNAL (K12-A5)
  - AUX FEED PUMP TRIP (K07-F7)

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

QID: 1097 **Rev:** 1 **Rev Date:** 7/27/16 **Source:** New **Originator:** Coble

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

Section: 4.1 **Type:** Generic EPE

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

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

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

Tier: 1 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 3

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

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

Given the following:

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

Without entering any additional Technical Specification LCOs, which one of the following would be the correct action to take to INITIALLY restore power to Unit 1 for these conditions?

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

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

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

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

Answer:

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

Notes:

"D" is correct based on the first direction and purpose of the Station Blackout Recovery Procedure 1202.008 Step 3.A. and B which has the unit recover one vital 4160 bus and then exit to the Degraded Power procedure 1202.007 to restore the rest of the busses.
"A" and "B" are incorrect due to the SU#2 Transformer is not available to Unit 1 since Unit 2 is aligned to this source (Refer to the Note on contingency step 8.C. of OP 1202.008) of power. Startup Transformer #2 is not designed to carry loads of both units so only one unit can be aligned to it.
"C" is incorrect because this would cross-tie both ESF buses. This cross-tie is procedurally allowed by 1107.002 but to do so requires entering Tech Spec LCO 3.8.9.A and the question therefore excludes this answer.

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

Added "All Unit 1 feeder breaker handswitches from SU#2 are in pull-to-lock." due to validator suggestion. JWC 7/27/16

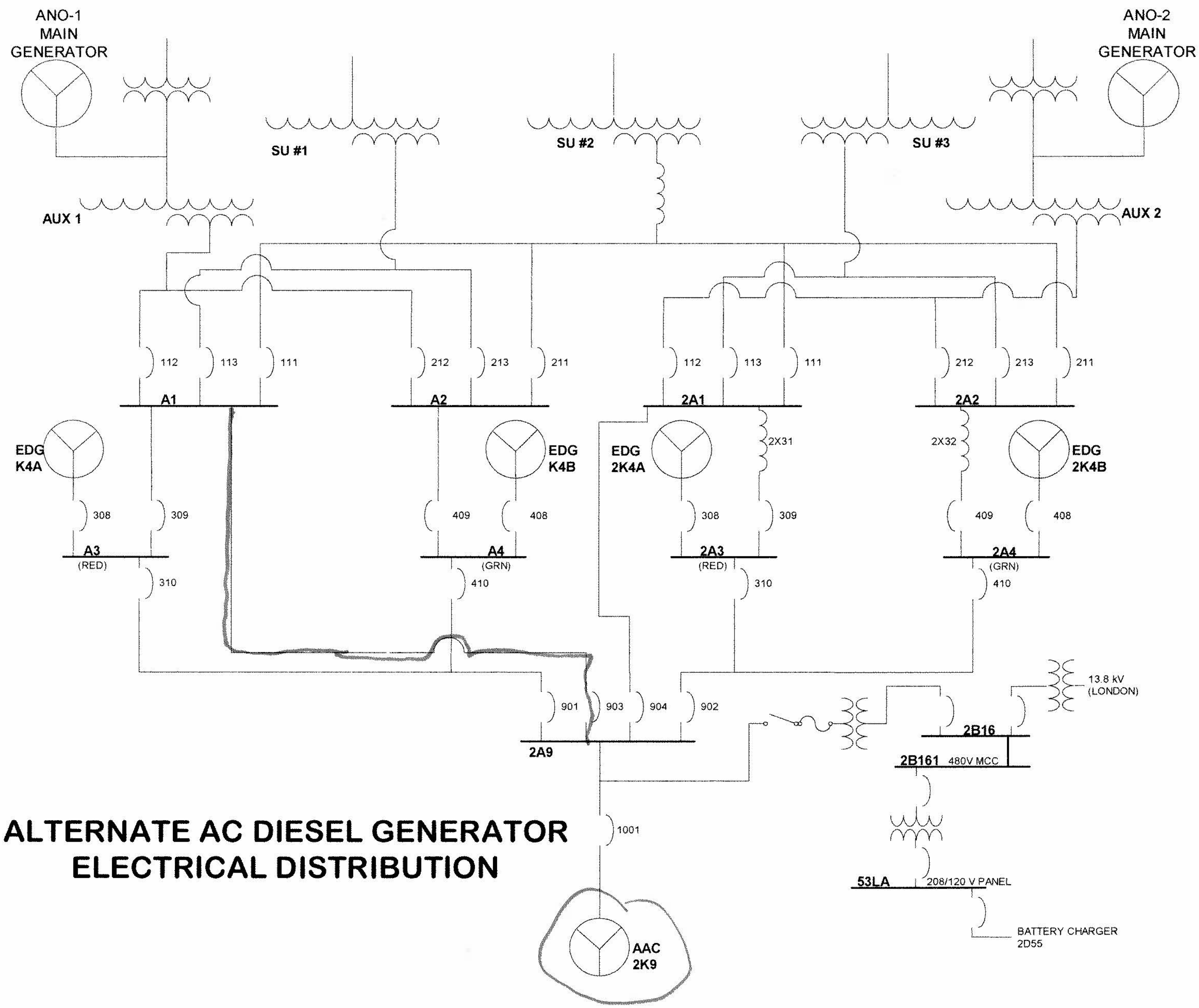
References:

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

History:

New question written for 2016 exam

INSTRUCTIONS	CONTINGENCY ACTIONS
3. Notify Unit 2 of need for AAC Gen (2K9) **AND** **attempt to restore EDG using Annunciator K01 Corrective Action (1203.012A) and Emergency Diesel Generator Operation (1104.036), while continuing with this procedure.** A. **IF** AAC Gen becomes available, **THEN** energize a vital bus using ES Electrical System Operation (1107.002), "Placing Alternate AC Generator on bus A3 (A4)" section. B. **IF** a vital bus becomes energized by EDG or AAC Gen, **THEN GO TO 1202.007, "DEGRADED POWER"** procedure.	
	NOTE RV Head voids can be identified during a Blackout using SPDS ICC2 display.
4. Check adequate SCM.	4. **IF** SCM is **not** adequate **AND** RV Head void is indicated, **THEN** perform rapid cooldown per step 55, while continuing with this procedure.



ALTERNATE AC DIESEL GENERATOR
ELECTRICAL DISTRIBUTION

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

Question: RO: 12 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

1305.006, Integrated ES System Test

History:

New question for 2016 exam.

Table 1
Odd Channel Acceptance Criteria

TEST QUANTITY	INSTRUMENT	MEASURED/ OBSERVED VALUES	LIMITING RANGE FOR OPERABILITY	IS DATA WITHIN LIMITING RANGE? (CIRCLE YES OR NO)	
Loop I SW Control Logic Test	N/A	Attachment 3	Attachment 3 satisfactory	YES	NO
Odd ES Channels Control Logic Test	N/A	Attachment 5	Attachment 5 satisfactory	YES	NO
DG1 Loaded	Clock	min.	Runs ≥1 hour @2600-2750 KW AND temperatures stabilize	YES	NO
DG1 (CH 1)	DAS Data From ESAS Actuation	sec.	At rated speed and voltage in ≤15 sec.	YES	NO
DG2 (CH 1)		sec.		YES	NO
Odd Channels ES Load Sequencing	A3 DAS Data from Loss of Power	Load shed logic	ES bus load shed on loss of power	YES	NO
		ES loads resequence	Resequence on buses	YES	NO
		HPI pump sec	4.7-5.3 sec	YES	NO
		LPI pump sec	9.6-10.4 sec	YES	NO
		SW pump sec	14.4-15.6 sec	YES	NO
		EFW pump sec	19.2-20.8 sec	YES	NO
		RBS pump sec	33.6-36.4 sec	YES	NO
		VSF-1A sec	48-52 sec	YES	NO
		VSF-1B sec	48-52 sec	YES	NO

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

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

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual control of components for which automatic control is lost.

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

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

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

Question: RO: 13 SRO:

Given the following conditions:

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

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

A. Presurizer Level Control Valve CV-1235

B, Pressurizer Heater Banks 3, 4 and 5

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

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

Answer:

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

Notes:

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

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

References:

1203.047 Loss of NNI Power Step 8. A.

History:

New question written for 2016 exam

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Check ICS and NNI Instrument Power Supply Status lights on C13 <u>AND</u> perform the following:	

NOTE

D11 breaker 25 supplies power to ICS and NNI Instrument Power Supply Status lights on C13.

INSTRUCTIONS	CONTINGENCY ACTIONS
A. <u>IF</u> all indications are normal other than loss of all ICS and NNI Instrument Power Supply Status lights on C13, **<u>THEN</u>** reset D11 breaker 25 using "Reclosing Tripped Individual Load Supply Breakers" section of Electrical System Operations (1107.001).	
B. Check all NNI X power available.	B. Perform the following: 1) <u>IF</u> any NNI Y power is also lost, **<u>THEN</u> GO TO step 2**. 2) <u>IF</u> only NNI X power is lost, **<u>THEN</u> GO TO step 6.**
C. Check all NNI Y power available.	C. **GO TO step 8.**

END

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

- Startup valve ΔP signals to MFW pumps fail low.
- Letdown Flow indication is lost.

8. IF only NNI Y power is lost, THEN perform the following:

A. **IF** MFW pump(s) are on ΔP control, **THEN** operate MFW pump(s) in HAND as necessary to control FW flow.

B. Align NNI instrument handswitches per Attachment 3, "Handswitch Alignment For Loss of NNI Y Power".

C. Check NNI Y AC power available.

NOTE

- Orifice Bypass (CV-1223) fails to 50%.
- Letdown Pressure indication is lost.

C. Perform the following:

1) Reset **AND** close NNI Y Cabinet C48 Normal Supply breaker (RS4 breaker 9) using "Reclosing Tripped Individual Load Supply Breakers" section of Electrical System Operations (1107.001).

2) Dispatch an operator to reset **AND** close NNI Y Cabinet C48 Backup Supply breaker (Y01 breaker 39) using "Reclosing Tripped Individual Load Supply Breakers" section of Electrical System Operations (1107.001), while continuing with this procedure.

(8. CONTINUED ON NEXT PAGE)

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

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

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

Section: 4.2 **Type:** Generic APE

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

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

K/A Number: AA2.03 **CFR Reference:** 41.7

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

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

Question: RO: 14 SRO:

Given:

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

1104.036, Emergency Diesel Generator Operation

History:

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

13.0 DG1 Start Without DC Control Power

CAUTION
If fault condition that caused loss of DC is not removed, a fault can still be present.

NOTE
Following sequence assumes no AC or DC is available.

13.1 IF known,
THEN remove fault condition that caused loss of DC.

13.2 Place DG1 Engine Control Selector Switch (HS-5234) in MAINT.

CAUTION
With loss of control power, the only functional DG protection is the mechanical overspeed device.

13.3 Open the following local breakers to prevent shutdown when DC power is restored:

- DG1 Local Field Flashing Power (D1116A).
(inside voltage regulator cabinet E11)

- DG1 Engine Control Power (D1114A).
(inside Engine Control Panel C107)

NOTE
"Breaker Local Operation Without DC Control Power", Exhibit G of Electrical System Operations (1107.001) contains instruction for manual operation of 4160 and 480 volt load center breakers.

CRITICAL STEP

13.4 To prevent full ES actuation upon restoration of power, de-energize ESAS digitals by opening following breakers:

- ESAS Panel C86 and C87 Breaker (RS1-4)

- ESAS Panel C91 and C92 Breaker (RS2-4)

NOTE

Breaker position can be verified by opening the breaker cabinet door and observing the OPEN/CLOSED flag on the breaker.

13.5 Trip OR verify open the following breakers by using the local mechanical trip pushbutton:

- A1 Feed to A3 (A-309)
- A3-A4 Crosstie (A-310)
- Service Water Pump P-4A Breaker (A-302)
- Service Water Pump P-4B Breaker (A-303)
- RB Spray Pump P-35A Breaker (A-304)
- LPI Pump P-34A Breaker (A-305)
- HPI Pump P-36A Breaker (A-306)
- HPI Pump P-36B Breaker (A-307)
- EFW Pump P-7B Breaker (A-311)

13.6 Override governor run solenoid as follows:

13.6.1 Lift red knurled knob on top of governor AND push in the locking pin.

13.6.2 While holding in the locking pin, lower the knurled knob until movement is restricted.

13.6.3 Release pin AND check knob stays in raised position.

NOTE

- SW to DG1 Coolers (CV-3806) requires DC power to open automatically.
- Manually opening CV-3806 will allow SW flow to DG1 once SW pump starts.

13.7 Manually open SW to DG1 Coolers (CV-3806).

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

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

Section: **Type:** Generic APEs

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

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

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

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

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

Question: RO: 15 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

1203.028, Loss of Decay Heat Removal

History:

New question for 2016 exam

SECTION 5 – LOSS OF SERVICE WATER FLOW

8. IF RCS press approaches the applicable limit listed below:

- **RCS loops not filled — 150 psig**
- **RCS loops filled — 250 psig**

THEN perform the following:

A. Cycle the ERV as necessary to maintain RCS press within limits.

B. **IF** RCS press can **not** be reduced below applicable limit,
THEN perform the following:

1) Stop the running DH pump.

2) Close at least one of the following Decay Heat Suction valves:

- CV-1050
- CV-1410
- CV-1404

3) **GO TO** applicable "Loss of Both DH Systems" section of this procedure.

9. Investigate cause of loss of heat sink.

CAUTION

If RCS temps are >200°F, it is possible for the SW side of the affected DH Cooler (E-35A or E-35B) to reach saturation temp due to lack of flow.

10. IF there is NO SW flow through DH cooler,
THEN perform the following:

A. Close applicable SW Inlet to E-35A or E-35B DH Cooler
AND immediately open associated supply breaker to prevent automatic re-opening:

Cooler	Valve	Breaker
E-35A	CV-3822	B5182
E-35B	CV-3821	B6183

B. Verify applicable Decay Heat Cooler E-35A/E-35B SW Outlet Isol valve is throttled as needed
AND NOT fully closed to avoid thermal-hydraulic lock:

Cooler	Valve
E-35A	SW-22A
E-35B	SW-22B

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

Question: RO: 16 SRO:

Given:
- Instrument Air leak is reported.
- Instrument Air header pressure low annunciator, K12-B3, alarmed.
- Instrument Air header pressure 35 psig and lowering rapidly.
- Breathing Air is being used to supply Instrument Air.
- Breathing Air is NOT in use by personnel for respiration
- OP-1203.024, Loss of Instrument Air AOP is in use.

Which of the following will be subsequently performed SPECIFICALLY to conserve instrument air?

A. Establish SG Pressure control using Atmospheric Dump Isolation valves

B. Isolate Breathing Air from the Instrument Air System.

C. Commence Plant S/D at 10% per minute.

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

Answer:

A. Establish SG Pressure control using Atmospheric Dump Isolation valves

Notes:

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

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

Revised per NRC examiner suggestion.

References:

1203.024, Loss of Instrument Air

History:

New for 2016 exam

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Throttling ATM Dump ISOL valve closed and opening ATM Dump CNTRL valve will minimize the amount of air used to position the control valve.

A.

40. Establish SG pressure control using ATM Dump ISOL valves as follows:

A. Verify ATM Dump Control system valves in MANUAL:

- CV-2618
- CV-2668
- CV-2619
- CV-2676

B. While maintaining 1000 to 1040 psig, slowly throttle ATM Dump ISOL valves closed **AND** open ATM Dump CNTRL valves.

C. Dispatch an operator to hand jack both ATM Dump CNTRL valves open. (Refer to Alternate Shutdown (1203.002), Exhibit A).

D. Maintain SG press 1000 to 1040 psig using ATM Dump ISOL valves.

41. Continue efforts to regain Instrument Air pressure using steps 6-11.

42. Check RC Pump Seals Total Inj Flow (CV-1207) bypassed.

Contingency action:

42. Perform the following to bypass seal injection:

A. Dispatch an operator with a radio to LNPR.

B. Direct dispatched operator to slowly open Seal INJ CV-1207 Bypass (MU-1207-3)
AND
slowly close Seal INJ CV-1207 Inlet (MU-1207-1) while maintaining RCP Seals Total INJ flow 30 to 40 gpm.

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

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

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

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

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

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

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

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

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

Given:

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

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

A. HPI cooling established for ≥ 120 minutes.

B. CET temperatures stable or dropping.

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

D. Subcooling margin is adequate

Answer:

B. CET temperatures stable or dropping.

Notes:

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

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

Revised stem per NRC examiner suggestion.

References:

1202.004, Overheating

History:

This question is a modified version of QID 335 which was used in 1999 and 2004 RO/SRO exam.

Modified for 2005 RO re-exam.
Revised for 2016 exam

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

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

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

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the (Inadequate Heat Transfer): 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:**	No	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	K

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

Given:

- Loss of all Feedwater ← Added SPDS indicator of inadequate Pri-Sec Heat Transfer
- HPI core cooling started
- ~~Only one HPI pump is running~~ ← Delete, adds nothing.

Prior to Revision

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~~. Subcooling margin adequate

Answer:

B. CET temperatures stable or dropping.

Notes:

"B" is correct since the only criteria for evaluation of adequacy of core cooling via HPI 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, Vol. 2, IV.B & VI

History:

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

INSTRUCTIONS	CONTINGENCY ACTIONS
8. <u>IF</u> Makeup Tank level drops below 18", <u>THEN</u> close Makeup Tank Outlet (CV-1275).	
9. Check Letdown in service.	9. <u>IF</u> conditions permit: • fuel damage does **<u>not</u>** exist • RCS to ICW leak is **<u>not</u>** suspected • SCM is adequate <u>THEN</u> restore Letdown (RT-13).
10. Control RCS press within limits of Figure 3 (RT-14).	
11. Check CET temps stable or dropping.	11. Perform the following: A. <u>IF</u> HPI flow is < full flow from one HPI pump, **<u>THEN</u> GO TO step 19.** B. Hold at this point until one of the following conditions is met: • <u>IF</u> EFW becomes available, **<u>THEN</u> GO TO step 14**. • <u>IF</u> Main or Aux Feedwater Pump becomes available, **<u>THEN</u> GO TO step 13**. • <u>IF</u> CET temps begin to drop, **<u>THEN</u> GO TO step 12.** • <u>IF</u> ≥ 120 minutes on HPI cooling elapse **<u>AND</u>** CET temps are still rising, **<u>THEN</u> GO TO step 19.** • <u>IF</u> CET temps are superheated **<u>AND</u>** moving away from the saturation line, **<u>THEN</u> GO TO 1202.005, "INADEQUATE CORE COOLING"** procedure.

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

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

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

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

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

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

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

Question: RO: 18 SRO:

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

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

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

Which of the following actions will stop these oscillations?

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

B. Place the EHC controls in Turbine Manual

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

D. Place both FW Loop Demands back in Automatic

Answer:

B. Place the EHC controls in Turbine Manual

Notes:

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

References:

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

History:

New for 2014 Exam.
Selected for 2016 exam

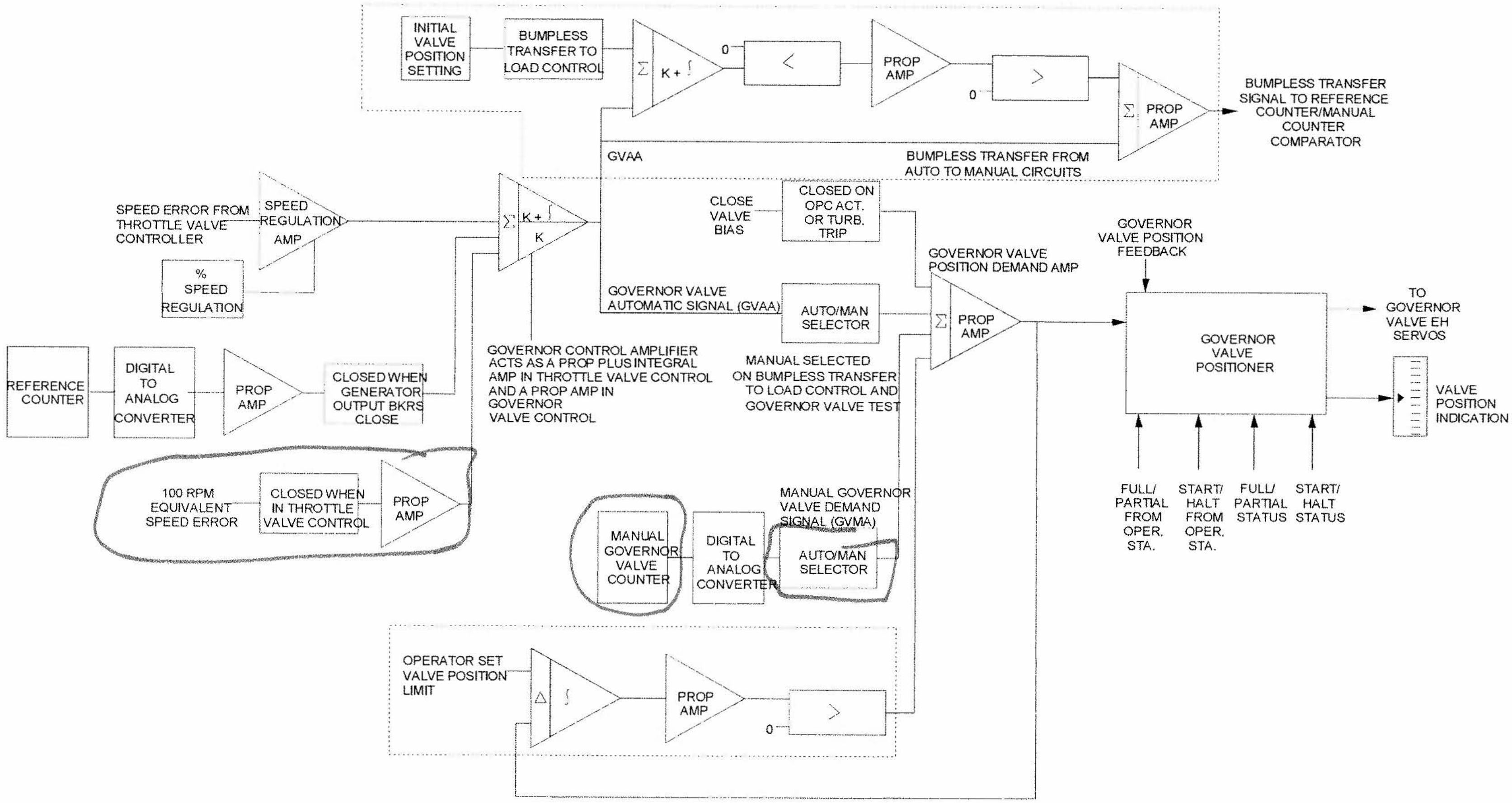
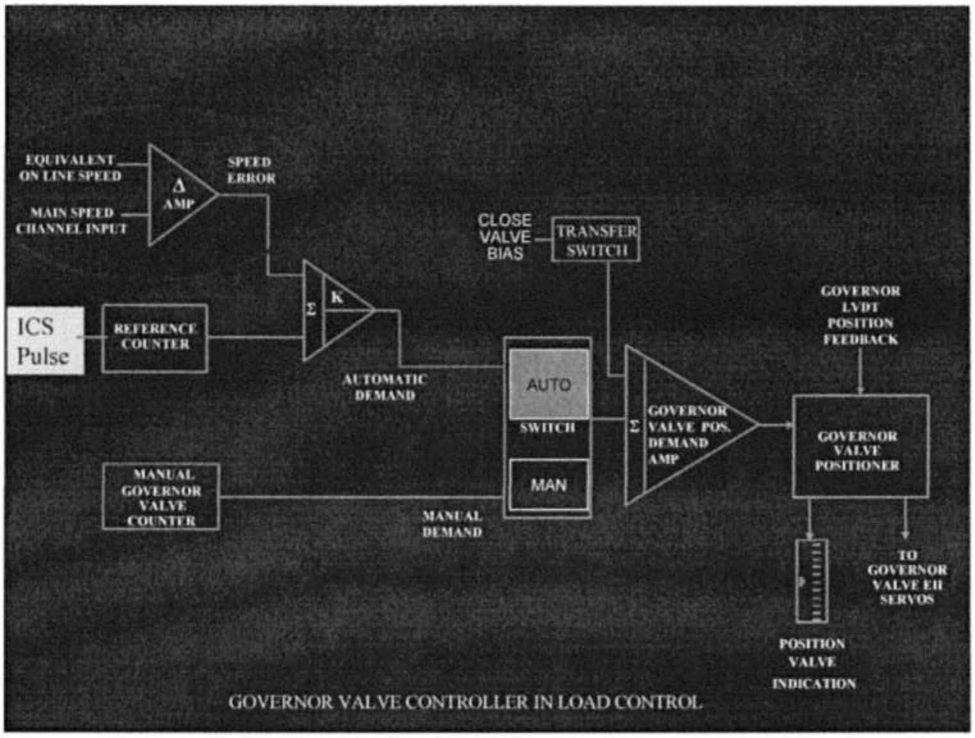
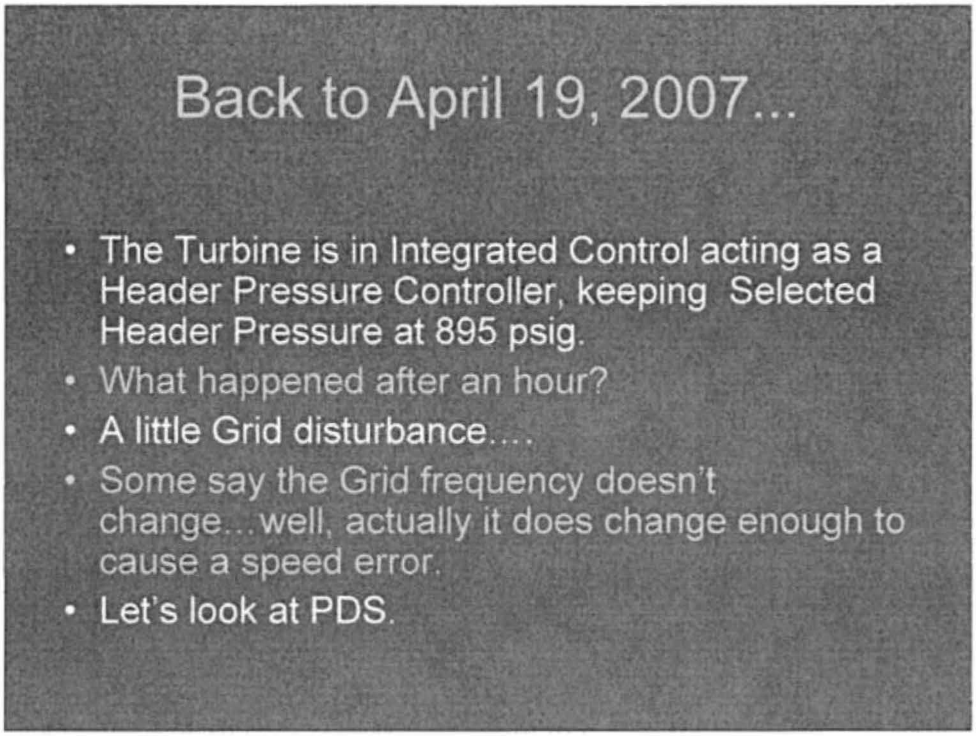


FIGURE 24.42: EHC GOVERNOR VALVE CONTROLLER

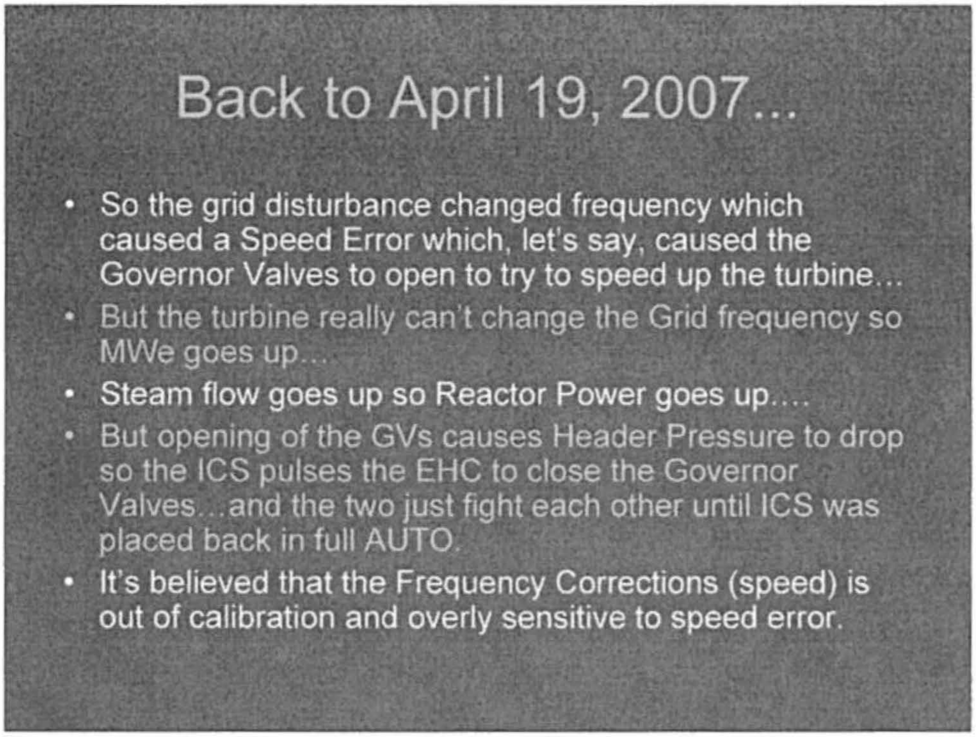


Discuss need for frequency correction if we lose a large part of the grid.

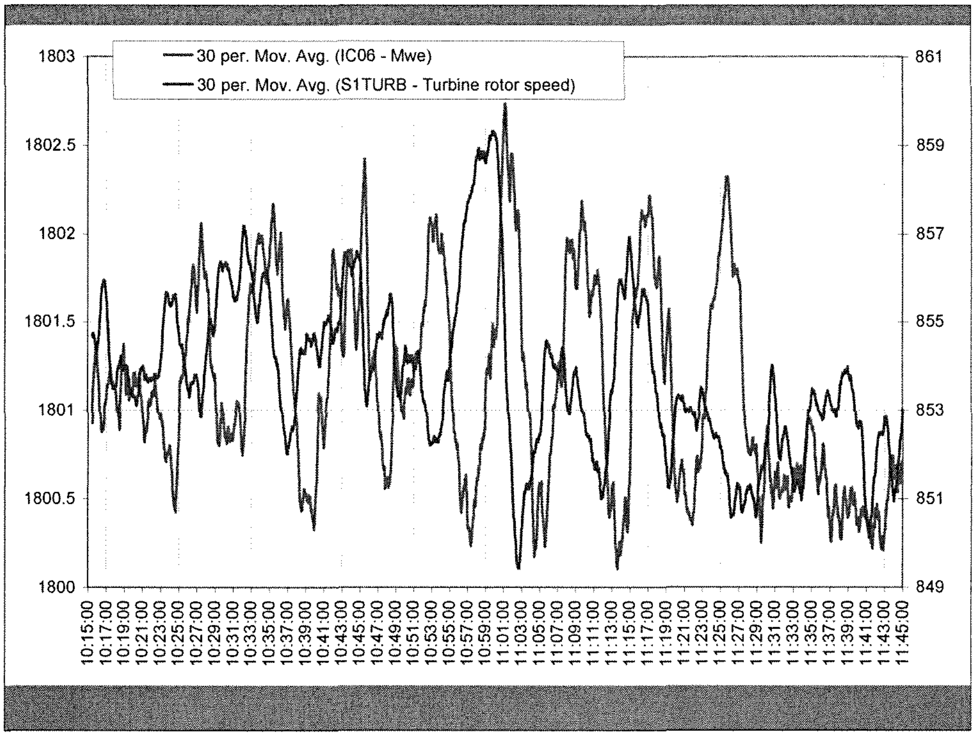
This page and the following pages are from AILP-RO-TURBC, Main Turbine Controls and Auxiliaries.



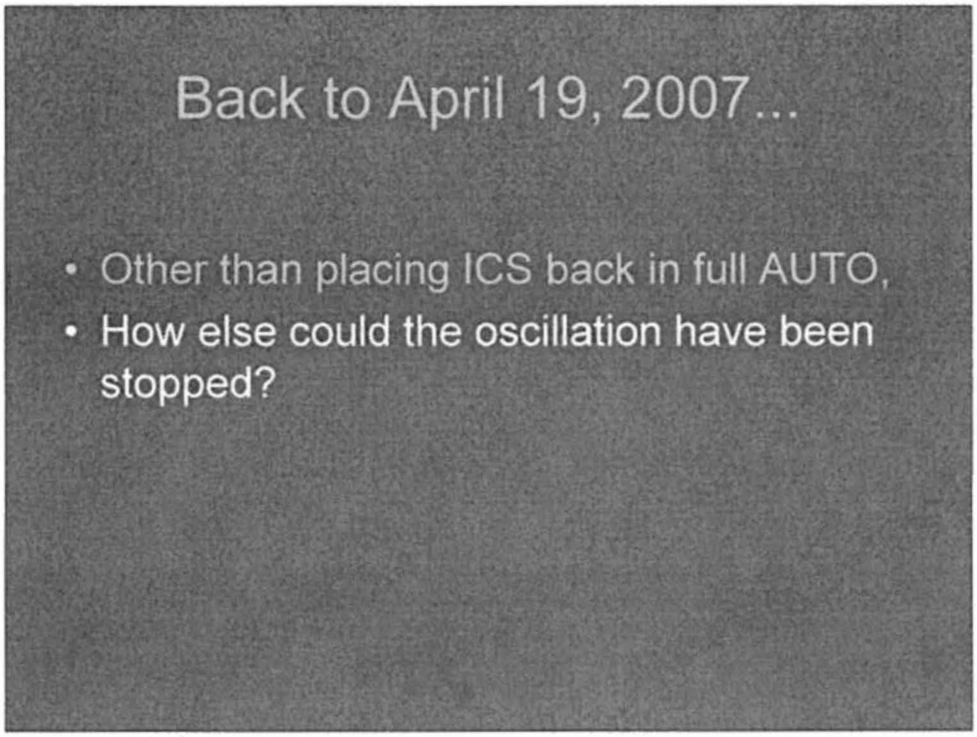
Show PDS display of EHC and how the frequency correction is continuously changing, so it is always active.



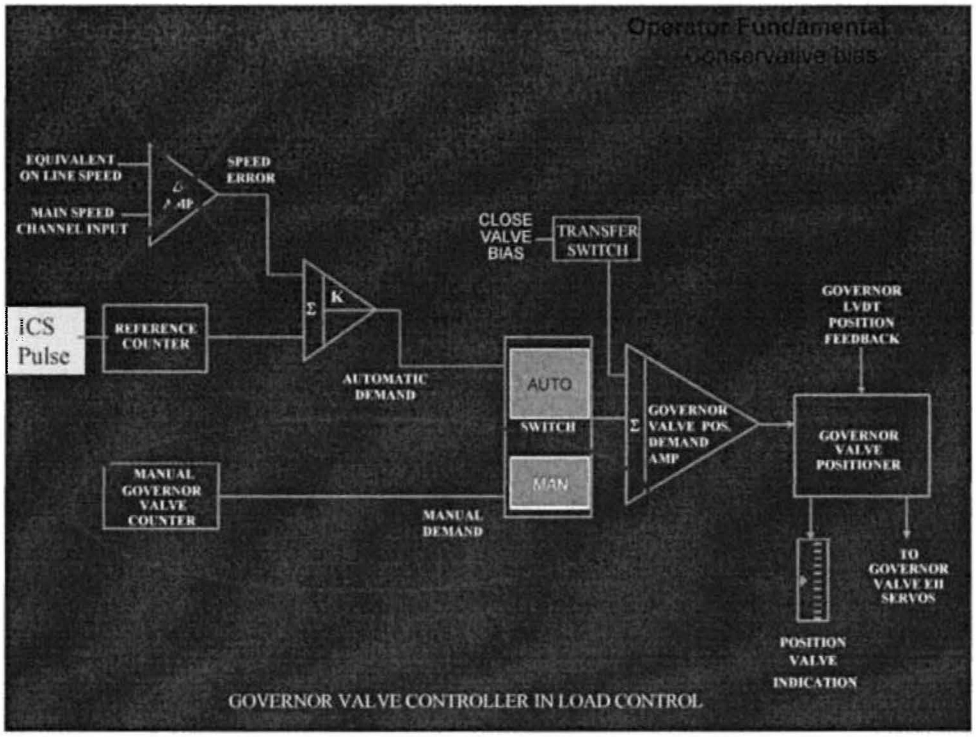
Discuss how the Frequency Correction will be checked for calibration and the possibility of changing the card to add a deadband.



This PDS trend shows how electrical load and turbine speed were in opposing oscillations.



Talk about going to Oper Auto, then to Turb Manual.



Turbine Manual removes frequency correction.

OF3 – Conservative bias (judgment). Maintain nuclear safety as your overriding priority.

RO

Tier 1

Group 2

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

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

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

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

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

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

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

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	Ap

Question: RO: 19 SRO:

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

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

What operator action is procedurally required?

A. Allow the runback to terminate normally.

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

C. Take manual control of SG/RX master.

D. Trip the reactor.

Answer:

C. Take manual control of SG/RX master.

Notes:

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

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

References:

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

History:

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

SECTION 2 - DROPPED ROD – REACTOR CRITICAL

3. <u>IF</u> a single rod drops,
<u>THEN</u> verify ICS runback to 40% of 902 MWe (~360 MWe)
<u>OR</u> current generator output is ≤ 40% of 902 MWe (~360 MWe).

<u>NOTE</u>

Instructions in CRD System Operating Procedure (1105.009) prefer NI power level <37% for recovery of a dropped rod.

A. Perform Rapid Plant Shutdown (1203.045) in conjunction with this procedure.

B. Adjust ICS demand as needed to reduce AND maintain the following conditions to clear the CRD Withdrawal Inhibited condition, and prevent Out Inhibit condition:

- <360 MWe
- <40% NI power

C. Operate as follows:

1) Operate IN LIMIT BYPASS when required to insert affected group. ←

2) <u>IF</u> dropped safety rod
<u>AND</u> required to place Letdown 3-way Valve (CV-1248) in BLEED, ←
<u>THEN</u> verify Batch Controller Outlet (CV-1250) closed.

Location: C13

Device and Setpoint: N/A

HI LOAD LIMIT IN EFFECT

Alarm: K07-C3

1.0 OPERATOR ACTIONS

1. Verify ICS in track AND running back to the maximum load limit setpoint.

2. IF high load limit clearly caused by an ICS failure OR an ICS input signal failure, THEN take manual control of affected ICS station(s) AND return plant to steady-state condition.

   A. Refer to ICS Abnormal Operation (1203.001).

3. WHEN runback is concluded, THEN check maximum load limit setpoint.

   A. IF necessary, THEN adjust to correct value.

2.0 PROBABLE CAUSES

1. Unit load demand is greater than maximum load limit.

3.0 REFERENCES

Schematic Diagram Annunciator K07 (E-457)

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

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

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

Section: 4.2 **Type:** Generic AOP

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

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

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

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	An

Question: RO: 20 SRO:

Given:

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

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

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

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

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

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

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

Answer:

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

Notes:

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

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

Revised at suggestion of NRC examiner.

References:

1203.021, Loss of Neutron Flux Indication

History:

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

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

QID: 0184 **Rev:** 0 **Rev Date:** 11/21/98 **Source:** Direct **Originator:** R. Fuller

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

Section: 4.2 **Type:** Generic AOP

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

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Startup termination on source-range loss.

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

Tier:	1	**RO Imp:**	3.2	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	An

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

Given:

- During a reactor startup with source range NI-2 and reactor power wide range recorder NR-502 inoperable, source range NI-1 fails to 10 E5.
- Intermediate range NI-3 indicates 5 E-11 amps
- Intermediate range NI-4 is off scale low.

Prior to Revision

What is required of the CBOR?

a. Continue the startup utilizing NI-3 until NI-4 comes on scale.

b. Perform a plant shutdown in accordance with normal operating procedures due to lack of proper overlap.

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

d. Hold power constant and perform an NI calibration.

Answer:

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

Notes:

[c] is correct per guidance in 1203.021, if the recorder NR.502 is inoperable AND no SR channel is >10 E5 cps AND no IR channel is > 1 E-10 amps AND 3/4 PR instruments are <10% power, then no on-scale flux indication exists and the reactor must be tripped.
[a] is incorrect, although NI-4 might come on scale, the startup should not be continued without valid neutron flux indication.
[b] is incorrect, although shutting down is conservative, per procedure the reactor must be tripped immediately.
[d] is incorrect, this action sounds like it could rectify this situation, however, it would be impossible to calibrate the NI's at this point and would be contrary to procedural guidance.

References:

1203.021 (Rev 007-02-0), Loss of Neutron Flux Indication, page 7, step 3.1

History:

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

SECTION 3 - LOSS OF ONE OR MORE SOURCE RANGE NI CHANNELS IN MODES 2 THROUGH 5

INSTRUCTIONS

NOTE

If all 4 of the following conditions apply, there is no on-scale indication of neutron flux:

- Three of four power range instruments are ≤5% power,
- No intermediate range instrument is $>10^{-10}$ amps,
- No source range instrument is $<10^{5}$ cps,
- Reactor Power Wide Range Recorder (NR-502) is inoperable.

1. IF no on-scale indication of neutron flux is available,
THEN trip reactor
AND perform Reactor Trip (1202.001) in conjunction with this procedure.

2. IF only one source range channel is operable,
THEN continue plant operations (TS 3.3.9).

A. Refer to TS 3.3.15.

3. IF both source range instruments fail,
AND at least one intermediate range channel indicates $> 10^{-10}$ amps,
THEN continue plant operations (TS 3.3.9).

A. Refer to TS 3.3.15.

4. IF both source range instruments fail,
AND both intermediate range channels indicate $\leq 10^{-10}$ amps,
THEN perform the following:

NOTE

Plant temperature changes which result in positive reactivity additions are allowed provided the temperature change is accounted for in the Shutdown Margin calculations.

A. Refer to TS 3.3.9 Condition A.

B. Immediately suspend operations involving positive reactivity changes.

C. Immediately initiate a shutdown and insert all control rods.

D. Within 1 hour verify CRD trip breakers open.

E. Within 1 hour and once per 12 hours thereafter, verify reactor >1.5% Δk/k shut down per Reactivity Balance Calculation (1103.015).

F. Refer to TS 3.3.15.

5. Notify Shift Manager to implement Emergency Action Level Classification (1903.010).

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

Tier:	1	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	An

Question: RO: 21 SRO:

Given:

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

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

A. 950 to 970 psig

B. 1000 to 1020 psig

C. 1030 to 1050 psig

D. 1060 to 1080 psig

Answer:

B. 1000 to 1020 psig

Notes:

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

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

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

References:

1202.006, Tube Rupture
Bases for 1202.006

History:

New question for 2016 exam.

Floating Steps

RCS Temp

- **WHEN** RCS T-hot is < 500°F,
  THEN maintain RCS cooldown rate as follows:

T-hot	Cooldown Rate
500°F to 300°F	≤ 100°F/hr
300°F to 170°F	≤ 50°F/hr

- **IF** RCS T-hot is < 490°F **AND** any of the following occur:
  - Bad SG Level approaches 410".
  - BWST Level reaches 23'.
  - Projected activity at the site boundary reaches Alert criteria.

  THEN perform **step 45.**

- **IF** four RCPs are running,
  THEN before RCS temp drops to 465°F trip RCP in loop with **bad** SG:

SG A	SG B
P32D	P32B

ESAS

- **IF** ESAS actuates,
  THEN perform **step 34.**

Secondary

- **IF** bad SG level is rapidly approaching 410"
  OR
  dose rate ≥ Alert criteria is projected at site boundary,
  THEN establish emergency cooldown rate of ≤ 240°F/hr to 500°F T-hot using **step 26**.

- **WHEN** good SG press is < 720 psig,
  THEN perform **step 42.**

- **WHEN** bad SG press is < 450 psig,
  THEN stop AUX Feedwater Pump (P75).

SF Pool Cooling

- **IF** Spent Fuel Pool cooling is **not** in service,
  THEN perform Unit 1 Spent Fuel Pool Emergencies (1203.050) in conjunction with this procedure.

INSTRUCTIONS	CONTINGENCY ACTIONS

45. WHEN RCS T-hot is < 490°F, THEN monitor for need to isolate bad SG as follows:

A. Check the following parameters remain within the specified limits:

SG level	≤ 410"
BWST level	> 23'
Off-site dose projection	< Alert criteria

A. Perform the following:

1) **IF** other SG is already isolated, **THEN** initiate HPI cooling (RT-4).

a) **IF no** HPI pumps are available, **THEN** allow ERV to cycle in AUTO.

(1) **IF** SCM is adequate, **THEN** trip running RCP.

(2) **IF** ERV fails open, **THEN** close Electromatic Relief ERV Isolation valve (CV-1000).

(3) **GO TO step 45.A.2).**

b) **IF** ERV **cannot** be opened, **THEN** verify HPI Recirc Blocks (CV-1300 and CV-1301) open.

1) Throttle HPI as necessary to maintain RCS press low within limits of Figure 3 (RT-14).

c) **IF** SG Tube-to-Shell ΔT reaches 60°F (tubes hotter)
AND
SCM is adequate,
THEN trip running RCP.

(1) Do **not** restart an RCP until SG Tube-to-Shell ΔT is ≤ 50°F (tubes hotter).

(45. CONTINUED ON NEXT PAGE)

INSTRUCTIONS

45. (Continued).

(45. CONTINUED ON NEXT PAGE)

CONTINGENCY ACTIONS

2) Verify **bad** SG Main Feedwater Isolation valve closed:

SG A	SG B
CV-2680	CV-2630

3) Verify **bad** SG EFW ISOL valves in MANUAL **AND** closed:

SG A	SG B
CV-2670 CV-2627	CV-2620 CV-2626

4) **IF** RCS press is > 1020 psig, **THEN** reduce RCS press to ≤ 1020 psig, while maintaining adequate SCM by any or all of the following:

- Maintain emergency cooldown rate of ≤ 240°F/hr to 500°F.
- Raise AUX Pressurizer Spray flow.
- Maximize Letdown flow.
- Verify HPI Recirc Blocks (CV-1300 and CV-1301) open and throttle HPI.
- Open High Point Vents:

A Loop	B Loop
SV-1081 SV-1082 SV-1083 SV-1084	SV-1091 SV-1092 SV-1093 SV-1094
Pressurizer	**Reactor Vessel**
SV-1077 SV-1079	SV-1071 SV-1072 SV-1073 SV-1074

- Verify Electromatic Relief ERV Isolation valve (CV-1000) open
  AND
  cycle ERV.

A OTSG Header	B OTSG Header	Setpoint	Accumulation
PSV-2699	PSV-2684	1050 psig	10%
PSV-2698	PSV-2685	1060 psig	9%
PSV-2697	PSV-2686	1070 psig	8%
PSV-2696	PSV-2687	1070 psig	8%
PSV-2695	PSV-2688	1090 psig	6%
PSV-2694	PSV-2689	1090 psig	6%
PSV-2693	PSV-2690	1100 psig	5%
PSV-2692	PSV-2691	1100 psig	5%

Accumulation is the pressure over lift setpoint at which the valve is flowing and will control flow. It is expressed as:

$$\%\ \text{Accumulation} = \frac{\text{Pressure(at design flow)} - \text{Pressure(lift)}}{\text{Pressure(lift)}} * 100\%$$

Each MSSV is rated for 6% blowdown. Blowdown is the pressure below lift pressure at which the valve will reseat. It is expressed as:

$$\%\ \text{Blowdown} = \frac{\text{Pressure(lift)} - \text{Pressure(close)}}{\text{Pressure(lift)}} * 100\%$$

So, assuming the MSSV seats are conditioned properly, the safeties should be leak tight up to 94% of lift pressure.

Main Steam Safety Valve position is monitored by the MSSV Position Indicating System. This system will be discussed in detail later in this STM in the Instrumentation section.

2.2.2 Atmospheric Dump Valves & ADV Block Valves

(Refer to Figure 15.11 & Table 15.4)

Each main steam header is fitted with an eight inch Atmospheric Dump Valve (ADV) upstream of the MSIV's. The ADV for the 'A" OTSG is CV-2668 and associated ADV Block Valve CV-2676. The ADV for "B" OTSG is CV-2618 and its ADV Block valve CV-2619.

Both the ADV's and Block Valves are controlled from the control room on panel C09. Valve position indication is provided above their associated controller or handswitch on C09. Atmospheric Dump Valve position is read on a 0 to 100% scale. ADV Block Valve position is by light indication only.

The ADV Block Valves are modulating control valves. This means that the valve can be throttled open or closed by the control room operator when controlling header pressure utilizing the ADV Block Valve.

The ADV's are designed to provide an alternate means of pressure control and heat removal from the steam generators in case

ANO1 EOP Step No.	B&W TBD Step No.	Explanation or Basis for Difference
43.	GEOG III.E 6.0	This step ensures ES does not inadvertently actuate during cooldown and depressurization.
44.	GEOG III.E 9.0, 13.0	This step ensures cooldown rate is reduced to within limits if emergency cooldown rate had been implemented.
		This step protects against thermal binding of the ERV Isolation (Ref. NRC Commitment P 14779).
45.	GEOG III.E 14.0, 17.0	This step provides for isolation of the leaking SG if necessary based on SG level, BWST level, or off-site dose.
	Volume 3 IV.B 2.A	This step ensures HPI cooling is initiated prior to isolating the last SG. HPI cooling is initiated to provide core cooling since both SGs will be unavailable for heat transfer.
	Volume 3 IV.E	This step ensures RCS pressure is maintained below MSSV setpoint when SG is isolated, to prevent lifting MSSV when SG fills solid. ADV is left unisolated in auto to provide an isolable relief path if RC pressure nears MSSV setpoint while SG is isolated. Although the value in the TBD of 1000 psig is not utilized, this is not a deviation as the value used (1020 psig) still meets the intent of TBD guidance to be below the lowest MSSV. This value was chosen as it matches the ADV automatic control setpoint.
		This step provides transition to HPI Cooldown (1202.011) if both SGs are isolated.
46.	GEOG III.E 20.0	This step protects against hydraulic lifting of fuel assemblies due to running four RCPs with high reactor coolant density.
47.	Volume 3 III.E 2.3, 3.3	This step is based on Tech Spec cooldown rate limits.
48.	GEOG III.E, 21.0	This step prevents unnecessary CFT discharge which would hamper depressurization.

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

Question: RO: 22 SRO:

Given:

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

Choose the correct procedural requirement:

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

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

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

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

Answer:

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

Notes:

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

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

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

Revised at suggestion of NRC examiner.

References:

1203.016, Loss of Condenser Vacuum

History:

Modified QID 10 for 2016 exam.

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

QID: 0010 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Direct **Originator:** GGiles

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

Section: 2.0 **Type:** Generic K&A

System Number: 2.4 **System Title:** Emergency Procedures / Plan

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:	3	**RO Imp:**	3.3	**RO Select:**	No	**Difficulty:**	2
Group:	G	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- K05-B2, CONDENSER VACUUM LO is in alarm
- K05-B3, VACUUM PUMP AUTO START is in alarm
- Power reduction is in progress due to rapidly lowering condenser vacuum.
- Plant is operating at 60% power
- E-11A North Waterbox is OOS for maintenance

Choose the appropriate operator actions:

PARENT

A. Trip the reactor and turbine if vacuum falls below 26.5 inches Hg.

B. Trip the reactor and turbine if vacuum falls below 24.5 inches Hg.

C. Trip the turbine only, if vacuum falls below 26.5 inches Hg.

D. Trip the turbine only, if vacuum falls below 24.5 inches Hg.

Answer:

B. Trip the reactor and turbine if vacuum falls below 24.5 inches Hg.

Notes:

"B" is the correct answer in accordance with 1203.016. 60% power is ~540 MW, therefore a turbine trip is required at 24.5 inches along with a reactor trip since power is >43% per the Reactor Trip EOP entry conditions. "A", "B" and "D" are incorrect because a reactor trip is required for a turbine trip above 43% and/or the wrong setpoint is given. A turbine trip at 26.5" Hg is only required if turbine load is 270 Mwe or less (~30% power).

References:

1203.016 Chg. 011-07-0
1203.012D Chg. 037-00-0

History:

Developed for 1998 RO/SRO Exam.
Modified for use in 2001 RO/SRO Exam
Selected for 2005 RO exam, but not used.
Selected for use on 2007 RO Exam.

ENTRY CONDITIONS

One or more of the following:

- Condenser vacuum degrading
- Any of the following annunciators in alarm:
  - VACUUM PUMP AUTO START (K05-B3)
  - CONDENSER VACUUM LO (K05-B2)
  - TURBINE TRIP (K04-A3)
  - TURBINE LO VACUUM TRIP (K05-A2)

INSTRUCTIONS

1. **Commence reducing turbine load to stabilize vacuum.**

   - **IF** MWe is >270 and vacuum is <24.5" Hg
     THEN trip the turbine.

   - **IF** MWe is <270 and vacuum is <26.5" Hg,
     THEN trip the turbine.

2. **Refer to Rapid Plant Shutdown (1203.045).**

3. **Verify proper condenser vacuum pump operation as follows:**

   A. Condenser Vacuum Pumps (C-5A and C-5B on C02) running.

      1) IF Condenser Vacuum Pump (C-5A/B) autostarts,
         THEN place handswitch in normal after start.

   B. Adequate Condenser Vacuum Pump (C-5A/B) Separator Tank (T-75A, T-75B) water level.

   C. Condenser Vacuum Pump Cooler (E-46A/B) ACW Outlet Temperature (TI-4020, TI-4022) normal.

> **NOTE**
>
> Under ideal conditions, the condenser vacuum pumps can only achieve approximately 26" Hg in the hogging mode of operation.

   D. **IF** Main Condenser vacuum continues to degrade below 26" Hg,
      THEN consider placing the local Condenser Vacuum Pump AUTO-HOG handswitches (HS-3636 and HS-3638) in HOG position, prior to going below 25" Hg.

   E. **IF** outside ambient temperature is below freezing,
      THEN check ambient temperature at the vacuum pumps is above freezing.

      1) IF ambient temperature at the vacuum pumps is NOT above freezing,
         THEN align vacuum pumps to the separators per Exhibit A and B of Vacuum System Operations (1106.010).

> **NOTE**
>
> The following step automatically sets the CONDENSER VACUUM LO (K05-B2) alarm setpoints to 24.7" or 26.7" Hg, depending upon MWe output to PMS.

4. **From PMS Alarm menu, set the Transient Low Vacuum Alarm:**
   "Y", Enter, F3 (save).

QID: 1096 **Rev:** 2 **Rev Date:** 7/28/16 **Source:** New **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

Question: RO: 23 SRO:

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

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

Which of the following procedure actions are required to be taken for the above conditions and why?

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

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

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

D. Verify closed, then open breakers for Hotwell makeup and Condensate reject valves to preclude creating an excess of contaminated water.

Answer:

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

Notes:

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

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

Added "A" SG pri-sec leak rate based on NRC examiner suggestion. JWC 7/15/16
Revised "D" based on NRC validation, stopping a trench release would have been a correct answer. JWC 7/28/16

References:

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

History:

New question for 2016 exam.

SECTION 1
HIGH GROSS GAMMA ACTIVITY
(continued)

NOTE

Letdown flow is limited by in-service components as follows:

- 80 gpm max per Makeup Filter (F-3A or F-3B)
- 87.5 gpm max per Letdown Cooler (E-29A or E-29B)
- 123 gpm max per Purification Demineralizer (T-36A or T-36B)

3.3 IF desired,
THEN maximize letdown using Orifice Bypass (CV-1223 on C04).

3.4 IF vital area access is jeopardized due to dose rates,
OR requested by TSC or RP supervision to minimize dose rates in other areas,
THEN isolate letdown by performing one of the following:

- Close Letdown Orifice Block (CV-1222)
AND Orifice Bypass (CV-1223) on C04 (both are air-operated).
- Close Letdown Coolers E-29A&B Outlet (RCS) (CV-1221) on C16 (MOV).
- Close Letdown Cooler E-29A Outlet (RCS) (CV-1214)
AND Letdown Cooler E-29B Outlet (CV-1216) on C18 (MOVs).

NOTE

- Minimum seal injection flow for each RCP is 2.5 gpm.
- With seal injection in service, a limited period of time is available prior to overfilling the pressurizer.

3.4.1 IF letdown is isolated
AND ICW is cooling to RCP seals,
THEN perform the following to maintain PZR level <290", while continuing with this procedure:

A. Perform ONE of the following:

- Place RCP Seal Injection Block (CV-1206) in OVRD
AND reduce RCP Seals Total INJ Flow to ~10 gpm.
- Close CV-1206.

B. Verify RCP seals are cooled by ICW.

3.5 Request Radiation Protection personnel monitor for changing radiological conditions in auxiliary building.

3.6 Monitor SPING 2 for rising count rate.

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

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

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

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

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

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

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

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

Group: 2 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** K

Question: RO: 24 SRO:

Reactor power is 90% and generated megawatts is 800.

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

A. 50%/min, 675 MWe

B. 50%/min, 360 Mwe

C. 30%/min, 675 MWe

D. 30%/min, 360 Mwe

Answer:

B. 50%/min, 360 MWe

Notes:

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

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

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

References:

1105.004, Integrated Control System

History:

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

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

QID: 0162 **Rev:** 0 **Rev Date:** 05/29/97 **Source:** Direct **Originator:** J. Cork

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

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

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

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

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

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

Question: RO: 24 SRO:

Reactor power is 90% and generated megawatts is 800.

After a runback for loss of one main feedwater pump, the ICS should run ~~stabilize~~ bak to the plant at ______ a rate of ______.

A. ~~360~~ 675 MWe ~~at~~ 50%/min

B. ~~340~~ 360 MWe 50%/min

C. ~~45% Reactor power~~ 675 MWe 30%/min

D. ~~50% Reactor power~~ 360 MWe 30%/min

Parent

Answer:

~~A. 360 Mwe~~

Notes:

[a] is correct as this question asks the trainee to recall the ICS runback limit for the loss of one MFW pump which is 360 MWe.
[b] is incorrect, number given is slightly incorrect
[c] and [d] are incorrect, the 360 MWe value is equivalent to 40% Generator output, not reactor output.

References:

~~1105.004, Chg. 023~~

History:

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

6.20 Low Load Feedwater Block Valve (CV-2624, CV-2674) Interlocks:

NOTE
Startup Valve (CV-2623, CV-2673) interlocks are based on valve limit switches.

6.20.1 If Startup Valve (CV-2623 or CV-2673) is >80% open, then the associated Low Load Feedwater Block Valve (CV-2624, CV-2674) automatically opens.

6.20.2 If Startup Valve (CV-2623 or CV-2673) is <50% open, then the associated Low Load Feedwater Block Valve (CV-2624, CV-2674) automatically closes.

6.20.3 Low Load Feedwater Block valve closes automatically upon reactor trip, even if ICS Control Override HS (C03) is in OVERRIDE.

6.20.4 With ICS Control Override HS in NORMAL, either of the following will cause Low Load Feedwater Block valve to close:

- Both Main Feedwater Pumps (P-1A and P-1B) trip
- All RC Pumps (P-32A thru P-32D) stopped

6.21 Feedwater Pumps Disch Crosstie (CV-2827) opens automatically on trip of either Main Feedwater Pump (P-1A or P-1B).

6.22 Main Feedwater Pump (P-1A, P-1B) trip rejects the associated MFW Pump Loop H/A station to HAND and runs demand to zero.

6.23 ICS Fixed Load Runbacks expressed as a percentage of 902 MWe:

Condition:	Run Back to:	Rate:
All RCPs running	103% (~930 MWe)	50%/min
Loss of 1 RCP	75% (~675 MWe)	50%/min
Loss of 2 RCPs (one in each loop)	If <55% Rx power, 45% (~405 MWe)	50%/min
Loss of 1 MFWP	40% (~360 MWe)	50%/min
Loss of 2 of 3 Condensate Pumps (P-2A, P-2B, P-2C)	40% (~360 MWe)	50%/min
Asymmetric rod	40% (~360 MWe)	30%/min
ULD >max load set	Max load set	Operator set rate of change
ULD <min load set	Run up to min load set	Operator set rate of change
Unit Load Demand in Tracking Mode	As established by equipment status	20%/min

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

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

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

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

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

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

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

Tier:	1	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 25 SRO:

Given:

- A loss of offsite power has occurred.
- Annunciator K01-A1, "EDG 1 AUTO START COMMAND", is in alarm.
- Annunciator K01-B1, "EDG 1 BRKR AUTO CLOSE FAILURE", is in alarm.
- No other alarms are in on EDG #1

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

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

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

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

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

Answer:

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

Notes:

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

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

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

References:

1203.012A, Annunciator K01 Corrective Action

History:

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

Location: C10

Device and Setpoint: see next page.

EDG 1 BRKR
AUTO CLOSE
FAILURE

Alarm: K01-B1

1.0 OPERATOR ACTIONS

1. IF bus A3 is de-energized,
THEN verify the following breakers open:

   - A1 Feed to A3 (A-309)
   - A3-A4 Crosstie (A-310)
   - A4-A3 Crosstie (A-410)

2. IF desired,
THEN perform the following to attempt to close A-308:

   A. Check K02-B6 (A3 L.O. RELAY TRIP) clear.

   B. Turn synchronize switch ON for DG1 Output breaker(A-308).

   C. Attempt to close A-308 from C10.

   D. IF breaker fails to close due to anti-pump feature,
   THEN perform the following:

      1) Depress A-308 control switch in the PULL-TO-LOCK position and release allowing switch to spring return to NORMAL-AFTER-TRIP position.

      2) Check A-308 auto closes.

   E. IF A-308 fails to close from C10,
   THEN close locally.

3. To clear alarm, remove A-308 HS from NORMAL-AFTER-TRIP position.

4. IF DG1 inoperable,
THEN verify proper MOD alignment for Service Water Pump (P-4B) and Makeup Pump (P-36B) per Makeup & Purification System Operation (1104.002) AND Service Water and Auxiliary Cooling System (1104.029).

2.0 PROBABLE CAUSES

Breaker A-308 tripped with an auto close signal.

3.0 REFERENCES

- Schematic Diagram Annunciator K01 (E-451)
- Schematic Diagram Diesel Generator ACB (E-100)

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

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

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

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

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

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

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

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

Question: RO: 26 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

1203.025, Natural Emergencies, Section 6, Flood

History:

New question for 2016 exam.

SECTION 6
FLOOD

ENTRY CONDITIONS

- Lake level > 340' and rising
- Forecasted lake level at site is > 350'
- Notification of any dam failure (potential or actual) upstream of Lake Dardanelle

NOTE

The Little Rock TOC Dispatcher will notify and call out personnel to install jumpers for breakers, switches and other equipment necessary for maintaining off-site power for shutdown and emergency operation.

12. Coordinate with Little Rock TOC Dispatcher and Unit 2 Control Room to initiate the following tasks:

NOTE

Jumpers are located at Air Break Tower (B1217).

A. Request Entergy Arkansas issue Switching Orders to perform the following:

1. De-energize Startup Transformer (SU-2).
2. Install jumper across Breaker B1218 to supply Startup Transformer (SU-2) directly to 161KV transmission line.
3. Isolate and bypass Startup 2 Voltage Regulator.
4. Disable Breaker B1218 Breaker Failure Scheme by opening 125VDC breaker 8-10 on Panel 15.
5. Disable Startup 2 Voltage Regulator protective trips by opening 125VDC breaker 8-24 on Panel 16R.
6. Open Breaker B1205 and associated manual switch(es) to limit load on installed jumpers to SU-2 load.
7. Open all test switches associated with TS-86ST2 to disable Startup Transformer #2 Primary trips (located on Panel 16R in Switchyard control house).
8. Open all test switches associated with TS-86ST2BU to disable Startup Transformer #2 Backup trips (located on Panel 16R in Switchyard control house).
9. De-energize Startup Transformer #2 (SU-2) auxiliary power by opening the following:
   - Startup #2 XFMR Auxiliaries Standby Pwr (2B42-E2)
   - Startup Transformer SU2 Normal Aux Supply (B3213B)
10. Energize Startup Transformer (SU-2).

SECTION 6 - FLOOD

16. Prior to flood waters exceeding elevation 354', perform the following:

A. Secure non-essential electrical loads.

B. Verify all necessary work is completed on SU 2 Xfmr.

NOTE

SU XFMR #2 load limits with no fans or oil pumps running apply:

- 161 KV winding: 95 amps at 161 KV (27 MVA)
- 6900V winding: 1255 amps at 6900V (15 MVA)
- 4160V winding: 1745 amps at 4160V (12.6 MVA)

C. Coordinate with Unit 2 Control Room to transfer plant auxiliaries to SU 2 Xfmr using applicable section of Electrical System Operation (1107.001).

D. Coordinate with dispatcher to de-energize SU 1 Xfmr.

17. IF it is determined that a potential or actual threat exists that would affect Spent Fuel Pool integrity, level or cooling capability,
THEN perform Unit 1 Spent Fuel Pool Emergencies (1203.050) in conjunction with this procedure. (INPO IER 11-2 Rec 4)

18. For each component verified in position Attachment B, install a Caution Tag stating, "This component is positioned for Unit 1 flooding concerns. Contact the Unit 1 Control Room prior to repositioning."

19. Annotate on the Shift Turnover Sheet that verification of Attachment B of 1203.025 is required daily while Lake Dardanelle is greater than 345 ft.

20. Conduct further operations as directed by plant management.

END

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

QID: 1105 **Rev:** 1 **Rev Date:** 7/20/16 **Source:** Modified **Originator:** NRC

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

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

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

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

K/A Number: 2.2.39 **CFR Reference:** 41.5 / 41.7 / 41.10

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

Group: 2 **SRO Imp:** 4.5 **SRO Select:** No **Taxonomy:** C

Question: RO: 27 SRO:

The plant has the following conditions:

- A LOCA has occurred
- An RCS cooldown is in progress per 1203.040, Forced Flow Cooldown
- The cooldown begins at 2300 with RCS temperature at 510 °F and RCS pressure is 1300 psig
- At 2330, RCS temperature is 465 °F, RCS Pressure is 1200 psig
- At 0000, RCS temperature is 440 °F, RCS Pressure is 1150 psig
- CBOT just completed SDM surveillance and reports the current SDM is -0.95% Δk/k

Based on the given conditions, what action is required to be taken FIRST and what is the MAXIMUM completion time for this action per Technical Specifications?

A. Initiate boration to restore SDM to within COLR limits within 30 minutes

B. Adjust the RCS cooldown rate to meet rate restrictions within 30 minutes

C. Initiate boration to restore SDM to within COLR limits within 15 minutes

D. Adjust the RCS cooldown rate to meet rate restrictions within 15 minutes

Answer:

C. Initiate boration to restore SDM to within COLR limits within 15 minutes

Notes:

"C" is correct. Per LCO 3.1.1 and the COLR, SDM shall be ≥ 1.0 % delta k/k and, if not, boration shall be initiated to restore SDM to within limits within 15 minutes.
"A" is plausible since it contains the correct action but is incorrect since the time requirement is twice that of the actual.
"B" is plausible in that the action and time requirement are correct per T.S. 3.4.3 condition B, but incorrect since cooldown rate has not been exceeded. "B" has additional credibility in that from 2300 to 2330 a step change of 45 °F occurred in that half hour period which exceeds the TS requirement when RCS temp is less than 280°F but is within the requirement of ≤50°F in any ½ hour period for the current temperature.
"D" is plausible for the same reason as "B", but the completion time stated is not in accordance with LCO 3.4.3, Condition B (but is correct for the SDM specification 3.1.1).

References:

1203.040, Forced Flow Cooldown
ANO Unit 1 Technical Specifications, LCO 3.4.3 and 3.1.1

History:

Modified QID 956 for 2016 Exam

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

QID: 0956 **Rev:** 0 **Rev Date:** 2/18/13 **Source:** Direct **Originator:** NRC

TUOI: **Objective:** **Point Value:** 1

Section: **Type:**

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

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

K/A Number: 2.2.39 **CFR Reference:**

Tier: 1 **RO Imp:** 3.9 **RO Select:** No **Difficulty:** 0

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

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

PARENT

The plant has the following conditions:

- A LOCA has occurred
- An RCS cooldown is in progress
- Calculations preparing for needed boration during the cooldown per procedure 1103.015 Attachments 4 and 5 show that the SDM at the beginning of the cooldown is -1.3% Δk/k
- The cooldown begins at 1400 with RCS temperature at 500 °F and RCS pressure is 1300 psig
- At 1430, RCS temperature is 440 °F, RCS Pressure is 1100 psig

Based on the given conditions, what action is required to be taken FIRST and what is the MAXIMUM completion time for this action per Technical Specifications?

A. Initiate boration to restore SDM to within COLR limits within 30 minutes

B. Adjust the RCS cooldown rate to meet rate restrictions within 30 minutes

C. Initiate boration to restore SDM to within COLR limits within one hour

D. Adjust the RCS cooldown rate to meet rate restrictions within one hour

Answer:

B. Adjust the RCS cooldown rate to meet rate restrictions within 30 minutes

Notes:

Per LCO 3.4.3, RCS cooldown rates shall be maintained within the limits specified in Figure 3.4.3-2. Note 4 on Figure 3.4.3-2 states the RCS cooldown rate and step cooldown rate limits. With RCS temperature greater than or equal to 280F, the maximum cooldown rate is 100F/hour, or a maximum step change of 50F in any ½ hour. From 1400 to 1430, the step cooldown rate change was 60F, and from 1430 to 1445 the change was 13F, which is still cooling down at a rate which exceeds the cooldown limits. Therefore, LCO 3.4.3, Condition B requires that the cooldown parameters be restored to within limits within 30 minutes. "B" is correct.

Answers "A" and "C" are required to be addressed per plant procedure 1103.015, Step 7.5.1, to restore the SDM to equal to or more negative than -1.5% Δk/k. However, the Technical Specifications require that the SDM be maintained at -1.0% Δk/k per procedure 1103.015, Step 5.5. Therefore, no action on this is required by the Technical Specifications with this condition. "A" and "C" are incorrect.
The action in answer "D" is correct, but the completion time stated is not in accordance with LCO 3.4.3, Condition B.

References:

1103.015 (Change 052),
ANO Unit 1 Technical Specifications, LCO 3.4.3 and 3.1.1 (Rev. 245)

1103.004 (Change 023),
1202.001 (Change 032),
1202.013 (Change 004),
1203.013 (Change 018)

History:

New for 2013 Exam

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limit specified in the COLR.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM greater than or equal to the limit specified in the COLR.	24 hours

SHUTDOWN MARGIN (SDM)

(Limits are referred to by Technical Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.8, 3.1.9, and 3.3.9)

Verify SHUTDOWN MARGIN per the table below.

APPLICABILITY	REQUIRED SHUTDOWN MARGIN	TECHNICAL SPECIFICATION REFERENCE
MODE 1*	≥ 1 %Δk/k	3.1.4, 3.1.5
MODE 2*	≥ 1 %Δk/k	3.1.4, 3.1.5, 3.3.9
MODE 3	≥ 1 %Δk/k	3.1.1, 3.3.9
MODE 4	≥ 1 %Δk/k	3.1.1, 3.3.9
MODE 5	≥ 1 %Δk/k	3.1.1, 3.3.9
MODE 1 PHYSICS TESTS Exceptions**	≥ 1 %Δk/k	3.1.8
MODE 2 PHYSICS TESTS Exceptions	≥ 1 %Δk/k	3.1.9

* The required Shutdown Margin capability of 1 %Δk/k in MODE 1 and MODE 2 is preserved by the Regulating Rod Insertion Limits specified in Figures 3-A&B, 4-A&B, and 5-A&B, as required by Technical Specification 3.2.1.

** Entry into Mode 1 Physics Tests Exceptions is not supported by existing analyses and as such requires <u>actual</u> shutdown margin to be ≥ 1 %Δk/k.

RO

Tier 2

Group 1

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

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

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

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

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

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

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

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: 28 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

Revised using NRC examiner suggestions.

References:

1203.031, Reactor Coolant Pump and Motor Emergency

History:

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

SECTION 1
SEAL DEGRADATION

NOTE

- Total seal outflow, ≥2.5 gpm could lead to overheating of seal, if seal injection were lost. 2.5 gpm is capacity of the seal cooling heat exchanger.
- RCP seal bleed off temperature is expected to rise to ~170°F on loss of seal injection to a running pump.

6. **<u>IF</u> any of the following conditions exist,**
   <u>THEN</u> raise monitoring frequency on the affected RCP seal:

   - RCP seal cavity pressure oscillations exceed 600 psi peak-to-peak.
   - ≥2.5 gpm total seal outflow, including seal bleedoff.
     (excluding shaft sleeve leakage)
   - RCP seal bleed off temperature >155°F.
   - Seal bleed off temp >50°F above 1st stage seal temp
   - Failure of one stage as indicated by zero or near zero stage ΔP

   A. <u>IF</u> another stage shows sign of failure,
      <u>THEN</u> consideration should be given to stopping pump per Reactor Coolant Pump Operation (1103.006).

(continued)

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

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

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

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

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

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

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

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

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

Question: RO: 29 SRO:

Given:

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

1104.002, Makeup and Purification System

History:

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

5.22 Maximum allowable flow through the seal injection filter (F-2) is 60 gpm.

5.23 Number of allowed successive starts with motor initially at ambient temperature is 2 starts. Number of allowed successive starts with motor initially at rated temperature is 1 start.

5.24 Due to the close tolerances in the makeup/HPI pumps (P-36A/B/C), operation for even a few seconds with their respective suction valve closed will result in damage to the pump.

5.25 Operation of makeup pump(s) with suction aligned from makeup tank (T-4) only and all inlets to T-4 isolated, will lead to vacuum being drawn in T-4 which will result in makeup pump damage.

5.26 Simultaneous operation of Aux Lube Oil Pump (P-64A,B,C) and HPI/MU Pump (P-36A,B,C) is minimized to limit oil leakage at pump outboard thrust bearing.

5.27 Only one train of HPI system will be tested at a time unless the reactor vessel head is removed or RCS temperature is above RTNDT to avoid possibility of low temperature overpressurization.

5.28 When operating two HPI pumps in parallel for testing, limit duration as necessary to keep makeup tank temperature < 120°F and RCP seal bleed-off flow > 1 gpm each.

5.29 If the sample return is lined up to the makeup tank, minor RCS boron dilution will occur during pressurizer steam space sampling or purging as unborated steam is condensed and returned as unborated water to the RCS. This will also result in the pressurizer at a higher boron concentration.

5.30 Above 300°F, RCS Makeup Flowrate should be maintained > 35 gpm during normal operations in order to minimize the potential for cracking of the "D" RCP HPI thermal nozzle (normal makeup flowpath).

5.31 When starting an idle MU/HPI pump, consideration should be given to the potential for adding water to the RCS which may have a different boron concentration than the RCS due to RCS boron changes over time with no flow through the idle pump. An idle MU/HPI pump water volume of 65 gallons should be assumed.

5.32 An OOS Aux Lube Oil Pump (P-64A,B,C) makes its associated HPI/MU pump (P-36A,B,C) inoperable.

5.33 If MU/HPI pump (P-36A,B,C) pump case has been drained or seals replaced, then seals will require venting by Mechanical Maintenance.

5.34 This procedure has been determined to have a Reactivity Impact. Applicable sections that actually have an impact contain a "Caution" at the beginning of the section as follows:

R A P

CAUTION

The following section has been determined to have a Reactivity Addition Potential (RAP) and this activity is classified as a Risk Level Rx (where "x" is a level 1, 2, 3, 4 or 5 per COPD-030).

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

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

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

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

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

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

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

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

Question: RO: 30 SRO:

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

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

Which of the following transients caused the above indications?

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

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

C. Letdown isolation due to high temperature.

D. Inadvertent closure of the Makeup Tank Outlet Isolation.

Answer:

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

Notes:

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

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

References:

1203.036, Loss of 125V DC

History:

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

The reactor trip results in turbine trip and generator lockout relays (286 G1-1, 286 G1-2, and 286 G1-3) actuation, which causes automatic transfer to offsite power. With a loss of control power to the even train breakers, these breakers will not operate. This results in a loss of AC power to the even train buses. DG2 will not start due to a loss of control power.

With no AC or DC power, inverters Y22, Y24 and Y25 will be lost resulting in loss of power to 120V AC Panels RS2 and RS4. Inverters Y11, Y13, and Y15 remain in a normal mode.

Loss of power to RS2 and RS4 results in EFIC actuation of MSL and EFW. With a loss of D21, control power to EFW Pump (P-7A) is lost and the turbine will trip on overspeed. EFW control valves associated with P-7A are failed full open (loss of RA2).

Loss of power to Y02 results in closure of Purification Demineralizer Inlet and Makeup Filter Inlet valves causing letdown relief valve to lift. Letdown must be isolated by closing LD Cooler E-29A Outlet MOV (CV-1214) and LD Cooler E-29B Outlet MOV (CV-1216).

Loss of DC control power to Condenser Vacuum Pump (C-5B), if operating, causes the Seal Recirc Pump (P-31B) to stop and vacuum pump inlet valve to close. Upon restoration of DC control power, the condenser vacuum pump will trip and must be restarted or will auto start on low vacuum.

If a valid ES signal is received, DH Cooler Bypass (CV-1432) will not reposition due to a loss of ES control (RA2 BKR 11).

Effects of Loss of Both D01 and D02

A complete loss of both bus D01 and D02 includes loss of:

- 125V DC Station Battery Bank to Bus D01 (D07)
- 125V DC Station Battery Bank to Bus D02 (D06)
- Battery chargers to D07 and D06

- D01 distribution system
- D02 distribution system

This loss results in the following conditions:

- Reactor trip
- Loss of power to main turbine trip solenoids (SV-8524 and SV-8527 and XZ-8524).
- Loss of power to EOS Overspeed Trip Protection.
- Loss of EOS Main Turbine Trip Solenoids (SV-6623 and SV-6624).

- Loss of power to generator lockout relays (286 G1-1, 286 G1-2, and 286 G1-3).
- Loss of power to distribution breaker control power.

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

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

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

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

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

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

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

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: 31 SRO:

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

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

A. BWST level will rise

B. Pressurizer level will drop

C. DH pump will be damaged

D. RCS cooldown rate will be exceeded

Answer:

B. Pressurizer level will drop

Notes:

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

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

References:

1104.004, Decay Heat Removal Operating Procedure

History:

New question for 2016 exam.

7.3.6 IF DH Suction RB Isolation (CV-1404) is closed, THEN perform the following to open CV-1404:

A. Unlock and close Decay Heat Suction from RCS CV-1404 breaker (B5651).

B. Make entries on Category E/Locked Component Log (E-DOC 1015.001H) for the following:

- CV-1404
- B5651

CAUTION

Aligning LPI/Decay Heat Pump (P-34A) suction to the RCS prior to securing pressure surge path (EC39303) will align the RCS to the BWST.

C. Open DH Suction RB Isolation (CV-1404).

7.4 IF placing LPI/Decay Heat Pump (P-34A) into service, THEN perform the following:

7.4.1 Notify RP of the following:

- Which Decay Heat Room Cooler(s) will be placed into service in the "A" Decay Heat Vault to identify potential contamination concerns
- "A" Decay Heat Pump will be started. Changing dose rates are possible in associated piping. IER L3-13-12

7.4.2 IF "A" LPI/DH Block Valve (CV-1401) Pressure Bleedoff or "A" Surge Chamber is aligned, THEN perform the following:

A. Perform "Reducing Pressure At "A" LPI/DH Line Pressure Upstream Check Valve DH-13A/17 (PI-1401)" Exhibit A of this procedure, to reduce pressure downstream of CV-1401 to minimum.

B. Perform "Securing "A" LPI/DH Block Valve (CV-1401) Pressure Bleedoff and Surge Chamber" section of this procedure.

7.4.3 Verify LPI/Decay Heat Pump (P-34A) oil levels normal per "LPI Decay Heat Pump and Motor P-34A Lube Oil Check and Add" Exhibit M of Electrical System Operations (1107.001).

7.4.4 Close DHR Cooler E-35A Outlet (CV-1428) (modulating valve).

7.4.5 Position E-35A Cooler Bypass (CV-1433) to 74-80% open.

7.4.6 Open LPI/Decay Heat Block (CV-1401) (modulating valve).

WARNING

At RC pressure >220 psig with DH flow throttled, the DH pump discharge relief can lift during pump start causing steam and water to exit floor drains.

CAUTION

- LPI/DH pump will be damaged if either Decay Heat Suction valve CV-1050 or CV-1410 closes while pump is operating.
- LPI/Decay Heat pump discharge relief setpoint is 445 +22.5/-13.35 psig. Discharge pressure is typically maintained <400 psig to prevent challenging the relief.

NOTE

- Decay Heat Suction (CV-1050) will close automatically if Core Flood Tank T-2A Outlet (CV-2415) comes off its closed seat or if RC pressure exceeds 320 psig.
- Decay Heat Suction (CV-1410) will close automatically if Core Flood Tank T-2B Outlet (CV-2419) comes off its closed seat or if RC pressure exceeds 385 psig.
- The auto close interlock is automatically reset when RCS pressure is <290 psig.
- SPDS Diagnostic display for LPI can be helpful in monitoring LPI/Decay Heat pump performance.

7.4.6 Perform the following to start LPI/Decay Heat Pump:

A. IF desired by CRS/SM,
THEN clear personnel from "A" DH Vault.

B. Make plant announcement for starting LPI/Decay Heat Pump P-34A.

C. Stabilize RCS temperature using Turbine Bypass Valves and/or Atmospheric Dump Valves.

D. Place LPI/Decay Heat Pump P-34A handswitch (HS-1417) to START.

7.4.7 WHEN LPI/Decay Heat Pump (P-34A) starts,
THEN verify the following valves open:

- LPI/Decay Heat Pump Brg Clr E-50A Inlet (CV-3840)
  - IF CV-3840 fails to open,
    THEN perform the following:
    A. Close local instrument air supply to CV-3840.
    B. Open vent petcock on bottom of actuator housing.
    C. Check CV-3840 indicates open on C18.
    D. Refer to Conduct of Operations (1015.001), "AOV Operations" section.
- DHR Clr Service Water E-35A Inlet (CV-3822)

7.4.8 After LPI/Decay Heat Pump (P-34A) start, monitor Pressurizer level for indication of discharge relief (PSV-1407) lifting.

A. IF Pressurizer level starts dropping,
THEN stop LPI/Decay Heat Pump (P-34A).

1. Check DH PP P-34A Disch Relief (PSV-1407) reseats.
2. Throttle E-35A Cooler Bypass (CV-1433) further open prior to attempting restart of LPI/Decay Heat Pump (P-34A).

7.4.9 Adjust E-35A Cooler Bypass (CV-1433) to establish Decay Heat Removal flow within the following limits:

- LPI/Decay Heat Pump (P-34A) flow ≤ 3500 gpm
- LPI/Decay Heat Pump (P-34A) discharge pressure <400 psig
- IF RCS is at atmospheric pressure or RCS has a N_2 overpressure
  AND decay heat load allows,
  THEN maintain total DH flow ≤ 2000 gpm.

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

QID: 0611 **Rev:** 2 **Rev Date:** 7/28/16 **Source:** Bank **Originator:** Cork/Pullin

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

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

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

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

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

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

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

Question: RO: 32 SRO:

Given:

- Plant is in Mode 5.
- P-34A Decay Heat Removal pump is in service.
- Annunciators K09-C7 "TRAIN A RCS LEVEL LO" and K09-D7 "TRAIN B RCS LEVEL LO" alarm.
- RB sump level 40% and rising. This is a step change of 2%.

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

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

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

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

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

Answer:

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

Notes:

Answer "B" is the correct response, an RCS leak of >20 gpm is indicated (2% step change in RB sump level = ~90 gallons), the pump should be secured and the suction from the RCS isolated.
Answer "A" is incorrect but plausible, this is the response to other problems with P-34A, this will not mitigate the low level.
Answer "C" is incorrect but plausible, closing CV-1434 will isolate the suction to P-34A from the RCS (this is required in Section 1 of 1203.028 for a leak <20 gpm) but isolating the entire DH system from the RCS is required.
Answer "D" is incorrect, although taking suction from the BWST and injecting to RCS would raise level (this is the LPI flowpath) and is plausible since it is an option to makeup for lost inventory later in the procedure, it is not an action to take "first" to mitigate the level loss.

Changed condition "A Decay Heat Removal system is in service" to "P-34A Decay Heat Removal pump is in service" per validator suggestion JWC 7/28/16

References:

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

History:

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

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

QID: 0611 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** Direct **Originator:** Cork/Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 025 **System Title:** Loss of Residual Heat Removal System (RHRS)

Description: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Increasing reactor building sump level

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

Tier:	1	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- Plant is in Mode 5.
- "A" Decay Heat Removal system is in service.
- RB sump level 40% and rising. (step change of 2%)

Annunciators K09-C7 "TRAIN A RCS LEVEL LO" and K09-D7 "TRAIN B RCS LEVEL LO" alarm.

Which of the following procedurally required actions should be performed?

Prior to Revision

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

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

C. Stop P-34A Decay Heat pump and close CV-1434.

Add noun names

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

Answer:

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

Notes:

Answer "B" is the correct response, an RCS leak is indicated, the pump should be secured and the suction from the RCS isolated.
Answer "A" is incorrect, although this is the response to other problems with P-34A, this will not mitigate the low level.
Answer "C" is incorrect, closing CV-1434 does not isolate the RCS drop leg only the suction to P-34A.
Answer "D" is incorrect, although this would raise level, it will not mitigate the level loss.

References:

1203.028, Loss of Decay Heat Removal

History:

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

Location: C14

Device and Setpoint:
Hot leg level less than operator-adjustable setpoint for TRAIN A ICCMDS LT-1195 RCS Level Low Annunciator

TRAIN A RCS LEVEL LO

Alarm: K09-C7

1.0 OPERATOR ACTIONS

NOTE
This alarm is disabled when Train A ICCMDS is in NORMAL mode.

1. Verify decay heat flow within limits of Decay heat Removal Operating Procedure (1104.004), Attachment B.

2. IF level continues to lower,
   THEN stop operating Decay Heat Pump (P-34A or P-34B).

3. Select LPI Diagnostic Instrumentation display on SPDS to monitor LPI pump performance.

4. Investigate cause of loss of RCS inventory.

5. Refer to Loss of Decay Heat Removal (1203.028).

NOTE
The setpoint should be set so that the alarm will alert the operator to an unexpected level reduction.

6. IF level reduction is intentional,
   THEN adjust setpoint of "RCS Level Low Annunciator" at Train A ICCMDS per Inadequate Core Cooling Monitor and Display (1105.008), "Change Setpoints" section to a value just below desired RCS level.

2.0 PROBABLE CAUSES

Abnormal decay heat flow

3.0 REFERENCES

Window Arrangement Annunciator K09 (E-459 sheets 1-4)

Location: C14

Device and Setpoint:
Hot leg level less than operator-adjustable setpoint for TRAIN B ICCMDS LT-1198 RCS Level Low Annunciator

TRAIN B RCS LEVEL LO

Alarm: K09-D7

1.0 OPERATOR ACTIONS

> **NOTE**
> This alarm is disabled when Train B ICCMDS is in NORMAL mode.

1. Verify decay heat flow within limits of Decay heat Removal Operating Procedure (1104.004), Attachment B.

2. IF level continues to lower,
   THEN stop operating Decay Heat Pump (P-34A or P-34B).

3. Select LPI Diagnostic Instrumentation display on SPDS to monitor LPI pump performance.

4. Investigate cause of loss of RCS inventory.

5. Refer to Loss of Decay Heat Removal (1203.028).

> **NOTE**
> The setpoint should be set so that the alarm will alert the operator to an unexpected level reduction.

6. IF level reduction is intentional,
   THEN adjust setpoint of "RCS Level Low Annunciator" at Train B ICCMDS per Inadequate Core Cooling Monitor and Display (1105.008), "Change Setpoints" section to a value just below desired RCS level.

2.0 PROBABLE CAUSES

Abnormal decay heat flow

3.0 REFERENCES

Window Arrangement Annunciator K09 (E-459 sheets 1-4)

SECTION 2 – LOSS OF INVENTORY >20 GPM

ENTRY CONDITIONS

One or more of the following:

- DECAY HEAT FLOW HI/LO (K09-A8) alarm
- TRAIN A RCS LEVEL LO (K09-C7) alarm
- TRAIN B RCS LEVEL LO (K09-D7) alarm
- TRAIN A DECAY HEAT ROOM FLOOD (K09-C4) alarm
- TRAIN B DECAY HEAT ROOM FLOOD (K09-D4) alarm
- Reduced level in any of the following:
  - Pressurizer
  - Hot leg
  - Tygon RCS level indication
  - **IF** Spent Fuel Transfer Tube (SF-45) open,
    THEN SF pool
- Possible rising level in any of the following:
  - RB sump
  - Aux Building sump
  - Aux Building Equipment Drain Tank (T-11)
  - Makeup Tank (T-4)
  - Dirty Waste Drain Tank (T-20A/B)

SECTION 2 – LOSS OF INVENTORY >20 GPM

INSTRUCTIONS

1. **Stop the running DH pump(s).**
2. **Close at least one Decay Heat Suction valve:**
   - CV-1050
   - CV-1410
   - CV-1404
3. **IF alternate purification flowpath is in-service, THEN close the following valves:**

   A. Makeup Prefilter F-25 Out to MU&P (MU-6)

   B. MU&P to Makeup Prefilter F-25 (MU-5)

   C. SF to DH Suction Header (SF-20)
4. **Notify Shift Manager to Implement Emergency Action Level Classification (1903.010).**
5. **Perform "Control Room Actions For Containment Closure And Evacuation" Attachment G of this procedure.**
6. **Commence plotting RCS temperature, RCS pressure and heatup rate every 15 minutes.**
7. **IF loss of inventory is due to Fuel Transfer Canal leak, THEN GO TO Refueling Abnormal Operations (1203.042), "Fuel Transfer Canal Leak" section.**

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

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

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

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

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

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

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

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 33 SRO:

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

A. B-8

B. B-6

C. B-4

D. B-2

Answer:

B. B-6

Notes:

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

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

References:

1107.002, ES Electrical System Operation

History:

Modified QID 903 for 2016 exam

ATTACHMENT B

480 V ES Load Center B6					
BREAKER NUMBER	DESCRIPTION Ref. Drawing (E-8)	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INIT
612	A4 Feed to B6 (E-105)	Closed			
613	B6-B5 Crosstie (E-106)	Open			
614	MCC B62 Supply (E-8)	Closed			
621	MCC B61 Supply (E-8)	Closed			
622	B6 Supply to MCC B56 (E-8)	Closed			
623	RB Cooling Fan VSF-1C (E-361)	Racked In			
624	Control Room Chiller VCH-2B (E-373)	Note 1 Closed			
631	CRD 480V Supply (E-431, E-256)	Note 2 Closed			
632	Switch Yard Supply (E-431)	Closed			
633	RB Cooling Fan VSF-1D (E-361)	Racked In			
634	MCC B65 Supply (E-8)	Closed			

Note 1: Fuses must be pulled to remove the 125VDC supply to the Control Room Chiller Control Cabinet (C-124). See (E-81).

Note 2: Closed per Plant Startup (1102.002).

INITIALS

3.0 Breaker Handling Jib Crane either NOT installed or secured in position with two locking bolts to the front rail. ______

ATTACHMENT C

BREAKER NUMBER	DESCRIPTION Ref. Drawing (E-18)	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INIT
6142	ERV Isolation Valve CV-1000 (E-199-2)	Closed			
	Local/Remote handswitch	REMOTE			
6143A	Spare	Open			
6143B	Normal Supply to Y02 Instrument AC Transformer X-61 (E-431)	Closed			
6144	PZR Proportional Heaters Bank 2 (E-203)	Closed			
6145A	Inverter Y24 (E-17)	Closed			
6145B	Spare	Open			
6146	North Battery Rm Exh Fan VEF-34 (E-365)	Closed			
6151	HPI to P-32B Discharge CV-1227 (E-219)	Closed			
6152	HPI to P-32A Discharge CV-1228 (E-219)	Closed			
6153	RCP Bleedoff Normal Return CV-1274 (E-208)	Closed			
6154	Letdown Coolers Outlet CV-1221 (E-216)	Closed			
6161	LPI/Decay Heat Block CV-1400 (E-183)	Closed			
6163	RB Sump -- Line B Outlet (Inside Sump) CV-1415 (E-182)	Closed			
6164	BWST T-3 Outlet CV-1408 (E-184)	Closed			
6166	RB Sump -- Line B Outlet CV-1406 (E-182)	Closed			
6171	R.B. Spray Block CV-2400 (E-219)	Note 1			
6172	SG-B ATM Dump Isol CV-2619 (E-442)	Closed			
6174	Core Flood Tank T-2B Sample CV-2418 (E-239)	Closed			

Note 1: This breaker will be closed by Plant Startup 1102.002.

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

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

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

Section: 3.5 **Type:** Containment Integrity

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

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

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

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

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

Question: RO: 34 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

1103.005, Pressurizer Operation

History:

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

WARNING

Opening the ERV causes a localized steam release at the pilot valve vent. This is a radiation and safety hazard.

CRITICAL STEP

7.2.5 Verify personnel are clear of the vicinity of the ERV.

CAUTION

- Pressurizer heatup rate limit is ≤ 100°F/hr.
- DH system maximum pressure is ≤ 250 psig.

7.2.6 WHEN RC pressure reaches 60-80 psig, THEN open the following valves to vent nitrogen from PZR to Quench Tank (T-42):

A. ERV Isolation Valve (CV-1000).

B. ERV (HS-1014 on C04).

7.2.7 Prior to RC pressure reaching 30 psig, close ERV (HS-1014 on C04).

7.2.8 IF this is a heatup following a refueling outage OR the ERV has NOT been exercised during cold shutdown within the last 92 days, THEN perform "Exercising of the Pressurizer Electromatic Relief Valve" Supplement 1 of this procedure.

NOTE

Venting and bubble formation is considered complete when both of the following conditions are met:

- Keeping the ERV open for three-minutes results in Quench Tank pressure rise of ≤ 1 psig.
- A saturation pressure/temperature relationship exists in the PZR.

7.2.9 WHEN RC pressure rises to 60-80 psig, THEN repeat steps 7.2.5 through 7.2.7 as necessary until bubble forms.

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

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

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

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

Description: Ability to monitor automatic operation of the CCWS, including: Requirements on and for the CCWS for different conditions of the power plant. .

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

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

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

Question: RO: 35 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

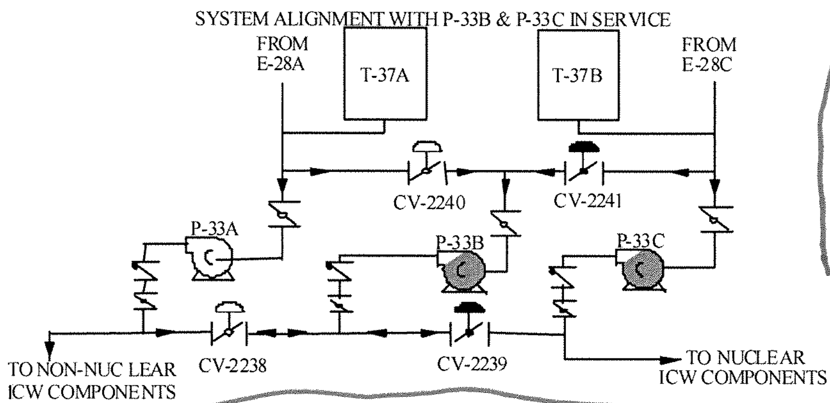
References:

STM 1-43, Intermediate Cooling Water

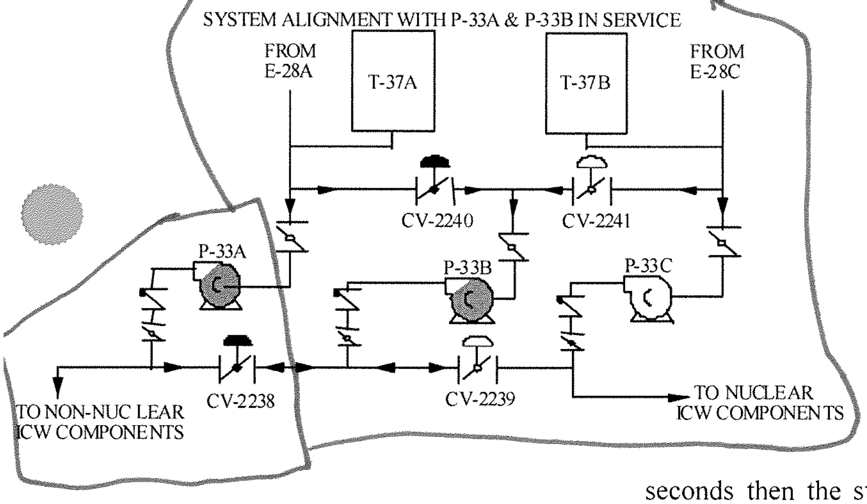
History:

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

(CV-2238 and CV-2239) are closed. P-33B handswitch will be in the normal-after-stop position.



In this condition, if P-33A discharge pressure switch, PS-2230 reaches alarm setpoint of <35 psig for > 10 seconds then P-33B will auto start. Non-Nuclear loop suction and discharge cross-connect valves, CV-2240 and CV-2238 open, aligning suction flow to P-33B and discharge flow from P-33B to the Non-Nuclear loop.



If P-33C discharge pressure switch, PS-2232 reaches alarm setpoint of <35 psig for > 10 seconds then P-33B will auto start. Nuclear loop suction and discharge cross-connect valves CV-2241 and CV-2239 open, aligning suction flow to P-33B and discharge flow from P-33B to the Nuclear loop.

With P-33B in service on either of the ICW loops and P-33B discharge pressure switch, PS-2231 reaches alarm setpoint of <35 psig for > 10 seconds then the standby pump (P-33A or P-33C) will auto start. The associated loop suction and discharge cross-connect valves will close isolating P-33B from that loop. If supplying Non-Nuclear ICW flow, CV-2240 and CV-2238 close on P-33B low discharge pressure. If supplying Nuclear ICW flow, then CV-2241 and CV-2239 close on P-33B low discharge pressure.

When operating with P-33B in service on either ICW loop and the opposite loops low discharge pressure is sensed, then the standby pump (P-33A or P-33C) will auto start and the suction / discharge cross-connect valves will re-align P-33B to the loop with the lowest pressure.

Example 1: If a low discharge pressure is sensed on P-33A with P-33A and P-33B in service then the following will occur.

* P-33C will start to supply Nuclear ICW loop flow.
* Nuclear ICW cross-connect valves CV-2241 and CV-2239 close separating the two ICW loops.

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

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

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

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

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

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

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

Tier: 2 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 4

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

Question: RO: 36 SRO:

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

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

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

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

A. ICW pump P-33B starts immediately.

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

C. ICW Booster pump P-114B starts.

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

Answer:

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

Notes:

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

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

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

Replaced D distracter per NRC examiner comment.

References:

1203.012K, Annunciator K12 Corrective Action

History:

Modified QID 94 for 2016 exam.

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

QID: 0094 **Rev:** 1 **Rev Date:** 11/4/98 **Source:** Direct **Originator:** JCork
TUOI: ANO-1-LP-AO-ICW **Objective:** 5.g **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems
System Number: 008 **System Title:** Component Cooling Water System (CCWS)
Description: Knowledge of the purpose and function of major system components and controls.

K/A Number: 2.1.28 **CFR Reference:** CFR: 41.7
Tier: 2 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3
Group: 3 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** C

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

HIGH MISS RATE WHEN USED!!

ICW Booster pump P-114A is in service and trips due to breaker overcurrent with a low suction pressure of 15 psig.

PARENT

How is seal cooling to RCPs maintained?

A. ICW pump P-33B will start due to low discharge pressure.

B. Bypass Control valve CV-2287 will open to maintain flow to seal coolers.

C. ICW Booster pump P-114B will start due to low discharge pressure.

D. ICW Booster pump P-114B will start due to low suction pressure.

Answer:

b. Bypass Control valve CV-2287 will open to maintain flow to seal coolers.

Notes:

With a common low suction pressure, the standby pump P-114B will not start due to suction pressure <45 psig even though discharge pressure is low (setpoint of <105 psig). The bypass control valve will open to maintain flow, therefore (b) is correct.
(a) alone will not ensure seal cooling due to increased DP through idle booster pumps. (c) and (d) are incorrect since suction and discharge pressure are low and will prevent start of P114B.

References:

STM 1-43, Intermediate Cooling Water, Rev. 3, page

History:

Developed for 1998 RO exam
Revised after 9/98 exam analysis review.

Location: C19

BOOSTER PUMP
DISCH PRESS
LO

Alarm: K12-C4

Device and Setpoint: (either of the following):
A. P-114A Disch Press (PS-2286), <105 psig
B. P-114B Disch Press (PS-2296), <105 psig, reset >125 psig

1.0 OPERATOR ACTIONS

NOTE

The standby RCP Seal Cooling Pump (P-114A or B) will auto start if the operating pump discharge pressure is <105 psig and suction pressure to the standby pump is >45 psig.

1. Verify start of standby RCP Seal Cooling Pump:
   - P-114A
   - P-114B

2. IF both P-114A AND P-114B are inoperable, THEN secure as follows:

   A. Place handswitch for P-114A and P-114B (C09) in PULL-TO-LOCK OR open breaker for P-114A (B-7146) and breaker for P-114B (B-7226).

   B. Verify RCP Seal Cooling Pump Bypass (CV-2287) open.

   C. Monitor seal return temperatures and RCP seal performance.

      1) IF RCP seal temperatures are NOT stabilizing, THEN perform the following:

         a. Minimize letdown by closing Letdown Orifice Bypass Valve (CV-1223).

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

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

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

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

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

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

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

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

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

Question: RO: 37 SRO:

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

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

A. Proportional heaters ON

B. Heater Bank 3 ON

C. Heater Bank 4 OFF

D. Heater Bank 5 OFF

Answer:

C. Heater Bank 4 OFF

Notes:

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

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

References:

1103.005, Pressurizer Operation

History:

New question for 2016 exam.

6.0 SETPOINTS

6.1 Electromatic Relief Valve

6.1.1 Normal operation: opens at 2450 psig
closes at 2395 psig

6.1.2 LTOP: opens at 400 psig
closes at 350 psig

6.2 Heater Banks 1 and 2 (proportional heaters)(SCR-1004, SCR-1005): have a variable output between 2135 psig (full on) and 2155 psig (full off).

6.3 Heater Bank 3 Pressure Switch (PS-1010): on at 2135 psig
off at 2155 psig

6.4 Heater Bank 4 Pressure Switch (PS-1006): on at 2120 psig
off at 2140 psig

6.5 Heater Bank 5 Pressure Switch (PS-1007): on at 2105 psig
off at 2125 psig

6.6 Pressurizer Level Switch (LS-1001)

6.6.1 Pressurizer lo lo level heater interlock:
Turns heaters off at ≤ 55"

6.6.2 PZR LEVEL LO LO (K09-A3): 55"

6.6.3 PZR LEVEL HI HI (K09-B3): 275"

6.7 Pressurizer Spray Valve (CV-1008):

6.7.1 Normal: opens at 2205 psig
closes at 2155 psig

6.7.2 > 80% with Main Feedwater pump trip: opens at 2080 psig
closes at 2030 psig

6.8 Pressurizer Level Indicator Switch (LIS-1002),
Pressurizer Level Recorder/Switch (LRS-1001)

6.8.1 PZR LEVEL LO (K09-C3): 200"

6.8.2 PZR LEVEL HI (K09-D3): 240"

6.9 Code Safeties (PSV-1001, PSV-1002): open at 2500 psig.

6.10 Quench Tank Level (LIS-1051)

6.10.1 QUENCH TANK LEVEL HI/LO (K09-B4): > 8212 gal
≤ 5071 gal

6.11 Quench Tank Pressure (PIS-1051)

6.11.1 QUENCH TANK PRESS HI (K09-A4): > 90 psig

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

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

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

Section: 3.7 **Type:** Instrumentation

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

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

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

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

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

Question: RO: 38 SRO:

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

A. Inverter Y22 from B65

B. Inverter Y22 from D02

C. Inverter Y24 from B61

D. Inverter Y24 from D02

Answer:

D. Inverter Y24 from D02

Notes:

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

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

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

References:

1107.003, Inverter and 120V Vital AC Distribution

History:

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

EXHIBIT D

1107.003 Exhibit D
PANEL RS4

REVISED 4/11/14

Power Source: Inverter Y24 or Y25
Location: Control room
Ref drawing: E-17-1A

NOTE: All breakers except spares should be closed.

1	"D" RPS Cabinet C44 (E-544-4)	2	Spare
3	Spare	4	Spare
5	CRD Pos System Logic Cabinet C51 (E-553-2)	6	Reactor Building Pressure Transmitter PT-2403 to "D" RPS (E-268-1, E-573-1, E-548-7)
7	ICC (Train B) Cabinet C554 (E-528-3) (1)	8	Rad Monitor Panel C25 Bay 3 (E-533-3)
9	NNI Y Cabinet C48 Normal Supply (E-547-3)	10	Cont Rad Mon RE-8061, EFW Test Flow FI-2888, T-41 Level LIT-4203 in C486-4 (E-410, E-331-33, E-331-34)
11	Spare	12	EFIC Channel "D" Panel C37-4 (E-597-8)
13	Vital Power to Radio (2)	14	Spare
15	Primary Power to "A" MFP Cntrls Secondary Power to "B" MFP C579 (E-556-5)	16	Spare
17	Spare	18	Spare

Note 1: If EC-44046 is implemented, then this breaker also supplies power to SFP level instrument LIT-2020-4 per E-259-11.

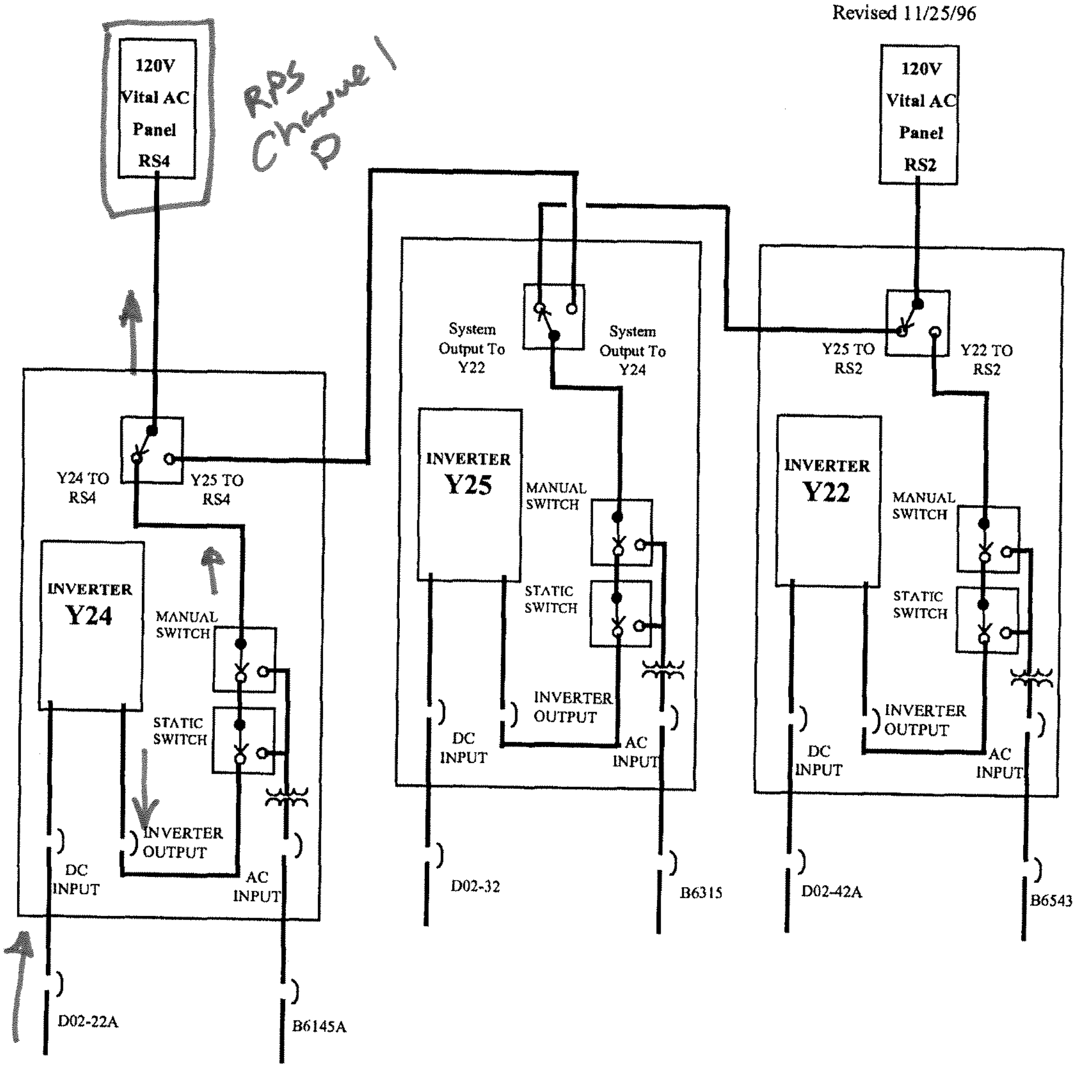
Note 2: If EC-46709 is implemented, then this breaker supplies power to "Vital Power to Radio UPS Outlet".

{4.3.1}
* De-energizing this circuit causes actuation of Control Room Isolation

1107.003
EXHIBIT L

Green Train Inverter One Line Diagram

Revised 11/25/96



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

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

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

Section: 3.7 **Type:** Instrumentation

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

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

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

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

Question: RO: 39 SRO:

PHOTO ON FOLLOWING PAGE

During a plant startup the following indications are observed:

- Rx power is 12%

- "A" MFP is operating.

- "B" MFP is tripped.

- All RPS alarms on K08 are clear.

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

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

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

A. 1 out of 3

B. 2 out of 3

C. 1 out of 4

D. 2 out of 4

Answer:

B. 2 out of 3

Notes:

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

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

References:

1102.002, Plant Startup

History:

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


36
TRIP
INPUT STATE
TRIP
MFP "A" TRIPPED
DEPRESSING
TEST (UPPER) SWITCH
WILL TRIP THIS
RPS CHANNEL
UNIT 1 OPS
MIS 99-012
INPUT STATE
RESET
MFP "A" RESET
885
CONTACT
BUFFER
DSI
TRIP
INPUT STATE
TRIP
MFP "B" TRIPPED
DEPRESSING
TEST (UPPER) SWITCH
WILL TRIP THIS
RPS CHANNEL
UNIT 1 OPS
MIS 99-012
DS2
RESET
INPUT STATE
RESET
MFP "B" RESET
885
CONTACT
BUFFER
MFP TRIP BYPASS
RESET
OUTPUT MEMORY
RESET
POINT
885
SOLID STATE
BISTABLE
DSI
MFP "A" TRIPPED
DS2
MFP "B" TRIPPED
DS3
MFP TRIP
DS4
MFP TRIP BYPASS
885
AUXILIARY
RELAY

NOTE
MFWP minimum speed should be ~3000 rpm.

17.15 Perform the following:

17.15.1 Verify Main Steam is aligned to E-2s prior to opening FW Pump recirc. ______

17.15.2 Perform "Startup of MFWP (P-1A) to Minimum Speed" or "Startup of MFWP (P-1B) to Minimum Speed" section of Condensate, Feedwater and Steam System Operation (1106.016), for the pump to go into service first. (Do NOT place the MFWP into service). ______

(circle one)

"A" / "B" MFWP minimum speed ________ rpm.

17.16 WHEN the second condensate pump is running, THEN perform the following in the order listed:

{4.3.4}

CAUTION
Placing MFP H/A station in auto prior to being called for in "Turbine S/U and Power Escalation to 25%" section of this procedure could result in MFP oscillations.

17.16.1 Place a Main FW Pump (P-1A or P-1B) into service and stop the Auxiliary Feedwater Pump per 1106.016, "Placing MFWP Into Service" section. ______

A. MFWP A/B (circle one) in-service. ______

B. Auxiliary Feedwater Pump secured. ______

17.16.2 Reset RPS Anticipatory Reactor Trip System (ARTS) for first MFW pump per Attachment E. ______

ATTACHMENT E

ANTICIPATORY REACTOR TRIP SYSTEM (ARTS) RESET

1.0 WHEN first Main Feedwater Pump is placed into service, THEN perform the following to reset RPS ARTS trip:

_______/_____
Date/Time

CAUTION
Depressing TRIP switch will trip the channel even if feed pump trip function is bypassed.

1.1 In RPS Channel A Cabinet (C41) perform the following:

1.1.1 Obtain SRO/RO Concurrent Verification of steps in this subsection.

1.1.2 Depress "test" switch labeled "RESET" on contact buffer lower module for the started Main Feedwater Pump.

1.1.3 Verify two red lights on contact buffer change state (top comes ON, bottom goes OFF).

1.1.4 Verify white light "MFWP" "A" ("B") "TRIPPED" (for the started MFWP) goes DIM.

1.2 In RPS Channel B Cabinet (C42) perform the following:

1.2.1 Obtain SRO/RO Concurrent Verification of steps in this subsection.

1.2.2 Depress "test" switch labeled "RESET" on contact buffer lower module for the started Main Feedwater Pump.

1.2.3 Verify two red lights on contact buffer change state (top comes ON, bottom goes OFF).

1.2.4 Verify white light "MFWP" "A" ("B") "TRIPPED" (for the started MFWP) goes DIM.

1.3 In RPS Channel C Cabinet (C43) perform the following:

1.3.1 Obtain SRO/RO Concurrent Verification of steps in this subsection.

1.3.2 Depress "test" switch labeled "RESET" on contact buffer lower module for the started Main Feedwater Pump.

1.3.3 Verify two red lights on contact buffer change state (top comes ON, bottom goes OFF).

1.3.4 Verify white light "MFWP" "A" ("B") "TRIPPED" (for the started MFWP) goes DIM.

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

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

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

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

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

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

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

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

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

Question: RO: 40 SRO:

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

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

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

A. Containment isolation would be challenged

B. Containment design pressure would be exceeded

C. Containment atmosphere iodine concentration would be higher.

D. Hydrogen production would be greater than design

Answer:

C. Containment atmposphere iodine concentration would be higher.

Notes:

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

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

Revised question based on NRC examiner suggestion.

References:

SAR, chapter 6

History:

New question for 2016 exam.

6 ENGINEERED SAFEGUARDS

Engineered safeguards are those systems and components designed to function under accident conditions to prevent or minimize the severity of an accident or to mitigate the consequences of an accident. In the event of a Loss of Coolant Accident (LOCA), the engineered safeguards act to provide emergency coolant to assure structural integrity of the core, to maintain the integrity of the reactor building, and to reduce the fission products expelled to the reactor building. Special precautions are taken to assure high quality in the components and in system design and to assure reliable and dependable operation.

The engineered safeguards include provisions for:

A. High Pressure Injection (HPI) of borated water by the Makeup and Purification System;

B. Low Pressure Injection (LPI) of borated water by the Decay Heat Removal System;

C. Core flooding with borated water by the Core Flooding System;

D. Reactor building cooling by the Reactor Building Cooling System;

E. Reactor building cooling with borated water spray by the Reactor Building Spray System;

F. Reactor building isolation by the Containment Isolation System and filtration of containment building leakage by the Penetration Room Ventilation System;

G. Removal of iodine fission products in the reactor building atmosphere by the Reactor Building Spray System; and,

Figure 6-1 schematically depicts the major engineered safeguards systems related to core and building protection. A general description of the engineered safeguards provisions is presented below and a more detailed description is presented in the latter portion of this section.

The systems above fulfill the functions ascribed to engineered safeguards in the FSAR. The Emergency Feedwater (EFW) System was subsequently upgraded to the standards of engineered safeguards equipment (see CLAPNR 1-120-02 dated 12/3/1980). Although the EFW is considered an ES system by design, the description of ES systems in this chapter and throughout the SAR generally refer to those systems listed above.

The HPI and LPI and the Core Flooding System are collectively designed as an Emergency Core Cooling System (ECCS) which, for the entire spectrum of Reactor Coolant System (RCS) break sizes, terminates the core thermal transient, limits the amount of zirconium-water reaction, and assures that the core integrity is maintained. Figure 6-1 shows the ECCS.

The HPI System is an integral part of the Makeup and Purification System which uses two of the three makeup pumps (P36A, P36B, P36C) for injection of coolant from the Borated Water Storage Tank (BWST) (T-3). The LPI system is an integral part of the Decay Heat Removal System which uses the two decay heat pumps (P34A, P34B) and two decay heat coolers (E35A, E35B) and has provision for coolant injection from the BWST or recirculation from the reactor building sump. (See Chapter 9 for a description of the Makeup and Purification and Decay Heat Removal Systems.) The Core Flooding System is composed of two separate pressurized tanks (T2A, T2B) containing borated water at reactor building ambient temperature. This passive system automatically discharges its contents directly into the reactor vessel at a preset RCS pressure without reliance on any actuating signal, or on any externally actuated component.

Reactor building integrity is insured by two independent pressure reducing systems operating on different principles; the Reactor Building Spray System and the Reactor Building Cooling System (refer to Figure 6-1). These systems have the redundancy required to meet the single failure criterion. These systems operate over the entire spectrum of RCS break sizes to rapidly reduce the driving force for leakage of radioactive materials from the reactor building. The Reactor Building Spray System also reduces the iodine fission product concentration in the reactor building atmosphere following a LOCA. This function is accomplished by adding alkaline sodium hydroxide to the borated water upstream of the spray headers. The Penetration Room Ventilation System further reduces post LOCA fission product releases by filtration of the penetration room atmosphere. This processes leakage from the containment into the penetration rooms.

Operability of engineered safeguards equipment is assured in several ways. Much of the equipment in these systems function during normal reactor operation thus providing a constant check on operational status. Where equipment is used for emergency functions only, such as in the Reactor Building Spray System, the systems have been designed to permit meaningful periodic tests. Operational reliability has been achieved by using proven component designs wherever possible and/or by conducting tests. Quality control and assurance requirements are implemented during the design, manufacture, and installation of the engineered safeguards components and systems to assure that a high quality level is maintained. The quality program is based upon the use of accepted industry codes and standards as well as supplementary test and inspections. The resultant high quality level of the components gives assurance that they will perform their intended function under the worst anticipated conditions following a LOCA. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a Design Basis Accident (DBA). All equipment must remain functional throughout the life of the plant. Certain safety-related equipment must operate during the design plant life as well as function as required during and following a DBA at the end of plant life.

This chapter describes the physical arrangement, design, and operation of the engineered systems as related to their safety function. Reactor building isolation is described in Chapter 5. Chapter 7 describes the actuation instrumentation for engineered safeguards systems. Table 6-12 gives actuation setpoints for all systems discussed. Chapter 14 describes the analysis of the engineered safeguards systems' capability to provide adequate protection during accident conditions. Chapter 9 discusses functions performed by these systems during normal operation and gives further design details and descriptive information.

6.1 EMERGENCY CORE COOLING SYSTEM

6.1.1 DESIGN BASES

The principal design for the Emergency Core Cooling System (ECCS) as described in the NRC General Design Criterion 35, "Emergency Core Cooling," is met. Protection for the entire spectrum of RCS break sizes is provided. Separate and independent flow paths are provided in the ECCS and redundancy in active components insures that the required functions are performed if a single failure occurs. Separate emergency power sources are supplied to the redundant active components and separate instrument channels are used to actuate the systems. Actuation pressures for the ECCS systems are shown in Table 6-12. The adequacy of the installed ECCS to prevent fuel and clad damage is discussed in Chapter 14.

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

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

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

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

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

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

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

Tier:	2	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 41 SRO:

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

What is the primary purpose of bistables?

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

B. Provide signals for computer and annunciator alarms

C. Provide communication links between analog and digital subsystems

D. Provide electrical isolation for signals outside of the system

Answer:

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

Notes:

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

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

References:

STM 1-65, Engineered Safeguards Actuation System

History:

New question for 2016 exam

4.2 PRESSURE TEST MODULES

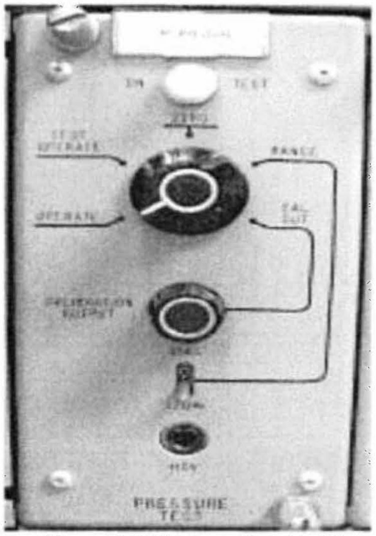


Figure 65-17 Pressure Test Module

Test modules provide for in-place testing of ESAS analog channel modules. The test modules provide multi-function-testing capabilities, including the generation of test signals.

Front plate layout provisions include:

Rotary test switch

Adjustment knob

Toggle switch

Indicating lamp

Test jack

The pressure test module is used to test the buffer amplifiers and bistables.

4.3 BISTABLES

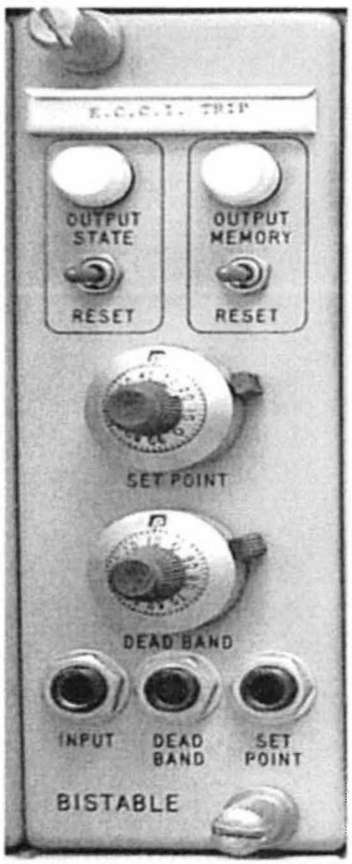


Figure 65.18 Bistable

Bistables are used in ESAS to convert analog input signals to digital output signals. The digital output signals are changes in voltage and current through operation of relay contacts. This occurs when reaching a setpoint value.

The bistable circuitry has two basic sections:

- Analog to digital
- Digital

The analog to digital section consists of the difference amplifier, the comparator; the dead band supply assembly and the setpoint supply assembly. The digital section consists of the relay coil drivers and the lamp state assembly.

Using the **difference amplifier**, the bistable compares two analog signals. One signal will be from a signal source external to the bistable. The second analog signal is an internal setpoint signal provided from the variable trip point power supply assembly.

An "output state" lamp provides indication of the actual output status of the bistable. If "on-dim," the bistable is not tripped. If "on bright," the bistable has tripped. This lamp will return to the "on-dim" condition when the bistable resets.

A bistable module is a standard 2 unit wide module. On the front plate are lights to indicate the Trip State of the bistable and the state of the bistable memory. There are two potentiometers with turn counting dials. One is for adjusting the setpoint. The other is for the reset deadband. Test jacks provide for measuring input, setpoint, and deadband.

Distracter "D"

4.4 CONTACT BUFFERS

Figure 65-19 Contact Bufffer

Contact buffers provide electrical isolation for signals originating outside ESAS. This ensures that faults in the external circuit will not adversely affect ESAS. This electrical isolation uses the principle of impedance in transformers. (See Discussion Box 65-2)

A contact in the secondary circuit of a transformer controls a relay in the primary side circuit. There is no direct electrical connection between the contacts and the relay.

The relay, in turn, operates contacts used to provide signals for ESAS in both the analog and digital channels.

Front plate layout includes two neon-indicating lamps and two toggle switches. There are a variety of uses for contact buffers in RPS and ESAS. Different internal wiring arrangements are available. Because of this, the "normal" state (lighted or not lighted) of the contact buffer lamp can vary from one application to the next. The primary purpose of the contact buffer lamp is to show the state of internal components during testing.

Distracter "C"

4.5 LOGIC BUFFERS

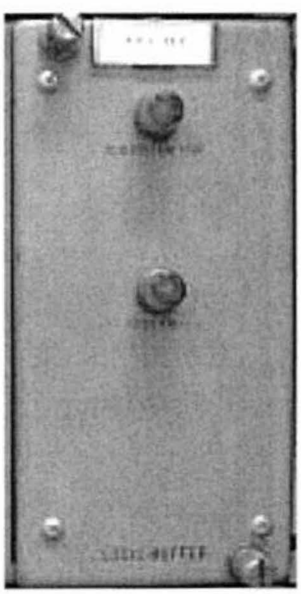


Figure 65.20 Logic Buffer Module

The logic buffer modules provide the communication links between analog and digital subsystems. The links are between analog channel bistable modules and digital channel actuation logic. (Figure 65-20)

There are five logic buffer modules in each of the ESAS analog channels. There is one in each channel for each of the engineered safeguard functions.

HPI and diverse containment isolation

LPI and diverse containment isolation

Reactor building cooling and isolation

Reactor building spray

Reactor building spray chemical addition

Each logic buffer module contains a normally energized trip relay. The trip relay is de-energized by contacts opening in the associated bistable or bistables. The relay closes two contacts when it de-energizes. This applies a signal to trip logic modules in two of the ten ESAS digital channels.

A normally energized module removal/test relay is in the logic buffer modules. (See discussion section on Module-Removal/Module-in-Test Interlock p. 15)

4.6 AUXILIARY RELAYS

Auxiliary relays "fan-out" signals to provide multiple external signals. Each lamp provides indication of the state of an internal relay. Each relay may operate multiple contacts.

Auxiliary relays also serve to electrically isolate ESAS from external circuits.

A common use of auxiliary relays in ESAS and RPS is to provide signals for computer and annunciator alarms

Distracter "C"

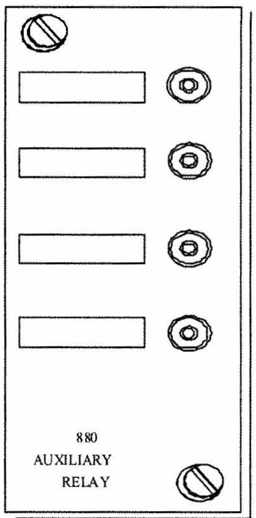


Figure 65-21 Auxiliary Relay - Face Plate

4.7 TRIP LOGIC MODULES

Trip logic modules contain the two-out-of-three coincidence logic circuitry. They also contain circuitry required for testing. (Figure 65-22 Trip Logic Module)

A trip logic module has nine lamps. One of these, the reset light, is normally lit (showing that the reset contact is closed). The lamp next to it lights if one or both of two manual trip relays operate to the manual trip condition. There are two rows of three lamps in each. The top row of (3) lights show which set of coincidence contacts has contacts closed when lit. With one analog channel tripped two lights are lit. The second row of three lamps indicates the status of the trip relays that operate the coincidence logic contacts. They light, indicating relay operation, on associated analog channel trip or when the test switch is depressed during channel testing. The ninth light indicates when the test switch is not in the operate position.

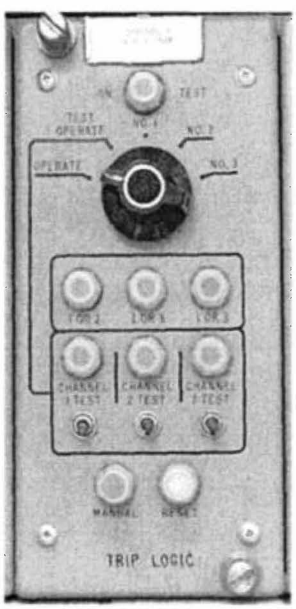


Figure 65.22: Trip Logic Module

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

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

Section: 3.5 **Type:** Containment Integrity

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

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

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

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

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

Question: RO: 42 SRO:

Complete the following statement:

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

STM 1-65, Engineered Safeguards Actuation System

History:

Modified 139 for 2014 Exam
Selected for 2016 exam

4.12.3 Reactor Building Cooling and Isolation

RB isolation and cooling (Channel 5 and 6 is initiated by high Reactor Building pressure of 4 psig, and as its name implies, its function is to isolate and cool the RB. The following equipment is actuated:

- CV-2234, 2235, 2220 and 2221 close to isolate the RC Pump Air/LO and CRD Coolers.
- CV-6205, CV-6202 and CV-6203 close to isolate the RB Chillers.
- The RB Coolers Inlet and Outlet Valves open to VCC 2A, B, C & D (CV-3812, CV-3814 and CV-3813, CV-3815).
- RB Cooling Fan "A", "B", "C" & "D" start and SV-7410, SV-7411, SV-7412 and SV-7413 (RB Bypass Dampers open.
- VEF-38A or B, Penetration Room Fans start.
- CV-2235, CRD Cooling Coil Inlet Isolation Valve closes.
- CV-1065, Quench Tank Cond. Isolation closes.

4.12.4 Reactor Building Spray

Reactor Building Spray and Chemical Addition components are actuated when RB pressure reaches 30 psig. The components actuated are:

- P35A & B RB Spray Pumps start.
- CV-2401 and 2400 RB Spray Blocks open.
- CV-1616 and 1617 open to supply Sodium Hydroxide to the Spray Pumps.

5.0 Technical Specifications

The Technical Specification requirements for the Engineered Safeguards Actuation System are found in:

- 3.5 Instrumentation Systems
  - ◊ 3.5.1 Operational Safety Instrumentation
    - ⇒ 3.5.1.1 Requirements of Table 3.5.1-1
    - ⇒ 3.5.1.2 Number of channels below that required.
  - ◊ Table 3.5.1-1 Instrumentation Limiting Conditions for Operation
  - ◊ 3.5.3 Safety Features Actuation Setpoints

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

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

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

Section: 3.5 **Type:** Containment Integrity

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

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

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

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

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

Question: RO: 43 SRO:

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

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

A. Reactor Building pressure less than 4 psig

B. Both LPI pump flows greater than 2800 gpm

C. NaOH Tank level less than 16 ft

D. Reactor Building sump level dropping

Answer:

D. Reactor Building sump level dropping

Notes:

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

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

References:

1202.010, ESAS

History:

New question for 2016 exam.

ATTACHMENT 1

1. (Continued)

(7) **<u>IF</u>** **<u>no</u>** LPI or LPI/HPI piggyback flow is recovered,
<u>THEN</u> perform the following:

(a) Stop HPI and LPI pumps.

(b) Depressurize RCS below 250 psig
<u>AND</u>
place DHR is service using Decay Heat Removal Operating Procedure (1104.004), "Decay Heat Removal During Cooldown" section, regardless of RCS temp.

d) **<u>IF</u>** both LPI trains were operating,
<u>THEN</u> depressurize RCS below 250 psig
<u>AND</u>
place DHR in service using Decay Heat Removal Operating Procedure (1104.004), "Decay Heat Removal During Cooldown" section, regardless of RCS temp.

2. <u>Check for indications of RB sump blockage as indicated by one or more of the following:</u>

- RB Sump level dropping
- Fluctuations in HPI, LPI or RB Spray parameters below:
  - Discharge press, suct press or flow on dedicated SPDS/PDS displays
  - Flow on C16/C18
  - LPI discharge press, suct press, or motor amps on dedicated PDS/PMS displays
  - Discharge press, suct press, flow or motor amps on the SPDS points listed below:

<u>SPDS Points to Monitor for RB Sump Blockage</u>

	LPI		RB Spray		HPI		
	P34A	**P34B**	**P35A**	**P35B**	**P36A**	**P36B**	**P36C**
Disch Press	P1404	P1405	P2426	P2425	P1241	P1242	P1243
Suct Press	P1407	P1408	P2429	P2428	P1246	P1247	P1248
Flow	F1401	F1402	F2401	F2400	F1228 F1230 F1231 F1232		F1209 F1210 F1211 F1212
Motor Amps	I1A305	I1A405					

ATTACHMENT 1

3. IF RB sump blockage is indicated, THEN perform all of the following:

A. Re-verify suction flowpath properly aligned as follows:

1) Verify the following valves open:

– RB Sump Outlets:

- ♦ CV-1405
- ♦ CV-1406
- ♦ CV-1414
- ♦ CV-1415

– P34A and P34B Suctions From BWST:

- ♦ CV-1436
- ♦ CV-1437

2) Verify BWST T3 Outlets closed:

- CV-1407
- CV-1408

B. Override **AND** throttle Low Pressure Injection (Decay Heat) Blocks to minimum flow listed below:

- CV-1400
- CV-1401

2 LPI pumps	1 LPI pump
≥ 2800 gpm/pump	≥ 3050 gpm

(3. CONTINUED ON NEXT PAGE)

3. (Continued)

C. **IF** both trains of RB Spray are operating, **THEN** perform the following:

1) **IF** there is **no** evidence of Containment breach, **THEN** perform the following:

a) Override **AND** stop both RB Spray pumps:

- P35A
- P35B

b) Override **AND** close both RB Spray Block valves:

- CV-2401
- CV-2400

c) **GO TO step 3.E.**

2) **IF** evidence of Containment breach exists, **THEN** perform the following:

a) Override **AND** stop one RB Spray pump (P35A or P35B).

b) Override **AND** close associated RB Spray Block valve.

P35A	P35B
CV-2401	CV-2400

c) **GO TO step 3.E.**

(3. CONTINUED ON NEXT PAGE)

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

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

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

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

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

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

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

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: 44 SRO:

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

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

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

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

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

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

Answer:

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

Notes:

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

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

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

References:

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

History:

Modified QID 910 (2014) for 2016 exam.

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

QID: 0910 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** New **Originator:** Possage

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

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

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

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

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

Tier:	2	**RO Imp:**	2.7	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

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

Given:
- RCS Temperature 500 °F
- Turbine Bypass Valves being used to control cooldown
- Plant shutdown in progress for 1R25

Per 1102.010, Plant Shutdown and Cooldown, what is the MAXIMUM cooldown rate and what is it based on?

PRINT

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

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

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

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

Answer:

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

Notes:

A is correct per 1102.010 and guidance from Framatome.
B is incorrect but plausible as this is the correct cooldown rate but has the basis for Tech Spec 3.4.3 limits.
C is incorrect but plausible as this has the correct reason but the cooldown rate is from Tech Spec 3.4.3.
D is incorrect but plausible as this is the Tech Spec 3.4.3 limit and the basis from Tech Specs.

References:

1102.010, Plant Shutdown and Cooldown

History:

New for 2014 Exam

ANO1 EOP Step No.	B&W TBD Step No.	Explanation or Basis for Difference
26.	GEOG III.E 9.0, 9.1, 14.0, 18.0	This step begins the RCS cooldown using the emergency cooldown rate if necessary. The reason for use of the emergency cooldown rate for these two criteria (high level and radiation release) is that several large tube leaks and/or a relatively high percentage of failed fuel already exists. Therefore, it is important to reduce RCS temperature as quickly as possible and isolate the affected SG.
27.	GEOG III.E 9.0, 14.0, 18.0	This step begins the RCS cooldown using the normal cooldown rate if the emergency cooldown rate was either not required or has been completed to 500°F T-hot.
28.	GEOG III.E 9.2	This step places P75 in service if available. This step bypasses steps for securing MFW pump until EFW is actuated if P75 is not available.
29.	GEOG III.E 9.2	This step ensures P75 provides adequate flow and secures the MFW pump(s).
30.	GEOG III.E 9.2	This step ensures EFW is off if normal feedwater is available, minimizing thermal stress on SG tubes.
31.	N/A	This step bypasses the step that manually actuates EFW if P75 is available. This step is not considered a deviation. The GEOG has no equivalent step, but this step is necessary due to the structure of the EOP to ensure that appropriate feedwater options are utilized.
32.	GEOG III.E 9.2	This step actuates EFW prior to securing MFW if P75 is unavailable.
33.	GEOG III.E 11.0 VI 1.0	This step provides actions for loss of SCM. NRC Commitment P 7612 This step restores RCP operation when SCM is restored. Forced flow is preferable to natural circulation, especially during Tube Ruptures where an expeditious cooldown is desired. Forced circulation cooldowns prevent void formation and their attendant complications and cooldown delays. In addition, forced circulation cooldowns provide pressurizer spray flow (optimizes RCS pressure control), lower RCS loop ΔTs (allows lower primary to secondary ΔPs which reduces tube leakage) and faster overall cooldown to DHR (minimizes integrated tube leak flow and radiation releases) assuming the condenser is available.

3.3.1.2 Tube-to-Shell ΔT

The normal tube-to-shell ΔT limit for cooldowns is 100°F (tubes colder) and, during an emergency cooldown (3.3.1.1) this limit may be increased to 150°F. Methods to control tube-to-shell ΔT are discussed in Chapter III.G.

This relaxation is allowed to facilitate an emergency cooldown should it be required. However, two important points should be considered:

a. Whenever tube-to-shell ΔT exceeds 100°F a post-transient stress evaluation will be required.

b. Higher tube-to-shell ΔTs will increase the tensile stresses on the tubes and may lead to higher leak flows. Indications of this occurring have been observed during actual tube leak transients.

Therefore, some judgment is required before a decision is made to increase tube-to-shell ΔT. Normally, it is recommended that tube-to-shell ΔT be kept much lower than the normal cooldown ΔT limit if at all possible. However, there may be cases where an increase in ΔT is necessary to accommodate an expeditious cooldown which may be accomplished with little or no risk (e.g., decision has already been made to isolate the affected SG and allow it to fill, thus increases in leak flow rate may not significantly impact the transient). As noted in section 3.3.1.1, the use of the emergency cooldown rate to 500°F should not result in excessive tube-to-shell ΔTs.

3.3.1.3 Cooldown Limits

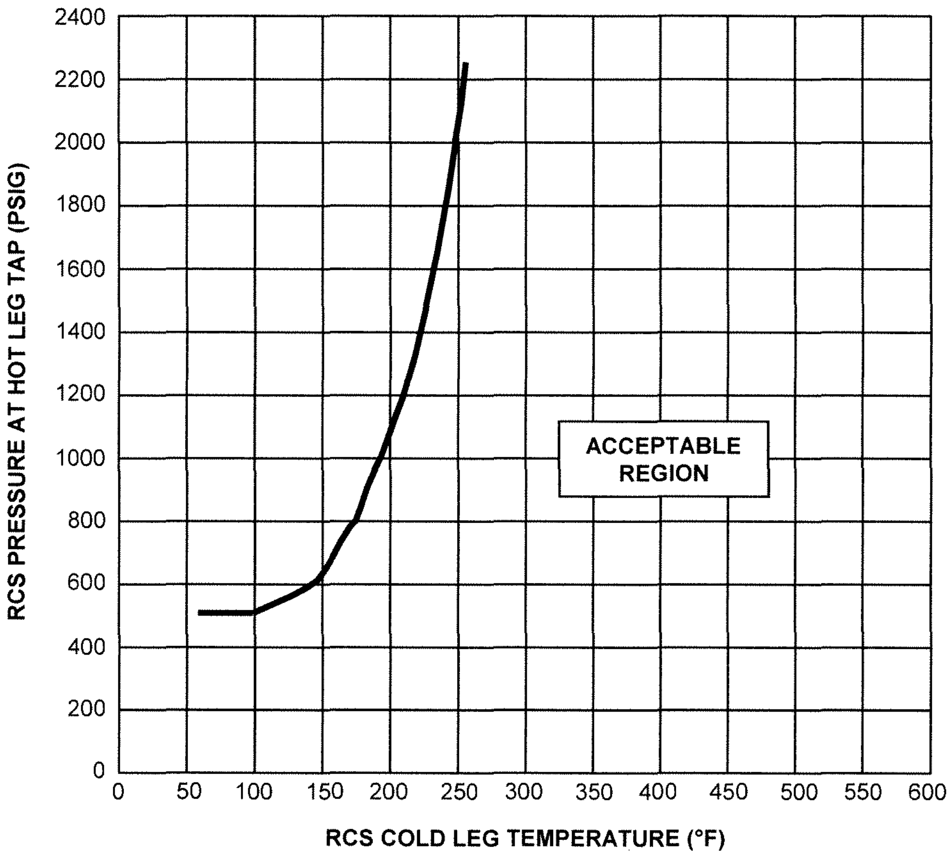
The normal cooldown limit is the Technical Specification limit. With the exception of section 3.3.1.1, this limit should not be exceeded during a plant cooldown when the RCS is subcooled. If the RCS is not subcooled, then this limit does not apply as discussed in Chapter III.B.

3.3.1.4 Summary of Limits During Cooldown

The following limits should be observed, if at all possible:

a. If section 3.3.1.1 applies, then above 500°F the cooldown rate limit is 240°F/hr

FIGURE 3.4.3-2
RCS Cooldown Limits to 54 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25 °F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255 °F	None
150 °F ≤ T ≤ 255 °F	≤ 2
T < 150 °F	No RCPs operating

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280 °F	100 °F/HR	≤ 50 °F in any 1/2 HR
280 °F > T ≥ 150 °F	50 °F/HR	≤ 25 °F in any 1/2 HR
T < 150 °F	25 °F/HR	≤ 25 °F in any 1 HR

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, and unit transients. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 contain P/T limit curves for heatup, cooldown, inservice hydrostatic testing, and physics testing at RCS temperatures ≤ 525 °F, and the maximum rate of change of reactor coolant temperature. The methods and criteria employed to establish operating pressure and temperature limits are described in BAW-10046A (Ref. 1). These limit curves are applicable through fifty-four effective full power years (EFPY) of operation. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for the various operating reactor coolant pump combinations.

Each P/T curve defines an acceptable region for normal operation below and to the right of the limit curve. The curves are used to develop operational guidance for use during heatup or cooldown maneuvering.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel. The vessel is the component most subject to brittle failure due to the fast neutron embrittlement it experiences during power operation, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials. Reference 2 requires an adequate margin to brittle failure during normal operation, abnormalities, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4). For the Linde 80 weld materials present in the ANO-1 reactor vessel beltline, an alternative approach was utilized for determining the adjusted reference nil ductility temperature as described in Topical Report BAW-2308, Revisions 1-A and 2-A (Ref. 12). The Master Curve methodology is accepted with exemption from the requirements of 10 CFR 50.61 (Ref. 13) and 10 CFR 50, Appendix G (Ref.2).

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

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

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

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

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

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

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

Tier:	2	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	An

Question: RO: 45 SRO:

Given:

- 100% power
- ICS in full automatic

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

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

STM 1-64, Integrated Control System

History:

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

action indicates that the neutron power is not able to satisfy its demand. Therefore, by modifying the feedwater demand signal with the neutron error, feedwater is held to within 5% of reactor power. Since the ICS is in Track, the turbine merely controls header pressure and thus the load can be no greater nor less than 5% of the neutron power.

2.6.2 Load Ratio (ΔTc) Control

The total feedwater flow demand signal is split by the ICS into loop "A" and "B" feedwater demand signals by adjustment of the value of a multiplier controller. This controller sets the value of loop "A" feedwater demand by multiplying the total flow demand by the value of the multiplier. If the multiplier is set at .5, half of the total feedwater flow demand signal becomes loop "A" feedwater demand. The loop "B" feedwater demand is determined by subtracting the loop "A" demand from the total demand. Changing the multiplier value will change the value of both loop demand signals. The maximum loop feedwater demand signal is 6 x 10^6 pounds mass per hour.

The value of the multiplier is set by the value of a control signal. This signal is the algebraic summation of two other signals. One of these signals is the RCS flow mismatch signal and will be zero when all four RCP's are properly operating. This signal will be described under "Three Pump Operations". The other signal is the ΔTc correction signal.

The control of the ratio of feedwater to each OTSG will determine the amount of heat that will be removed from the primary water in the reactor coolant system (RCS) and the relative amount of loading that each OTSG will carry. Therefore, the loading of the OTSGs can be indicated by the relative RCS return temperatures to the reactor (Tc's). If the difference in the Tc's (ΔT_c) is controlled near zero, then each OTSG will be loaded properly for the RCS flow through it. A trip of one RCP would give an immediate re-ratioing. An important benefit of keeping ΔTc low is that quadrant tilts within the reactor may be kept to a minimum.

The actual ΔTc is compared to the ΔTc setpoint. The difference (ΔTc Error) is used to generate the ΔTc correction signal. A zero ΔTc correction signal will split the signal equally between the loops.

The operator may choose to manually control the ΔTc correction signal by placing the Load Ratio Hand/Automatic Station in hand. The only difference between this station and the other feedwater hand/auto stations is the additional dial and knob located under the meter. This provides the ΔTc setpoint for automatic operation. The setpoint may be varied from 0% to 100% which corresponds to -10°F to +10°F. The normal value is 50% (0°F).

~~When position~~ is selected on this station, the ΔTc correction signal is indicated on the meter. If the meter indicates 50%, the correction signal is zero (loop "A" multiplier set at .5) and loop demand signals are equal. If the meter indication is above 50%, then loop "A" demand is > loop "B" demand. The opposite is true if the indication is < 50%.

When measured variable is selected on this station, the difference between the actual • Tc and the • Tc setpoint (• Tc error) is indicated on the meter. • Tc = "A" Loop Tc - "B" Loop Tc. The meter scale is ±10•F. Positive reading means that "A" loop is hotter. A bumpless transfer from hand to auto may take place when the • Tc error equals zero (50% on meter). If the • Tc does not equal zero, adjustment to zero may be accomplished by adjusting the manual output of the station or by changing the • Tc setpoint.

If both loop demand stations are placed in hand, this station rejects to hand and can not be placed in auto.

2.6.3 Feedwater Flow Control

The method of flow control used by the feedwater system is dependent upon the plant power level. (refer to figure 64.24) At low power feedwater flow is controlled by the startup and low load control valves with the main feedwater block valve shut and the feedwater pumps operating to maintain 70 psid across the feed valves. The valves are sequenced into operation so that the startup valve opens first followed by the low load control valves then the main FW block valves. As plant load is increased, feedwater flow control will be shifted from the valves to the pumps. This is accomplished by opening the main block valves and controlling the speed of the feedwater pumps to control flow.

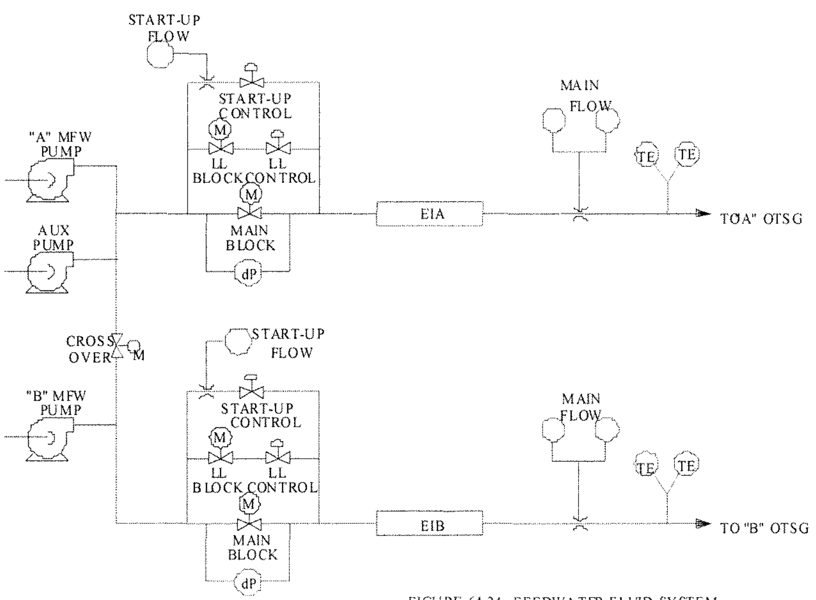


FIGURE 64.24: FEEDWATER FLUID SYSTEM

Below ~2% reactor power, the auxiliary feedwater pump has enough capacity to feed the OTSG's. Above ~2%, an indication that the auxiliary feedwater pump is not able to provide sufficient feedwater is, the startup and low load valves are full open and still not able to maintain OTSG level.

The flow control signal is obtained by comparing loop feedwater demand to loop feedwater flow. Actual flow is provided by the selected main FW flow instrument. The developed flow error signal is applied through proportional plus integral control to establish the valve demand signal for the startup and low load valves. The feedwater pump control signal is developed by having a feedforward control signal created by calculating a ball park base pump speed from each loop demand. The feedforward control signal will provide the coarse adjustment for the feed pump speed. • P or flow error will provide the fine tuning.

With the loop feedwater demand < 50%, the flow to each OTSG is controlled by modulation of startup and low load control valves.

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

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

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

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

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

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

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

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

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

Question: RO: 46 SRO:

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

A. Condensate Storage Tank T-41

B. EFW Condensate Storage Tank T-41B

C. Service Water System Loops I and II

D. ECP via FLEX transfer pump

Answer:

C. Service Water System Loops I and II

Notes:

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

Revised question to eliminate two implausible distractors.

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

References:

1106.006, Emergency Feedwater Pump Operation

History:

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

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

QID: 0269 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-EFW **Objective:** 4 **Point Value:** 1

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

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

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

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

Tier:	2	**RO Imp:**	3.6	**RO Select:**	No	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	K

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

The assured water source to the Emergency Feedwater System is:

a. Main Feedwater

b. Condensate Storage Tank

c. Circulating Water System

d. Service Water System

PREVIOUS REVISION

Answer:

d. Service Water System

Notes:

Service water is the assured source of water to the Emergency Feedwater System, therefore "a", "b", and "c" are incorrect.

References:

1106.006 Rev 059-02-0

History:

Used in 1999 exam.
Direct from ExamBank, QID# 91

Although the normal suction is from the "Q" EFW Condensate Storage Tank (T-41B), the assured source of emergency feedwater is the Service Water system. When T-41B is depleted, the system can be manually aligned to the "non-Q" Condensate Storage Tank (T-41), if available. This arrangement is made in order to normally supply the best quality water available to the steam generators. Switch over to the service water system is made after all condensate is depleted by opening the Service Water Loop Isolations (Loop I - CV-3850, Loop II - CV-3851) and by placing selector switches (HS-2800 & HS-2802) to the SW position. When this is done, the Service Water supply valves (Loop I - CV-2803, Loop II - CV-2806) open and the Condensate Storage Tank (CST) supply valves close (CV-2800 & CV-2802).

The emergency feedwater system is automatically started on the following conditions:

- Low steam generator level
- Low steam generator pressure
- Loss of all Reactor Coolant pumps
- Loss of both main feed pumps
- ESAS channel 3 and 4
- DROPS/AMSAC actuation

Automatic steam generator level control is provided by EFIC to control steam generator level at one of the following setpoints:

- Low level control, 31"
- Natural circulation, 312"
- Reflux boiling, 378"

Steam generator level fill rate is also controlled by the EFIC system for the natural circulation and reflux boiling setpoint to prevent overcooling.

Flow control for the EFW system is provided by 4" modulating solenoid valves:

- EFW P7A to SG-B Control (CV-2647)
- EFW P7B to SG-B Control (CV-2648)
- EFW P7A to SG-A Control (CV-2645)
- EFW P7B to SG-A Control (CV-2646)

These valves receive signals from the EFIC system. The valves are pilot-operated and require system pressure for operation. The valves fail open on a loss of electrical power and cannot be operated locally at the valve. Position indication of the solenoid-operated pilot is provided in the control room. See Emergency Feedwater Initiation and Control (1105.005) for a more detailed description of the EFIC system.

Controllers for the valves may be operated as follows:

- In HAND, the controller can be adjusted to position the valve as desired. Upon receiving an EFW initiate signal (vector enable), the controller automatically switches to AUTO. After a short time delay, the controller may be placed in HAND (even with the initiate signal present) if desired. This condition is annunciated by EFW TRAIN A NOT IN AUTO (K12-D5) or EFW TRAIN B NOT IN AUTO (K12-D6).

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

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

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

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

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

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

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

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

Question: RO: 47 SRO:

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

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

A. 4 to 4.5 "/min

B. 6 to 6.5 "/min

C. 7 to 7.5 "/min

D. 7.6 to 8 "/min

Answer:

C. 7 to 7.5 "/min

Notes:

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

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

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

References:

1105.005, Emergency Feedwater Initiation and Control

History:

Modified QID 270 for 2016 exam

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

QID: 0270 **Rev:** 1 **Rev Date:** 11/8/05 **Source:** Direct **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:**	No	**Difficulty:**	2.5
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	Ap

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

The EFIC automatic fill rate is designed to prevent overcooling.

With the plant in a degraded power condition and given a SG pressure of 885 psig.

Determine the proper OTSG fill rate by EFIC for the EFW system:

Parent ////

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. 032

History:

Used in 1999 exam.
Direct from ExamBank, QID# 92 used in class exam
Selected for 2005 RO re-exam.
Selected for 2010 RO/SRO exam

6.0 SETPOINTS

6.1 Initiation Setpoints

- EFW low level initiate ~ 11", delayed 9.9 seconds
- MSLI and EFW initiate on low SG pressure ~ 600 psig
- Loss of both MFW pumps with reactor power >7%
- ESAS Channel 3 or Channel 4 trip
- MFW Flow in both loops <15% with reactor power >45%. (AMSAC)
- All RCPs OFF (May be bypassed at <10% Power)

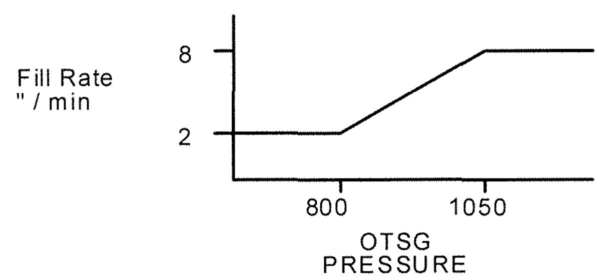
6.2 Control Setpoints

6.2.1 SG level

- Low level control ~ 31"
- Natural circulation control ~ 312"
- Reflux boiling control ~ 378"

6.2.2 Rate of SG level rise when RCPs are off is variable from 2"/minute at 800 psig to 8"/minute at 1050 psig See figure 6.2.2.

Figure 6.2.2 – SG Fill Rate with RCPs Off



6.2.3 SG ΔP ~ 100 psi determines good (unaffected) SG to allow EFW flow and isolates bad (affected) SG on MSLI actuation.

6.2.4 Atmospheric dump control valves will control SG pressure at ~ 1020 psig at all times if not isolated.

6.3 Low condenser vacuum interlock opens atmospheric dump isolation valves at ~ 21" Hg.

6.4 MSLI actuation opens affected SG atmospheric dump isolation valve.

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

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

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

Section: 3.6 **Type:** Electrical

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

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

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

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

Question: RO: 48 SRO:

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

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

A. Startup #2 Transformer Lockout

B. Main Turbine Lockout

C. Main Generator Lockout

D. Main Generator Backup Reverse Power

Answer:

C. Main Generator Lockout

Notes:

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

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

Revised stem per NRC examiner suggestion.

References:

STM 1-32, Electrical Distribution

History:

New question for 2016 exam

3.3 BREAKER LOGIC

3.3.1 Unit Auxiliary Feeder (A-112, A-212, H-14, H-24)

(Refer to Figure 32.62)

The unit auxiliary 6900V and 4160V breakers supply busses H1, H2, A1, A2 during power operations from the output of the main generator via the Unit Auxiliary Transformer. There is no auto close function for these breakers. The breaker may be closed from the control room as long as:

- "Remote" selected (front of breaker)
- SYNC select switch "On" (on C10)
- No main generator L.O. relay trip (286-G1-2)
- No bus L.O. relay trip (186-A1/A2/H1/H2)

The Unit Auxiliary feeder breakers to A and H buses will trip when:

- *6900/4160V bus locks out (186-A1/A2/H1/H2)
- *Main generator lock out (286-G1-2)
- 4160V A1 bus only ES ch. 1 (ESX-A3)
- 4160V A2 bus only ES ch. 2 (ESX-A4)
- 6900/4160V undervoltage (127-A1/A2/H1/H2)

If selected to remote, the breaker will trip on manual transfer to SU-1 or SU-2 when:

- SU-1/SU-2 (C10) handswitch in "normal after" position
- SU-1/SU-2 synch selector switch "On"
- SU-1/SU-2 feeder breaker to bus closed (152-113 or 152-213 "a" contact)

*Circuit contains a test switch to defeat protective function.

3.3.2 SU1 Feeder Breaker (A-113, A-213, H-15, H-25)

(Refer to Figure 32.63)

SU1 transformer is the normal supply to A and H buses when the main generator is *not* operating. A and H buses may be supplied from SU-1 (either automatically or manually) provided the transformer is available. The SU-1 transformer is available if:

- Breaker control selected to remote (front of breaker)
- Normal voltage on SU-1 secondary (127-113/213/15/25 relay NOT tripped)
- NO SU1 lock out relay tripped (186-ST1-2)
- The breaker control switch on C-10 is NOT in pull-to-lock.

With the transformer available and the bus lockout relay NOT tripped, the A and H buses may be manually supplied from SU-1 transformer by turning the sync selector switch "on" and placing the C-10 control switch to close.

The A and H buses will automatically transfer to SU-1 (SU-1 bus feeder breaker closes). This transfer can be either **"fast"** if the buses are in synchronization with each other (within 5-10 cycles, .085 to .17 seconds) or **"slow"** if the bus is de-energized as indicated by trip of its undervoltage relay.

If the main generator lock-out relay trips with SU-2 feeder breaker open or SU-2 lock-out relay trips with the Unit Aux feeder breaker open, then this ***automatic*** transfer will occur if the following conditions are also met:

- SU-1 must be available as described above,
- No associated bus lock outs,
- SU-1 selected or SU-2 not available,
- Synch check relay sensing SU-1 and associated bus are in phase (**"fast"** transfer), or
- Associated bus undervoltage relay tripped (**"slow"** transfer).

Note: Again, for **either** a fast or slow transfer to take place the main generator lockout relay has to be tripped with the SU2 feeder breaker open **or** SU2 lockout relay tripped with the Unit Aux feeder breaker open.

The SU-1 transformer feeder breakers to the A and H buses will trip when:

- *SU-1 transformer secondary output undervoltage
- *Associated bus lock out relay tripped
- *SU-1 transformer lock out relay tripped

If selected to remote, the breaker will trip on manual transfer to Unit Aux or SU-2 when:

- Unit Aux/SU-2 (C10) hand switch in normal after position
- Unit Aux/SU-2 synch selector switch on
- Unit Aux/SU-2 feeder breaker to bus closed

*Circuits contain a test switch to defeat protective function.

So you see for the A1/A2/H1/H2 busses that a Main Generator Lockout will send a signal to trip the Unit Aux feeder breaker at the same time it sends a signal to the Startup feeder breakers to close (which one closes depends on the selector switch on C10). The key to the fast transfer is the sync switch contacts which will maintain a sync permissive for approximately 20 cycles (0.34 seconds). The **"fast"** transfer should occur within 5 to 10 cycles but if it doesn't, then the **"slow"** transfer will occur after the bus is de-energized (the "27" or undervoltage relay trips). If a Main Generator Lockout signal is not present, such as in the case of an inadvertent ES actuation, then neither fast nor slow transfer will occur, the ES will trip the Unit Aux feeder, the affected bus will be de-energized and must be manually re-energized.

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

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

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

Section: 3.6 **Type:** Electrical

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

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

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

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 49 SRO:

Initial conditions:

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

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

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

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

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

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

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

Answer:

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

Notes:

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

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

References:

1203.026, Loss of Reactor Coolant Makeup

History:

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

INSTRUCTIONS

SECTION 1 -- LOSS OF HPI PUMP

NOTE

Indications of loss of HPI suction are:

- Erratic flow, and
- Erratic discharge pressure, and
- Control valves stable

1. **IF HPI pump has lost suction,**
   THEN stop the HPI pump.

2. **Isolate letdown by performing one of the following:**

   - Close Letdown Coolers Outlet (CV-1221)
   - Close both of the following on C18:
     - Letdown Coolers Outlet (RCS) (CV-1214)
     - Letdown Coolers Outlet (RCS) (CV-1216)

NOTE

- With HPI pump off, ICW cooling of RCP seals should provide adequate time to correct HPI pump or control problems, providing no pre-condition exists, such as excessive RCP shaft sleeve leakage. HPI can provide necessary makeup for normal operations or plant shutdown.
- Reactor Coolant Pump and Motor Emergency (1203.031), Attachment A can be used as an aid to assess seal parameters.

3. **Verify RC pump seals are being cooled by ICW.**

   A. IF ICW to RCP seals is NOT available,
   THEN perform Reactor Coolant Pump and Motor Emergency (1203.031), "Simultaneous Loss of Seal Injection and Seal Cooling Flow" section.

4. **Prepare to restart an HPI pump as follows:**

   A. IF OP HPI pump is unavailable
   AND STBY HPI pump is unavailable,
   THEN dispatch an operator to re-align the ES HPI pump per Attachment A of this procedure.

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

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

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

Section: 3.6 **Type:** Electrical

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

Description: Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights.

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

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	An

Question: RO: 50 SRO:

PHOTO ON FOLLOWING PAGE

A malfunction of the red train Vital 125V DC electrical system has occurred.

Using the attached photograph, determine which of the following local alarms would accompany the indications shown:

A. Local annunciator for D01, "BLOWN FUSE"

B. Local annunciator for Charger D03A, "DC OUTPUT BREAKER OPEN"

C. Local annunciator for D01, "BATTERY DISCONNECT OPEN"

D. Local annunciator for Charger D03A, "HIGH DC FLOAT VOLTAGE"

Answer:

B. Local annunciator for Charger D03A, "DC OUTPUT BREAKER OPEN"

Notes:

"B" is correct. The photo shows that D-01 Amps are approximately 100. Normally, the amps on the battery are "zero" indicating that the battery charger is carrying the load and the battery is in standby. The amp meter is indicative of the battery automatically picking up the load due to failure of the battery charger which would be indicated by the local alarm "DC Output Breaker Open".
"A" is incorrect but plausible if the candidate does not know what this alarm means and believes the photo is indicative of the battery charger carrying the load vs. the battery. The blown fuse this alarm refers to is the one between the D07 battery and bus D01.
"C" is incorrect but plausible if the candidate believes the photo is indicative of the battery charger carrying the load vs. the battery which would be the case if the battery disconnect were open.
"D" is incorrect but plausible if the candidate believes that the voltage indicated is abnormal. The system is called 125 volt but with the battery charger connected it is normally 130 volts.

References:

E-17, Red Train Vital AC and 125V DC Single Line and Distribution
1203.012, Annunciator K01 Corrective Action

PHOTO (separate file) MUST FOLLOW THIS QUESTION ON THE EXAM!!!!!!

History:

New question for 2016 exam

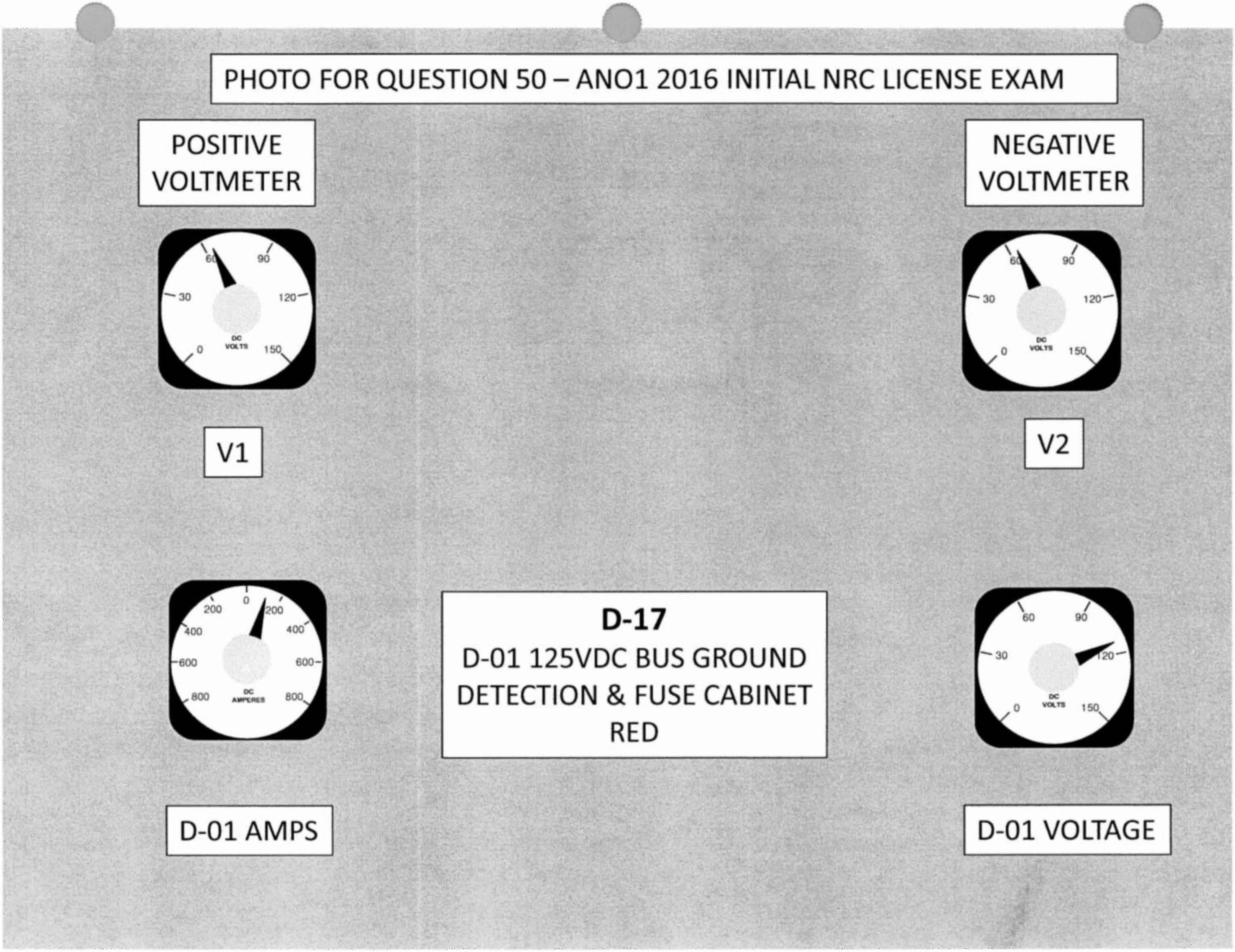


PHOTO FOR QUESTION 50 – ANO1 2016 INITIAL NRC LICENSE EXAM
POSITIVE VOLTMETER
NEGATIVE VOLTMETER
0 30 60 90 120 150
DC VOLTS
V1
V2
200 0 200 400 400 600 600 800 800
DC AMPERES
D-17
D-01 125VDC BUS GROUND DETECTION & FUSE CABINET RED
D-01 AMPS
D-01 VOLTAGE

Location: C10

Device and Setpoint:

N/A

D01
CHARGER
TROUBLE

Alarm: K01-E7

NOTE

This annunciator has multiple inputs with reflash capability.

1.0 OPERATOR ACTIONS

1. Determine which Battery Charger (D-03A or D-03B) is supplying bus D01.

2. Dispatch Operator to local alarm panel (K1650 or K1651) on the battery charger to determine cause of alarm.

3. Verify Alarm To Control Room toggle switch for idle charger is OFF AND for in-service charger is ON.

4. For AC power failure, check breaker B-5145 for Battery Charger D-03A or breaker B-5733 for Battery Charger D-03B.

5. Either attempt to restore battery charger to service OR place other charger into service on bus D01 using "Battery Charger Operation" section of Battery and 125V DC Distribution (1107.004).

6. IF a battery charger is not supplying the battery,
THEN notify electricians to monitor battery for operability once an hour until charger is in-service AND the battery is operable.

7. IF applicable,
THEN initiate steps to have the failed charger repaired.

8. IF alarm is due to LOW CURRENT LIMIT SELECTED,
THEN return the current limit setting toggle switch inside the charger cabinet to the normal position. It is not expected that the low current limit be used.

9. Reference TS 3.8.4, TS 3.8.5, TS 3.8.9 and TS 3.8.10.

2.0 PROBABLE CAUSES

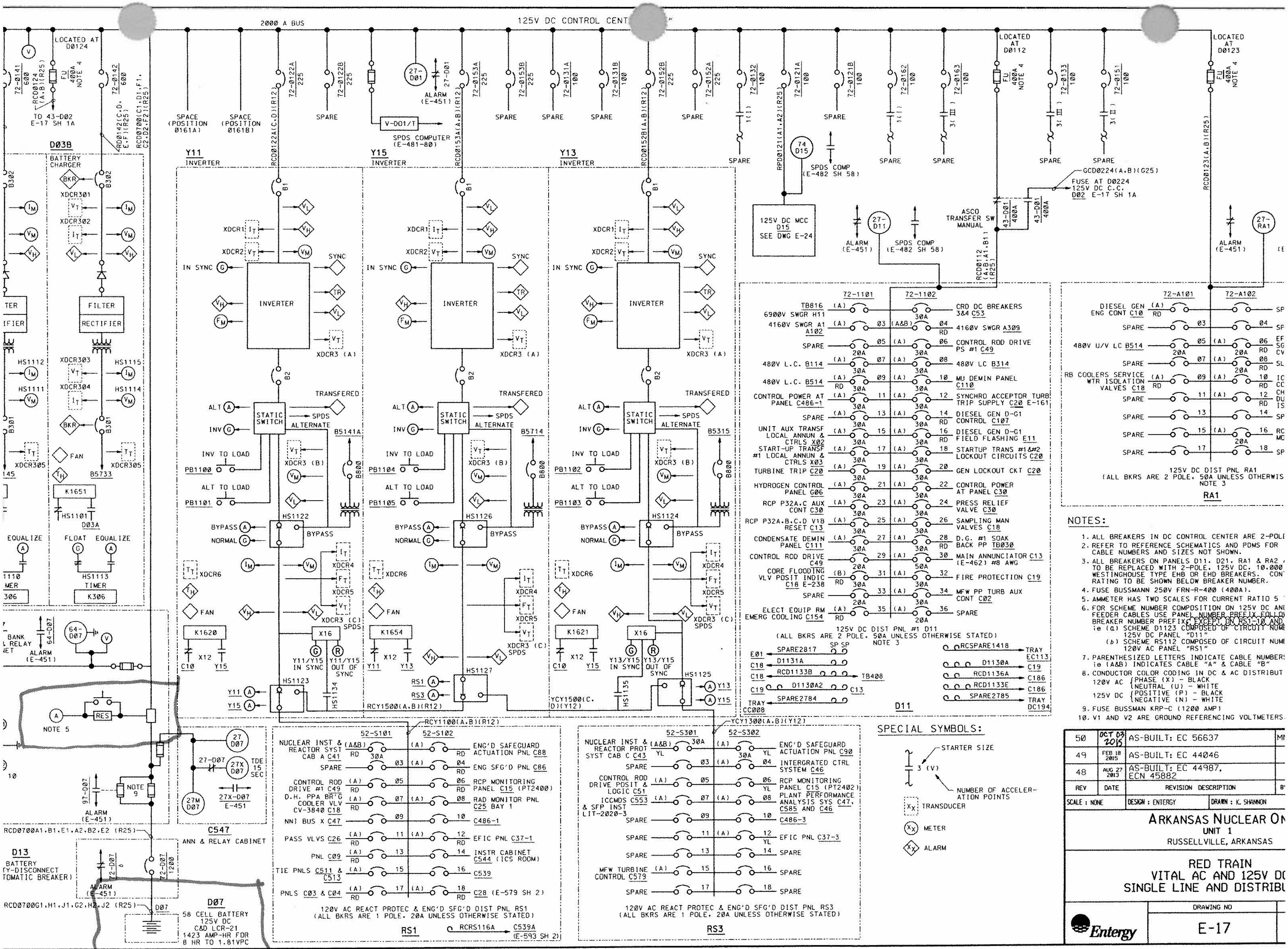
NOTE

Low float and high float voltage alarms are local to K1650 or K1651 only.

- Fan failure
- AC power failure
- DC Output breaker open
- Low current limit selected

3.0 REFERENCES

Schematic Diagram Annunciator K01 (E-451)



NOTES:

1. ALL BREAKERS IN DC CONTROL CENTER ARE 2-POLE
2. REFER TO REFERENCE SCHEMATICS AND PDMS FOR CABLE NUMBERS AND SIZES NOT SHOWN.
3. ALL BREAKERS ON PANELS D11, D21, RA1 & RA2 TO BE REPLACED WITH 2-POLE, 125V DC, 10,000 WESTINGHOUSE TYPE EHB OR EHD BREAKERS. CON RATING TO BE SHOWN BELOW BREAKER NUMBER.
4. FUSE BUSSMANN 250V FRN-R-400 (400A).
5. AMMETER HAS TWO SCALES FOR CURRENT RATIO 5
6. FOR SCHEME NUMBER COMPOSITION ON 125V DC AND FEEDER CABLES USE PANEL NUMBER PREFIX FOLLO BREAKER NUMBER PREFIX EXCEPT ON RS1-10 AND ie (a) SCHEME D1123 COMPOSED OF CIRCUIT NUM 125V DC PANEL "D11" (b) SCHEME RS112 COMPOSED OF CIRCUIT NUMB 120V AC PANEL "RS1"
7. PARENTHESIZED LETTERS INDICATE CABLE NUMBERS ie (A&B) INDICATES CABLE "A" & CABLE "B"
8. CONDUCTOR COLOR CODING IN DC & AC DISTRIBUT 120V AC {PHASE (X) - BLACK, NEUTRAL (U) - WHITE} 125V DC {POSITIVE (P) - BLACK, NEGATIVE (N) - WHITE}
9. FUSE BUSSMAN KRP-C (1200 AMP)
10. V1 AND V2 ARE GROUND REFERENCING VOLTMETERS.

REV	DATE	REVISION DESCRIPTION
50	OCT 03 2015	AS-BUILT: EC 56637
49	FEB 18 2015	AS-BUILT: EC 44046
48	AUG 27 2013	AS-BUILT: EC 44987, ECN 45882

SCALE : NONE | DESIGN : ENTERGY | DRAWN : K. SHANNON

ARKANSAS NUCLEAR ON
UNIT 1
RUSSELLVILLE, ARKANSAS

RED TRAIN
VITAL AC AND 125V DC
SINGLE LINE AND DISTRIBU

Entergy

DRAWING NO
E-17

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

QID: 0792 **Rev:** 1 **Rev Date:** 7/26/16 **Source:** Bank **Originator:** S. Pullin

TUOI: A1LP-RO-EDG **Objective:** 19 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators (ED/G)

Description: Knowledge of the physical connections and / or cause-effect relationships between the ED/G system and the following systems: Starting air system.

K/A Number: K1.05 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

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

Question: RO: 51 SRO:

Given:

- Plant at 100%
- CBOT is performing #1 EDG monthly surveillance per 1104.036 Supplement 1.

The CBOT presses the start pushbutton on C10.
A short time later annunciator K01-B2 "EDG 1 OVERCRANK" alarms

What is the cause of the alarm and how long did the starting air system attempt to start the engine?

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

B. #1 EDG did not exceed 300 rpm in 8 seconds and air start motors engaged for 2.5 seconds.

C. #1 EDG did not exceed 30 rpm in 15 seconds and air start motors engaged for 8 seconds.

D. #1 EDG did not exceed 30 rpm in 15 seconds and air start motors engaged for 2.5 seconds.

Answer:

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

Notes:

A is correct, following a start signal one bank of the air start system will engage and crank the engine. If the EDG does not reach 30 rpm in 2.5 seconds, then that air start system is disengaged and the other engaged (it will crank for 2.5 seconds). If the EDG does not achieve 30 rpm after 8 seconds, then all cranking is stopped. A timer will cause the Overcrank alarm if the EDG does not achieve 300 rpm in 45 seconds. This 45 seconds allows time for the EDG to continue attempting to start in case it was sputtering and trying to start.
"B" is incorrect but plausible since this is a combination of the correct RPM for the overcrank, the 8 second time limit for cranking, and the time limit for one bank.
"C" is incorrect but plausible since this is a combination of the RPM for the air start timer, the time required by Tech Specs for EDG start, and the time limit for cranking.
"D" is incorrect but plausible since this is a combination of the RPM for the air start timer, the time required by Tech Specs for EDG start, and the time limit for one bank.

This question matches the K/A as it requires the candidate to recall the relationship between the EDG and the starting air system (times and sequence of air start motors and banks).

Revised question per validator comment that both A and C could be correct. Revised C and D. JWC 7/26/16

References:

STM 1-31, Emergency Diesel Generators
1203.012A, Annunciator K01 Corrective Action

History:

New 2010 RO/SRO exam
Selected for 2016 exam

Location: C10

Device and Setpoint:

EDG 1
OVERCRANK

Alarm: K01-B2

1.0 OPERATOR ACTIONS

1. Place DG1 lockout switch in LOCKOUT position.

2. Reference TS 3.8.1, TS 3.8.2 and TS 3.8.3 for operability requirements.

3. Initiate action to determine cause of over-crank.

4. Operate fuel oil priming pump and verify return-fuel sightglass (sightglass nearest the engine) is full.

5. WHEN cause of over-crank is corrected,
THEN prove DG1 operable using Emergency Diesel Generator Operation (1104.036), Supplement 1.

6. IF DG1 inoperable,
THEN verify proper MOD alignment for Service Water Pump (P-4B) and Makeup Pump (P-36B) per Makeup & Purification System Operation (1104.002) AND Service Water and Auxiliary Cooling System (1104.029).

7. Alarm may be cleared by ANY of the following methods:

   - Place DG1 lockout switch in LOCKOUT position
   - Depress local RESET button
   - Place Local/Maint/Remote switch in MAINT
   - Place DG1 Output (A-308) in PULL-TO-LOCK

2.0 PROBABLE CAUSES

- DG1 did not reach minimum speed within 45 seconds
- Loss of fuel oil pump prime

3.0 REFERENCES

- TS 3.8.1, TS 3.8.2 and TS 3.8.3
- Schematic Diagram Annunciator K01 (E-451)
- Schematic Diagram Diesel Generator Engine Control (E-102)

Pressure relief valves are installed on the starting air compressor outlet and on the air receiver tanks. The setpoints of the relief valves are 250 and 225 psig, respectively.

3.6 OPERATION

3.6.1 Normal Operation

When an EDG start signal is received, the engine control circuits energize the air start solenoid. If the engine does not reach 30 RPM within 2.5 seconds, the engine control circuits will switch to the other air start system. This alternation will continue for 8 seconds. The starter motors are disengaged when the engine reaches 300 RPM, main oil pump discharge pressure reaches 20 psig, or 8 seconds has elapsed. The receiver tank pressure is maintained by the starting air compressor.

3.6.2 Abnormal Operation

Each redundant starting air system is capable of starting the EDG five times. A loss of one of the systems would not prevent starting the EDG.

The air start solenoid can be overridden to start the EDG in the event of a loss of DC power. To override the air start solenoid, depress the "T" handle on the solenoid.

4.0 FUEL OIL SYSTEM

4.1 SYSTEM FUNCTION

The fuel oil system stores and transfers fuel oil to the EDG. At the engine, equipment is provided to filter and inject the fuel oil into the engine cylinders.

4.2 DESIGN BASIS

The emergency diesel fuel tanks and day tanks are located in seismic class 1 structures of plant buildings. The emergency diesel fuel tanks are located in the underground fuel oil vault. The fuel oil day tanks are located in the EDG skids.

The T.S. minimum volume of fuel oil supplied in one emergency diesel fuel tank and day tank is sufficient for 3.5 days of operation. Enough time should be available to provide additional fuel oil for continued EDG operation.

4.3 SYSTEM DESCRIPTION

4.3.1 Description

The EDG fuel oil system is comprised of two systems: the fuel oil supply and engine fuel oil system. The fuel oil supply system consists of the equipment necessary to store and transfer fuel oil to the engine. The engine fuel oil system consists of the equipment necessary to deliver the fuel oil to the cylinders.

4.3.2 Flowpaths

(Refer to figure 10 for layout, figures 8 & 9 for flowpaths)

The emergency diesel fuel tanks (T-57) are gravity filled from the bulk fuel oil storage tank. The fuel oil is filtered by F-27. The emergency diesel fuel oil transfer pumps take suction on the emergency diesel fuel tanks and transfer fuel oil to the emergency

***NOTE:** These percentages of motor rated voltage are nominal. Refer to the Unit 1 Technical Specifications for range of permissible values.

When a start signal is received, the engine start pilot relay (K-8) is energized. An auto-start signal will also energize the engine auto-start relay (K-21). The engine auto-start relay starts the fuel oil priming pump and gives the EDG auto-start annunciator in the control room. The engine start pilot relay energizes the governor run solenoid (allows the governor to position the fuel racks to deliver fuel oil to the engine cylinders), starts the governor boost motor (provides high pressure oil for governor operation), and energizes the air start motor circuitry.

The air start motor circuitry consists of an alternator, a speed switch assembly, and time delay relays to disengage the air start motors and annunciate the failure of the EDG to start.

The alternator will energize the air start solenoid valve for air start system #1. The #1 air start system air start motors will begin cranking the engine. The engine speed should be >30 RPM in 2.5 seconds. If engine speed is at least 30 RPM, air start system #1 will stay in service. When the next start signal is received, the alternator will energize the air start solenoid valve for air start system #2.

If engine speed remains less than 30 RPM for 2.5 seconds, the alternator circuit will shift to air start system #2. The air start solenoid valve for air start system #2 is energized. The #2 air start system air start motors will begin cranking the engine. Alternation between the two air start systems will continue until engine speed is >30 RPM.

The air start motor circuit uses two parameters to indicate that the engine is running. When engine speed >300 RPM or the main oil pump discharge pressure is >20 psig, the air start solenoid valve is de-energized and the air start motors will disengage. When engine speed >300 RPM, the engine run relay energizes. The engine run relay gives the run indication in the control room, and de-energizes the governor boost motor and fuel oil priming pump.

In the event that the engine fails to start, two time delay relays are used to stop engine cranking and to annunciate the failure of the EDG to start. An eight second and a 45 second time delay relay are energized when the start signal is received. After the eight seconds, the air start solenoid valve is de-energized and engine cranking stops. After 45 seconds, the EDG overcrank alarm is annunciated in the control room. These relays are blocked when engine speed reaches 300 RPM. The engine reset switch resets the start logic.

The EDG will tie on to the respective bus if:

- the respective bus voltage is <75%
- the normal supply breaker (A309/A409) is open
- one of the cross tie breakers is open
- EDG voltage is normal
- the EDG output breaker is not in pull-to-lock

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

QID: 1065 **Rev:** 1 **Rev Date:** 7/13/16 **Source:** New **Originator:** Cork

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

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring

Description: Knowledge of the operational implications of the following concepts as they apply to the PRM system: Radiation theory, including sources, types, units, and effects.

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

Tier:	2	**RO Imp:**	2.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.0	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 52 **SRO:**

Given:
- Plant heatup is in progress per 1102.002, Plant Startup.
- RCS Tcold is ~480°F.
- RCS pressure 2150 psig.
- Fourth RCP was started an hour ago.

The Process Monitor Radiation High annunciator alarms.
The plant computer indicates the failed fuel ratio has dropped from 21.49 to 12.72.

What is the cause of this alarm and what operational implication does this have?

A. Crud burst from starting the fourth RCP is releasing activated iron and nickel isotopes, letdown flow must be raised to increase RCS filtration.

B. Iodine portion of the failed fuel detector is failing low, a mode change is not allowed.

C. In-service letdown demineralizer is exhausted, and must be swapped.

D. RCS activity due to release of fission products is rising, a reactor startup may not commence.

Answer:

D. RCS activity due to release of fission products is rising, a reactor startup may not commence.

Notes:

"D" is correct, a marked drop in gross/iodine ratio (failed fuel ratio) indicates a rise in fission products in the RCS. As the iodine portion of the failed fuel monitor's output rises, this is compared with gross activity (all activity) and the gross to iodine ratio gets smaller (both rise but the Iodine rises by a larger percentage). This indicates the amount of fission fragments such as I-131 is rising in the RCS, an indicator of failed fuel.
"A" is incorrect but plausible in that starting RCPs often produces crud bursts but this will result in the gross activity rising due to activated corrosion products and this will cause the failed fuel ratio to rise, not lower.
"B" is incorrect but plausible as this will result in the failed fuel ratio changing but again this will cause the failed fuel ration to rise, not lower.
"C" is incorrect but plausible as letdown demineralizers are often changed based on activity but this would be an activity differential across the demineralizer, not the RCS as a whole.

This question matches the K/A since the failed fuel ratio comes from the Failed Fuel Monitor, a process rad monitor on letdown. The question asks for an operational implication and the implication is that failed fuel is present due to the type of activity being seen on the failed fuel monitor.

Revised based upon NRC examiner suggestion.

References:

1203.019, High Activity in Reactor Coolant

History:

New question for 2016 exam.

SECTION 2
FAILED FUEL

1.0 SYMPTOMS

1.1 PROC MONITOR RADIATION HI (K10-B2) alarm.

1.2 HIGH ALARM on Failed Fuel Iodine (RI-1237S) monitor.

1.3 Marked drop in gross/iodine ratio.

1.4 RCS sampling indicates rise in fission products.

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

NOTE
Selecting PDO will initiate a lengthy report that is used by Reactor Engineering. Selecting this option multiple times has the potential to lose the original report.

3.1 IF Plant Computer is available,
THEN from NASP menu initiate Plant Data Output program by selecting PDO.

NOTE
The following points are available on the Plant Computer:

- Failed Fuel Gross - R1237
- Failed Fuel Iodine - R1237S
- Calculated Failed Fuel Gross/Iodine Ratio - R1237R

3.2 IF failed fuel ratio drops by 40% as indicated by WCO Logsheet (OPS-A3) or Plant Computer,
THEN reduce reactor power by 50% of present power level as follows:

3.2.1 Commence power reduction per Power Reduction and Plant Shutdown (1102.016) using applicable section(s).

3.2.2 Contact the duty Reactor Engineer and Operations Manager.

3.3 Instruct Chemistry to sample based on the following:

3.3.1 IF letdown in service,
THEN obtain high pressure letdown sample (preferred).

3.3.2 IF letdown NOT in service,
THEN obtain RCS Hot Leg sample.

QID: 1079 **Rev:** 0 **Rev Date:** 4/26/16 **Source:** Bank **Originator:** Cork

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

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

System Number: 076 **System Title:** Service Water System (SWS)

Description: Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Automatic start features associated with SWS pump controls.

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

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

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

Question: RO: 53 SRO:

The plant is operating at 100% with the following conditions:
- P-4A and P-4C SW pumps running
- P-4B SW pump is aligned to A3 but is tagged out for bay maintenance.
- SW valve alignment is normal otherwise.
- B55/56 is aligned to B5.

Subsequently, ESAS actuates on low RCS pressure with a concurrent Loss of Offsite Power.

#2 EDG fails to start.

What will the service water pump alignment be?

A. P-4A to P-4B crosstie valves CV-3644 & CV-3646 CLOSED;
P-4C to P-4B crosstie valves CV-3640 & CV-3642 OPEN;
ACW isolation CV-3643 CLOSED.

B. P-4A to P-4B crosstie valves CV-3644 & CV-3646 OPEN;
P-4C to P-4B crosstie valves CV-3640 & CV-3642 CLOSED;
ACW isolation CV-3643 OPEN.

C. P-4A to P-4B crosstie valves CV-3644 & CV-3646 OPEN;
P-4C to P-4B crosstie valves CV-3640 CLOSED & CV-3642 OPEN;
ACW isolation CV-3643 OPEN.

D. P-4A to P-4B crosstie valves CV-3644 OPEN & CV-3646 CLOSED;
P-4C to P-4B crosstie valves CV-3640 CLOSED & CV-3642 OPEN;
ACW isolation CV-3643 CLOSED.

Answer:

D. P-4A to P-4B crosstie valves CV-3644 OPEN & CV-3646 CLOSED;
P-4C to P-4B crosstie valves CV-3640 CLOSED & CV-3642 OPEN;
ACW isolation CV-3643 CLOSED.

Notes:

"D" is correct, with ESAS, no offsite power, and failure of #2 EDG the green train crosstie valves will remain open (all of the valves mentioned are open at the beginning of the event) and the red train crosstie valves will close to ensure train separation (because they have power). All of the crosstie valves re-position (or not) based upon which SW pump is running. B55/56 is aligned to B5 (red train) so the ACW isolation will close on ESAS as designed to ensure SW flow goes to ESF components and the sole SW pump P-4A is not operating in a runout condition.
"A", "B", and "C" are all alternate combinations which are plausible since they would be correct for different combinations of SW pumps running and different alignment of B55/56, but they are incorrect for this set of conditions.

This question matches the K/A due to the components are all SW components and the question asks if the candidate understands how the crosstie valves and ACW isolation will atuomatically align based upon which SW pump auto-starts (auto start features).

References:

1104.029, Service Water & Auxiliary Cooling Water
STM 1-42, Service & Auxiliary Cooling Water

History:

Selected regular exam bank question QID ANO-OPS1-05891a for 2016 exam a door hinge with a spring assembly that maintains the flappers separated during normal flow. ER-ANO-2004-0321-000 removed the rubber seats that were originally installed in the SW pump check valves (SW-1A, B, & C).

2.3.7 SW Pump Discharge Valves

(Refer to Figure 42.01)

Each SW pump is provided with a manually operated, 18-inch butterfly valve used to isolate the SW pump. The SW pump discharge valves are "Category E" controlled valves, normally locked open. The discharge valves are designated as SW-2A, SW-2B, and SW-2C.

2.3.8 SW Crosstie Valves

(Refer to Figures 42.01 & 42.06 - 42.09 and Table 42.3)

The crosstie valves in the service water pump's discharges are used to permit 100% operation of both service water loops and the auxiliary cooling water (ACW) loop with any two of the three SW pumps running. The crosstie valves are Enertech, 18 inch, motor-operated, triple offset, rotary disk valves having a metal to metal seating surface. SW crosstie valves are located on the first floor of the intake structure. The valve operators for each valve are located on the second floor of the Intake Structure, which ensures valve operation if flooding occurs.

The four SW crosstie valves are controlled by handswitches located on panel C-16 or C-18. To ensure SW systems are isolated during an ESAS actuation with a loss of offsite power and failure of one of the EDG's to start, the four cross-ties are powered in a "Red / Green" manner. This means that one of the two crossties for P-4A to P-4B is red powered (EDG1) and the other crosstie is green powered (EDG2). The same is true for the P-4B to P-4C crossties. In this condition the SW pump associated with the operable diesel would be isolated from the opposite system by its diesel-supplied crosstie.

Table provided below contains crosstie HS location, type of controls and power supply breaker for each valve.

Equip ID	Component	Power Supply	HS #	HS Location	Remarks
CV-3640	P4B to P4C	B-5223	3647	C-18	(1)
CV-3642	P4B to P4C	B-6224	3642	C-16	(1)
CV-3644	P4A to P4B	B-6223	3644	C-16	(1)
CV-3646	P4A to P4B	B-5224	3646	C-18	(1)
(1) Handswitch: Open/Close, spring return to center. Indication above HS.					

Prior to 1R12 the SW crossties valves were not configured as described above. Both crosstie valves for P-4A to P-4B were red powered and P-4B to P-4C crossties were both green powered. This condition would have required immediate operator actions to isolate the operable SW pump if the events occurred as described in the previous section. The changes to crosstie power supplies and valve logic were incorporated per LCP-94-5022.

The four crosstie valves receive a signal from either ES channel 1 or channel 2, which determines valve position upon ESAS actuation. Additional factors that determine valve position are SW pump configuration and valve position. SW pump configuration is determined by breaker position (open or closed) and valve position provided by valve open or close limit switches.

Common factors that affect valve opening or closing are:

* Valve open or close position.
* Motor overload.
* High opening or closing torque.

Each SW crosstie valve set is provided with a "manual" / "auto" pushbutton, which determines valve control during an ESAS actuation. When an ESAS actuation occurs its associated ES channel will control the crosstie valves, "auto" pushbutton will be backlit white. In this condition valve position is based on valve logic and operation from the valves handswitch will not be available. Valve operation using their associated handswitch requires the crosstie valve to be placed in "manual" by pushing the manual pushbutton (backlit white). When placed back in "auto", the crosstie valve will reposition to the ES desired position based on valve logic. For additional information refer to 1105.003, Engineered Safeguards Actuation System procedure.

SW crosstie valve logic when operating with P-4A and P-4C in service following an ESAS actuation (Channels 1 & 2) will be as follows:

* P-4A to P-4B crosstie valves CV-3646 and CV-3644 will close.
* P-4B to P-4C crosstie valves CV-3640 and CV-3642 will close.

In this alignment with P-4A and P-4C in service, ACW loop isolation valve CV-3643 will also close, resulting in an isolated portion of service water piping susceptible to pressure locking. To prevent this potential occurrence, a vent and capillary tubing to the floor drain is provided in this piping with a continuous minimum flow. (Modification added during 1R20)

SW crosstie valve logic when operating with either combination of P-4A and P-4B or P-4B and P-4C, the crosstie valves will position as follows upon channel 1 & 2 actuation.

* With P-4A and P-4B in service, the crosstie valves for P-4A to P-4B, CV-3646 and CV-3644 will close and crosstie

5.25 Operation with generator hydrogen pressure <50 psig and ACW aligned to the generator can cause condensation and/or ACW leakage into the generator.

5.26 Loss of Service Water (1203.030) lists the Unit 1 TS associated with an inoperable service water loop. Loss of Service Water cooling must be evaluated under the Safety Function Determination Program.

5.27 If P-4B Disch Isol (SW-2B) must be closed, then crossover capillary tubing should be venting (not clogged) to remove risk of hydraulic lock of crossover valves in the event of an ES actuation.

5.28 Due to potential issues identified with a degraded ability to open against Maximum Expected Differential Pressure the inner (with respect to proximity to P-4B) cross-tie valves, P-4A to P-4B Crosstie (CV-3644) and P-4B to P-4C Crosstie (CV-3640), should be opened first and the outer cross-tie valves P-4A to P-4B Crosstie (CV-3646) and P-4B to P-4C Crosstie (CV-3642) should be opened second. (CR-ANO-1-2013-1261)

5.29 Alignments of the Service Water system which restrict SW system return such as Temporary Modification to align E-28C discharge to the ACW piping, can result in excessively high pressures and component damage if SW system loads are quickly adjusted or secured (CR-ANO-1-2002-1371).

5.30 Valve stem damage can occur to Service Water to ICW Cooler Supply valves (CV-3811 and CV-3820) if opened with Service Water loop pressure greater than 91 psig.

5.31 Valve stem damage can occur to the Service Water Cross-tie valves if Loop I or Loop II pressure is greater than 51 psig and the area between CV-3640 and CV-3644 is depressurized.

5.32 When MCCs B55/56 are NOT aligned to the same power supply as P-4B, a single failure of a loss of the opposite train ES power during a Design Basis Accident (DBA) could result in the remaining Service Water pump potentially driven to a runout condition supplying all its loop loads and Auxiliary Cooling Water. CR-ANO-1-2013-2671

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

QID: 0227 **Rev:** 2 **Rev Date:** 7/13/16 **Source:** Bank **Originator:** Cork

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

Section: 3.8 **Type:** Plant Services Systems

System Number: 078 **System Title:** Instrument Air System (IAS)

Description: Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Cross-tied units.

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

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: 54 SRO:

Given:
- Both units are at 100% power.
- Unit 2 2C28A Instrument Air Compressor is out of service.
- Instrument Air pressure has dropped to 68 psig.
- Field operators can not find an Inst. Air leak on Unit One.
- Instrument Air pressure is now at 58 psig.

Which of the following is the procedurally required response per 1203.024, Loss of Instrument Air, to restore or conserve Instrument Air pressure?

A. Dispatch operator to take manual control of Pzr level control valve CV-1235.

B. Trip Reactor, actuate EFW and MSLI on both SGs.

C. Close Unit 1 to Unit 2 Instrument Air cross-connect.

D. Isolate Seal Injection by closing CV-1206.

Answer:

C. Close Unit 1 to Unit 2 Instrument Air cross-connect.

Notes:

"C" is correct, per 1203.024, the U1 to U2 cross connect should be closed if instrument air pressure drops below 60 psig.
"A" is incorrect, but plausible as it is an action in 1203.024 however this does not occur until pressure is less than 35 psig.
"B" is incorrect, but plausible as it is an action in 1203.024 however this would not be done unless pressure was less than 35 psig.
"D" is incorrect, but plausible as it is an action in 1203.024 however this would not be done unless necessary to maintain PZR level <290".

Revised "A" distracter as it is no longer in procedure.

This question matches the K/A since it involves the Instrument Air system and it requires the candidate to exhibit knowledge of when the cross-tie between the units is closed.

Revised question per NRC examiner suggestion.

References:

1203.024, Loss of Instrument Air

History:

Developed for 1998 RO/SRO Exam QID 0102.
Modified for 98 RO Re-exam
Modified for 2005 RO exam.
Selected for 2011 RO Exam.
Selected for 2016 exam.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

- Unit 1 Instrument Air Header Pressure can be monitored using U1 PMS point P5409
- Unit 2 INST Air Main Supply PRESS can be monitored using U2 PMS point P3013

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Verify available standby IA Compressor (C-28A/B) running.	
2. Dispatch an operator to determine specific compressor, air dryer, and filter condition.	
3. Check Instrument Air <u>not</u> supplying air for respiration.	3. Notify Radiation Protection of the loss of Instrument Air pressure **<u>AND</u>** direct the following: • Workers on Instrument Air must secure work in progress • Isolate the Instrument Air supply
4. Check both of the following conditions exist: • **low Instrument Air header pressure is due to loss of Instrument Air on Unit 2** • **Unit 1 and Unit 2 Instrument Air systems are cross-connected**	4. **GO TO step 5**.
A. Check Unit 1 Instrument Air Header PRESS remains > 60 psig.	A. Direct Unit 2 to terminate Instrument Air cross-connect.
B. **GO TO step 6**.	

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0104 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** GGiles

TUOI: A1LP-RO-EOP10 **Objective:** 15.5 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment System

Description: Ability to monitor automatic operation of the containment system, including: Containment isolation.

K/A Number: A3.01 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** K

Question: RO: 55 SRO:

Following an ESAS actuation the CBOT is directed to perform RT-10 to verify proper actuation.
The RT instructs you to verify each component properly actuated on C16, C18, and C26.

How is this accomplished for containment isolation valves (assume no components have been overridden)?

A. Verify all containment isolation valve "closed" indication lights are illuminated.

B. Verify containment isolation valve positions are in positions marked with green dots.

C. Verify containment isolation valve positions are in position marked by green or red background.

D. Verify containment isolation valves are in position marked with black tape background.

Answer:

D. Verify containment isolation valves are in position marked with black tape background.

Notes:

"D" is the correct response. A black tape background identifies the proper actuation position of ES components.
"A" is incorrect but plausible since the verification is for containment isolation valves, however not all containment penetration valves will be closed.
"B" is incorrect but plausible since Reg Guide 1.97 instrumentation is identified in this manner.
"C" is incorrect but plausible since this is the method used by Unit 2 for ESFAS actuated components.

This question is a direct match for the K/A as it requires the candidate to know how to properly monitor for automatic containment isolation.

Revised due tor NRC examiner comments.

References:

1015.018 , Plant Labeling
1202.012, Repetitive Tasks, RT-10 "Verify Proper ES Actuation"

History:

Developed for 1998 RO Exam.
Selected for use in 2002 RO/SRO exam.
Used on 2004 RO/SRO Exam.
Selected for 2016 exam

VERIFY PROPER ESAS ACTUATION

4. Verify proper ESAS Channels tripped:

Condition	Channels Actuated
RCS press ≤ 1550 psig	1,2,3,4
RB press ≥ 18.7 psia	1,2,3,4,5,6
RB press ≥ 44.7 psia	7,8,9,10

5. Perform the following:

A. Verify each component properly actuated on C16 and C18, **except** those overridden in previous steps.

B. Verify proper ES system flow rates.

NOTE

- During ESAS actuation, low LPI flow is expected until RCS depressurizes below LPI pump shutoff head.
- During large break LOCAs, high LPI flow can be experienced. Flow must be throttled to ensure ECCS flows are maintained within assumptions of calculations.

1. IF any of the following conditions exist:

   - A HPI FLOW HI/LO (K11-A4)
   - B HPI FLOW HI/LO (K11-A5)
   - A LPI FLOW HI/LO (K11-B4)

   - B LPI FLOW HI/LO (K11-B5)
   - A RB SPRAY FLOW HI (K11-C4)
   - B RB SPRAY FLOW HI (K11-C5)

   THEN use Annunciator K11 Corrective Action (1203.012J) to clear unexpected alarms.

C. IF only one train of HPI is available
AND
RCS press is > 600 psig,
THEN throttle HPI Block valve with the highest flow to within 20 gpm of the next highest flow.

7.2.6 The following colors will designate awareness groupings for Unit 1/2:

RED - HIGH AWARENESS (Unit 1/2)

Red annunciators are those where the operator should immediately evaluate the need for response or where immediate operator awareness may be required. RED annunciators have the potential to impact unit safety, unit availability, or safety system operation.

GREEN - MEDIUM AWARENESS (Unit 2)

Green annunciators are those where the operator should evaluate the need for prompt action or raised operator awareness. These annunciators could potentially result in associated RED annunciators in alarm, major process equipment trouble, or a radiation release.

WHITE - GENERAL AWARENESS (Unit 1/2)

White annunciators are those were the operator should evaluate the need for timely action or raised operator awareness may be required. These alarms would be addressed as time permits during an event as directed by CRS or S/S. All Unit 1 awareness groupings will be white with the exception of the following which are red:

- K04-A3 - turbine trip
- K08-A3 - Reactor trip

7.3 Other Special Labels / Tags

The following specifications are for the use of other special labels, in particular labels used to indicate emergency system actuation.

7.3.1 Unit 1 ESAS Actuation Indication

On Unit 1 a BLACK box made from plastic labeling material (210-121 Gravoply 1/32" thick, New Hermes stock number) or electrical tape is used to indicate whether a component receives an ESAS actuation signal. This box is cut to fit around the status indicating light for the appropriate actuation position (closed, open, etc.).

7.3.2 Unit 2 ESFAS Actuation Signal

On Unit 2 a colored box made from plastic labeling material or electrical tape is used to indicate if a component receives an ESFAS actuation signal. This box is cut to fit around the status indicating light for the appropriate actuation position (closed, open, etc.).

A GREEN box made from 258-121 Gravoply, 1/32" thick, (New Hermes stock number) or electrical tape should be used to indicate a close or start actuation signal.

A RED box made from 248-121 Gravoply, 1/32" thick,(New Hermes stock number) or electrical tape should be used to indicate a open or stop actuation signal.

RO

Tier 2

Group 2

QID: 0674 **Rev:** 1 **Rev Date:** 7/28/16 **Source:** Bank **Originator:** Possage

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive System

Description: Ability to (a) predict the impacts of the following malfunction or operations on the CRDS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rod-misalignment alarm.

K/A Number: A2.17 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	Ap

Question: **RO:** 56 **SRO:**

Given:
- Approach to criticality is in progress.
- Groups 1 - 5 are fully withdrawn.
- Group 6 is at 45% withdrawn.
- Group 6 Rod 4 drops into the core.
- K08-C2, "CONTROL ROD ASYMMETRIC" in alarm.

Which of the following procedural actions is required for the given conditions?

A. Relatch Group 6 Rod 4 and withdraw to 30% in increments of <25%.

B. Insert Group 6 rods and verify reactor remains subcritical.

C. Insert Group 5 and Group 6 rods in sequence.

D. Trip the reactor and perform 1202.001.

Answer:

C. Insert Group 5 and Group 6 rods in sequence.

Notes:

"C" is correct. If a dropped rod exists and NI power is <2%, then per Section 5 of 1203.003 the starup is terminated by inserting regulating rods in sequence.
"A" is incorrect but plausible since recovery of a single dropped rod is allowed in Mode 1, however recovery of a dropped rod from a subcritical condition can result in uncontrolled criticality and unanalyzed control rod configurations.
"B" is incorrect, but plausible since the rod is in Group 6 and the reactor is not yet critical, however procedure direction is to insert ALL regulating control rods.
"D" is incorrect, but plausible since a Reactor trip is required for two dropped rods if NI power is greater than or equal to 2%.

This question matches the K/A since it involves the Control Rod Drive System and requires candidate to predict the impacts of a malfunction of the CRD (dropped rod generating the rod misalignment alarm) by analyzing the given conditions and to recall procedure requirements for these conditions. This is acceptable per ES-401, D.2.a, 2nd paragraph.

Changed Group 6 to 45% withdrawn due to validator comment that Asymmetric alarm would not come in with Grp 6 at 30% with a dropped rod. JWC 7/28/16

References:

1203.003, Control Rod Drive Malfunction Action

History:

New for 2007 RO Exam. KA 2.4.50
Selected for 2016 exam

SECTION 5
DROPPED ROD – REACTOR SUBCRITICAL
ENTRY CONDITIONS

One or more of the following:

- Dropped Rod/Rods
  - On Diamond panel, green IN LIMIT lamp on
  - On PI panel, green IN LIMIT (0%) lamp on
  - On PI panel, amber FAULT lamp on
  - CONTROL ROD ASYMMETRIC (K08-C2) annunciated
  - On Diamond panel, amber ASYMM FAULT
- Dropped Rod/Rods with Control Rod Bayonet Coupling Failure --- Leadscrew Separation

SECTION 5
DROPPED ROD – REACTOR SUBCRITICAL
INSTRUCTIONS

CAUTION

Recovery of dropped rod/rods from a subcritical condition can result in uncontrolled criticality and unanalyzed control rod configurations.

NOTE

Per Reactor Engineering, a dropped rod is defined as a rod that has sudden, instantaneous inward motion of greater than 6.5% from its previous position.

1. **IF either of the following conditions exist:**
   - **dropped rod(s) exist**
   - **control rod(s) did not latch and failed to withdraw resulting in asymmetric conditions**

   THEN perform the following:

   A. Insert all regulating control rods in SEQ.

   B. Stabilize plant in Mode 3, >525°F per the applicable steps of Power Reduction and Plant Shutdown (1102.016) or Plant Startup (1102.002).

2. **IF desired,**
   THEN operate IN LIMIT BYPASS as required to insert affected group.

3. **IF dropped safety rod**
   AND required to place Letdown 3-way Valve (CV-1248) in BLEED,
   THEN verify Batch Controller Outlet (CV-1250) closed.

{1}
4. **IF dropped safety rod,**
   THEN insert safety groups per "Control Rod Insertion" section of CRD System Operating Procedure (1105.009).

5. **WHEN asymmetric condition is clear,**
   THEN depress FAULT RESET on Diamond panel.

6. **Initiate the following actions:**
   - Initiate Condition Report
   - Contact Operations Management
   - Contact Reactor Engineering

END

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0193 **Rev:** 3 **Rev Date:** 7/28/16 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-NI **Objective:** 8 **Point Value:** 1

Section: 3.2 **Type:** Reactor Coolant System Inventory Control

System Number: 002 **System Title:** Reactor Coolant

Description: Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship between reactor power and RCS differential temperature.

K/A Number: K5.10 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.1 **SRO Select:** No **Taxonomy:** Ap

Question: RO: 57 SRO:

As a reactor operator it is important to ensure indicated reactor power is accurate.

Which of the following sets of parameters would correspond to 85% power range NI power?

A. Thot 593 degrees, Tcold 566 degrees

B. Thot 596 degrees, Tcold 561 degrees

C. Thot 598 degrees, Tcold 559 degrees

D. Thot 599 degrees, Tcold 557 degrees

Answer:

C. Thot 598 degrees, Tcold 559 degrees

Notes:

Normal Thot is 602 and normal Tcold is 556 at 100%. This equates to a delta T of 46 degrees. 46 x 0.85 = 39
"C" is correct. Delta T is 39 degrees.
"A" is incorrect, this is a delta T of 27 degrees. This is the previous correct answer and is thus plausible.
"B" is incorrect. Delta T is 35 degrees, indicative of ~76% power.
"D" is incorrect. Delta T is 42 degrees and indicates ~91% power.

Revised question due to implausible distracter and to make correct answer "more" correct. Eliminated Tave in answers as it did not add anything.
Revised question at suggestion of NRC examiner to make power in stem 85% and changed "C" parameters to make it correct and changed "B" and "D" distracters to bound correct answer. Question thus meets definition of modified. JWC 7/14/16
Revised parameters slightly to ensure Tave remained within limits, DT's did not change. This was at suggestion of validators. JWC 7/28/16

This question matches the K/A since it involves RCS parameters (Thot and Tcold) and requires candidate to exhibit knowledge of the relationship between these parameters and reactor power.

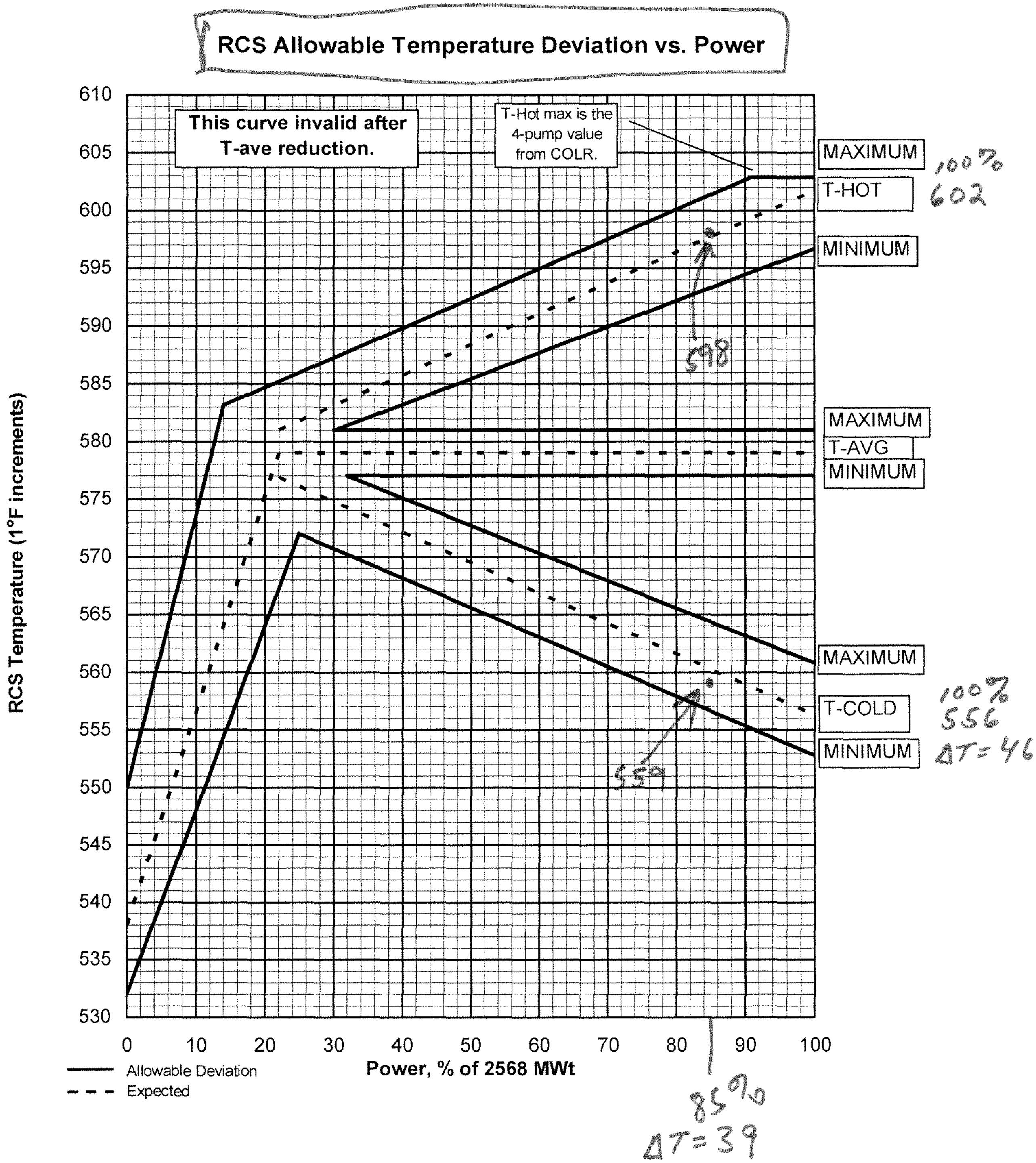
References:

1102.004, Power Operation

History:

Developed for use in 98 RO Re-exam
Used in 2001 RO/SRO Exam.
Selected for 2005 RO re-exam.
Modified for 2016 exam.

ATTACHMENT D



QID: 1066 **Rev:** 0 **Rev Date:** 4/15/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NI **Objective:** 4 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation

Description: Knowledge of bus power supplies to the following: NIS channels, components, and interconnections.

K/A Number: K2.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** K

Question: RO: 58 SRO:

Which of the following supplies power to Power Range channel NI-7?

A. RS-1

B. RS-2

C. RS-3

D. RS-4

Answer:

C. RS-3

Notes:

"C" is the correct power supply for Power Range NI channel 7.
"A", "B", and "D" are the other possible choices and are thus plausible (but incorrect) if one does not know the Power Range NI channel arrangement.

This question matches the K/A since it requires candidate to exhibit knowledge of NIS channels, specifically NI-7.

References:

1105.001, NI & RPS Operating Procedure
1107.003, Inverter and 120V Vital AC Distribution

History:

New question for 2016 exam.

16.6 IF "B" Reactor Protection System will be de-energized, THEN perform one of the following:

- Direct I&C to de-energize "B" Reactor Protection System.
- Inside C42 perform the following:

A. Place the Contact Monitor P.S. (Auxiliary Power Supply) switch in OFF.

B. Place the NI-6 Detector Power Supply switch in OFF.

C. Place the +15 VDC power supply breaker in OFF.

D. Place the -15 VDC power supply breaker in OFF.

E. Place System Fan Left breaker in OFF.

F. Place System Fan Right breaker in OFF

G. Place System AC Power breaker in OFF.

16.7 IF "C" Reactor Protection System will be de-energized, THEN perform one of the following:

- Direct I&C to de-energize "C" Reactor Protection System.
- Inside C43 perform the following:

A. Place the Contact Monitor P.S. (Auxiliary Power Supply) switch in OFF.

B. Place the NI-7 Detector Power Supply switch in OFF.

C. Place the +15 VDC power supply breaker in OFF.

D. Place the -15 VDC power supply breaker in OFF.

E. Place System Fan Left breaker in OFF.

F. Place System Fan Right breaker in OFF

G. Place System AC Power breaker in OFF.

EXHIBIT C

1107.003 Exhibit C
PANEL RS3

REVISED 4/11/14

Power Source: Inverter Y13 or Y15
Location: Control Room
Ref drawing: E-17-1

NOTE: All breakers except spares should be closed.

1	"C" RPS Cabinet C43 (E-544-3)	2	ESAS Analog-3 Panel C90 (E-537-3)
3	Spare	4	ICS Cabinet C46 (E-545-3)
5	CRD Pos System Logic Cabinet C51 (E-553-2)	6	Reactor Building Pressure Transmitter PT-2402 to "C" RPS (E-268-1, E-573-1, E-548-5)
7	ICC (Train A) Cabinet C553 (E-528-1) (1)	8	Plant Performance Analysis System (C47, C46, C585) (E-546-3)
9	Spare	10	Cont. Rad Monitor RE-8060 and EFW PIT-2888 in cabinet C486-3 (E-410, E-331-35)
11	Spare	12	EFIC Channel "C" Panel C37-3 (E-597-6)
13	Spare	14	Spare
15	Primary Power to "B" MFP Cntrls Secondary Power to "A" MFP C579 (E-331-47, E-556-5)	16	Spare
17	Spare	18	Spare

Note 1: If EC-44046 is implemented, then this breaker also supplies power to SFP level instrument LIT-2020-3 per E-259-10.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1067 **Rev:** 1 **Rev Date:** 7/20/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP05 **Objective:** 11 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-Core Temperature Monitor

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations:
Core damage.

K/A Number: A2.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: 59 SRO:

Given:
- An overheating event has been in progress.
- 1202.005, Inadequate Core Cooling, is in use.
- Core Exit Thermocouples = 1460 degrees F (average) and rising.
- RCS pressure = 2350 psig
- All actions have been performed for the current Region.

Critical parameters have been updated by the ATC:
- Core Exit Thermocouples = 1520 degrees F (average) and rising.
- RCS pressure = 2400 psig

CBOT reports multiple CETs are alarming on the plant computer with the status "INVALID" or "FAIL_LO".

What is occurring and what procedural action is required for the above conditions?

A. CETs are experiencing thermionic emission, trip all running RCPs.

B. CETS are failing due to short circuits, trip all running RCPs.

C. CETs are experiencing thermionic emission, use ADVs to reduce SG T-sat to ~100°F below current value.

D. CETS are failing due to short circuits, use ADVs to reduce SG T-sat to ~100°F below current value.

Answer:

B. CETS are failing due to short circuits, trip all running RCPs.

Notes:

Answer "B" is correct, Region 4 of Inadequate Core Cooling has been entered, all running RCPs should be tripped per step 12. CETs fail low when they have open circuits which could occur due to the extremely high temps experienced in Region 4.
"A" is incorrect, although plausible since thermionic emission affects the other type of Incore instrument - Incore Self Powered Neutron Detectors (SPNDs), this phenomenon does not affect CETs. The action given is correct.
"C" is incorrect, although plausible since thermionic emission affects the other type of Incore instrument - Incore Self Powered Neutron Detectors (SPNDs), this phenomenon does not affect CETs. The action is incorrect, this action is performed after entering Region 3.
"D" is incorrect, but plausible since this is the failure mechanism for the CETs. The action is incorrect however, this action is performed after entering Region 3.

This question matches the K/A since it involves the In-Core Temperature Monitor (CETs in the ICCMDS at ANO-1), requires the candidate to predict the impact of core damage on the CETs, and to recall the specific procedural direction required for this region in the ICC procedure.

Per NRC examiner suggestion added Region 3 conditions to question. JWC 7/20/16

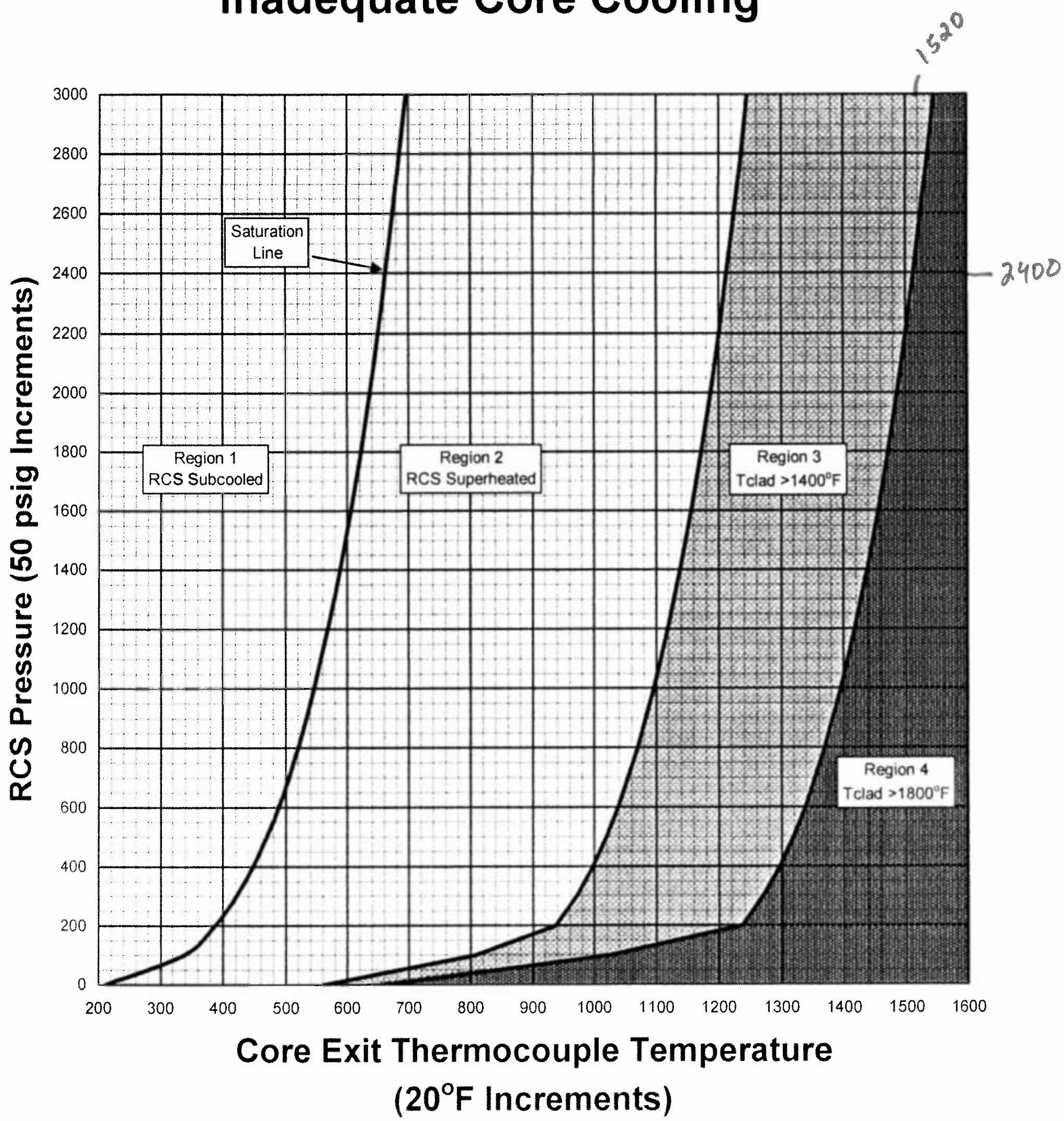
References:

1202.005, Inadequate Core Cooling
1202.013, EOP Figures, Figure 4
1105.008, Inadequate Core Cooling Monitor and Display

History:

New question for 2016 exam.

FIGURE 4
Core Exit Thermocouple for Inadequate Core Cooling



INSTRUCTIONS	CONTINGENCY ACTIONS
12. IF Region 4 of Figure 4 is entered, THEN perform the following:	
A. Trip all running RCPs.	
B. Notify Shift Manager to coordinate with TSC to implement Severe Accident Management Guidelines.	
C. **IF** ERV is open, **THEN** leave open until directed otherwise by TSC.	
D. **IF** High Point Vents are open, **THEN** leave open until directed otherwise by TSC.	
E. Proceed as directed by TSC.	

END

Reactor Vessel Level Monitoring Sensors (RVLMS)

Two level probes, each having nine level sensors and an absolute thermocouple near the top, are installed in the reactor vessel through the head at the center CRDM location (center CRD no longer used). Level is sensed at approximately 2' intervals from the top of the dome to near the top of the fuel assemblies.

A level sensor consists of two thermocouples connected internally to provide a signal proportional to the temperature difference. One thermocouple is heated by an internal heater element in the probe. The area around the heated thermocouple has a different heat transfer coefficient to the surrounding RCS and, therefore, has a different sensitivity to water or steam. As the water level drops below the level sensor, its ΔT changes and provides wet or dry indication.

The absolute thermocouple provides head fluid temperature indication from near the top of the head.

TS 3.3.15 includes the Reactor Vessel Level Monitor Sensors (RVLMS).

Core Exit Thermocouples (CET)

Twenty-four qualified core exit thermocouples provide temperature indication in a range of 50°F to 2300°F. These instruments are part of the incore detector system and are installed through the bottom of the reactor vessel through the incore instrument guide tubes. All valid CETs are averaged, and each CET is compared to the average. If a significant deviation exists, the CET is flagged SUSPECT. Failed or suspect CETs are automatically excluded from the average. TS 3.3.15 includes the core exit thermocouples.

Hot Leg Level Sensors

Each of the two RCS hot legs is instrumented with differential pressure transmitters. These provide one wide range (top to bottom) level indicator and four narrow range level indicators. The narrow range indicators cover top to bottom with four non-overlapping ranges.

Each level transmitter signal is compensated for changes in reference leg temperature and changes in RCS temperature. RTDs strapped on the reference leg tubing are used for reference leg temperature. Core exit thermocouple average temperature is used for RCS temperature. If the CET average value is invalid, such as when CETs are removed during refueling, an addressable default value, normally 110°F, is substituted.

TS 3.3.15 includes the hot leg level sensors.

RCS Pressure Input

RCS pressure input for subcooling margin calculations is provided by PT-1041 and PT-1042. These pressure transmitters provide wide range (0-3000 PSIG) RCS pressure indication on C04.

Reactor Coolant Pump Contacts

RC pumps ON/OFF status is provided by pump breaker contacts.

NOTE

- "FAIL_HI" or "FAIL_LO" indicates the input parameter is outside electrical limits. This point is automatically deleted from calculations.
- "INVALID" indicates a parameters reading is not accurate. Value is automatically deleted from calculations.
- "DEL" indicates the input parameter is not enabled.
- "HIGH" indicates a high alarm setpoint has been reached.
- "LOW" indicates a low alarm setpoint has been reached.
- "BAD" indicates a communication failure has occurred with the input parameter.
- "OTD" indicates an open thermocouple exists.

2.2 **SELECT** each of the displays specified in the Acceptance Criteria.

2.3 **RECORD** the displayed information.

NOTE

Any RC Pump ON causes Hot Leg Level and lower five Reactor Vessel Level indicators to be flagged INVALID.

2.4 <u>IF</u> any measured value actuates one of the following flags,

- INVALID
- FAIL_LO
- FAIL_HI
- DEL
- HIGH
- LOW
- OTD
- BAD

<u>AND</u> is <u>NOT</u> an expected result of plant conditions,
<u>THEN</u>:

2.4.1 **RECORD** flag with value in section 4.0.

2.4.2 **ENSURE** Condition Report has been initiated for repair or deletion from scan, as applicable.

2.4.3 **RECORD** Condition Report number section 4.0.

2.4.4 Using touch screen, **RETURN** system to desired display.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0200 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** B. Short

TUOI: A1LP-RO-SFC **Objective:** 8 **Point Value:** 1

Section: 3.8 **Type:** Plant Services Systems

System Number: 033 **System Title:** Spent Fuel Pool Cooling System

Description: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Maintenance of spent fuel level.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 60 SRO:

A break has occurred on the discharge line downstream of the discharge valve of the in service Spent Fuel Cooling Pump (P-40A).
The pump is stopped and the discharge valve (SF-5A) is closed.

What should happen with Spent Fuel Pool level?

A. The SFP will drain to 2' above fuel assemblies due to elevation of bottom of tilt pit gate.

B. The SFP will drain to point of uncovering the spent fuel assemblies.

C. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

D. The SFP level will drop ~3 feet to the bottom of the suction pipe.

Answer:

C. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

Notes:

"C" is correct. The discharge pipe has the siphon break holes located at the normal pool level.
"A" is incorrect but plausible if the candidate doesn't know the answer and believes the design will drain quite a bit of the pool but not uncover the assemblies due to bottom of tilt pit gate elevation, which is ~2' above top of fuel racks.
"B" is incorrect, but plausible since one discharge line goes all the way to the bottom of the pool. However with no operator action at all, the lowest the level would go is ~3 feet to the bottom of the suction pipe. This is still ~ 20 feet above the fuel.
"D" is incorrect but plausible, the suction pipe bottom is at ~3 feet, however, with the discharge valve closed the pool will stop draining out the break at the normal pool level due to the siphon holes on the discharge pipe.

This question matches the K/A since it is about knowledg of the spent fuel cooling system and requires the candidate to recall the design feature which maintains spent fuel level despite a break in the return line.

Revised question by re-wording stem, distractor B, and adding discharge valve designator per request of NRC examiner.

References:

STM 1-07, Spent Fuel Cooling System

History:

Selected for 2011 RO Exam.

Selected for 2016 exam.

Pool. During refueling the minimum boron concentration is determined by RCS requirements. Additional shutdown margin is required during core reload assuming the most reactive fuel assembly is placed in the worst core location.

Cooling and purification of the pool water is accomplished by recirculating it through heat exchangers, with a bypass flow through the lead Spent Fuel Pool Filter (F-4A), to Spent Fuel Pool Demineralizer (T-5), and then through the lag filter (F-4B). Water enters the pool at two separate nozzles, each equipped with a manually-operated globe valve for flow control. One nozzle discharges at the bottom of the pool and the other at the surface. The bottom discharge impacts turbulence to the pool and tends to keep particles in suspension, thereby increasing the likelihood of their being recirculated and removed by the SF filters. In addition, the bottom discharge promotes pool circulation through the stored spent fuel assemblies to improve cooling. The suction nozzle for the spent fuel pool circulating pumps is located at the opposite end of the pool from the discharge nozzles and is near the surface. This arrangement provides thermal mixing and insures uniform water temperature.

The SF Pool Skimmer is not normally used and requires installation of equipment before use. The skimmer system is designed to remove floating debris from the surface of the SF pool. Skimmer suction is provided to either the Borated Water Recirc Pump (P-66) or the header for the Spent Fuel Pool Circulating Pumps (P-40A or B).

To prevent inadvertent draining of the SF Pool below the stored fuel elements, pool drains are not provided and the suction line for the pumps is located three feet below the normal water level (centerline of the suction line is at elevation 397'0"). The two discharge lines have siphon breaker holes drilled into them at the normal pool level to prevent a break in the return piping from siphoning the water from the pool. A portable pump is required for complete draining of the SF pool (no fuel elements in pool).

2.1.2 Spent Fuel Racks

The assemblies are stored in stainless steel racks arranged in 44 rows of 22 elements, for a capacity of 968 spent fuel assemblies. One space is available for storage of a failed fuel assembly and its special container. The parallel rows are designed to ensure a center to center distance of 10.65 inches between fuel assemblies is maintained in all directions. This spacing is sufficient to maintain a Keff of less than 1.00 even if the pool is flooded with unborated water. A fuel rack has been fitted with a fuel assembly upper end fitting for load testing of the bridge grapple and/or storage of a control component.

The spent fuel rack designs described employ three separate and different arrays which will be considered as three separate spent fuel racks. All three storage arrays are designed on the basis of the currently accepted NRC guidance on spent fuel rack design, with consideration of the changes in fuel and fission product inventory resulting from depletion in the reactor core. Although all three storage racks types differ in design, they take credit for the reduction in reactivity associated with fuel burnup. Criticality safety is assured

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1094 **Rev:** 2 **Rev Date:** 7/28/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP03 **Objective:** 15 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 035 **System Title:** Steam Generator

Description: Ability to manually operate and/or monitor in the control room: Fill of dry S/G.

K/A Number: A4.02 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 2.8 **SRO Select:** No **Taxonomy:** An

Question: RO: 61 SRO:

Given:
- Recovery from an Overcooling condition is in progress.
- Reactor Coolant Pumps are not runnning.
- Auxiliary Feedwater Pump, P-75 is the only available source of water.

- "A" SG level is 10 inches and stable
- "B" SG level is 21 inches and lowering
- "A" SG shell temperature is 485 °F
- "B" SG shell temperature is 445 °F
- CET Temperatures are ~425 °F
- RCS Pressure is 1950 psig

Steam leak has been isolated locally.
RT-16, Feeding Intact SG, is in use.

Which of the following is the procedure action required by RT-16 for the above conditions?

A. Feed only "B" SG, reduce flow to ≤ 450 gpm due to Tube-to-Shell DT limit has been exceeded.

B. Feed both SGs, reduce flow to ≤ 200 gpm due to Tube-to-Shell DT limit has been exceeded.

C. Feed both SGs at ≤ 0.2 x 10e6 lbm/hr to establish 300 to 340" level while maintaining Tube-to-Shell DT limits.

D. Feed only "B" SG at ≤ 0.2 x 10e6 lbm/hr to establish 300 to 340" level while maintaining Tube-to-Shell DT limits.

Answer:

C. Feed both SGs at ≤ 0.2 x 10e6 lbm/hr to establish 300 to 340" level while maintaining Tube-to-Shell DT limits.

Notes:

"C" is correct, subcooling margin is adequate, no tube-to-shell DT limits have been exceeded, thus AFW should be fed until 300 to 340" is established in each SG while maintaining tube-to-shell DT limits. ANO-1 has been analyzed for AFW flow to a dry S/G so no specific requirements (other than the EOP limits) are in effect. Flow must be less than 0.2 x 10e6 lbm/hr per RT-16.
"A" is incorrect but plausible as this action would be correct if tube-to-shell DT had been exceeded. Candidate has to recall tube-to-shell DT limits of 100°F tubes colder and 60°F tubes hotter. Indications given show that tube-to-shell DT is 60°F tubes colder. Also both SGs can be fed, not just "B". Flow limit of ≤ 450 gpm is for feeding with EFW (not AFW) with any RCP running and primary to secondary heat transfer not yet established.
"B" is incorrect but plausible as this action would be correct if tube-to-shell DT had been exceeded. Candidate has to recall tube-to-shell DT limits of 100°F tubes colder and 60°F tubes hotter. Indications given show that

tube-to-shell DT is 60°F tubes colder. Flow limit of ≤ 200 gpm is for feeding with EFW (not AFW) with all RCP's off and primary to secondary heat transfer not yet established.
"D" is incorrect but plausible as this action would be correct if this answer stated to feed both SGs.

This question is based upon QID 662 which assumes an Overheating condition and RT-16 in use with AFW, this new question assumes an Overcooling condition with RT-16 in use with AFW.

This question matches the K/A as it involves operating SG feedwater controls within procedural limits with indications of a dry S/G (A).

Revised question based upon NRC examiner comments. JWC 7/14/16
Changed "as necessary" in C & D to "at ≤ 0.2 x 10e6 lbm/hr" due to validator comment. JWC 7/28/16

References:

1202.003, Overcooling
1202.012, Repetitive Tasks, RT-16 Feeding Intact SG

History:

New question for 2016 exam

INSTRUCTIONS	CONTINGENCY ACTIONS

25. WHEN overcooling is terminated, THEN perform the following for each SG:

A. IF MSIV is open
AND
TURB BYP Valves are available,
THEN operate TURB BYP Valves as necessary to prevent RCS heatup.

Contingency: A. Perform the following:

1) Operate ATM Dump Control System as necessary to prevent RCS heatup:

SG A		SG B
CV-2691	**MSIV**	CV-2692
CV-2676	**ATM Dump ISOL**	CV-2619
CV-2668	**ATM Dump CNTRL**	CV-2618

a) Place TURB BYP Valves for SG(s) with closed MSIV in HAND
AND
close to prevent loss of condenser vacuum.

B. Check feedwater aligned for each SG.

CAUTION

Drying out a SG can cause the 60°F (tubes hotter) Tube-To-Shell ΔT limit to be exceeded due to ambient temperature losses.

NOTE

Methods of limiting Tube-To-Shell ΔT (tubes hotter) are listed in order of preference.

Contingency: B. Perform the following, while continuing with this procedure:

1) **IF** steam leak has been isolated locally,
THEN refill intact SG (RT-16).

(25. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. IF feed source is MFW or AFW, THEN perform the following:

A. Verify affected SG(s) Main and Low Load Feedwater Block closed:

SG A		SG B
CV-2625	**Main Feedwater Block**	CV-2675
CV-2624	**Low Load Feedwater Block**	CV-2674

B. Place affected SG(s) Startup valve in HAND **AND** close:

SG A		SG B
CV-2623	**Startup**	CV-2673

C. Verify at least one Condensate pump running.

D. Verify affected SG(s) Main Feedwater Isolation open:

SG A		SG B
CV-2680	**Main Feedwater Isolation**	CV-2630

E. Verify Feedwater Pumps DISCH Crosstie (CV-2827) open.

(2. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. (Continued)

F. **IF** AUX Feedwater Pump (P75) is available,
THEN perform the following:

1) Dispatch an operator to verify AUX FW Pump RECIRC to E-11A Isolation (FW-1) open.

2) WHEN FW-1 open,
THEN verify Aux Feedwater Pump (P75) running.

3) **GO TO step 2.H.**

G. **IF** MFW pump is available,
THEN verify MFW pump running.

1) Place RFR Override handswitch in OVERRIDE.

H. **IF** SCM is **not** adequate, N/A
THEN establish **AND** maintain SG levels 370 to 410" within 25 minutes of SCM loss using Startup valve H/A stations in HAND.

1) **IF** SCM becomes adequate prior to establishing 370 to 410",
THEN GO TO step 2.I.

2) **IF** any good SG press drops below 720 psig,
THEN perform the following:

a) Bypass MSLI by momentarily placing SG Bypass toggle switch on each EFIC cabinet Initiate module in BYPASS.

- C37-3
- C37-4
- C37-1
- C37-2

(2. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. (Continued)

N/A 3) **WHEN** SG level is 370 to 410"
THEN check primary to secondary heat transfer in progress indicated by all of the following:

- T-cold tracking associated SG T-sat (Fig. 2)
- T-hot tracking CET temps
- T-hot/T-cold ΔT stable or dropping

a) **IF** primary to secondary heat transfer is **not** in progress,
THEN raise primary to secondary ΔT to 40 to 60°F as follows:

(1) **IF** SG press drops below 720 psig during the following steps,
THEN on Initiate module in each EFIC cabinet, place each SG Bypass toggle switch in BYPASS and release:

- C37-3
- C37-4
- C37-1
- C37-2

(2) Adjust TURB BYP or ATM Dump Control System to establish SG press within limits of Figure 5 "SG Pressure to Establish 40 to 60°F Primary to Secondary ΔT".

b) Re-check primary to secondary heat transfer in progress per step 3) above.

(2. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. (Continued)

N/A

c) **IF** primary to secondary heat transfer is established,
THEN adjust TURB BYP or ATM Dump Control System to stabilize RCS temp.

d) **IF** primary to secondary heat transfer is **not** in progress,
THEN raise primary to secondary ΔT to 90 to 110°F as follows:

(1) Adjust TURB BYP or ATM Dump Control System to establish SG press within limits of Figure 6 "SG Pressure to Establish 90 to 110°F Primary to Secondary ΔT".

(2) **IF** primary to secondary heat transfer is established,
THEN adjust TURB BYP or ATM Dump Control System to stabilize RCS temp.

4) **WHEN** primary to secondary heat transfer is established,
THEN adjust affected SG(s) Startup valve(s) to maintain the following:

SG A		SG B
CV-2623	**Startup**	CV-2673

- Adequate SCM
- ≤ 100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)
- Desired cooldown rate

(2. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. (Continued)

CAUTION

Excessive FW flow can result in loss of SCM due to RCS shrinkage.

I. **IF** SCM is adequate,
THEN adjust associated Startup valve(s) as necessary to maintain the following:

SG A		**SG B**
CV-2623	**Startup**	CV-2673

- MFW Loop flow ≤ 0.2×10^{6} lbm/hr
- Adequate SCM
- ≤ 100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)

1) **IF** RCPs are off,
THEN check primary to secondary heat transfer in progress indicated by all of the following:

- T-cold tracking associated SG T-sat (Fig. 2)
- T-hot tracking CET temps
- T-hot/T-cold ΔT stable or dropping

(2. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. (Continued)

2) **WHEN** primary to secondary heat transfer is established,
THEN adjust associated Startup valve(s) to maintain the following:

SG A		SG B
CV-2623	**Startup**	CV-2673

- Adequate SCM
- ≤ 100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)
- Desired cooldown rate

3) **IF** TURB BYP Valves are **not** available,
THEN operate ATM Dump Control System to establish desired SG press:

SG A		SG B
CV-2676	**ATM Dump ISOL**	CV-2619
CV-2668	**ATM Dump CNTRL**	CV-2618

4) **IF** associated MSIV is open and TURB BYP Valves are available,
THEN operate TURB BYP Valves to establish desired SG press:

SG A		SG B
CV-2691	**MSIV**	CV-2692
CV-6689 CV-6690	**TURB BYP Valves**	CV-6687 CV-6688

(2. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. (Continued)

5) **WHEN** SG level is ≥ 20",
THEN perform the following:

a) **IF** any RCP is running,
THEN perform the following:

(1) **WHEN** SG level is near midpoint for 20 to 40",
THEN place associated Startup valve(s) in AUTO:

SG A		**SG B**
CV-2623	**Startup**	CV-2673

(2) Verify SG level maintained 20 to 40".

b) **IF no** RCPs are running,
THEN continue to raise SG level as necessary to establish and maintain 300 to 340", while maintaining the following:

- MFW Loop flow ≤ 0.2×10^6 lbm/hr until primary to secondary heat transfer is established
- Adequate SCM
- ≤ 100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)
- Desired cooldown rate

END

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1080 **Rev:** 0 **Rev Date:** 4/27/16 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-ICS **Objective:** 17 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 041 **System Title:** Steam Dump System and Turbine Bypass Control

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure.

K/A Number: A1.02 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** C

Question: RO: 62 SRO:

Given:
- Plant startup in progress
- Turbine control is in ICS Auto
- Generated MW was 140 but has lowered to 130 Mwe
- Turbine Bypass Valves (TBVs) are closed

If turbine header pressure setpoint is 895 psig, then the TBVs would be expected to begin opening at _________ psig.

A. 895

B. 905

C. 945

D. 995

Answer:

B. 905

Notes:

"B" is correct, once 15% (135 MW) power is reached, a 50 psig bias is applied to the TBVs. However, if power lowers to less than 15%, then the bias is removed and the TBVs open when the setpoint is exceeded by 10 psig.
"A" is incorrect but plausible since the TBVs will control at setpoint prior to initially reaching 15% power.
"C" is incorrect but plausible since this is the header pressure setpoint plus the 50 psig bias.
"D" is incorrect but plausible as this is the header pressure setpoint plus the 100 psig bias applied following a reactor trip (to limit cooldown).

This question was modified from QID ANO-OPS1-345 by adding that generated MW had gone above 15% but had lowered to less than 15%. Changed header pressure setpoint to 895 psig which necessitated modifying ALL of the answer choices.

This question matches the K/A since it requires the candidate to recognize (monitor) when the Turbine Bypass Valves will open (to control steam pressure) if a transient were to occur.

References:

STM 1-64, Integrated Control System

History:

Modified regular exam bank question (ANO-OPS1-345) for 2016 exam

Given:

- Turbine control is in ICS auto.
- Generated output is 100 megawatts.
- The turbine bypass valves are full shut.

If turbine header pressure setpoint is 900 psig, the turbine bypass will open at _____ psig.

A. 870
B. 910
C. 950
D. 1010

Answer: B

PARENT

Question Comments:

Image Reference: None

QuestionID: ANO-OPS1-345

Objectives:

1. CourseID: A1LP-RO-ICS Objective: 17

KA References:

1. 041 A1.02 Steam pressure [3.1/3.2]

References:

Training Programs:

Categories:

1. Continuing Training - Part B
2. spullin09

Systems:

Task References:

Cognitive Level: 2: Comprehension or Analysis

Point Value: 1.0

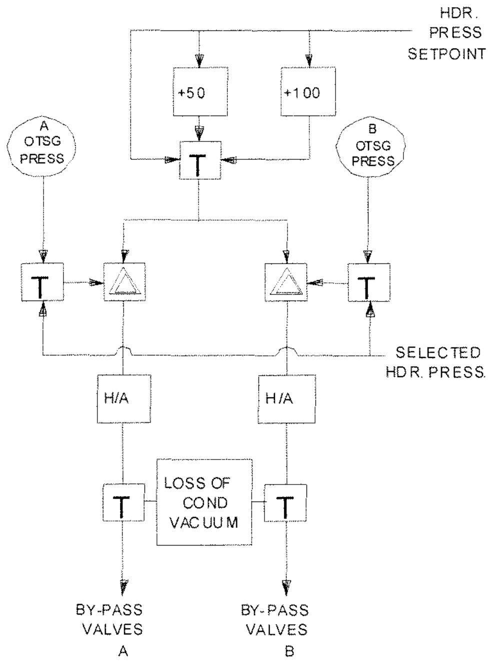
Exam Bank: OpsUnit1

Review Status: Reviewed

Comments:

generators. The turbine bypass valves are modulated to maintain turbine header pressure at setpoint. Setpoint is 600 - 1200 psig by setpoint station. (Normally set at 895 psig.)

FIGURE 64.19 TURBINE BYPASS VALVE CONTROL



The control signal to the bypass valves is derived by comparing either, each loop's OTSG pressure or selected header pressure, to the turbine header pressure setpoint. (Refer to figure 64.19) The pressure input selection is dependent upon the turbine controls. When the turbine goes from throttle to governor valve control (TV-GV transfer), the input signal for bypass valve control is changed automatically from OTSG pressure to selected turbine header pressure. Since two controllers are then trying to control the same header pressure, the stronger controller will predominate, driving the other controller and associated TBVs in the closed direction. This asymmetric condition will remain until the 50 bias is applied as outlined below.During a plant startup, as reactor power is increased, the turbine bypass valves will open to maintain OTSG pressure at 895 psig, thus holding T_{sat} at 532^0 F (refer to section on Principles of Heat Transfer). At ~7% reactor power, the turbine generator is initially rolled. The turbine bypass valves will compensate for steam used in the turbine by reducing the amount of steam going to the condenser. Once the turbine generator is loaded, the following actions are taken by the operator and the ICS in order to prevent the turbine and the bypass valves from competing for control of header pressure. With the turbine generator loaded, the operator forces the turbine bypass valves closed by increasing the steam flow to the turbine generator which increases the generators megawatt output. When one of the following conditions occur, the turbine header pressure setpoint to the turbine bypass valves will be biased by +50 psig:

1. All turbine bypass valves closed and turbine header pressure not greater than setpoint by 10 psig.
2. Load demand in the integrated master subsystem >15% (135 megawatts).

It should be noted that after the valves are biased shut by condition 1, if generated power is less than 135 megawatts and header pressure exceeds stepoint by 10 psig, then the 50 psig bias from condition 1 will be removed and the turbine bypass valves will open. If header pressure returns to within 10 psig of setpoint, the bias will be reinstated.

The purpose of these 50 psig biases is to keep the turbine bypass valves from competing with the turbine governor valves for header pressure control when the turbine generator is placed in "Integrated Control" control. When the turbine is in Integrated Control, the turbine governor valves are modulated to maintain pressure equal to the setpoint of 895 psig.

A bias of +100 psig will be added when the Reactor trips to limit the RCS cooldown. A pressure setpoint of 895 psig corresponds to a RCS

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0470 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Modified **Originator:** J.Cork

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 3.9 **Type:** Radioactivity Release

System Number: 071 **System Title:** Waste Gas Disposal System (WGDS)

Description: K4.04 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Isolation of waste gas release tanks.

K/A Number: K4.04 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 63 SRO:

Given:
- Waste Gas Decay Tank T-18A release is in progress.
- Shortly afterwards RE-4830 Gaseous Waste Discharge Process Monitor goes into high alarm.

Which of the following will occur as a result of this actuation?

1. C-9A and C-9B Waste Gas Compressors stop.
2. Gaseous Waste Discharge Isolation valve (CV-4830) closes.
3. Aux. Building Vent Header (CV-4806) diverts to the in-service Waste Gas Decay Tank.
4. Gaseous Radwaste Discharge Header flow control valve (CV-4820) closes.

A. 1 and 4

B. 2 and 3

C. 3 and 4

D. 2 and 4

Answer:

D. 2 and 4

Notes:

Answer "D" is correct, on high radiation signal from RI-4830 the discharge isolation CV-4830 closes, flow control CV-4820 closes, and CV-4806 opens to allow flow to divert to the Waste Gas Surge Tank.
"A" is incorrect but plausible since CV-4820 does close and the compressors do have auto stop functions but not on high radiation.
"B" is incorrect but plausible since CV-4830 does close and CV-4806 diverts but a check valve arrangement prevents it from diverting to the Waste Gas Decay Tanks.
"C" is incorrect but plausible since CV-4820 does close and CV-4806 diverts but a check valve arrangement prevents it from diverting to the Waste Gas Decay Tanks.

Modified question by revising the stem into an operational context. Revised choices 1-4 so that they all had component IDs. Added choice 2 and removed "old" 2 so that "D" is now the correct answer. Changed A and D answers.

This question matches the K/A since it reqiures knowledge that the gaseous waste rad monitor has a design feature which will isolated the waste gas decay tanks on a high alarm.

References:

1104.002, Gaseous Radwaste System

History:

Direct from regular ExamBank QID ANO-OPS1-1399.
Selected for use on 2007 RO Exam.
Modified for use in 2016 exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0470 **Rev:** 0 **Rev Date:** 10/4/2003 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 3.9 **Type:** Radioactivity Release

System Number: 071 **System Title:** Waste Gas Disposal System (WGDS)

Description: Knowledge of the physical connections and/or cause-effect relationships between the Waste Gas Disposal System and the following systems: ARM and PRM systems.

K/A Number: K1.06 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	3.1	**RO Select:**	No	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** **SRO:**

When a high radiation condition occurs in the Waste Gas Discharge Header, the radiation monitor will cause what automatic actions to occur?

1. C-9A and C-9B Waste Gas Compressors power supply breakers will trip open.
2. The Aux. Building Vent Header diverts to the Waste Gas Surge Tank.
3. The Waste Gas Decay Tank effluent control valve (CV-4820) shuts.
4. The Gas Collection Vent Header diverts to the Waste Gas Decay Tank in service.

Prior to Revision

A. 1 and 2

B. 2 and 3

C. 3 and 4

D. 1 and 4

Answer:

B. 2 and 3

Notes:

Answer "B" is correct, on high radiation signal from RI-4830 the discharge isolation CV-4820 closes, CV-4830 closes, and CV-4806 opens to allow flow to diver to the Waste Gas Surge Tank. All other answers are combinations of related but incorrect choices.

References:

1203.006 Chg 010-02-0

History:

Direct from regular ExamBank QID #1399.
Used on 2004 RO exam Was KA A4.25
Selected for use on 2007 RO Exam.

6.2.7 WASTE GAS DECAY TK T18A PRESS HI (K115-A4)
T-18A Press Alarm (PS-4807): 85 psig

WASTE GAS DECAY TK T18B PRESS HI (K115-B4)
T-18B Press Alarm (PS-4808): 85 psig

WASTE GAS DECAY TK T18C PRESS HI (K115-C4)
T-18C Press Alarm (PS-4809): 85 psig

WASTE GAS DECAY TK T18D PRESS HI (K115-D4)
T-18D Press Alarm (PS-4810): 85 psig

6.2.8 WASTE GAS HEADER FLTR ΔP HI (K115-A5)

- F-19 ΔP Alarm (PDS-4835): 14" H_2O (~0.5 psid)

6.2.9 WASTE GAS DISCH LINE RAD HI (K115-B5) from Gaseous Waste Disch Process Monitor (RE-4830).

A. Upon alarm, opens ABVH Diversion to T-17 (CV-4806) and closes the following valves:

- Gaseous Waste Disch Isol (CV-4830)
- T-18s Discharge to Gaseous Radwaste Discharge Header Flow Control (CV-4820)

6.2.10 WASTE GAS DECAY TK HDR PRESS HI (K115-C5)

- WGDT Disch Press (PS-4826): 21 psig

6.2.11 GAS SAMPLING PANEL C119 TROUBLE (K115-D5)
H_2O_2 Analyzer Panel (C119)

GAS SAMPLING PANEL C119A TROUBLE (K115-D6)
H_2O_2 Analyzer Panel (C119A)

6.2.12 WASTE GAS COMPR C9A LEAK HI (K115-A6)
C-9A Leak Detector (PS-4870): 10 psig

WASTE GAS COMPR C9B LEAK HI (K115-C6)
C-9B Leak Detector (PS-4860): 10 psig

6.2.13 WASTE GAS DISCH FLTR ΔP HI (K115-B6)

- F-16 ΔP Alarm (PDS-4830): 14" H_2O (~0.5 psid)

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0379 **Rev:** 2 **Rev Date:** 4/27/16 **Source:** Bank **Originator:** J.Cork

TUOI: A1LP-WCO-ARMS **Objective:** 7 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 072 **System Title:** Area Radiation Monitoring (ARM) System

Description: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: radiation levels.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: 64 SRO:

During performance of 1305.001, Supplement 6, Area Radiation Monitor Monthly Alarm Check, you discover Relay Room Area Monitor, RI-8002, high alarm setpoint is greater than the maximum allowable value.

What are the required actions?

A. Record the value found, and document set-point drift in Section 3.0 of the surveillence test.

B. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

C. Record the value found, then have I&C make the required adjustment under a "blanket" Work Order.

D. Adjust the setpoint to twice the background reading, then record the As-Left Setpoint.

Answer:

B. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

Notes:

Answer [B] is correct per the procedure supplement as it maintains the system alarms within the design criteria of the system.
Answer [A] is incorrect but would be correct for discrepancies not governed by a procedural response.
Answer [C] is incorrect but this is how it was handled in the past.
Answer [D] is incorrect but this is how adjustments are made for process rad monitors in Supplement 5.

Revised choice D to make it plausible.

This question matches the K/A since it requires the candidate to know how to adjust alarm setpoints on area radiation monitors, setting the high alarm setpoint to less than or equal to the procedural max high alarm setpoint ensures the area monitor will alarm when radiation levels change slightly.

References:

1305.001, Radiation Monitoring System Check and Test

History:

Modified regular exambank QID #2645
Selected for the 2008 RO Exam.

Revised for the 2016 exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0379 **Rev:** 1 **Rev Date:** 11/15/00 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-WCO-ARMS **Objective:** 7 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 072 **System Title:** Area Radiation Monitoring (ARM) System

Description: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: radiation levels.

K/A Number: A1.01 **CFR Reference:** 41.5/45.5

Tier: 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** C

Question: **RO:** **SRO:**

During performance of 1305.001, Supplement 6, Area Radiation Monitor Monthly Alarm Check, you discover Relay Room Area Monitor, RI-8002, high alarm setpoint is greater than the maximum allowable value.

What are the required actions?

A. Record the value found, and document set-point drift in Section 3.0 of the surveillence test.

B. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

C. Record the value found, then have I&C make the required adjustment under a "blanket" Work Order.

D. Record the value found and continue, nothing else needs to be done since RI-8002 is not a Tech Spec required monitor.

Prior to Revision

Answer:

B. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

Notes:

Answer [B] is correct per the procedure as it maintains the system alarms within the design criteria of the system.
Answer [A] would be correct for discrepancies not governed by a procedural response.
Answer [C] is how this was handled in the past.
Answer [D] is how an incompetent operator might proceed.

References:

1305.001, Radiation Monitoring System Check and Test

History:

Modified regular exambank QID #2645
Selected for the 2008 RO Exam.
Selected for the 2016 exam

NOTE
Monitors may be tested in any order. Test steps shall be followed in order.

2.3 For each remaining monitor, perform the steps as applicable.

2.3.1 IF desired, THEN test RI-8002 Relay Room monitor as follows:

A. Place Alarm Setting switch in WARNING position.

NOTE
The Min/Max Setpoint value is based on the design value for this area during normal operation (assuming 1% failed fuel) according to ANO-1 SAR.

B. Check Warning setpoint is 1 mR/HR.

NOTE
Warning alarm setpoints should be adjusted as necessary to a value high enough to preclude any warning actuations due to electrical noise deflections but low enough to detect rising radiation levels as early as possible.

- IF necessary, THEN adjust per step 2.4.

C. Record As-Left Setpoint in Table 1.

D. Place Alarm Setting switch in HIGH position.

E. Check Hi Alarm setpoint is ≤ 2 mR/HR.

NOTE
High alarm setpoints are based, with the noted exceptions, on minimizing spurious alarms due to transient radiation level rises but low enough to provide early detection of abnormal radiological conditions in the area.

- IF necessary, THEN adjust per step 2.5.

F. Record As-Left Setpoint in Table 1.

2.4 IF the background for the monitor is such that the warning setpoint needs to be adjusted,
THEN perform the following to adjust the setpoint:

2.4.1 Slide Area Monitor drawer out to gain access to Alarm Setting potentiometers.

2.4.2 While holding the Alarm Setting switch in the WARNING position, adjust the warning potentiometer to the desired setpoint.

2.4.3 Slide Area Monitor drawer back to the normal position and secure.

2.4.4 Reset alarms if applicable.

NOTE

Except for brief periods during evolutions such as Dry Fuel movement, the high alarm setpoint shall NOT exceed Maximum High Alarm Setpoint listed in Table 1. The high alarm setpoint should be adjusted slightly below or equal to the Maximum High Alarm Setpoint listed in Table 1.

2.5 IF Alarm setpoint exceeds max allowable value
OR if high alarm setpoint must be adjusted for any reason,
THEN perform the following to adjust the setpoint:

2.5.1 Slide Area Monitor drawer out to gain access to Alarm Setting potentiometers.

2.5.2 While holding the Alarm Setting switch in the HIGH position, adjust the HIGH potentiometer to the desired setpoint.

2.5.3 Slide Area Monitor drawer back to the normal position and secure.

2.5.4 Reset alarms if applicable.

2.5.5 Inform SM/CRS of any abnormal findings.

2.5.6 Using "DBM" function of Plant Monitoring System (PMS), verify Area Monitor alarm setpoints indicate same values as monitor setpoints.

2.6 Perform the following to check Unit 1 Control Room Vent Supply Radiation Monitor (2RITS-8001A) high alarm setpoint:

2.6.1 At Unit 1 Control Room Vent Supply Radiation Monitor (2RITS-8001A), press MODE to display high alarm setpoint.

- Record high alarm setpoint in Table 1.

3.0 ACCEPTANCE CRITERIA

3.1 Compare the As-Left Setpoint to the Maximum Normal Setpoint and the As-Left High Alarm Setpoint to the Max High Alarm Setpoint.

NOTE

- Warning alarm setpoints should be adjusted as necessary to a value high enough to preclude any warning actuations due to electrical noise deflections but low enough to detect rising radiation levels as early as possible.
- High alarm setpoints are based, with the noted exceptions, on minimizing spurious alarms due to transient radiation level rises but low enough to provide early detection of abnormal radiological conditions in the area.

Table 1

	WARNING SETTING				ALARM SETTING		
Monitor Indicator Number/ Description	Minimum Setpoint	As-Left Setpoint	Maximum Normal Setpoint	Is As-Left Setpoint ≤ Max Normal Setpoint?	As-Left High Alarm Setpoint	Maximum High Alarm Setpoint	Is As-Left High Alarm Setpoint ≤ Max High Alarm Setpoint?
{4.3.1} {4.3.3} **RI-8001 Control Room**	**1 mR/HR**		**1 mR/HR (2)**	**YES NO**		**7 mR/HR (1)**	**YES NO**
RI-8002 Relay Room	1 mR/HR		1 mR/HR (2)	YES NO		2 mR/HR	YES NO
RI-8003 Machine Shop	1 mR/HR		1 mR/HR (2)	YES NO		2 mR/HR	YES NO
RI-8004 Elev 317	1 mR/HR		10 mR/HR	YES NO		20 mR/HR	YES NO
RI-8005 Sample Room	1 mR/HR		2.5 mR/HR	YES NO		7.5 mR/HR	YES NO

{4.3.1} {4.3.2} **Note (1) This is based on the Control Room being designed for continuous occupancy at a maximum of 5 Rem for the duration of a maximum hypothetical accident (30 days continuous occupancy) according to ANO-1 SAR. It was calculated as follows:**

$$\frac{5 \text{ X } 10^{3} \text{ mR}}{(30 \text{ days})\ (24 \text{ hr/day})} \cong 7 \text{ mR/HR}$$

Note (2) This is based on the design value for this area during normal operation (assuming 1% failed fuel) according to ANO-1 SAR.

(continued next page)

QID: 0309 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** Possage

TUOI: A1LP-RO-ICS **Objective:** 40 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation System

Description: Knowledge of the effect that a loss or malfunctions of the NNIS will have on the following: MFW system.

K/A Number: K3.04 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.7	**SRO Select:**	No	**Taxonomy:**	An

Question: RO: 65 SRO:

Given:

- The plant is operating at 100% power.
- Loop "A" T-cold Narrow Range Temperature instrument fails HIGH.

If this instrument was hard selected by the SASS selector switch, what ICS HAND/AUTO stations should be placed in HAND per 1203.001, ICS Abnormal Operation?

A. Both Feedwater Loop Demands, Reactor Demand and Diamond Panel.

B. SG/Rx Master, Loop Delta Tc and Reactor Demand

C. Both Feedwater Loop Demands, SG/Rx Master and Loop Delta Tc.

D. Both MFW Pumps, Loop Delta Tc and Turbine (EHC).

Answer:

A. Both Feedwater Loop Demands, Reactor Demand and Diamond Panel.

Notes:

A cold leg temperature instrument failure causes the reactor demand signal to drive rods inward due to a high indicated Tave. Feedwater flows are changed to balance loop cold leg temperatures.
"A" is correct. Both feedwater loop demand stations reactor demand and diamond panel must be taken to manual.
"B" is incorrect because feedwater is affected downstream of the SG/Rx Master, but plausible because it includes reactor demand.
"C" is incorrect because reactor demand is affected downstream of the SG/Rx Master, but plausible because it includes both FW loop demands
"D" is incorrect because the turbine is not affected but plausible since it includes loop delta Tc and a failure of a Tc instrument is given.

This matches the K/A since it involves a failure of an NNI instrument (A Tc) and evaluates knowledge of ICS handstations to take to "hand" to mitigate this failure on feedwater and the reactor.

Added AOP to stem per suggestion from NRC examiner. JWC 7/14/16

References:

1203.001, ICS Abnormal Operation

History:

2007 RO Exam.
Selected for 2016 exam

SECTION 3 – RCS T-cold High/Low

NOTE

Excore Nuclear instrumentation is inaccurate when RCS temperatures are different than when NIs were calibrated.

INSTRUCTIONS

1. **Place the following in HAND:**
   - Feedwater Loop A Demand H/A
   - Feedwater Loop B Demand H/A
2. **Place the following in HAND:**
   - Diamond Panel
   - RX Demand H/A
3. **Adjust the following as necessary to stabilize RCS temperature and maintain power < 100%:**
   - Diamond Panel
   - Feedwater Loop A Demand H/A
   - Feedwater Loop B Demand H/A
4. **Lower the Feedwater Loop H/A with the highest FW Flow to maintain $\Delta Tc < 5°F$.**
5. **Select the good RCS T-cold instrument for indication.**
   - Loop A T-Cold:
     - TT-1015
     - TT-1018
   - Loop B T-Cold:
     - TT-1044
     - TT-1048
6. **Proceed as directed by CRS/SM.**

END

RO

Tier 3

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1083 **Rev:** 1 **Rev Date:** 7/28/16 **Source:** New **Originator:** Cork

TUOI: ASLP-RO-OPSPR **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of operator responsibilities during all modes of plant operation.

K/A Number: 2.1.2 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.4 **SRO Select:** No **Taxonomy:** K

Question: RO: 66 SRO:

The Annunciator Control Periodic Review is performed to verify the continued need for each annunciator that has been removed from service or modified.

What positions are responsible for completing this review in accordance with 1015.001, Conduct of Operations?

A. Control Board Operator or I&C Superintendent

B. System Engineer or I&C Superintendent

C. Control Board Operator or STA

D. STA or System Engineer

Answer:

C. Control Board Operator or STA

Notes:

"C" is correct per 1015.001, Conduct of Operations, form 1015.001C. The form states this review shall be performed by a licensed operator or STA.
"A" is incorrect since the I&C Superintendent does not have this responsibiilty per 1015.001C but this is plausible due to the position and department, the System Engineer is notified whenever an annunciator is removed from service.
"B" is incorrect since neither of these positions are responsible for the review but plausible in that a System Engineer is notified whenever an annunciator is removed from service and I&C Supt. is plausible due to the position and department.
"D" is incorrect since the System Engineer (SE) does not have this responsibiilty per 1015.001C but plausible since the SE is notified whenever an annunciator is removed from service and the STA is one of the responsible positions.

This question matches the K/A since it involves a licensed operator responsibility which may be performed during any mode of plant operation.

NRC validation week: deleted "two" prior to "positions" from stem. JWC 7/28/16

References:

1015.001, Conduct of Operations

History:

New question for 2016 exam

ANNUNCIATOR CONTROL PERIODIC REVIEW

☐ UNIT 1 ☐ UNIT 2
(check one)

This review is completed by a Licensed Operator or STA to verify the continued need for each annunciator that has been removed from service or modified. (PMRQ (U1) 9670, (U2) 9671)

1.0 Review all Annunciator Removal From Service or Modification Sheets and perform the following. Record any corrective actions taken in the "Comments" section.

- IF required,
  THEN verify annunciator markers still in place.
- IF applicable,
  THEN verify WR/WO still active on Passport.
- IF applicable,
  THEN verify WO is flagged CRA (Control Room Alarm) as required by EN-FAP-OP-006.
- IF alternative method of monitoring listed,
  THEN verify method valid for present plant condition.
- IF any conditions no longer needed,
  THEN initiate action to restore annunciator.
- Verify Annunciator Out of Service Index current.

2.0 IF annunciator out of service > 60 days,
THEN perform the following:

2.1 Request a PAD review of affected annunciator(s) be completed before 90 days and attached to the applicable 1015.001B form in Annunciator Out of Service Binder.

2.2 IF annunciator out of service > 90 days
AND PAD review is not completed,
THEN initiate a Condition Report documenting that the PAD has not been completed
AND document the Condition Report number on the applicable step of the associated Annunciator Out of Service Form 1015.001B.

Comments: ____

____

____

____

Performed by ______________________________ Date ______________

Approved by ______________________________ Date ______________
CRS/SM

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0838 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** Bank **Originator:** J. Cork

TUOI: ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledge and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

K/A Number: 2.1.4 **CFR Reference:** 41.10 / 43.2

Tier: 3 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

Question: RO: 67 SRO:

For the purpose of maintaining an NRC operator's license, which of the following should be reported to the NRC?

A. A change in marital status.

B. A traffic citation for speeding.

C. A new diagnosis for high blood pressure.

D. An audit by the IRS of previous year's tax return.

Answer:

C. A new diagnosis for high blood pressure.

Notes:

Only "C" is requred to be reported per EN-NS-112 and 1063.008.
The others are plausible situations which can occur in life that are not required to be reported as part of an operator's license.

This question matches the K/A since it relates to an individual licensed operator responsibility to maintain an active license.

References:

1063.008, Operations Training Sequence

History:

New for 2011 RO Exam.
Selected for 2016 exam.

C. A licensed individual shall, as soon as possible, notify the Manager, Operations if during the term of the license the individual develops a physical or mental condition that could adversely affect the performance of assigned operator duties or cause operational errors. The facility shall notify the NRC within thirty (30) days of learning of the diagnosis.

The individual should then be directed to the Medical Review Officer for evaluation. Based on this evaluation, or items identified during normal license physical examinations, the Medical Review Officer should make any required restrictions known to the Superintendent, Operations Training, for submittal to the NRC for evaluation. This restriction should be reported to the regional office on NRC form 396, "Certification of Medial Examination by Facility Licensee" for review. This submittal will include a copy of all supporting medical information and recommended wording for the conditional license to be issued to the affected operator.

D. A licensed individual shall notify the NRC within 30 days about a conviction for a felony

E. An individual whose SRO license has become inactive may reactivate that license for the purpose of supervising refueling operations by completing all the requirements identified on form 1063.008B.

6.10.9 Monitoring for marginal performance

A. Guidance for monitoring student performance is provided in EN-TQ-114, Licensed Operator Requalification Training Program Description.

{3.2.20} **B. When a student's performance reaches criteria established in EN-TQ-114, Remedial Training Plans are developed and Academic Review Boards conducted IAW EN-TQ-201-04, SAT-Implementation Phase as necessary.**

1. Remedial Training plans are developed and documented using TQF-201-IM05, Remedial Training Plans.

2. Academic Review Boards are documented using TQF-201-IM06, Academic Review Board Recommendation.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1084 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** New **Originator:** Cork

TUOI: ASLP-RO-COPD **Objective:** DD **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

K/A Number: 2.1.37 **CFR Reference:** 41.1 / 43.6 / 45.6

Tier: 3 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.6 **SRO Select:** No **Taxonomy:** K

Question: RO: 68 SRO:

As a licensed operator you are responsible for compliance with COPD-030, ANO Reactivity Management Program.

Which of the following activities would require a Reactivity Management Brief prior to performance of the activity?

A. Raising the seal injection flow rate to RCP P-32A.

B. Bypassing the E-3/4A Feedwater Heaters.

C. Adding nitrogen to the Makeup Tank T-4.

D. Adjusting the reactive loading on the Main Generator.

Answer:

B. Bypassing the E-3/4A Feedwater Heaters.

Notes:

"B" is correct based on Att. 9.3 in COPD-030. Changing Feedwater flow rate or temperature will affect secondary power and thus will affect reactor power.
"A" is incorrect but plausible since this evolution will increase the amount of fluid going into the RCS, but seal injection is coming from the Makeup Tank so reactivity will not be affected.
"C" is incorrect but plausible since the Makeup Tank is part of Makeup and Purification which makes up to the RCS but changing Makeup Tank pressure will not affect reactivity.
"D" is incorrect but plausible as this contains the word "reactive" but changing reactive load will not change secondary power so there is no reactivity affect.

This question matches the K/A since it requires knowledge of what activity requires a RM brief per ANO's procedure for reactivity management.

Replaced A distractor at request of NRC examiner. JWC 7/14/16

References:

COPD-030, ANO Reactivity Management Program

History:

New question for 2016 exam

Section 1 — RMI Determination

1.0 Using the following table, determine whether the activity has a potential Reactivity Management Program Impact. Check any impacts that are applicable.

Reactivity Management Program Impact	
Procedure/Activity/Other Action ____________________	
Impact	**Screening Criteria**
☐	Affect nuclear fuel or the way nuclear fuel is operated, handled, or stored?
☐	Affect core components such as control rods, neutron sources, fuel rod storage baskets, or fuel rod encapsulation tubes?
☐	Affect control rod drive mechanisms or position indication?
☐	Change input instrumentation, addressable constants, or software that provides heat balance calculations, core thermal limit calculations, or core power distribution calculations?
☐	Affect reactivity calculations such as shutdown margin (SDM) or estimated critical configuration (ECC)?
☐	Affect incore or excore nuclear instrumentation, power monitoring ability (SRM, IRM, APRM, LPRM, RBM, or WRNM), or Power Monitoring Programs used by the operator (WRNM, Process Computer, Core Monitoring Code such as Powerplex / 3D-Monicore / PDMS etc)?
☐	Affect reactor manual control systems or components?
☐	Affect Reactor Protection System (RPS) components?
☐	Affect boration/dilution systems, chemicals, or components?
☐	Change the boron concentration in Reactor Coolant System (RCS), makeup systems, Emergency Core Cooling Systems (ECCS), or Spent Fuel Pool?
☐	Change the parameters or the indications Main Steam flow, Main Feedwater flow, Heater Drains flow, Main Feedwater temperature, Spent Fuel Pool Level, or Spent Fuel Pool temperature?
☐	Temperature changes of the reactor coolant by affecting Heater Drain Flow or Steam Flow?
☐	~~Core Flow with the vessel by affecting, Control Rod Drive System~~, or Reactor Water Cleanup Systems?
☐	Control Rod Position by affecting Rod Position Indication, Rod Positioning via RMCS, RWM, Rod Scrams via RPS?
☐	Change T_{HOT} or T_{COLD} or indications?
☐	Change steam generator pressure or indications?
☐	Affect main turbine control?
☐	Affect Steam Bypass Control System (SBCS) or the Atmospheric Dump Valves (ADVs)?
☐	Change core monitoring system or core protection systems software?
☐	Affect Reactor Power Cutback (RPC) function or group selection?
☐	Change Pressurizer or Reactor Coolant System pressure?
☐	Affect the concentration of reactor poisons (e.g., xenon, samarium, gadolinium, boron) in the Reactor fuel?
☐	Affect any reactor system procedures?
☐	Other, list details:

2.0 IF no items are checked in table above,
THEN the activity is not an RMI and no further actions are required.

3.0 IF any items are checked in table above,
THEN an SRO on the associated unit shall determine whether the activity actually has a Reactivity Management Impact, and if applicable, the associated "R" level.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0231 **Rev:** 2 **Rev Date:** 7/14/16 **Source:** Bank **Originator:** J.Cork

TUOI: FLP-OPS-ESOMS **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of tagging and clearance procedures.

K/A Number: 2.2.13 **CFR Reference:** 41.10 / 45.13

Tier:	3	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	4.3	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 69 SRO:

Which of the following conditions is correct with regard to preparation and installation authorization of a common unit tagout?

A. Installation may be authorized by either the Unit 1 or the Unit 2 Operations Supervisor.

B. Preparer may be non-licensed as long as the opposite unit reviewer is licensed.

C. Preparer and reviewer may be non-licensed if installation is authorized by a Unit Operations Supervisor from each unit.

D. Preparer and reviewer must include a licensed operator from each unit.

Answer:

D. Preparer and reviewer must include a licensed operator from each unit.

Notes:

"D" is correct because the procedure requires both the preparer and the reviewer preparing the tagout to be licensed on their respective unit one on Unit 1 and one on Unit 2.
"A" is incorrect but plausible because a common unit tagout requires both Unit's Operations Supervisors to approve it not just one.
"C" is incorrect, but plausible because both Unit Ops Supervisors must approve but the preparation & review must be done by licensed operators even though the non-licensed operator is qualified to do so.
"B" is incorrect but plausuble because the preparation & review must be done by licensed operators on their respective units even though the non-licensed operator is qualified to do so.

This question matches the K&A because to answer this question correctly requires knowledge of the tagging and clearance procedure.

Revised B and C choices at request of NRC examiner. JWC 7/14/16

References:

EN-OP-102, Attachement 9.5 Sheet 1.

History:

Developed for use in 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.
Selected for use on 2007 RO Exam.
Selected for use on 2016 RO Exam.

1.0 Arkansas Nuclear One

1.1 Common Tagouts (ANO 1 and ANO 2)

NOTE

The determination of whether a TAGOUT should be considered a COMMON TAGOUT is based upon whether the SSC may normally be operated by either unit (both units train and qualify on the system). Examples are not all inclusive.

- The following examples should be considered a COMMON TAGOUT based upon common operator qualification:
  - Primary Hydrogen System
  - Generator Hydrogen System
  - Liquid Nitrogen System
  - T-41B
  - Cardox System (components tagged on both units)
  - Vendor Supplied Demineralized Water Trailers
- The following examples by the shared nature of the systems should be considered a COMMON TAGOUT:
  - Instrument Air cross-ties
  - Turbine Building Crane
  - Fuel Handling Crane (L-3 and 2L-35)
  - MCC B81 (power to both units condensate vacuum degasifiers

1.1.1 IF a Tagout is determined to be Common, THEN Respond Yes to the COMMON TAGOUT Attribute.

1.1.2 IF a Tagout is determined to be Common, THEN a LICENSED OPERATOR from each unit shall review it. (For example: The preparer shall be Licensed Operators on one Unit and the reviewer shall be Licensed Operator on the other Unit.)

AND

Both unit OPERATIONS SUPERVISORS must authorize installation. Opposite Unit Supervisors Should sign into the eSOMS Clearance module and select their name from list in the Opposite Unit Supervisor Attribute.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1082 **Rev:** 0 **Rev Date:** 4/29/16 **Source:** New **Originator:** Cork

TUOI: ASLP-RO-PRCON **Objective:** 1 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for making changes to procedures.

K/A Number: 2.2.6 **CFR Reference:** 41.10 / 43.3 / 45.13

Tier:	3	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 70 SRO:

Being a part of Operations might require you to make a procedure change.

Which of the following would be regarded as a change to the INTENT of a procedure?

A. Adding text to clarify the purpose of a procedure step.

B. Changing the title of a position to correspond to corporate heirarchy.

C. Deleting a QC hold point in a procedure section for a filter change.

D. Adding a step to close an open configuration control loop.

Answer:

C. Deleting a QC hold point in a procedure section for a filter change.

Notes:

"C" is the correct answer per 1000.006, definition 4.9.2.
"A", "B", and "C" are common procedure changes and thus plausible, but do not constitute intent changes per 1000.006.

This question matches the K/A since it requires the candidate to recall part of the process of making a procedure change: the definition of an intent change.

References:

1000.006, Procedure Control

History:

New question for 2016 exam

shall reflect administrative guidelines established in Station Administrative or Departmental Administrative procedures.

4.8 INDEPENDENT VERIFICATION - See EN-HU-102. For additional guidance and requirements, refer to Attachment 10.

4.9 INTENT CHANGE - An intent change exists when it involves the following items:

4.9.1 <u>Changes the Purpose of the Procedure</u>

A purpose change would allow the procedure to be used to perform a function that was not intended by the originator, the OSRC or the cognizant authority. Changes to the purpose section of the procedure do not necessarily change the purpose of the procedure but if the change allows performance of a new, unrelated function, the change is an intent change.

4.9.2 <u>Changes the Scope of the Procedure</u>

A scope change would allow the purpose of the procedure to be applied to a component, subsystem or system for which it was not originally intended, or deletion of an activity applied to a component, subsystem, or system to maintain operability.

A scope change also involves: (1) Any deletion of a hold point or step(s) that have previously required a hold point, or (2) Addition of steps or methods that may require hold points such as the addition of a step(s) requiring verification activity.

4.9.3 <u>Degrades Controls Prescribed in Administrative Procedures</u>

A change that contradicts a station or departmental administrative procedure.

4.9.4 <u>Reduces the Level of Nuclear Safety</u>

A change that reduces the level of nuclear safety (regardless of whether margin still exists).

4.9.5 <u>Degrades Acceptance Criteria</u>

A change in the non-conservative direction (this may be both directions) or deletion such that the resultant set point could reduce the level of nuclear safety.

4.10 INTERIM APPROVAL - Approval process that may be used for revisions that do not involve an intent change or 50.59 Evaluation and are stopping work in progress. Refer to Attachment 1, Interim Approval Process.

Procedure No. ____________________ Revision No. ________________

Title ____

Originator ______________________________ Date _____________

The following are types of procedure changes that do not require a PAD Review. Documentation is established by indicating on the appropriate 1000.006 form that a PAD Review is not required and this form will be attached to the procedure change package. It is not necessary to complete any EN-LI-100 forms if a PAD Review is not required.

NOTE

All other changes, not programmatically excluded, require a PAD Review per EN-LI-100.

☐	1	Correcting grammar or spelling errors.
☐	2	Corrections to the numbering of steps, sections, forms, attachments, exhibits or pages without changing sequence.
☐	3	Addition/modification of text to improve clarity without changing process or intent.
☐	4	Correcting reference to step or section numbering (alpha/numeric) within the procedure.
☐	5	Correcting references to procedure titles, numbers, sections or steps of another procedure.
☐	6	Correcting obvious clerical/typographical errors that reference incorrect equipment/component designations/stock numbers (letters or numbers).
☐	7	Correcting references to equipment location, room number, general direction (north, south, etc.), elevation, or cabinet number.
☐	8	Cosmetic changes (i.e., affecting appearance only) that do not affect process or intent.
☐	9	Changing previously approved organization or individual titles.
☐	10	Adding/correcting references in the reference section or in the body of the procedure or adding a procedural step that references the use of another procedure.
☐	11	Incorporating information from approved Engineering Processes as long as the process has received a PAD Review in accordance with EN-LI-100, and the PAD/50.59 review(s) encompasses the changes being made. Reference and attach PAD/50.59 Review(s) for Engineering Process used: ____________________.
☐	12	Adding steps for gathering or disseminating information, e.g., recording data, making plant announcements, making calls to get information, etc.
☐	13	Adding steps to close configuration control loops where steps were obviously intended to exist.
☐	14	Adding, modifying or deleting steps or information in a procedure that have been evaluated or incorporated into another procedure.
☐	15	Adding the initial level of use designator to a procedure, changing the format or location of the level of use designator in accordance with approved procedures or changing the level of use designation.
☐	16	Adding, modifying or deleting IPTE requirements.
☐	17	Administrative changes made as part of the reactivity impact program.
☐	18	Adding, modifying or deleting portions of the Inservice Inspection (ISI) and Inservice Testing (IST) Programs that are controlled in accordance with 10CFR50.55a (e.g. changing acceptance criteria values for surveillances, etc.) Engineering process used (ECN, SEP, etc.):___________

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1081 **Rev:** 1 **Rev Date:** 7/14/16 **Source:** New **Originator:** Cork

TUOI: ESLP-GET-RWT01.07 **Objective:** 44 **Point Value:** 1

Section: 2.0 **Type:** Generic K&A

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

K/A Number: 2.3.12 **CFR Reference:** 41.12 / 45.9 / 45.10

Tier:	3	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 71 SRO:

You have been directed to perform a task in the Makeup Tank Room, a Locked High Radiation Area (LHRA).

Which of the following is a requirement per EN-RP-101, Access Control for Radiologically Controlled Areas, SPECIFICALLY for entry into the LHRA?

A. Red trip ticket

B. Continuous RP coverage

C. Approval by on-watch Shift Manager

D. Double PC garments

Answer:

B. Continuous RP coverage

Notes:

"B" is the correct answer per EN-RP-101, continuous RP coverage is required for workers in a field dose rate greater than or equal to 1R/hr which is the definition of an LHRA.
"A" is incorrect but plausible as this is required for HRA (high radiation area) as well as LHRA.
"C" is incorrect but plausible as this is required for entry into VHRA (very high radiation area).
"D" is incorrect but plausible, this may be required for highly contaminated areas but is not peculiar to LHRA entry.

This question matches the K/A since it requires the candidate to recall an essential and unique requirement for entry into a locked high radiation area.

Revised C &D, and stem per NRC examiner request. JWC 7/14/16

References:

EN-RP-101, Access Control for Radiologically Controlled Areas

History:

New for 2016 exam

5.3 RADIATION AREA ACCESS CONTROL

[1] Specific monitoring and radiological controls for access to Radiation Areas shall be listed on the appropriate RWP.

[2] As a minimum, each person entering a "Radiation Area" shall have:

- Dosimeter of Legal Record (DLR)
- Direct reading dosimeter
- Approved RWP
- White Trip Ticket

5.4 HIGH RADIATION AREA (HRA) ACCESS CONTROL

[1] High Radiation Area entry points require a barricade to prevent inadvertent access.

[2] **IF** the barricade for an HRA must be temporarily removed, **THEN**, an RP Technician may maintain direct "line-of-sight" surveillance of the access to the HRA until the access/barrier is re-secured and verified.

[3] Specific monitoring and radiological controls for access to High Radiation Areas are listed on the appropriate RWP.

[4] As a minimum, each person entering a "High Radiation Area" shall have :

- DLR
- Alarming direct reading dosimeter (Electronic Dosimeter)
- Stay Time (**IF** greater than 500 mrem per entry expected)
- Approved RWP
- Pre-Job briefing on radiological conditions in the area utilizing Attachment 9.9, "Typical High Radiation Area Brief Checklist"
- Red Trip Ticket

5.5 LOCKED HIGH RADIATION AREA ACCESS CONTROL

[1] Barricades and blocking devices shall be a minimum of 6 feet in height **AND** Installed in a manner such that they prevent unauthorized access.

[2] No ladders, equipment or material shall be stored around or used in a manner that would allow unauthorized access over the enclosure.

[3] **IF** a change in plant layout or radiological conditions occurs which results in areas with dose rates in excess of 1000 mRem/hr at 30 cm from the source of radiation or any surface that the radiation penetrates, **THEN** evaluate the use of locking gates **OR** "cocooning" in the affected area(s) to enhance access control **AND** to prevent unauthorized entry.

[4] **WHEN** using the cocooning method, **THEN** a sign on the barrier must be used to inform the radiation worker of the purpose of the barrier **AND** of the hazards if the barrier is removed or altered to gain access to the area.

[5] All entrances or access points to Locked High Radiation Areas shall be locked with a distinct LHRA lock for the area or room.

[6] Entrances or access points to LHRAs shall remain locked EXCEPT during periods of access by personnel under an approved RWP. The following guidelines shall be used:

(a) Lock each access to a LHRA, **OR**

(b) Establish an Access Control Guard to prevent unauthorized entry following the guidelines of section 5.10, **OR**

(c) Control access to an LHRA through the use of a barricade and red flashing light, subject to the conditions described and with RPM approval, as follows:

(1) **IF** no enclosure exists for purposes of locking a LHRA located within a large area such as containment, **AND** an enclosure can not be reasonably constructed, **THEN**

a. Ensure the provision for the use of flashing light is specified within the site's Technical Specifications for LHRAs.

b. Obtain written approval of the Radiation Protection Manager, or designee, using Attachment 9.3, "Approval for Locked High Radiation Area Deviations", for use of a barricade and red flashing lights to control access.

c. Once approved, barricade **AND**

d. Conspicuously post the area.

5.5 continued

e. Activate the red flashing light(s) as a warning device.

f. Instruct personnel working or traversing in the vicinity of these alternative controls as to their meaning and significance.

[7] Use a ladder lock, if appropriate, to control access to an LHRA.

(a) The ladder lock, if used shall be a minimum of 6 feet in length as per step 5.5[1].

(b) **WHEN** ladder locks are used to prevent unauthorized access to LHRAs, **THEN** ensure that BOTH sides of the ladder are blocked to prevent unauthorized access.

[8] Control LHRA shielded containers such as floor plugs, rad waste cubicles, filter housings, and outside shielded liner storage containers when the following are met:

(a) Dose rates greater than1R/hr at 30 cm, **AND**.

(b) Contents can be accessed through the use of local installed lifting devices or readily available mobile cranes, **AND**.

(c) Bolting is not in place to prevent access without tools.

(d) Controls may include:

- De-energize cranes with RP admin control of tag out;
- Use of RP-controlled locks on chain hoists;
- Use of RP-controlled locking nuts on plug bolts;
- Removal of lifting lugs used to remove plug and lugs are controlled by RP.

[9] Specific monitoring and radiological controls for access to Locked High Radiation Areas shall be made by RP Personnel and listed on the appropriate RWP.

[10] As a minimum, each person entering a Locked High Radiation Area shall have:

- DLR
- Alarming direct reading dosimeter (Electronic Dosimeter)
- Approved RWP
- RP Lead technician or RPS approval

5.5 continued

- **IF** workers are in a field dose rate greater than or equal to 1R/hr, OR worker dose is expected to be greater than 500 mrem per entry, **THEN** continuous RP coverage with the use of EN-RP-141, Attachment 9.1 "Radiological Stay Time Verification Sheet" is required.
- Radiation Protection Manager's approval for entry into LHRAs with general area dose rates greater than 1.5 Rem/hr in the actual work area.
- Documented pre-job brief for entry, given by RP personnel. RPS performs the pre-job brief for entry into LHRAs with general area dose rates greater than 1.5 Rem/hr in the actual work area. This brief shall cover:
    - Radiological conditions in the immediate work areas using most recent survey data available **AND**
    - The scope of the work to be perfomed
- Red Trip Ticket

[11] While LHRAs are open, the access to the LHRA shall be controlled in accordance with site-specific Technical Specifications.

[12] Turnover of radiological coverage by RP personnel during Locked High Radiation Area work should be avoided.

[13] **WHEN** transfer of the LHRA key is required, **THEN** perform in accordance with Section 5.11.

[14] The following verification shall follow the initial check by the access control guard or RPT and be documented on Attachment 9.6, "LHRA / VHRA Key Log."

(a) Upon re-establishing any LHRA boundary controls following work that required access into these areas, a second person shall verify the access point(s) are securely locked.

(b) This verification shall consist of ensuring the locking mechanism has been replaced, where removed, **AND** that the access point is shut **AND** locked.

(c) **IF** the person who performed the initial check was NOT an RPT, THEN the person performing the verification shall be an RPT.

(d) **WHEN** keys are required to lock doors, **THEN** verify that the door is closed **AND** secured/locked with a physical challenge of the door.

5.5 continued

(e) **IF** the door is locked using a padlock and chain or cable, **THEN** inspect the lock and chain or cable for defects, **AND** physically challenge the lock.

(f) **IF** deficiencies are found during this inspection **THEN** immediately report the deficiency RP Supervision **AND** document the deficiency in a Condition Report.

[15] **IF** a new, unanticipated LHRA is discovered, **THEN** perform the following:

(a) Ensure all personnel are immediately evacuated from the area and direct them to report to RP.

(b) Guard the area and prohibit unauthorized access.

(c) Maintain control of the area at all times. **DO NOT** leave the area unguarded for any reason until proper procedural radiological controls have been established.

NOTE

Decisions regarding operation of plant equipment and systems are the sole responsibility of licensed Operations personnel. Due to plant conditions, it may not be possible or advisable for Operations to implement requests for equipment or system status changes.

(d) **IF** a request is made to Operations to secure equipment or a system lineup to address LHRA conditions, **THEN** confirm the action has occurred and verify the LHRA condition has been eliminated prior to leaving the area unguarded.

(e) Initiate a Condition Report to document this occurrence.

5.6 VERY HIGH RADIATION AREA ACCESS CONTROL

CAUTION

To the extent possible, entry into a VHRA should be forbidden unless there is a sound operational or safety reason for entering.

Without proper controls and monitoring, personnel entering these areas could receive radiation exposure with severe or life-threatening consequences.

[1] Barricades and blocking devices shall completely enclose the Very High Radiation Area sufficient to thwart undetected circumvention of the barrier.

[2] Fencing or walls around a Very High Radiation Area should extend to the overhead and preclude anyone from climbing over the barricade.

[3] All entrances or access points to Very High Radiation Areas shall be locked with a unique lock **AND** conspicuously posted.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0751 **Rev:** 1 **Rev Date:** 7/11/2008 **Source:** Bank **Originator:** Spullin

TUOI: ASLP-RO-RADPRO **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledge's and Abilities

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier: 3	**RO Imp:** 3.2	**RO Select:** Yes	**Difficulty:** 3
Group:	**SRO Imp:** 3.7	**SRO Select:** No	**Taxonomy:** K

Question: RO: 72 SRO:

Which of the following exposure limits are
Entergy's Routine Annual Administrative Guidelines?

A. TEDE 2000 mrem per year; SDE, WB= 40 rem; and LDE= 12 rem

B. TEDE 5000 mrem per year; SDE, WB= 40 rem; LDE= 12 rem

C. TEDE 5000 mrem per year; SDE, WB= 50 rem; LDE= 15 rem

D. TEDE 2000 mrem per year; SDE, WB= 50 rem; LDE= 15 rem

Answer:

A. TEDE 2000 mrem per year; SDE, WB= 40 rem; and LDE= 12 rem

Notes:

A. is correct, the limits are Entergy Routine Annual Administrative Guidelines
B. is incorrect but plausible as these are the Maximum Annual Administrative guidelines for Entergy
C. is incorrect, but plausible as these are the Annual Regulatory limits
D. is incorrect, but plausible as they are a mix of different limits

This question matches the K/A since it requires the candidate to recall one of the normal radiation exposure limits.

References:

EN-RP-201, Dosimetry Administration

History:

New for the 2008 RO exam
Selected for 2016 exam

5.3, continued

[2] Maximum Annual Administrative Guidelines

- TEDE = 4.5 rem
- LDE = 12 rem
- SDE, WB = 40 rem
- SDE, ME = 40 rem
- Declared Pregnant Woman (DPW) TEDE = 50 mrem/month, 450 mrem/gestation period.
- Minors TEDE – Minors are not allowed access to RCAs.
- Unmonitored individual TEDE = 100 mrem/year
- Members of the Public TEDE = 100 mrem/year

[3] Routine Annual Administrative Guidelines

- TEDE = The lesser of:

  2000 mrem per year **OR**

  5000 mrem – (1250 mrem x UQ per year)

  Where UQ = the number of undocumented quarters for the current year

  (EXCEPT when Lifetime TEDE is greater than or equal to the individuals age x 1 rem in which case the annual TEDE guideline will be set to 1 rem.)
- LDE = 12 rem
- SDE, WB = 40 rem
- SDE, ME = 40 rem
- TODE = 40 rem

QID: 0242 **Rev:** 0 **Rev Date:** 9-1-99 **Source:** Bank **Originator:** D. Slusher

TUOI: A1LP-RO-NNI **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Ability to identify post-accident instrumentation.

K/A Number: 2.4.3 **CFR Reference:** CFR: 41.6/45.4

Tier: 3 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

Group: G **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** K

Question: RO: 73 SRO:

What instruments are marked with a green dot?

A. Instruments designated for use during an alternate shutdown.

B. Instruments that should be reliable during accident conditions.

C. Instruments the Shift Engineer uses after a reactor trip.

D. Instruments designated as important to the Emergency Plan.

Answer:

B. Instruments that should be reliable during accident conditions.

Notes:

"B" is correct because instruments which are reliable and to be used for accident conditions are marked by a green dot as required by Reg Guide 1.97 and IAW OP 1305.028 Section 3.0.
"A" is incorrect but plausible because SPDS instrumentation is designated for the alternate shutdown AOP OP-1203.002 Attachment 10.
"C" is incorrect but plausible because System Engineering instruments used after a reactor trip are designated by the Post Transient Procedure Admin Procedure OP-1015.037 Attachment I.
"D" is incorrect but plalusible since equipment important to the Emergency Plan is identified in 1903.069 but there are no specific instrument markings for control room instrumentation.

This Question matches the K&A because the candidate must have the ability to identify which instruments he can use post accident and the ones he will use have the green dots on the indicator.

References:

1305.028, Reg Guide 1.97 Instrumentation Verification

History:

Developed for 1999 exam.
Selected for the 2010 RO/SRO Exam
Selected for the 2016 RO/SRO Exam

1.0 PURPOSE

To provide a listing of Unit 1 Reg Guide 1.97 instruments.

To provide test supplements for ensuring the instrumentation used for accident mitigation is identified and available.

2.0 SCOPE

This verification satisfies in part an NRC commitment to control and maintain identification of Reg Guide 1.97 post accident monitoring instrument indicators.

Attachment A contains a list of Reg Guide 1.97 instruments that are channel checked by either Operations Logs or surveillance procedures.

This procedure contains the following test supplements:

- Supplement 1 - Checks those indicators required to be marked in the Unit 1 Control Room on a refueling frequency.
- Supplement 2 - Verifies computer and panel indications available on a monthly frequency. This test satisfies SR 3.3.15.1 (17a, 17b, 18, 19 and 20).

3.0 DESCRIPTION

Attachment A provides a list which may be consulted prior to altering other tests or surveillances so that the required checks of these instruments continue.

Reg Guide 1.97 states that, "The instruments designated as Type A, B, and C, and Categories 1 and 2 should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions."

For Unit 1, instrument indicators designed to be relied upon in post accident conditions are to be conspicuously labeled with a green dot. These indicators consist of Reg Guide 1.97, Category 1 and 2 indicators and some additional environmentally qualified (EQ) instruments. Only those instruments which supply analog information, that is actual parameter values will be labeled. On/off and open/close light indications will not be labeled.

3.1 Supplement 2 contains channel checks on EFW, HPI, LPI, and RB Spray flow instrumentation.

It is expected that these checks will be made when there is no flow present. When actual flow is zero, SPDS can indicate that the instrument is bad. Therefore, the channel checks use the Point Maintenance function, checking that the transducer voltage signal is reading a live zero. This is ~1 vdc on 1 to 5 volt instruments. Zero volts is an indication of a failed point.

If flow is present, the channel check ensures available instrumentation reading the same parameter compares favorably.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0051 **Rev:** 2 **Rev Date:** 7/14/2016 **Source:** Bank **Originator:** GGiles

TUOI: ASLP-RO-EPLAN **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of general operating crew responsibilities during emergency operations:

K/A Number: 2.4.12 **CFR Reference:** 41.10 / 45.12

Tier: 3 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** K

Question: RO: 74 SRO:

A General Emergency has been declared on Unit 1.

The Shift Manager has announced a plant evacuation.

Which of the following actions should be performed?

A. All Operations personnel on watch should report to the Control Room.

B. All on watch Operations personnel, with the exception of the control room staff, should report to the Operations Support Center (OSC).

C. All non-watchstanding Operations personnel (training/support) should report to the Technical Support Center (TSC).

D. All Operations personnel, with the exception of the on watch Operations personnel, should evacuate the plant.

Answer:

A. All Operations personnel on watch should report to the Control Room.

Notes:

"A" is correct IAW OP-1903.030 Step 6.2.2.B.
"B" is incorrect but plausible because on shift crew members should report to the control room and all other operation personnel should report to the OSC IAW OP-1903.030 Step 6.2.2.D.
"C" is incorrect but plausible because all other shift operations personnel should report to the OSC not the TSC.
"D" is incorrect but plausible because Non-essential personnel will be evacuating the plant.

This question matches the K&A because it describes the general responsibilities of an operating crew during a plant emergency evacuation.

Changed "on duty" to "on watch" or to "non-watchstanding" where applicable, per NRC examiner suggestion. JWC 7/14/16

References:

1903.030, Evacuation, Sections 6.2.2.B, & C

History:

Developed for 1998 SRO Exam.
Used on the 2016 NRC Exam.

6.2 IMMEDIATE ACTIONS

6.2.1 Localized Evacuation

A. **IF** the conditions listed in section 6.1.1 are observed,
THEN consider a localized evacuation of the affected area(s). Use Form 1903.030C, "Localized Evacuation Checklist", to perform a localized evaluation.

B. A brief description of the control room hand switch settings for the Evacuation Alarm System is outlined below:

1. Unit 1 - Three (3) position hand switch:

a. **REAC. BLD**. - Activates the Unit 1 Reactor Building Alarm System only

b. **OFF** - Deactivates the Evacuation Alarm System

c. **PHY. PLT**. - Activates the Evacuation Alarm System for the entire physical plant including surrounding buildings located within the protected area.

2. Unit 2 - Four (4) position hand switch:

a. **OFF** - Deactivates the Evacuation Alarm System

b. **CTMT** - Activates the Unit 2 Reactor Building Alarm system only

c. **CTMT AUX** - Activates only the Unit 2 Reactor Building and Unit 2 portion of the Auxiliary Building Alarm System

d. **PLANT** - Activates the Evacuation Alarm System for the entire physical plant including surrounding buildings located within the protected area.

6.2.2 Plant Evacuation

A. Use Form 1903.030B, "Plant Evacuation Checklist", to determine if a plant evacuation is advisable. For a plant evacuation based on a declaration of a SAE or GE, use forms 1903.011P, Q, R, S, T, or U.

B. On duty Shift Operations personnel should report to the Control Room.

C. All other Shift Operations personnel (training/support) should report to the OSC Assembly Area (located in the Maintenance Facility) and report to the OSC Manager.

D. The following groups should immediately report to their designated emergency work location and notify their immediate supervisor of their location for accountability purposes.

1. Emergency Response Organization (ERO) personnel to the following:
    - EOF
    - TSC
    - OSC
    - JIC

2. Maintenance, Chemistry and Radiation Protection craft personnel should report to the OSC Assembly Area located in the Maintenance Facility.

E. Maintenance personnel who are not a part of the OSC staff or are not assigned a position or task in the Emergency Response Organization, should report to the EOF and standby for instructions (See Procedure 1903.067, "Emergency Response Facility - Emergency Operation Facility (EOF)", for assembly location).

F. System Engineering should report to the EOF.

6.2.3 Exclusion Area Evacuation

A. When any of the conditions listed in step 6.1.3 are detected an exclusion area evacuation should be initiated using Form 1903.030B, "Plant Evacuation Checklist".

B. Coordinate with Security Personnel for the evacuation and control of the exclusion area (including the Generation Support Building).

1. Under most conditions, the exclusion area should be evacuated within one hour.

2. The Exclusion Area should be patrolled at least once every 2 hours thereafter (if conditions allow).

C. Non-Entergy personnel evacuating from the affected areas of the Exclusion Area follow designated evacuation routes under the supervision of Security personnel while on site. Once offsite, personnel proceed as directed by State and local authorities.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0848 **Rev:** 3 **Rev Date:** 7/28/16 **Source:** Modified **Originator:** J. Cork

TUOI: A1LP-RO-FPS **Objective:** 10 **Point Value:** 1

Section: 2 **Type:** Generic K/A's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of fire protection procedures.

K/A Number: 2.4.25 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	3	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	K

Question: RO: 75 SRO:

You are standing watch and it is now 1630 on 08/26/2016.

Which of the following would be considered a fire system impairment in accordance with 1000.120, ANO Fire Impairment Program?

A. P-6A Electric Fire Pump non-functional due to on-going surveillance since 0400.

B. A smoke detector string in Corridor 98 defeated for PMs since 0900.

C. Fire hose station in Aux Bldg 335 elevation isolated since 1030 for hose replacement.

D. Control Room Halon System #3 defeated for corrective maintenance since 0830.

Answer:

A. P-6A Electric Fire Pump non-functional due to on-going surveillance at 0600.

Notes:

A is correct since per 1000.120 this non-functionality has not been corrected prior to the end of the night shift and has carried over to day shift.
B, C, and D are incorrect but plausible since they are non-functional but any systems out of service for less than one shift for surveillances, corrective maintenance, or PMs are not considered an impairment per 1003.002.

Revised conditions and times to make "A" correct, this is a modified question.

This question matches the K/A as it applies to a Tier 3 topic: it requires the candidate to recall portions of a fire protection procedure (1000.120) and to ascertain which condition requires the performance of the administrative task of reporting a fire impairment.

Changed A time to 0400 and all "at" before times to "since" per request of NRC examiner. JWC 7/14/16
Changed time now from 1300 to 1630 to ensure 12 hour difference for correct answer due to validator suggestion. JWC 7/28/16

References:

1000.120, ANO Fire Impairment Program

History:

New question for 2011 exam.
Modified for 2016 exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

History:

New, not used. Used on 2011 Audit exam.

QID: 0848 **Rev:** 0 **Rev Date:** 06/24/11 **Source:** New **Originator:** J. Cork

TUOI: A1LP-RO-FPS **Objective:** 10 **Point Value:** 1

Section: 2 **Type:** Generic K/A's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of fire protection procedures.

K/A Number: 2.4.25 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 3 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 2

Group: G **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** C

Question: **RO:** **SRO:**

You are standing watch and it is now 1500 on 08/26/2011.

Which of the following would be considered a fire system impairment in accordance with 1003.002, Insurance Impairment Reporting?

A. P-6A Electric Fire Pump inoperable due to corrective maintenance at 0800.

B. A smoke detector string in Corridor 98 defeated for PMs at 0900.

C. Leaking fire hose station in Aux Bldg 335 elevation isolated at 1030.

D. Control Room Halon System #3 failed surveillance at 0530.

Answer:

D. Control Room Halon System #3 failed surveillance at 0530.

Notes:

D is correct since this is an impairment..
A, B, and C per 1003.002 are incorrect since any systems out of service for less than one shift for surveillances, corrective maintenance, or PMs are not considered an impairment.

References:

1003.002, Chg. 004

History:

New question created for 20011 RO Exam.

PRIOR TO REVISION

receive communications directly from the Control Room in case Control Room Isolation is required. A Control Room Operator cannot serve the function of a CRHB Watch.

4.6 ELECTRONIC DATABASE- An electronic tool to assist with the implementation of the requirements of this procedure.

4.7 FIRE IMPAIRMENT

4.7.1 Fire Protection System impairments exist when a fire protection feature is not operable or otherwise available to fulfill its design function due to failure; OR, when maintenance, surveillance or test of a fire protection feature lasts greater than 1 shift. Fire suppression or detection systems that are out of service for less than one shift due to routine surveillances, corrective maintenance or PMs are not considered to be a reportable impairment.

Failure of any of the components or support systems listed below constitutes a reportable system impairment.

A. Fire pump
B. Sprinkler system (wet or dry)
C. Deluge water spray system
D. Loop sectionalizing valve
E. Halon, FM-200 and CO2 systems
F. Fire hydrant
G. Fire hose station
H. Fire water piping
I. Any detection system
J. Fire barriers, Fire doors, and Fire barrier penetration seals

4.7.2 Fire Protection Program impairment establishes a compensatory measure to maintain the administrative elements of the Fire Protection Program. Examples would include fire watch posting per EN-DC-161, "Control of Combustibles," requirements or by other measures as required by the Fire Marshal or designee to compensate for non-compliant conditions.

SRO

Tier 1

(all)

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1100 **Rev:** 0 **Rev Date:** 6/17/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP02 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 009 **System Title:** Small Break LOCA

Description: Ability to determine or interpret the following as they apply to a small break LOCA: Adequate core cooling.

K/A Number: EA2.39 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 4.3 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** C

Question: RO: SRO: 76

Given:
- A small break LOCA has occurred.
- ESAS actuated on low RCS pressure.
- Subcooling margin (SCM) was lost and all RCP's were tripped.
- 1202.002, Loss of Subcooling Margin is in use.

The break has been isolated.

Current plant conditions are:
- RCS pressure1800 psig and slowly rising
- CET average 552 °F (Thot temps ~ the same) and lowering
- Thot/Tcold delta temperature dropping
- SG pressures are being controlled by ADVs and ATC operator
- Tcold tracking SG Tsat

Which procedure should be transitioned to given the above conditions?

A. Small Break LOCA Cooldown, 1203.041

B. Reactor Trip, 1202.001

C. ESAS, 1202.007

D. Natural Circulation Cooldown, 1203.013

Answer:

B. Reactor Trip, 1202.001

Notes:

"B" is correct. The condtions given show the loss of adequate SCM has been corrected, SCM is adequate based on RCS pressure and CETs, and primary to secondary heat transfer is in progress. In accordance with step 19 of 1202.002, Loss of Subcooling Margin, the CRS will transition to 1202.001, Reactor Trip, to complete analysis of plant status.
"A" is incorrect, but plausible since step 19 contingency action of 1202.002 would require a transition to 1203.041 if an uncontrolled RCS cooldown was occurring due to HPI break flow.
"C" is incorrect but plausible since ESAS had actuated.
"D" is incorrect but plausible since step 19 contingency action of 1202.002 would require a transition to 1203.013 if RCS leak were unisolable, RCS press was > 150 psig, and RCPs were not available.

This question is SRO only since it relates to 10CFR55.43(b)(5), assessment of facility condtions and selection of procedures. The candidate is presented with facility conditions and must be able to select the appropriate procedure to transition to.

This question matches the K/A as it requires the candidate to have the ability to determine if core cooling is adequate following a small break LOCA.

References:

1202.002, Loss of Subcooling Margin

History:

New question for 2016 SRO exam

INSTRUCTIONS	CONTINGENCY ACTIONS
19. <u>IF</u> cause of loss of adequate SCM is corrected <u>AND</u> SCM is adequate <u>AND</u> primary to secondary heat transfer is in progress, <u>THEN</u> GO TO 1202.001, "REACTOR TRIP" procedure.	19. Perform the following: A. <u>**IF**</u> an uncontrolled RCS cooldown is occurring due to HPI/break flow, regardless of SG status, <u>**THEN**</u> **GO TO Small Break LOCA Cooldown (1203.041)** procedure. B. <u>**IF**</u> RCS leak is un-isolable <u>**AND**</u> RCS press remains ≥ 150 psig, <u>**THEN**</u> perform the following: 1) <u>**IF**</u> all RCPs are off, <u>**THEN**</u> perform the following: a) <u>**IF**</u> RCPs are available, <u>**THEN**</u> start one RCP in each loop (RT-11). b) <u>**IF**</u> RCPs are <u>**not**</u> available, <u>**THEN**</u> **GO TO Natural Circulation Cooldown (1203.013)** procedure, "Offsite Power Available" section. 2) <u>**IF**</u> RCPs are running, <u>**THEN**</u> **GO TO Forced Flow Cooldown (1203.040)** procedure.

(19. CONTINUED ON NEXT PAGE)

QID: 0639 **Rev:** 1 **Rev Date:** 5/4/16 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-ADHR **Objective:** 9 **Point Value:** 1

Section: 4.2 **Type:** Generic Abnormal Plant Evolutions

System Number: 025 **System Title:** Loss of Residual Heat Removal System (RHRS)

Description: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere.

K/A Number: AA2.02 **CFR Reference:** 43.5

Tier: 1	**RO Imp:** 3.4	**RO Select:** No	**Difficulty:** 3
Group: 1	**SRO Imp:** 3.8	**SRO Select:** Yes	**Taxonomy:** C

Question: **RO:** **SRO:** 77

Given:
- Plant is in Mode 5, cooldown in progress for refueling outage
- Both Decay Heat Removal trains are in service
- Both Decay Heat Removal flows are steady at ~1900 gpm.

Then the following occurs:
- K10-B2 "PROCESS MONITOR RADIATION HIGH" alarms
- CBOT reports RI-3809, Loop A DH Process Rad Monitor is in alarm

For the above conditions which of the following actions are required and which section of 1203.028, Loss of Decay Heat Removal, should be in use?

A. Stop P-34A DH pump and close P-34A suction valve from RCS (CV-1434) per Section 3, DH Removal System Leak >20 GPM

B. Stop both P-34A and P-34B DH pumps and close DH suction valve (CV-1050) per Section 9, Loss of Both DH Systems - RCS Pressure Boundary Intact

C. Stop both P-34A and P-34B DH pumps and close DH suction valve (CV-1050) per Section 3, DH Removal System Leak >20 GPM

D. Stop P-34A DH pump and close P-34A suction valve from RCS (CV-1434) per Section 9, Loss of Both DH Systems - RCS Pressure Boundary Intact

Answer:

A. Stop P-34A DH pump and close P-34A suction valve from RCS (CV-1434) per Section 3, DH Removal System Leak >20 GPM

Notes:

Answer "A" is correct, the alarm response procedure directs a transition to 1203.028 and Section 3 of the AOP contains actions for a DH cooler leak into Service Water at step 14 and only the affected pump should be stopped and its associated suction valve closed.
"B" is incorrect, but plausible since stopping both pumps is appropriate IF the operator closes common suction valve CV-1050. This section would be used for a condition affecting both pumps but should not be used for a condition only affecting the A pump.
"C" is incorrect, but plausible since this is the correct action but Section 9 will have the operator close a common suction isolation such as CV-1050.
"D" is incorrect, but plausible since this is the correct action but the wrong procedure section.

This question is SRO only since it meets 10CFR55.43(b)(5): the question requires the candidate to evaluate the conditions given and to select the appropriate procedure and action within that procedure which would assist in mitigating the event. It cannot be answered solely by knowing the major mitigative strategy nor can it be answered solely by knowing entry conditions

This question meets the K/A since it requires the candidate to assess the malfunction of the DH system, determine that it is DH cooler leakage and select the procedure section and appropriate action to mitigate the malfunction.

References:

1203.028, Loss of Decay Heat Removal
1203.012I, Annunciator K10 Corrective Action

History:

New, created for 2007 SRO exam.
Selected for 2011 SRO Exam.
Modified extensively for 2016 SRO exam due to previous version no longer meeting SRO Only question standards.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0639 **Rev:** 0 **Rev Date:** 10/18/2006 **Source:** Direct **Originator:** Cork/Possage
TUOI: A1LP-RO-ADHR **Objective:** 9 **Point Value:** 1

Section: 4.2 **Type:** Generic Abnormal Plant Evolutions
System Number: 025 **System Title:** Loss of Residual Heat Removal System (RHRS)
Description: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.
K/A Number: 2.4.4 **CFR Reference:** 41.10 / 43.2 / 45.6

Tier:	1	**RO Imp:**	4.5	**RO Select:**	No	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** **SRO:**

Given:
- Plant is in Mode 5
- "A" Decay Heat Removal is in service
- P-34A Decay Heat Removal flow is steady at 1900 gpm.
- K10-B2 "PROCESS MONITOR RADIATION HIGH" in alarm
- RI-3809, Loop A DH Process Rad Monitor in alarm

Prior to Revision

What procedure should be used to address the above conditions?

A. 1203.014, Control of Secondary System Contamination

B. 1203.028, Loss of Decay Heat Removal

C. 1203.030, Loss of Service Water

D. 1203.039, Excess RCS Leakage

Answer:

B. 1203.028, Loss of Decay Heat Removal

Notes:

Answer "B" is correct Loss of Decay Heat Removal procedure will address the cooler leak into Service Water.
Answer "A" is incorrect, although a leak of RCS via the DH cooler would cause a contamination concern 1203.014 deals with a SG tube leak.
Answer "C" is incorrect, although the leak will be into the service water system, 1203.030 is not needed.
Answer "D" is incorrect, although RCS leakage is present and 1203.039 does deal with intersystem LOCA's, it does not address a decay heat cooler leak.

References:

1203.012I, Chg. 048

History:

New, created for 2007 SRO exam.
Selected for 2011 SRO Exam.

	PROCESS/AREA MONITOR		SERVICE WATER		CFT	MAKEUP AND PURIFICATION		
A	RDACS RADIATION HI PAGE 2 •		SERV WATER PUMP TRIP PAGE 28	SW BAY LEVEL LO PAGE 40 •	CFT A PRESS HI/LO PAGE 53	HPI PUMP TRIP PAGE 58		LETDOWN TEMP HI PAGE 74
B	AREA MONITOR RADIATION HI PAGE 3	PROC MONITOR RADIATION HI PAGE 10 •	SW PUMP DISCH PRESS HI PAGE 33	SW PUMP BRG TEMP HI PAGE 42 •	CFT B PRESS HI/LO PAGE 54	MU FLOW HI PAGE 60	MU TANK LEVEL HI/LO PAGE 67	MU TANK PRESS HI/LO PAGE 75
C	RADIATION MONITOR TROUBLE PAGE 5 •	PROC MONITOR FLOW TROUBLE PAGE 25 •	SW PUMP P4A STRAINER ΔP HI PAGE 35	SW PUMP MTR WDG TEMP HI PAGE 46 •	CFT A LEVEL HI/LO PAGE 55	HPI PUMP P36A OIL PRESS LO LO PAGE 61	HPI PUMP P36B OIL PRESS LO LO PAGE 68	HPI PUMP P36C OIL PRESS LO LO PAGE 77
D	CONTROL ROOM SUPPLY DUCT RADIATION HI PAGE 8	TB DRAIN RAD MONITOR FLOW TROUBLE PAGE 27	SW PUMP P4B STRAINER ΔP HI PAGE 38	SW PUMP P4C STRAINER ΔP HI PAGE 50	CFT B LEVEL HI/LO PAGE 56	HPI PUMP P36A OIL PRESS LO PAGE 62	HPI PUMP P36B OIL PRESS LO PAGE 69	HPI PUMP P36C OIL PRESS LO PAGE 78
E					ISOL VLV OPEN RCS PRESS LO PAGE 57	HPI PUMP/MTR BRG TEMP HI PAGE 64 •	HPI PUMP MTR WDG TEMP HI PAGE 71	
F	RAD MONITOR TEST IN PROGRESS PAGE 9					MU SYS F3A/B FILTER Δ P HI PAGE 66	MU SYS F25 FILTER Δ P HI PAGE 73	HPI/PZR AUX SPRAY PIPING TEMP CHANGE PAGE 80
	1	2	3	4	5	6	7	8

• SIGNIFIES REFLASH CAPABILITY.

Location: C16

Device and Setpoint:

Any process monitor in Radiation Monitoring System Panel (C25 Bays 1 thru 3)	HIGH ALARM or loss of power	PROC MONITOR RADIATION HI Alarm: K10-B2
OR		
Turb Bldg Drn Rad Monitor (RI-5641)	HIGH ALARM or loss of power	

Monitors are listed in step 4.

1.0 OPERATOR ACTIONS

1. Check panels C486-2 and C25 (Bays 1, 2, 3) to determine which process monitor is in alarm.

   A. IF alarm is on RB Atmos Gaseous Monitor (RI-7461), THEN GO TO step 13.

2. Confirm alarm as follows:

   A. Verify drawer has power.

      1) IF Turb Bldg Drn Rad Monitor (RI-5641) is de-energized, THEN initiate steps to have problem investigated and corrected.

      2) IF process monitor on C25 is de-energized, THEN GO TO RADIATION MONITOR TROUBLE (K10-C1).

   B. Verify FAILURE ALARM light is off.

   C. Compare counts to alarm setpoint.

   D. Verify drawer fasteners are secure.

> **NOTE**
> Instantaneous spiking for the purposes of this procedure is the step rise and subsequent fall in process monitor count rate that is NOT indicative of an upward trend.

3. IF alarm was caused by instantaneous spiking, THEN reset alarm by performing the following:

   A. IF Gaseous Radwaste (RI-4830), THEN GO TO step 15.

(Step 3 continued next page)

(step 3, continued)

B. IF any other alarm [Not Gaseous Radwaste (RI-4830)], THEN perform the following to reset alarm:

1) IF background values are fluctuating, causing frequent alarms AND it is desired to adjust the setpoint, THEN adjust alarm setpoint to value determined by CRS, NOT to exceed the applicable process monitor acceptable limit found in Radiation Monitoring System Check and Test (1305.001), "Process Monitor Monthly Alarm Check" supplement.

2) IF the alarm setpoint was adjusted in the previous step OR instantaneous spike without desire to adjust setpoint, THEN select "ALARM RESET" on the appropriate drawer AND ensure bay door fasteners are properly secured.

3) Update the Plant Computer as applicable with the new alarm setpoint using the DBM function and exit this procedure.

4. Locate applicable monitor below AND proceed as directed:

- IF Liquid Radwaste (RI-4642), THEN GO TO Liquid Waste Discharge Line High Radiation Alarm (1203.007).
- IF Failed Fuel Gross/Iodine (RI-1237), THEN GO TO High Activity in Reactor Coolant (1203.019).
- IF Service Water Loop 1 (RI-3814), THEN GO TO step 5.
- IF Service Water Loop 2 (RI-3815), THEN GO TO step 6.
- IF Decay heat Loop A (RI-3809), THEN GO TO step 7.
- IF Decay Heat Loop B (RI-3810), THEN GO TO step 8.
- IF Nuc ICW Monitor(RI-2236), THEN GO TO step 9.
- IF Non-Nuc Monitor(RI-2237), THEN GO TO step 10.
- IF Main Condenser (RI-3632), THEN GO TO step 11.
- IF Discharge Flume (RI-3618), THEN GO TO step 12.

(Step 4 continued next page)

(step 6, continued)

CAUTION

A break in SW piping to RB cooling inside RB would dilute boron concentration in RB sump.

NOTE

Transmitters for loop I and loop II flow are not environmentally qualified for LOCA conditions.

C. Monitor service water flow to RB cooling loop and RB pressure on SPDS:

- RB Cooler SW Flow Loop I (F3816)
- RB Cooler SW Flow Loop II (F3817)

D. IF large SW line break is indicated by very low flow in loop II as compared to loop I, THEN isolate VCC-2C and VCC-2D by closing the following valves from C16:

- RB Cooling Coils Inlet (CV-3813)
- RB Cooling Coils Outlet (CV-3815)
- CV-3813 Bypass (SV-3813)

E. Notify Chemistry that a potentially radioactive discharge to lake may have occurred.

7. IF Decay Heat Loop A (RI-3809, Bay 1) radiation is high, THEN perform the following:

A. IF cooling in decay heat removal mode, THEN GO TO applicable section of Loss of Decay Heat Removal (1203.028).

B. IF ESAS is actuated AND DH Cooler (E-35A) is required to remain in-service, THEN perform "Transferring Service Water Bays from Lake to Emergency Cooling Pond" section of Service Water and Auxiliary Cooling System (1104.029).

C. IF "A" DHR train is NOT required to remain in-service, THEN secure LPI Pump (P-34A) and DH Cooler (E-35A) per appropriate section of Decay Heat Removal Operating Procedure (1104.004).

D. Initiate steps to have Service Water samples taken.

E. Initiate steps to have repairs made.

SECTION 3 – DH REMOVAL SYSTEM LEAK >20 GPM

INSTRUCTIONS

1. **IF leak is associated with a single DH pump AND both DH pumps in service, THEN GO TO step 14.**

2. **Stop the running DH pump(s).**

3. **Close at least one Decay Heat Suction valve:**

   - **CV-1050**
   - **CV-1410**
   - **CV-1404**

"C" Distracte

4. **IF alternate purification flowpath is in-service, THEN close the following valves:**

   A. Makeup Prefilter F-25 Out to MU&P (MU-6)

   B. MU&P to Makeup Prefilter F-25 (MU-5)

   C. SF to DH Suction Header (SF-20)

5. **Notify Shift Manager to implement Emergency Action Level Classification (1903.010).**

6. **Perform "Control Room Actions For Containment Closure And Evacuation", Attachment G of this procedure.**

7. **Commence plotting RCS temperature, RCS pressure and heatup rate every 15 minutes.**

8. **IF maintenance activities in the Reactor Building could be affected by RCS level rise, THEN perform local evacuation of the affected areas.**

9. **IF RCS temp exceeds 280°F, THEN GO TO applicable "Loss of Both DH Systems" section of this procedure.**

10. **IF leak is common to both DH systems AND is unisolable, THEN GO TO applicable "Loss of Both DH Systems" section of this procedure.**

SECTION 3 – DH REMOVAL SYSTEM LEAK >20 GPM

14. IF leak is associated with a single DH pump AND both DH pumps in service, THEN perform the following to isolate applicable DH System:

A. Verify applicable DH Pump is stopped.

B. IF "A" DH system is to be isolated, THEN verify the following valves closed:

- LPI Block (CV-1401)
- P-34A Suction from RCS (CV-1434)
- Decay Heat Cooler E-35A Outlet (CV-1428)
- Decay Heat Cooler Bypass (CV-1433)
- P-34A Suction from BWST (CV-1436)
- Rack down LPI Pump P-34A breaker (A-305)

C. IF "B" DH system is to be isolated, THEN verify the following valves closed:

- LPI Block (CV-1400)
- P-34B Suction from RCS (CV-1435)
- Decay Heat Cooler E-35B Outlet (CV-1429)
- Decay Heat Cooler Bypass (CV-1432)
- P-34B Suction from BWST (CV-1437)
- Rack down LPI Pump P-34B breaker (A-405)

15. IF two DH Removal systems are required to be operable by Tech Specs, THEN perform the following:

- **immediately initiate corrective action to return required coolant loops to operable status**
- **refer to Decay Heat Removal and LTOP System Control (1015.002)**

16. Initiate action to repair leak.

A. WHEN leak is repaired, THEN return applicable DH Pump(s) to operation per Decay Heat Removal Operating Procedure (1104.004).

END

SECTION 9 – LOSS OF BOTH DH SYSTEMS-RCS PRESSURE BOUNDARY INTACT

INSTRUCTIONS

1. **Notify Shift Manager to implement Emergency Action Level Classification (1903.010).**

2. **Perform "Control Room Actions For Containment Closure And Evacuation" Attachment G of this procedure.**

3. **Commence plotting RCS temperature, RCS pressure and heatup rate every 15 minutes.**

4. **IF either DH system becomes available,**
   THEN return to applicable section of this procedure.

5. **IF Tygon level instrument is in-service,**
   THEN isolate Tygon level instrument.

6. **Cycle ERV as necessary to maintain RCS press below NDTT limits of Plant Startup (1102.002), Attachment A.**

A

7. **Verify the following:**

   "B" and

   A. DH pump is off.

   "D" Distracter

   B. Close at least one of the following Decay Heat Suction valves:

      - CV-1050
      - CV-1410
      - CV-1404

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1085 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** Cork

TUOI: ASCBT-EP-A0081 **Objective:** 5 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 029 **System Title:** Anticipated Transient Without Scram (ATWS)

Description: Ability to determine or interpret the following as they apply to a ATWS: Reactor nuclear instrumentation

K/A Number: EA2.01 **CFR Reference:** 43.5

Tier:	1	**RO Imp:**	4.4	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 78

Given:
- Plant startup is in progress.
- Reactor power is 46%.

While going through the Main FW Block valves a MFW transient occurs:
- Total Main FW flow lowers to 0.0 lbm/hr,
- Generated MWe goes to zero,
- EFW actuated on both trains,
- RCS pressure rising rapidly,
- Reactor power 5% and dropping rapidly,
- All Safety Groups full out,
- All Regulating Groups fully inserted.

Subsequent actions taken by the ATC successfully trip all CRDs, Power Range channels indicate 0%.

Which of the following Emergency Action Level classifications should be declared?

A. Alert due to loss of all Main Feedwater

B. Alert due to failure of RPS

C. Site Area Emergency due to loss of all Main Feedwater

D. Site Area Emergency due to failure of RPS

Answer:

B. Alert due to failure of RPS

Notes:

"B" is correct per 1903.010, an automatic trip failed to shutdown the reactor and manual actions successfully shutdown the reactor as indicated by reactor power <5%. Indications are that AMSAC actuated on low MFW flow tripping the turbine, then DSS tripped the CRD regulating groups but not the safety groups, indicating a failure of RPS to trip the reactor, and Alert is the correct EAL classification.
"A" is incorrect but plausible since a loss of all Main Feedwater is given and Alert is the correct classification but there is not an Alert classification for loss of all main feedwater.
"C" is incorrect but plausible since a loss of all Main Feedwater is present but there is not an SAE classification for loss of all main feedwater.
"D" is incorrect but plausible since RPS failed to trip as explained above but an SAE is not correct since actions in the Control Room successfully tripped the reactor. An SAE would be proper if the Control Room actions were not successful.

This question is SRO level because it meets 10CFR55.43(b)(5) assessment of facility condtions and selection of appropriate procedures. Classification of emergencies is a specific SRO only responsibility.

This question meets the K/A since the candidate must assess the conditions given, determine that an ATWS has occurred, and then using the nuclear instrumentation parameters given must determine the EAL classification that is appropriate.

Lowered Main FW flow to 1.4 x e 6 at suggestion of NRC examiner. JWC 7/15/16

References:

1903.010, Emergency Action Level Classification
1203.012G, Annunciator K08 Corrective Action
1202.001, Reactor Trip

History:

New question for 2016 SRO exam.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
SYSTEM MALFUNCTION – Failure of Reactor Protection System			
SG3 1 2 Automatic trip and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor. **AND** b. All manual actions do not shutdown the reactor as indicated by reactor power ≥ 5%. **AND** c. Either of the following exist or have occurred due to continued power generation: • CET temperatures at or approaching 1200 °F. **OR** • Feedwater flow rate less than: **Unit 1:** 430 gpm **Unit 2:** 485 gpm	**SS3** 1 2 Automatic trip fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor. **AND** b. Manual actions taken at panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) do not shutdown the reactor as indicated by reactor power ≥ 5%.	**SA3** 1 2 Automatic trip fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor **Emergency Action Level(s):** 1. a. An automatic trip failed to shutdown the reactor as indicated by reactor power ≥ 5%. **AND** b. Manual actions taken at panel C03 (Unit 1) or panels 2C03/2C14 (Unit 2) successfully shutdown the reactor as indicated by reactor power < 5%.	

Location: C13

Device and Setpoint: see below

AMSAC
TRIP

Alarm: K08-B5

1.0 OPERATOR ACTIONS

1. IF alarm is valid,
THEN GO TO Reactor Trip (1202.001).

2. WHEN cause of alarm is corrected,
THEN alarm may be reset by depressing RESET pushbuttons at both Diverse Reactor Overpressure Prevention System (DROPS) panels in C498.

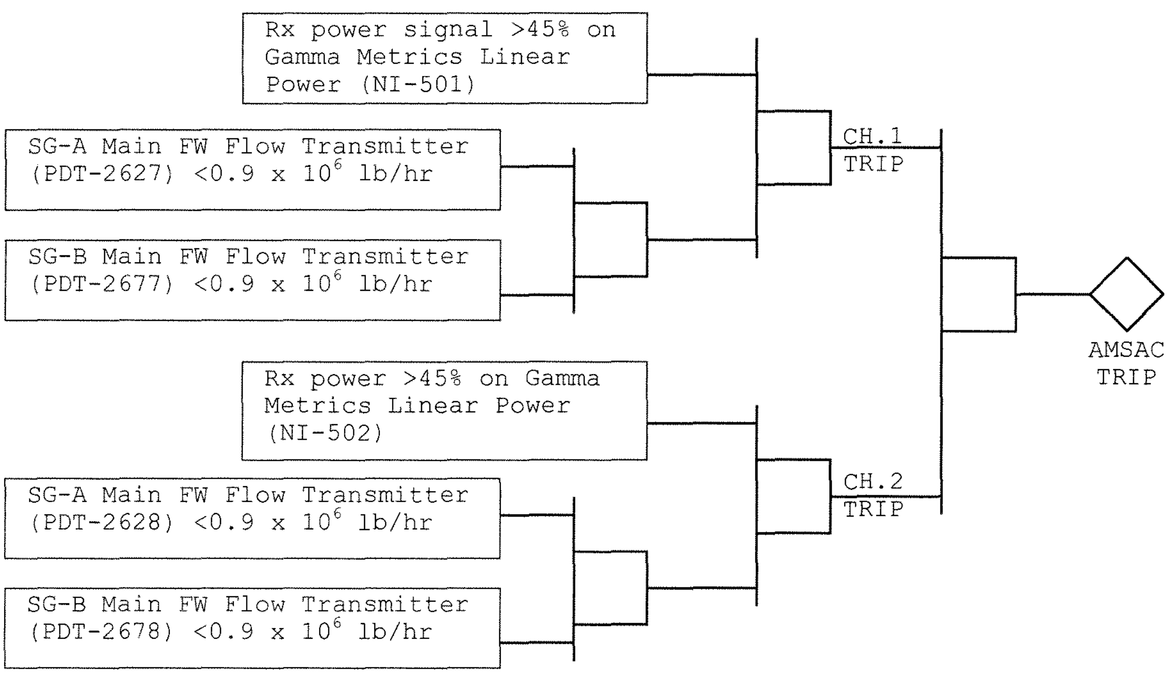
2.0 PROBABLE CAUSES

ATWS Mitigation System Actuation Circuitry (AMSAC) trip confirm due to Rx power >45% and both feedwater loop flows <0.9 x 10^6 lb/hr.

3.0 REFERENCES

Schematic Diagram Annunciator K08 (E-458 sheet 2)

Setpoint logic:



INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0584 **Rev:** 1 **Rev Date:** 5/9/16 **Source:** Bank **Originator:** Cork

TUOI: A1LP-RO-EOP03 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 040 **System Title:** Steam Line Rupture

Description: Knowledge of symptom based EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 43.2

Tier:	1	**RO Imp:**	3.7	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	An

Question: RO: SRO: 79

A steam line rupture has occurred in the Reactor Building with the following conditions now present:
- ESAS actuated on channels 1 thru 6.
- All RCPs secured per RT-10.
- RB pressure 3 psig and dropping.
- RCS pressure is 1050 psig.
- T-hot is 390°F.
- HPI has been throttled.
- EOP actions have terminated the overcooling.

The STA recommends to the CRS to restore normal operating pressure per RT-14 in order to reset ESAS and re-start RCPs.

As CRS, does this recommendation follow the EOP mitigation strategies, and why or why not?

A. Yes, the overcooling event has been terminated.

B. No, this action could overstress reactor vessel.

C. Yes, adequate SCM exists so this is allowable.

D. No, RB pressure is not within normal limits.

Answer:

B. No, this could overstress reactor vessel.

Notes:

"B" is correct, trainee must recognize that with RCPs secured and HPI having been initiated that Pressurized Thermal Shock (PTS) limits apply until an evaluation is performed prior to returning to normal pressure. PTS limits prevent overstressing reactor vessel.
"A" is incorrect, yes this is plausible as the overcooling has been terminated but normal operating pressure would violate procedure.
"C" is incorrect, but plausible as adequate subcooling margin is present but normal operating pressure would violate procedure.
"D" is incorrect, but plausible: RB pressure is a concern and is outside normal limits, but the overriding concern is with PTS.

Revised the T-hot value given from 490°F to 390°F so that raising pressure from 1050 to 2155 would definitely violate the NDTT limit. Revised the "C" distractor by simply stating SCM exists since it was implausible that SCM would have been lost. Removed "due to existence of adequate SCM" since the candidate should evaluate RCS pressure-temperature conditions independently.

This question is SRO level, it meets 10CFR55.43(b)(2) since it requires knowledge of the Technical Specification bases.

This question matches the K/A since it has direct ties to the EOP mitigation strategy followin an RCS cooldown caused by a steam line rupture to limit RCS pressure low within limits of Figure 3 if PTS limits apply.

References:

1202.012, Repetitive Tasks, RT-14
Technical Specification Bases, 3.4.3
1202.013, EOP Figures, Figure 3

History:

New for 2005 SRO exam.
Selected for the 2010 SRO exam
Selected for the 2016 SRO exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0584 **Rev:** 0 **Rev Date:** 5/20/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-EOP03 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 040 **System Title:** Steam Line Rupture

Description: Knowledge of symptom based EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.7 **SRO Select:** No **Taxonomy:** An

Question: **RO:** **SRO:**

A steam line rupture has occurred in the Reactor Building with the following conditions now present:
- ESAS actuated on channels 1 thru 6.
- All RCPs secured per RT-10.
- RB pressure 19 psig and dropping.
- HPI throttled due to existence of adequate SCM.
- RCS pressure is 1050 psig.
- T-hot is 490°F.
- EOP actions have terminated the overcooling.

The SE recommends to the CRS to restore normal operating pressure per RT-14 in order to reset ESAS and re-start RCPs.

As CRS, does this recommendation follow the EOP mitigation strategies?

A. Yes, overcooling event has been terminated.

B. No, this could overstress reactor vessel.

C. Yes, adequate SCM has been restored.

D. No, RB pressure is not within normal limits.

Prior to Revision

Answer:

B. No, this could overstress reactor vessel.

Notes:

"B" is correct, trainee must recognize that with RCPs secured and HPI having been initiated that PTS limits apply until an evaluation is performed prior to returning to normal pressure. PTS limits prevent overstressing reactor vessel.
"A" is incorrect, yes the overcooling has been terminated but normal operating pressure would violate procedure.
"C" is incorrect, subcooling margin was never lost but normal operating pressure would violate procedure.
"D" is incorrect, although RB pressure is a concern the overriding concern is with PTS concerns.

THIS QUESTION IS TIED to 43.1

References:

1202.012, chg. 004-03-0, RT-14

History:

New for 2005 SRO exam.
Selected for the 2010 SRO exam

CONTROL RCS PRESS

NOTE

- PTS limits apply if any of the following has occurred:
  - HPI on with all RCPs off
  - RCS C/D rate > 100°F/hr with Tcold < 355°F
  - RCS C/D rate > 50°F/hr with Tcold < 300°F
- Once invoked, PTS limits apply until an evaluation is performed to allow normal press control.
- When PTS limits are invoked OR SGTR is in progress, PZR cooldown rate limits do not apply.
- PZR cooldown rate <100°F/hr.

1. IF PTS limits apply or RCS leak exists,
THEN maintain RCS press low within limits of Figure 3.

2. IF RCS press is controlled AND will be reduced below 1650 psig,
THEN bypass ESAS as RCS press drops below 1700 psig.

3. IF PZR steam space leak exists,
THEN limit RCS press as PZR goes solid by one or more of the following:

A. Throttle makeup flow.

B. IF SCM is adequate,
THEN throttle HPI flow by performing the following:

1) Verify both HPI Recirc Blocks open:

- CV-1300
- CV-1301

2) Throttle HPI.

C. Raise Letdown flow.

1) IF ESAS has actuated,
THEN unless fuel damage or RCS to ICW leak is suspected, restore Letdown per RT-13.

D. Verify Electromatic Relief ERV Isolation open (CV-1000)
AND cycle Electromatic Relief ERV (PSV-1000).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, and unit transients. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 contain P/T limit curves for heatup, cooldown, inservice hydrostatic testing, and physics testing at RCS temperatures ≤ 525 °F, and the maximum rate of change of reactor coolant temperature. The methods and criteria employed to establish operating pressure and temperature limits are described in BAW-10046A (Ref. 1). These limit curves are applicable through fifty-four effective full power years (EFPY) of operation. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for the various operating reactor coolant pump combinations.

Each P/T curve defines an acceptable region for normal operation below and to the right of the limit curve. The curves are used to develop operational guidance for use during heatup or cooldown maneuvering.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel. The vessel is the component most subject to brittle failure due to the fast neutron embrittlement it experiences during power operation, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials. Reference 2 requires an adequate margin to brittle failure during normal operation, abnormalities, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4). For the Linde 80 weld materials present in the ANO-1 reactor vessel beltline, an alternative approach was utilized for determining the adjusted reference nil ductility temperature as described in Topical Report BAW-2308, Revisions 1-A and 2-A (Ref. 12). The Master Curve methodology is accepted with exemption from the requirements of 10 CFR 50.61 (Ref. 13) and 10 CFR 50, Appendix G (Ref.2).

LCO (continued)

The heatup and cooldown rates stated are intended as the maximum changes in temperature in one direction in the stated time periods. The actual temperature linear ramp rate may exceed the stated limits for a shorter time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the stated time period.

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first fifty-four EFPY. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The magnitude of the departure from the allowable operating P/T regime or the magnitude of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

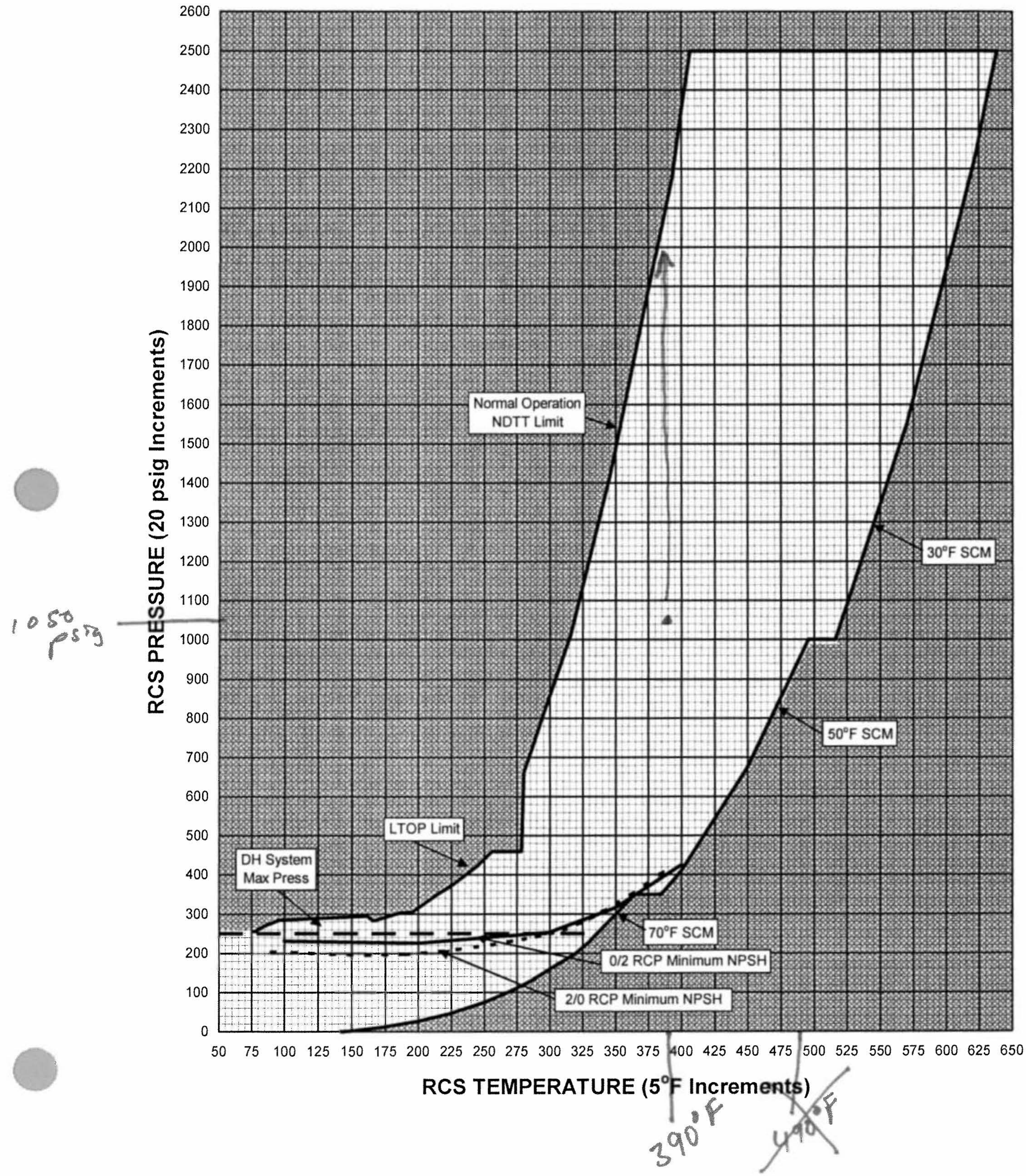
The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or inservice hydrostatic testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

ACTIONS

A.1 and A.2

With RCS pressure and temperature not within criticality limit of Figure 3.4.3-1 during PHYSICS TESTS with RCS temperature ≤ 525 °F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If RCS pressure and temperature can be restored within the 30 minute time period, shutdown is not required.

FIGURE 3
RCS Pressure vs Temperature Limits



INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0586 **Rev:** 1 **Rev Date:** 5/9/16 **Source:** Modified **Originator:** S.Pullin

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 056 **System Title:** Loss of Offsite Power

Description: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

K/A Number: 2.2.36 **CFR Reference:** 43.2

Tier:	1	**RO Imp:**	3.1	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: RO: SRO: 80

REFERENCE PROVIDED

Given:
- Plant is at 100% power with no failed equipment.
- A loss of the 161 KV ring bus occurs (de-energized).
- Autotransformer is energized from the 500 KV ring bus.

Entergy Arkansas states that maintenance to repair the 161 KV ring bus will take 1 to 5 days.

Which of the following is the maximum time allowed before the plant is required to be in Mode 3?

A. 30 hours

B. 72 hours

C. 78 hours

D. 84 hours

Answer:

C. 78 hours

Notes:

Answer "C" is correct, knowledge of the switchyard layout is required to know that the auto transformer supplies SU Transformer #1 (for Unit 1) and the 161KV ring bus supplies SU Transformer #2. With the 161KV ring bus de-energized so will SU Transformer #2 be de-energized , thus 1 of the 2 required offsite power sources is inoperable. The time for Required Action A.3 must be added to Required Action F.1 to arrive at the correct time limit.
"A" is incorrect, but plausible since 24 hours is the completion time for Required Action C.2 added to Required Action F.1.
"B" is incorrect but plausible as 72 hours is the completion time for Required Action A.3 alone.
"D" is incorrect but plausible as this is the completion time for Required Action A.3 added to the tjime for Required Action F.2.

This question is considered modified since the Tech Spec it is based on has changed since the last time the question was used. This changed the correct answer and one distracter.

This question is SRO level because it meets 10CFR55.43(b)(2), facility operating limitations in the Technical Specifications, specifically 3.8.1.

The question meets the K/A since it presents the candidate with a degraded offsite power source requiring maintenance, and requires the candidate to apply the Technical Specifications to these conditions and arrive at a correct answer.

References:

Technical Specification 3.8.1

This reference must be included in the student's exam handout!!!

History:

Selected for 2011 SRO Exam.
Modified for 2016 SRO exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0586 **Rev:** 0 **Rev Date:** 5/31/05 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-TS **Objective:** 5 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 056 **System Title:** Loss of Offsite Power

Description: Ablity to apply Technical Specifications for a system.

K/A Number: 2.1.12 **CFR Reference:** 43.2 / 43.5 / 45.3

Tier:	1	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	Ap

Question: **RO:** **SRO:**

Prior to Revision

OPEN REFERENCE

Given:
- Plant is at 100% power with no failed equipment.
- A loss of the 161 KV ring bus occurs (de-energized).
- Autotransformer is energized from the 500 KV ring bus.

Providing the 161 KV ring bus remains de-energized, when is the plant required to be in Mode 3?

A. Within 24 hours

B. Within 36 hours

C. Within 72 hours

D. Within 84 hours

Answer:

D. Within 84 hours

Notes:

Answer "D" is correct, the time for Required Action A.3 must be added to Required Action F.1 to arrive at the correct time limit.
Answer "A" is incorrect, 24 hours is the completion time for Required Action A.2.
Answer "B" is incorrect, 36 hours is the completion time for Required Action A.2 added to the time for Condition F.
Answer "C" is incorrect, 72 hours is the completion time for Required Action A.3 alone.

References:

T.S. 3.8.1

This reference must be included in the student's exam handout!!!

History:

Direct from regular exambank, QID #3073
Selected for 2005 SRO exam
Selected for 2011 SRO Exam.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour AND Once per 12 hours thereafter
	AND	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 ------------NOTE----------------- Startup Transformer No. 2 may be removed from service for up to 30 days for preplanned preventative maintenance. This 30 day Completion Time may be applied not more than once in any 10 year period. -------------------------------------	
	Restore required offsite circuit to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 12 hours thereafter
	AND	
	B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND	
	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	OR	

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG. AND B.4 Restore DG to OPERABLE status.	24 hours 7 days AND 10 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND C.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours
D. One required offsite circuit inoperable. AND One DG inoperable.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.6, "Distribution Systems – Operating," when Condition D is entered with no AC power source to any train. -- D.1 Restore required offsite circuit to OPERABLE status. OR D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 ---------------NOTE--------------- LCO 3.0.4.a is not applicable when entering Mode 4. ------------------------------------- Be in MODE 4.	6 hours 12 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	-----------------------------NOTE----------------------------- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. -- Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves "ready-to-load" conditions.	 31 days

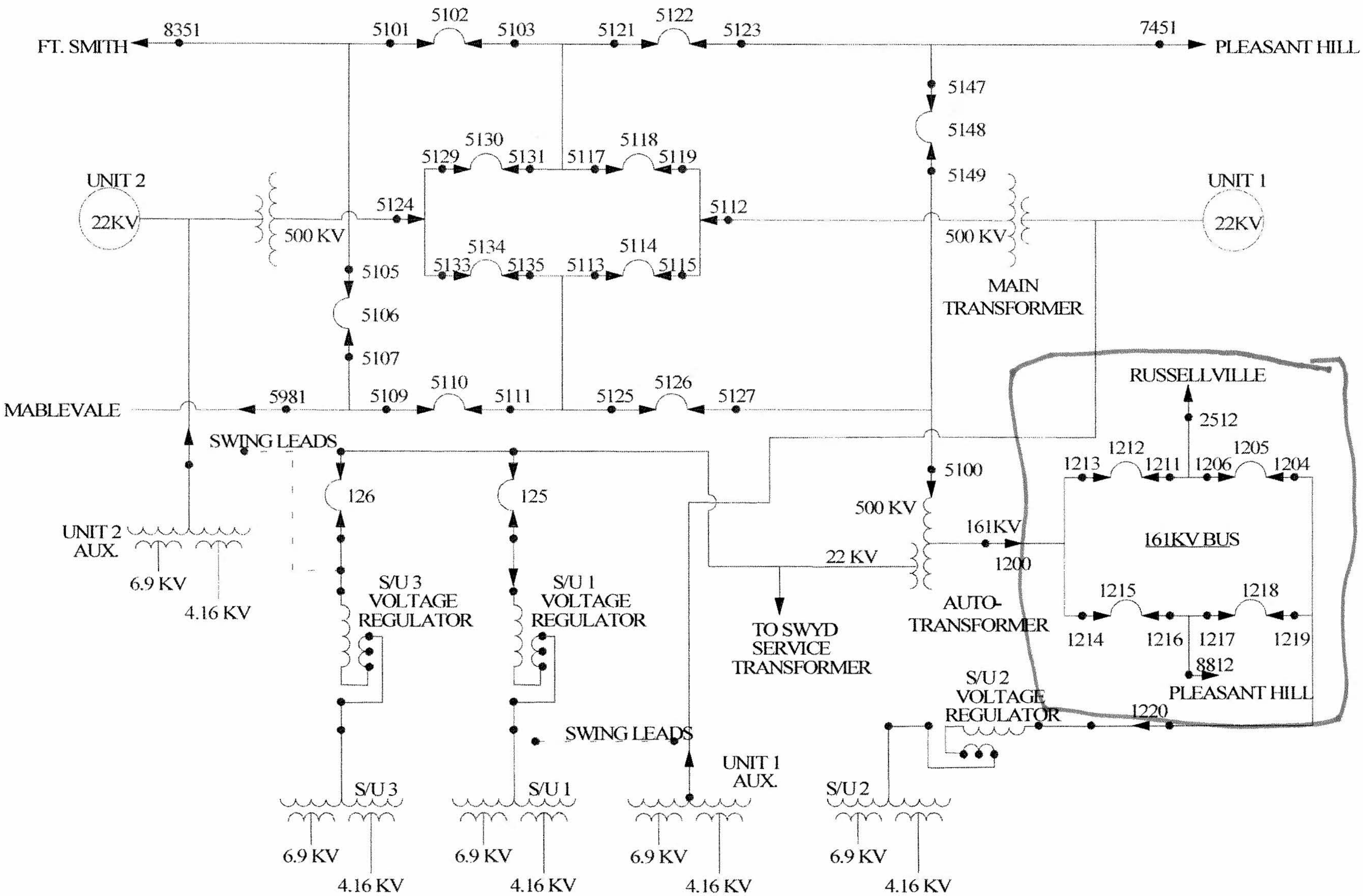


FIGURE 32.01: SWITCH YARD ONE LINE DIAGRAM

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1050 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-RO-EOP04 **Objective:** 10 **Point Value:** 1

Section: 4.3 **Type:** B&W EPE/APE

System Number: E04 **System Title:** Inadequate Heat Transfer

Description: Ability to determine and interpret the following as they apply to the (Inadequate Heat Transfer): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

K/A Number: EA2.2 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** An

Question: RO: SRO: 81

Given:
- Reactor tripped due to loss of both MFW Pumps at time 1020.
- P-75 AFW pump is tagged out for maintenance.
- Bus A3 is locked out.
- P-7A EFW pump trips.
- P-32B and P-32D RCPs have been tripped, P-32A and P-32C are running.
- RCS Thot temperatures are 620°F and rising.
- RCS pressure is 2320 psig and rising.
- SG Tube-to-Shell Delta-T is 65°F, tubes hotter.

It is now 1021 and based on the above current conditions, the procedurally required action is to _______.

A. Trip P-32A and P-32C RCPs in accordance with Loss of Subcooling Margin (1202.002).

B. Initiate HPI Cooling per RT-4 in accordance with Overheating (1202.004).

C. Trip P-32A and P-32C RCPs in accordance with Overheating (1202.004).

D. Initiate Full HPI per RT-3 in accordance with Loss of Subcooling Margin (1202.002).

Answer:

C. Trip P-32A and P-32C RCPs in accordance with Overheating (1202.004).

Notes:

C" is correct, in accordance with the Overheating EOP the running RCPs should be tripped if tube-to-shell DT exceeds 60°F tubes hotter.
"A" is incorrect since the Overheating EOP would be entered first and not exited to Loss of SCM, it is plausible since the conditions indicate extremely hot conditions close to but not inadequate SCM.
"B" is incorrect in accordance with step 1 and step 5 of 1202.004, Overheating. This distracter is incorrect since conditions are close to the ERV lifting but it has not reached the auto open setpoint, it pressure had reached 2450 psig, then a transition to RT-4 is made to initiate HPI Cooling.
"D" is incorrect, but plausible since the conditions indicate extremely hot conditions (close to but not inadequate SCM) and the Loss of SCM EOP does initiate full HPI early in the procedure.

Added times to question based on discussion with Chief Examiner. Rev. 1 - moved time to loss of both MFW pumps per NRC examiner request. JWC 7/15/16

This question is SRO only since per 10CFR55.43(b)(5) it requires the candidate to evaluate the conditions given and to select the appropriate procedure and action within that procedure which would mitigate the conditions with the highest priority.

This question meets the intent of the K/A as the candidate must be able to determine which procedure is

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Leaving P-32A and P-32C running if available provides enhanced PZR spray and one RCP running per loop.

4. Reduce running RCPs to one per loop.

A. **IF** SG Tube-to-Shell ΔT reaches 60°F (tubes hotter)
AND
SCM is adequate,
THEN trip running RCP(s).

1) Do **not** restart an RCP until SG Tube-to-Shell ΔT is $\leq$ 50°F (tubes hotter).

5. IF overheating has been corrected, THEN GO TO 1202.001, "REACTOR TRIP" procedure.

5. **IF** any of the following criteria is met:

- ERV opens
- RCS press $\geq$ 2450 psig
- RCS press approaches NDTT Limit (Figure 3)
- Secondary feed **not** expected to become available
- Overheating causes SCM to become inadequate

THEN while continuing attempts to restore secondary feed, perform the following:

A. Initiate HPI cooling (RT-4).

1) Record time full HPI flow initiated for reference in step 11: __________

(5. CONTINUED ON NEXT PAGE)

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0347 **Rev:** 1 **Rev Date:** 3/28/16 **Source:** Bank **Originator:** G. Alden

TUOI: A1LP-RO-FH **Objective:** 1.4 **Point Value:** 1

Section: 4.3 **Type:** B&W APEs

System Number: A08 **System Title:** Refueling Canal Level Decrease

Description: Ability to determine and interpret the following as they apply to the (Refueling Canal Level Decrease) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A Number: AA2.2 **CFR Reference:** 43.7

Tier:	1	**RO Imp:**	3.8	**RO Select:**	No	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question: RO: SRO: 82

The main fuel bridge has a grappled spent fuel assembly and is indexed over the core when an NI seal plate cover failure occurs.

Water level in the canal is falling at two inches per minute.

As SRO in Charge of Fuel Handling you should direct the main fuel bridge operator to:

A. Continue to the upender and place the assembly in the upender.

B. Leave the fuel assembly in the mast and evacuate the area.

C. Place the assembly in the fuel rack in the deep end of the canal.

D. Return the assembly to an available location in the reactor vessel.

Answer:

D. Return the assembly to an available location in the reactor vessel.

Notes:

In this scenario the fuel transfer canal level is dropping and the fuel assembly must be placed in an area that will remain covered with water after the canal is drained.
Therefore, "D" is the correct answer in accordance with 1203.042, Section 2, step 3.A. With Refueling Canal level at minimum (400') there is 23.5 feet to the top of the reactor vessel (bottom of refueling canal) so at 2"/min there are 282" of water which gives 141 minutes to take action before reaching the top of the vessel. At one ft/min there would only be 23.5 minutes to take action. The fuel mast can move at 20 ft/min in fast speed down to ~12" above the core then shifts to slow speed (5 ft/min) to the bottom of the core, so it would take ~4 minutes to place the assembly in the core. A seal plate failure cannot drain the RCS level below the top of the vessel. There are still ~9.5 ft from the top of the vessel to the top of the fuel assemblies.
"A" is incorrect but plausible if the assembly could not be returned to the reactor vessel.
"B" is incorrect but plausible per the Note before step 3 in Section 2 but conditions are not given to indicate that dose levels in the area are hazardous.
"C" is incorrect but plausible if the SRO in Charge of Fuel Handling determined that level was dropping too fast to transfer the assembly back to the vessel, but level is dropping slowly enough to allow returning the assembly to the vessel.

This question is SRO level since it meets 10CFR55.43(b)(7) and concerns fuel handling procedures.

This question meets the K/A since the conditions given meet the entry condtions for a Refueling Canal Level Decrease section in ANO-1's Refueling Abnormal Operations procedure and requires the candidate to know which is the proper course of action given the conditions.

References:

1203.042, Refueling Abnormal Operations

History:

Last used 2011 SRO exam.
Selected for 2016 SRO Exam.

SECTION 2 -- TRANSFER CANAL LEAK

ENTRY CONDITIONS

- Reactor Building sump level rising.

NOTE

RB refueling deck elevation is 401'6". Spent Fuel Pool deck is 404'.

- Spent Fuel Pool/Fuel Transfer Canal level dropping.
- Report of water running down primary shield wall into Reactor Building lower elevations.

SECTION 2 -- TRANSFER CANAL LEAK

INSTRUCTIONS

1. **Perform the following while continuing with this section:**

   - Commence "Setting Containment Closure" Attachment K of Decay Heat Removal and LTOP System Control (1015.002). Utilize CRS Admin and Outage management when manned.

   - Perform "Control Room Actions For Containment Closure And Evacuation" Attachment G of Loss of Decay Heat Removal (1203.028).

2. **Notify Shift Manager to perform the following:**

   - Implement Emergency Action Level Classification (1903.010).

   - Notify Operations Manager.

   - Notify Outage Desk, if manned.

   - Notify Reactor Engineering.

NOTE

Maintaining the bridge controls attended while a load is suspended ensures proper controls of load and prompt response to events. It is acceptable to leave the suspended load unattended when radiological hazards dictate an immediate departure from the area.

3. **IF any fuel assemblies or control components in RB are stored outside Reactor Vessel**
   OR in transit,
   THEN perform the following:

   A. Return any fuel assemblies or control components to an available position in the Reactor Vessel (Ref. Tech Spec 3.9.6).

   B. IF fuel handling in progress
   AND level dropping so fast that a significant loss of shielding can occur before the assemblies can be moved to the Reactor Vessel,
   THEN SRO in Charge of Fuel Handling may evaluate placing fuel assembly(ies) in the safest location, unless radiological hazards dictate an immediate departure.

   C. IF fuel assemblies and control components **cannot** be returned to the Reactor Vessel,
   THEN transfer applicable components to SF Pool as follows:

      1) Verify Fuel Transfer Canal level is being maintained with the decay heat system, if necessary, at approximately the same level as the SF Pool.

      2) Transfer assemblies or components to SF Pool for storage.

(continued)

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1086 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** Cork

TUOI: **Objective:** **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 059 **System Title:** Accidental Liquid Radwaste Release

Description: Ability to control radiation releases.

K/A Number: 2.3.11 **CFR Reference:** 43.4

Tier:	1	**RO Imp:**	3.8	**RO Select:**	No	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.3	**SRO Select:**	Yes	**Taxonomy:**	C

Question: RO: SRO: 83

Given:
- Unit 1 is at 100% power.
- Chemistry reports the "A" OTSG secondary leak rate is 25 gpd.
- A report from the Inside AO reveals a Turbine Building Trench continuous release is in progress.
- Discharge Flume (RI-3618) subsequently goes into high alarm.

Which of the following actions is required for the above conditions?

A. Obtain a grab sample and perform gamma and I-131 analysis per ODCM L2.3.1.

B. Suspend the release and initiate a special report per Technical Specifications 3.7.4, Secondary Activity.

C. Obtain a grab sample and perform gamma and I-131 analysis per Technical Specifications 3.7.4, Secondary Activity.

D. Suspend the release and initiate a condition report per ODCM L2.3.1.

Answer:

D. Suspend the release and initiate a condition report per ODCM L2.3.1.

Notes:

"D" is correct per ODCM L2.3.1 B.1 and B.2. This action is also consistent with guidance in 1104.044, Turbine Building Draining System, and 1203.014, Control of Secondary System Contamination when a primary to secondary tube leak is in progress (also refer to 1203.023, Small Steam Generator Tube Leaks). During tube leaks only batch releases are allowed. The ODCM spec for continuous releases is not applicable since the release should be terminated immediately. Only batch releases are allowed and the sampling requirements are not met for a batch release so the release should be suspended and a condition report initiated.
"A" is incorrect but plausible as this is the correct action (and correct licensing basis document) for exceeding I-131 limits for continuous releases of secondary coolant but continuous releases should be isolated per 1203.014, Control of Secondary System Contamination when a primary to secondary tube leak >15 gpd is in progress.
"B" is incorrect but plausible as this is the correct action but the incorrect licensing basis document. Technical Specifications adds to the plausibility since during a SG tube leak TS LCO 3.4.13 for primary-to-secondary leakage and 3.7.4 for secondary system activity could apply.
"C" is incorrect but plausible correct action for exceeding I-131 limits for continuous releases of secondary coolant but this distracter gives the wrong licensing basis document.

This question is SRO level because it meets 10CFR55.43(b)(4) as it addresses radiation hazards that may arise during normal and abnormal situations.

This question meets the K/A since it concerns accidental liquid radwaste releases (an alarming radiation monitor and tube leak constitute an accidental liquid release since the sampling and analysis requirements are not met) and the question concerns the ability to control radiation releases.

Revised question per NRC examiner suggestions. JWC 7/15/16

References:

Offsite Dose Calculations Manual, L2.3.1
1203.023, Small Steam Generator Tube Leaks
1203.014, Control of Secondary System Contamination

History:

New question for 2016 SRO exam

L 2.3 RADIOACTIVE LIQUID EFFLUENTS

L 2.3.1 Radioactive material released to the discharge canal shall:

a. For dissolved or entrained noble gases, be limited to a total concentration of $\leq 2 \times 10^{-4}$ μCi/ml.

b. For radioactive nuclides other than dissolved or entrained noble gases, be limited to the concentration specified in 10 CFR 20, Appendix B, Table II, Column 2.

c. During any calendar quarter, result in a dose commitment to a MEMBER OF THE PUBLIC of ≤ 1.5 mrem to the total body and ≤ 5 mrem to any organ.

d. During any calendar year, result in a dose commitment to a MEMBER OF THE PUBLIC of ≤ 3 mrem to the total body and ≤ 10 mrem to any organ.

e. Be processed by a LIQUID RADWASTE TREATMENT SYSTEM when accumulative dose during a calendar quarter is projected to exceed 0.18 mrem to the total body and/or 0.625 mrem to any organ.

APPLICABILITY: At all times.

ACTIONS

------------------------------NOTE------------------------------
Separate Condition entry is allowed for each Limitation L 2.3.1.a through L 2.3.1.e above and for each BATCH RELEASE and CONTINUOUS RELEASE Surveillance requirement not met.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any limit listed in L 2.3.1.a through L 2.3.1.e not met.	A.1 Initiate action to restore to within limit.	Immediately
	<u>AND</u>	
	A.2 Initiate a condition report to document the condition, determine any limitations for the continued effluent release operations, and track the condition for inclusion in the Radioactive Effluent Release Report pursuant to TS 5.6.3 (ANO-1) / TS 6.6.3 (ANO-2).	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. --------------NOTE-------------- Only applicable to BATCH RELEASE. ------------------------------------ Sampling and/or analysis requirements not met.	B.1 Verify associated effluent release suspended. AND B.2 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately Immediately
C. --------------NOTE-------------- Only applicable to CONTINUOUS RELEASE of secondary coolant. ------------------------------------ Secondary coolant dose equivalent I-131 (DEI) > 0.01 µCi/ml.	C.1 Obtain a grab sample of the associated secondary coolant. AND C.2 Perform gamma isotopic and I-131 analysis of sample.	12 hours 12 hours following sample acquisition
D. Annual dose limits of L 2.3.1.d projected to exceed 40 CFR 190 limits.	D.1 Apply for a variance from the NRC to permit releases in excess of 40 CFR 190 limits.	Prior to exceed 40 CFR 190 limits Immediately
E. Required Action(s) and/or Completion Time(s) of Conditions C and/or D not met. OR Sampling and/or analysis requirements not met.	E.1 Initiate a condition report to document the condition and determine any limitations for the continued effluent release operations.	Immediately

SECTION 1 -- SG-A Tube Leak

NOTE

- ANO has committed to the NRC that unit shutdown will occur with SG-A tube leakage of ≥15 gpd.
- Monitoring of leakage spikes or step changes is allowed for up to 1 hour. The 1 hour allowed monitoring period begins when 15 gpd is surpassed for the first time. If leak rate remains above 15 gpd or is spiking above 15 gpd at the end of 1 hour, then leakage is assumed to be ≥15 gpd.

8. IF SG-A leak rate is ≥15 gpd,
THEN Initiate controlled shutdown at 10% - 30%/hr per Power Reduction and Plant Shutdown (1102.016) to be in Mode 3 as expeditiously as possible
AND within 24 hours.

A. GO TO step 10 while continuing with plant shutdown and cooldown.

9. IF SG-A leak rate is ≥10 gpd,
THEN monitor N-16 detectors and secondary activity levels.

A. Proceed as directed by Ops Manager, while continuing with this procedure.

B. **GO TO** step 11.

10. Perform Control of Secondary System Contamination (1203.014).

NOTE

Only the MGP N-16 Radiation Monitoring System is qualified to meet the minimum requirements specified by EPRI Guidelines.

11. IF the MGP N-16 Radiation Monitoring System is or becomes unavailable to SG-A,
THEN perform Attachment 2, "No Operable Continuous Radiation Monitor" section.

12. Raise monitoring of radiation monitors to once every 15 minutes using Attachment 3.

NOTE

- The remainder of the steps in this procedure should be performed by Operations personnel other than Control Room personnel.
- Unit 1 Control Room should be notified of equipment status changes as they occur.

**5. IF trench dump is in progress,
THEN stop trench dump by placing the following handswitches in OFF:**

- Trench Sump Pump (P-122A) (HS-5635)
- Trench Sump Pump (P-122B) (HS-5636)
- Emergency Trench Sump Pump (P-97) (HS-3613)

6. Secure systems as follows:

- Perform "Removing MSR DI from Service" section of MSR Drain Demineralizer Operation (1106.031).
- IF the plant is being shut down,
  THEN perform "Securing Zinc Injection" section of Chemical Addition (1104.003).

**7. IF plant shutdown is required,
THEN perform the following:**

A. Align Condensate Polishers to prevent wide spread contamination of polisher resin and reduce secondary system activity level as follows:

1) Inform Control Room personnel of intent to remove all but two polishers from service.

NOTE

To minimize radiation exposure to personnel at the polisher controls and in the train bay, it is preferred that C & D polishers remain in service.

2) IF only one polisher is in service,
AND flow can be maintained >1500 gpm/polisher with two polishers,
THEN perform the following:

a. Place an idle polisher in service per Condensate Demineralizer System Operation and Regeneration (1106.024), "Placing Standby Polisher in Service Without Using Recycle Method" Section.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1045 **Rev:** 0 **Rev Date:** 3/17/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-RO-FPS **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 067 **System Title:** Plant Fire On Site

Description: Knowledge of fire protection procedures.

K/A Number: 2.4.25 **CFR Reference:** 43.2

Tier: 1 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** An

Question: RO: SRO: 84

REFERENCE PROVIDED

Given:
- Unit 1 is at 100% power.
- Annunciator K12-D1 "FIRE PROT SYSTEM TROUBLE" goes into alarm.
- ATC observes "C463 PANEL TROUBLE" LED illuminated on K125 on C19.
- CBOT investigates and reports yellow trouble LED illuminated for Upper North Electrical Penetration Room (UNEPR) smoke detector and that no other alarms are in.
- Inside AO reports that one smoke detector in UNEPR has red LED lit on base. Smoke detector will not reset.
- STA reports that his review of e-prints show there are six smoke detectors in the UNEPR detector string.

Which of the following actions are required to comply with plant procedures and regulatory requirements?

A. Submit condition report on the smoke detector, the detection string is functional.

B. Establish a one hour roving fire watch within one hour for the UNEPR.

C. Determine any limitations for continued operation of the plant within 24 hours.

D. Establish a continuous fire watch within one hour for the UNEPR.

Answer:

D. Establish a continuous fire watch within one hour for the UNEPR.

Notes:

"D" is the correct answer, the examinee must recognize that a trouble LED on the UNEPR smoke detector string renders the entire detector string inoperable since the trouble alarm will stay in and another will not be received. The 1203.009 fire alarm response procedure directs declaring the detection non-functional. Since the smoke detector string actuates the sprinkler system for the UNEPR, the sprinkler system is thus non-functional and TRM 3.7.9 applies. One of the actions for this specification is to establish a continuous fire watch for the UNEPR within one hour (TRM 3.7.9.A.2).
"A" is incorrect but plausible since the conditions state that there is a problem with only 1 detector and 6 detectors are in the string, thus it might appear that 50% of the detection is available. Per the explanation for "D" above, only one trouble alarm will be received thus the entire string must be declared inoperable.
"B is the other required action for 3.3.6.A (inoperable detection and thus plausible) but a one hour roving fire watch is not an adequate compensatory measure for an inoperable sprinkler system as TRM 3.7.9 requires a continuous fire watch.
"C" is the standard action whenever required actions and completion times are not met (thus plausible), it is present in TRM 3.7.9 but it does not apply since the conditions do not indicate it should be applicable.

This is an SRO level question as it meets 10 CFR 55.43(b)(1), conditions and limitations in the facility license. The administration of fire protection program requirements is a specific example of an SRO level question in ES-401, Att. 2, page 17. ANO does not expect Ros to be familiar with TRM requirements NOR does it require SROs to commit TRM fire protection specifications to memory despite the one hour time requirements. The TRM fire protection specifications are too numerous and too complex.

This question matches the K/A since it involves fire protection procedures which are used to ensure fire protection systems are functional to detect a plant fire on site. This question requires the candidate to know the alarm response procedure requires declaring the detection string to be non-functional when a trouble alarm occurs.

References:

TRM 3.7.9 - THIS REFERENCE ALONG WITH 3.3.6 MUST BE IN EXAMINEE'S HANDOUT.
1203.009, Fire Protection System Annunciator Corrective Action

History:

New SRO question for 2016 exam.

Location: C19

Device and Setpoint: see page 3 of 3.

FIRE PROT
SYSTEM
TROUBLE

Alarm: K12-D1

1.0 OPERATOR ACTIONS

1. Check Trouble Lights (K125) on C19 for source of trouble alarm.

2. IF C463 PANEL TROUBLE,
THEN check Pyrotronics and Notifier C463 for yellow trouble LED.

   A. IF yellow trouble LED is on,
   THEN refer to corrective actions for yellow trouble LED in Attachment A of this procedure.

NOTE

- Symptoms of loss of power to Fire Indicating Unit C461 (AB 354') are trouble alarms on the detection strings fed by C461.
- Attachment B of this procedure shows the Fire Indicating Units and their associated Zone Indicating Units and monitored areas.

   B. IF the yellow trouble LEDs are on for detection strings associated with Fire Indicating Unit C461 (AB 354'),
   THEN declare associated Zone Indicating Units and fire detection strings non-functional.

3. IF FIRE WATER FLOW (K12-A2) is also alarmed,
THEN check Fire Water Flow Indicating Lights on C19.

   A. IF flow is indicated,
   THEN GO TO K12-A2 corrective actions.

4. IF TURB SAMPLE ROOM is lit,
THEN verify UAV-5605 Turb Sample Rm Deluge Isol (FS-52) is fully open.

NOTE

- Both Fire Indicating Unit C180 (LSEPR) and Fire Indicating Unit C190 (Shift Managers office) have a power available light on the front.
- Attachment B of this procedure shows the Fire Indicating Units and their associated Zone Indicating Units and monitored areas.

5. IF C180/C190 LOSS OF POWER (K125-7 on C19) is in alarm,
THEN determine panel(s) with loss of power
AND declare associated Zone Indicating Units and fire detection strings non-functional.

6. IF FUEL VAULT SYSTEM TROUBLE (K125-8 on C19) is lit,
THEN dispatch an operator to Fuel Oil Vault Notifier (C467) to report cause of alarm.

ATTACHMENT A

PANEL C463 TROUBLE ACTIONS AND CIRCUIT RESTORATION

NOTIFIER C463 TROUBLE Page 107 of 111	NOTIFIER C467 TROUBLE Page 109 of 111

A1

CONTROL UNIT		
• POWER • ALARM • TROUBLE	• GROUND FAULT • AUDIBLE CIRCUIT	
RESET LAMP TEST	ALARM SILENCE	TROUBLE SILENCE

NOTE: Rocker switches are behind panel door.

A2

AUXILIARY BUILDING ELECTRICAL PENETRATION ROOMS				REACTOR BUILDING ELECTRICAL PENETRATION AREAS			
UNEPR TROUBLE ZONE 149-E	UNEPR UAV-5615 TRIP ZONE 149-E	LNEPR TROUBLE ZONE 112-I	LNEPR ZONE 112-I	RB LNEP ZONE 32-K			
	LNEPR UAV-5625 TRIP ZONE 112-I		UNEPR ZONE 149-E	RB UNEP ZONE 32-K			
1	2	3	4	5	6	7	8
Page 8 of 111	Page 9 of 111	Page 10 of 111	Page 11 of 111	Page 13 of 111			

A3

AUXILIARY BUILDING ELECTRICAL PENETRATION ROOMS				REACTOR BUILDING ELECTRICAL PENETRATION AREAS			
USEPR TROUBLE ZONE 144-D	USEPR UAV-5616 TRIP ZONE 144-D	LSEPR TROUBLE ZONE 105-T	LSEPR ZONE 105-T	RB LSEP ZONE 33-K			
	LNEPR UAV-5626 TRIP ZONE 105-T		USEPR ZONE 144-D	RB USEP ZONE 33-K			
1	2	3	4	5	6	7	8
Page 15 of 111	Page 16 of 111	Page 17 of 111	Page 18 of 111	Page 20 of 111			

ATTACHMENT A

A2-4 (U & L)
Zone 112-I Lower North Elect Penetration Rm
Zone 149-E Upper North Elect Penetration Rm

1.0 CAUSES

- Zone 112-I Lower No Elect Penetration Rm (A2-4U) red alarm LED:
  - Smoke detector actuation in room
- Zone 149-E Upper No Elect Penetration Rm (A2-4L) red alarm LED:
  - Smoke detector actuation in room
- A2-4L or A2-4U yellow trouble LED:
  - Break in detection circuit continuity

2.0 ACTION REQUIRED

2.1 A2-4U red alarm LED: Refer to FIRE (K12-A1).

2.2 A2-4L red alarm LED: Refer to FIRE (K12-A1).

2.3 A2-4L or A2-4U yellow trouble LED:

2.3.1 Declare affected zone of fire detection instrumentation non-functional.

2.3.2 Perform required actions of U1 TRM 3.3.6 and U1 TRM 3.7.9 AND report fire system impairment if required.

2.3.3 Review Unit 1 Fire Impairment Database to determine if additional fire protection controls are required per Unit 1 TRM.

2.3.4 <u>IF</u> a fire detection system is non-functional
<u>AND</u> an hourly fire watch is already monitoring the area because of a non-functional fire barrier,
<u>THEN</u> station a continuous fire watch per ANO Fire Impairment Program (1000.120).

2.3.5 Submit a WR/WO.

3.0 TO CLEAR ALARM

3.1 A2-4U red alarm LED:

3.1.1 Clear smoke from area.

3.1.2 Reset Fire Indicating Unit C180 (LSEPR).

3.1.3 Reset C463 using Reset Lamp Test Switch on control unit.

TRM 3.3 INSTRUMENTATION

TRM 3.3.6 Fire Detection System Instrumentation

TRO 3.3.6

-----NOTE-----

1. Reactor Building smoke detectors are not required to be FUNCTIONAL during Type A Integrated Leak Rate Testing.
2. All non-functional detectors specified in TRM Table 3.3.6-1 will be tracked.

The following heat/smoke detectors in the locations specified in TRM Table 3.3.6-1 shall be FUNCTIONAL:

1. A minimum of 50% of the heat/smoke detectors in locations outside the Reactor Building, and,
2. All heat/smoke detectors located inside the Reactor Building.

APPLICABILITY: At all times

ACTIONS

-----NOTE-----

1. Separate Condition entry is allowed for each location specified in TRM Table 3.3.6-1.
2. In lieu of Required Actions establishing a fire watch or requiring equipment restoration, the licensee may choose to establish compensatory measures commensurate with the evaluated risk for continued operation with non-functional detectors. All other Required Actions are applicable regardless of compensatory measures established.
3. Entry into Condition A or C requires documentation of a Fire System Impairment, except when the non-functional detector is a result of maintenance or testing lasting less than 12 hours.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. -----NOTE----- Not applicable to Reactor Building fire detectors. -----			
Less than 50% of the detectors in the locations specified in TRM Table 3.3.6-1 FUNCTIONAL.	A.1	Establish a 1-hour roving fire watch.	1 hour
	<u>AND</u>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Condition A (continued)	A.2	Restore at least 50% of the detectors in the locations specified in TRM Table 3.3.6-1 to FUNCTIONAL status.	14 days
B. One or more detectors in the locations specified in TRM Table 3.3.6-1 non-functional that result in complete loss of automatic actuation function of a fire suppression system.	B.1	Declare the associated Fire Suppression Sprinkler/Halon System non-functional and enter applicable Conditions and Required Actions of TRO 3.7.9 and/or 3.7.10.	Immediately
C. One or more Reactor Building fire detectors non-functional.	C.1	-------------NOTE-------------- Only required in Mode 1 and 2, or when Required Action C.2 cannot be performed. -----------------------------------	
		Monitor and record Reactor Building temperature.	Once per hour
	AND		
	C.2	-------------NOTE-------------- Only required in Modes 3, 4, 5, 6 and defueled when environmental and radiological conditions permit unescorted entry. -----------------------------------	
		Verify fire watch patrol of the affected area.	Once per 8 hours
D. Required Actions and associated Completion Time for Condition A, B, or C not met.	D.1	Initiate a condition report.	Immediately
	AND		
	D.2	Determine any limitations for continued operation of the plant.	24 hours

TRM Table 3.3.6-1

SAFETY-RELATED AREAS PROTECTED BY HEAT/SMOKE DETECTORS

Protected Area Description	Fire Zone	Elevation	Controls Suppression System
Spent Fuel Area	159-B	404'	N/A
Computer Transformer Room	167-B	404'	N/A
Upper North Reactor Building Cable Spreading Area	32-K	401'	FS-5643
Upper South Reactor Building Cable Spreading Area	33-K	401'	FS-5644
Controlled Access Area	128-E	386'	N/A
Main Control Room Ceiling	129-F	386'	Halon System #3
Auxiliary Control Room Ceiling	129-F	386'	Halon System #2
Auxiliary Control Room Floor	129-F	386'	Halon System #1
Upper South Electrical Penetration Room	144-D	386'	UAV-5616
Upper North Electrical Penetration Room	149-E	386'	UAV-5615
Lower South Electrical Penetration Room	105-T	374'	UAV-5626
Lower North Electrical Penetration Room	112-I	373'	UAV-5625
Lower North Reactor Building Cable Spreading Area	32-K	373'	FS-5642
Lower South Reactor Building Cable Spreading Area	33-K	373'	FS-5645
South Switchgear Room	100-N	372'	N/A
South Inverter Room	110-L	372'	N/A
South Battery Room	110-L	372'	N/A
Cable Spreading Room	97-R	372'	UAV-5638
Hallway	98-J	372'	UAV-5639
North Switchgear Room	99-M	372'	N/A
4160 VAC Switchgear Area	197-X	372	N/A
West Heater Deck Area	197-X	372	N/A
North Emergency Diesel Generator Room	86-G	369'	UAV-5602
South Emergency Diesel Generator Room	87-H	369'	UAV-5601
Electrical Equipment Room.	104-S	368'	N/A
North Upper Piping Penetration Room	79-U	360'	UAV-5654
South Upper Piping Penetration Room	77-V	356'	N/A
Tank Room	68-P	354'/374'	N/A
Intake Structure	INTAKE	354'/366'	N/A
Laboratory And Demineralizer Access Area	67-U	354'	N/A
Condensate Demineralizer Area	73-W	354'	N/A
Compressor Room.	76-W	354'	N/A
Bowling Alley (Near Train Bay)	197-X	354	N/A
Pipe Area	40-Y	341'	N/A

TRM 3.7 PLANT SYSTEMS

TRM 3.7.9 Fire Suppression Sprinkler System

TRO 3.7.9

-----------------------------------NOTE-----------------------------------

Fire Suppression Water System sectionalized, loop, or sprinkler system valves may be closed to support system testing provided an individual is stationed at the valve with direct communication with the control room, such that the valve can be re-opened without delay if needed.

The Fire Suppression Sprinkler Systems specified in TRM Table 3.7.9-1 shall be FUNCTIONAL.

APPLICABILITY: At all times

ACTIONS

-----------------------------------NOTE-----------------------------------

1. Separate Condition entry is allowed for each sprinkler system specified in TRM Table 3.7.9-1.
2. In lieu of Required Actions establishing a fire watch, verifying FUNCTIONAL smoke and/or heat detection for the affected areas, establishing backup suppression equipment, or returning non-functional fire suppression sprinkler systems to FUNCTIONAL status, the licensee may choose to establish compensatory measures commensurate with the evaluated risk for continued operation with non-functional Fire Suppression Sprinkler Systems. All other Required Actions are applicable regardless of compensatory measures established.
3. Entry into Condition A requires documentation of a Fire System Impairment, except when non-functionality is a result of maintenance or testing lasting less than 12 hours.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Fire Suppression Sprinkler Systems specified in TRM Table 3.7.9-1 non-functional.	A.1.1 Establish a continuous fire watch in the affected area.	1 hour
	<u>OR</u>	
	A.1.2 Verify FUNCTIONAL smoke and/or heat detection for the affected area with control room alarm.	1 hour
	<u>AND</u>	

TRM Table 3.7.9-1

SAFETY-RELATED AREAS PROTECTED BY SPRINKLER SYSTEMS

Suppression Sprinkler Systems	Fire Zone	Elevation	Control Valve / Flow Switch
Upper North Reactor Building Cable Spreading Area	32-K	401'	FS-5643
Upper South Reactor Building Cable Spreading Area	33-K	401'	FS-5644
Decon Room and Hot Mechanic Shop*	149-E	386'	FS-5630
Upper South Electrical Penetration Room	144-D	386'	UAV-5616
Upper North Electrical Penetration Room	149-E	386'	UAV-5615
Lower South Electrical Penetration Room	105-T	374'	UAV-5626
Lower North Electrical Penetration Room	112-I	373'	UAV-5625
Lower North Reactor Building Cable Spreading Area	32-K	373'	FS-5642
Lower South Reactor Building Cable Spreading Area	33-K	373'	FS-5645
Cable Spreading Room	97-R	372'	UAV-5638
Hallway	98-J	372'	UAV-5639
North Emergency Diesel Generator Room	86-G	369'	UAV-5602
South Emergency Diesel Generator Room	87-H	369'	UAV-5601
Laboratory and Demineralizer Access Area*	67-U	354'	UAV-5628
Condensate Demineralizer Area	73-W	354'	UAV-5627
Intake Structure	INTAKE	354'	FS-5600
EFW Pump Room, P7A	38-Y	335'	UAV-5607
T-57A Diesel Generator Fuel Vault	251	328'	UAV-5609
T-57B Diesel Generator Fuel Vault	252	328'	UAV-5610

* Area is covered by a Sprinkler system without a corresponding Detection System.

ACTIONS (continued)

B.1

Some detectors are used to automatically actuate fire suppression sprinkler/halon systems. In these cases, the loss of the detector will prevent actuation of the fire suppression sprinkler/halon system. Therefore, it is necessary to declare the associated fire suppression sprinkler/halon system non-functional when a detector that supports actuation of the system is non-functional. This Required Action refers the user to the applicable TRM for fire suppression sprinkler and/or halon systems in order to ensure additional corrective actions and/or compensatory measures are implemented in a timely fashion.

C.1

This Required Action permits monitoring RB temperature when RB fire detection instrumentation is non-functional. The preferred instruments for monitoring RB temperature during periods when RB fire detectors may be non-functional are computer points T6278 and T6279, and must be recorded once per hour. Establishing a computer point with alarm may be used to meet the monitoring and recording requirement.

A fire in the RB will normally result in a sudden rise in temperature. A rapid increase in temperature permits prompt operator action to align fire water to the RB. Other indications that may be considered for more frequent monitoring include Reactor Coolant Pump motor amps, equipment vibration, etc.

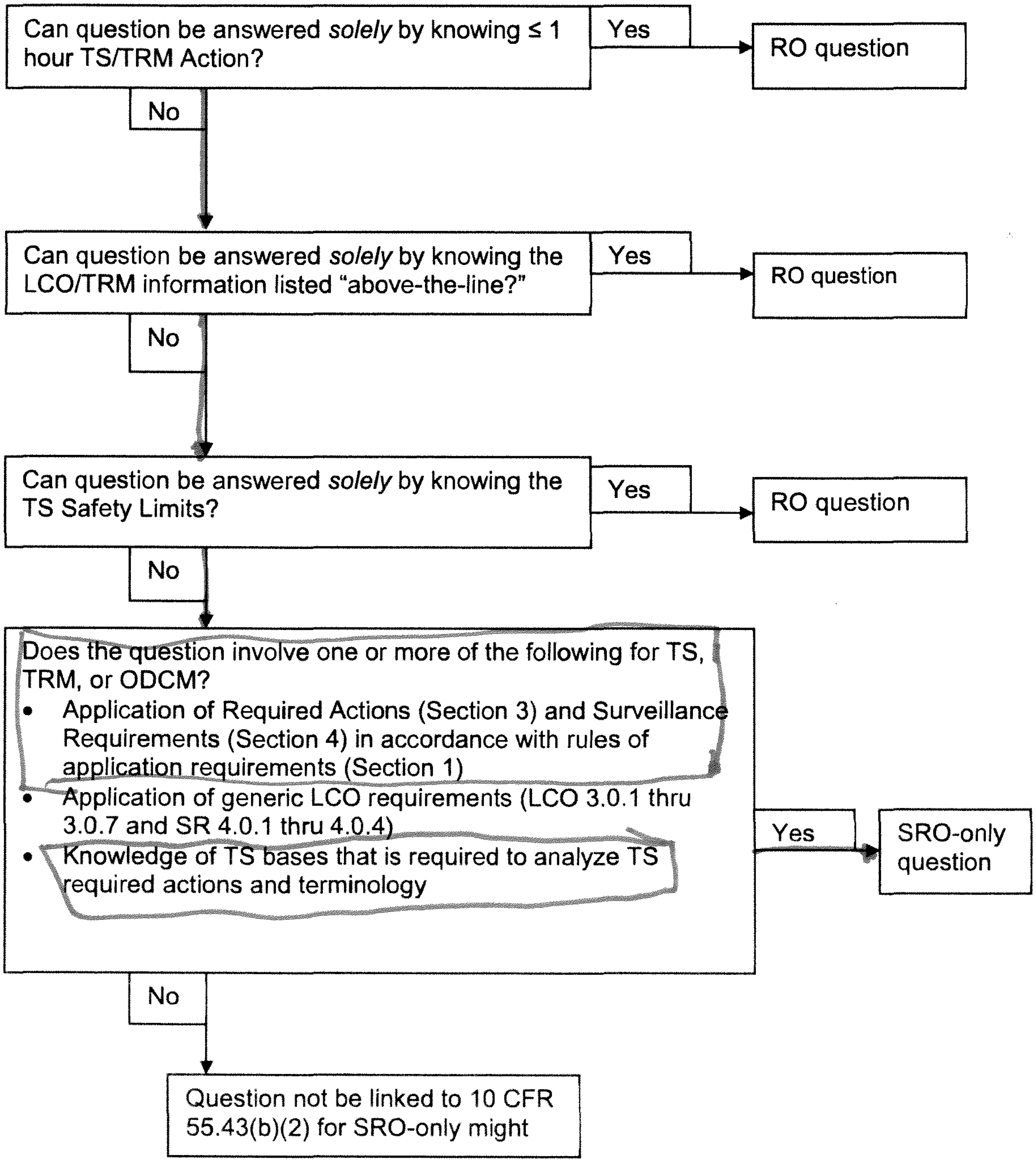
C.2

During operation while shutdown, it is prudent to establish a roving fire watch during periods when RB fire detection instrumentation is non-functional since the possibility of increased maintenance activities within the RB also can increase the possibility of fire. The CT and inspection interval are reasonable based on the low probability of a fire occurring in the RB within any 8-hour period and given the monitoring of RB temperature as required by Required Action C.1.

D.1

If the Required Actions and associated CTs cannot be met, then additional measures may be necessary to ensure continued safe operation or to reduce overall station risk. Therefore, a condition report must be initiated immediately to assess the impact on continued operation given the degraded condition of the fire detection system instrumentation.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0737 **Rev:** 2 **Rev Date:** 6/2/2008 **Source:** Bank **Originator:** Steve Pullin

TUOI: A1LP-RO-EOP **Objective:** 06 **Point Value:** 1

Section: 2 **Type:** Generic K&A

System Number: E09 **System Title:** Natural Circulation Cooldown

Description: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A Number: 2.2.44 **CFR Reference:** 43.5

Tier: 1 **RO Imp:** 4.2 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** Ap

---

Question: RO: SRO: 85

Given:
- Reactor tripped due to Degraded Power condition.
- "B" OTSG has been isolated due to leaking MSSV.

- Plant is cooling down on "A" OTSG.
- Tube to Shell delta T is 110 °F tubes colder.
- Subcooling Margin is adequate.

Which procedural action is correct for this condition?

A. Reduce cooldown rate per 1202.007, Degraded Power.

B. Establish 40-60 °F primary to secondary delta T per 1203.013, Natural Circulation Cooldown.

C. Reduce cooldown rate per 1203.013, Natural Circulation Cooldown.

D. Establish 40-60 °F primary to secondary delta T per 1202.007, Degraded Power.

Answer:

C. Reduce cooldown rate per 1203.013, Natural Circulation Cooldown.

Notes:

"C" is correct, a transition is made to Natural Circulation Cooldown from Degraded Power, and 1203.013 states that when cooling down with one dry SG and tube to shell delta temperature exceeds 100°F, then reduce cooldown rate.
"A" is incorrect but plausible since Degraded Power conditions are stated, however Degraded Power sends one to Natural Circulation Cooldown.
"B" is incorrect but plausible since the delta T reduction is an action for overheating but this action is not in 1203.013, it would be in Degraded Power if one returned to that EOP from Natural Circulation Cooldown due to inadequate SCM.
"D" is incorrect but plausible since Degraded Power conditions are given but the delta T reduction is an action for overheating, not excessive tube to shell delta T.

Revised question by changing order of correct answer, was A, now C. Removed "on Natural Circulation" to given conditions by stating plant is cooling down to remove cueing. Student should recognize plant is in Degraded Power.

This question is SRO level because it meets 10CFR55.43(b)(5) assessment of facility condtions and selection of appropriate procedures. This question cannot be answered solely by knowing a major mitigative strategy nor solely by knowing entry conditions to EOP/AOP. Candidate must know that one transitions to 1203.013 from 1202.007 with the given conditions.

This question meets the K/A since the candidate must interpret the control room conditions given to ascertain the status of the steam generators, and know the correct actions to direct to affect plant conditions positively, all during a natural circulation cooldown.

References:

1203.013, Natural Circulation Cooldown

History:

New for 2008 SRO Exam.
Selected for 2016 SRO exam

INSTRUCTIONS

35. (Continued)

Distracter

CONTINGENCY ACTIONS

B. Raise primary to secondary ΔT to 40 to 60°F as follows:

1) Adjust ATM Dump Control System to establish SG press within limits of Figure 5 “SG Pressure to Establish 40 to 60°F Primary to Secondary ΔT”.

SG A		**SG B**
CV-2676	**ATM Dump ISOL**	CV-2619
CV-2668	**ATM Dump CNTRL**	CV-2618

a) **IF** ATM Dump ISOL valve is de-energized,
THEN dispatch an operator with a radio to hand jack associated ATM Dump CNTRL valve to establish SG press within limits of Figure 5 (Refer to Alternate Shutdown (1203.002), Exhibit A).

C. **IF** primary to secondary heat transfer is established,
THEN adjust ATM Dump Control System as necessary to stabilize CET temp.

(35. CONTINUED ON NEXT PAGE)

INSTRUCTIONS	CONTINGENCY ACTIONS
15. Check ESAS ACTUATION alarms clear on K11.	15. Verify proper ESAS actuation (RT-10).
16. Check Spent Fuel Pool cooling in service.	16. Perform Unit 1 Spent Fuel Pool Emergencies (1203.050) in conjunction with this procedure. [INPO IER L1 11-2]
17. Maximize RB cooling (RT-9).	
18. Check adequate SCM. How to get to 1203.013	18. Perform the following: • IF SCM is less than adequate AND not caused by overheating, THEN **GO TO step 24.** • IF SCM is less than adequate, AND caused by overheating, THEN **GO TO step 55**.
19. Check RCS T-cold ≥ 540°F.	19. IF RCS T-cold is < 540°F and dropping, THEN **GO TO step 40.**
20. Check SG press ≥ 900 psig.	20. **GO TO step 40.**
21. Check CET temps < 610°F.	21. **GO TO step 55.**
22. Check SG tube integrity (RT-18).	22. IF SCM is adequate OR no other LOCA is indicated (RB and Aux Bldg Sump levels are stable), THEN **GO TO 1202.006, "TUBE RUPTURE"** procedure while attempting to restore buses per step 74 of this procedure.
23. GO TO step 74.	

END

INSTRUCTIONS	CONTINGENCY ACTIONS

52. Consult with Operations Manager to determine if cooldown is required.

A. IF cooldown is required, THEN perform the following:

1) Verify steps 2 through 22 of this procedure complete.

2) Perform cooldown using Natural Circulation Cooldown (1203.013), Section 1, "Degraded Power", while attempting to restore buses per step 74 of this procedure.

Transition

53. IF cooldown is not required, THEN return to Mode 3, > 525°F as follows:

A. Operate Pressurizer Heaters to maintain PZR heatup rate ≤ 100°F/hr.

B. Adjust ATM Dump Control System to establish the following, while continuing with this procedure:

- RCS heatup rate ≤ 50°F/hr
- RCS temp 535 to 545°F

SG A		SG B
CV-2676	**ATM Dump ISOL**	CV-2619
CV-2668	**ATM Dump CNTRL**	CV-2618

Contingency action B. IF ATM Dump ISOL valve is de-energized, THEN dispatch an operator with a radio to hand jack associated ATM Dump CNTRL valve to maintain RCS heatup rate to ≤ 50°F/hr
AND
establish CET temps 535 to 545°F while continuing with this procedure.
(Refer to Alternate Shutdown (1203.002), Exhibit A).

54. Verify steps 2 through 22 have been completed
AND
GO TO step 74.

END

CAUTION

If only one SG is in service, steam bubble may form in idle loop.

NOTE

- If SG is out of service, depressurization may be limited by ambient loss cooldown of the idle loop.
- Steam formation in idle loop would be indicated by drop in hot leg level indication accompanied by rapid rise in PZR level if depressurizing, or drop in PZR level if pressurizing.

3. IF cooling down with one SG dry, THEN perform the following:

A. IF Tube-to-Shell ΔT reaches 100°F (tubes colder),
THEN unless SCM is less than adequate, reduce cooldown rate as necessary to maintain Tube-to-Shell ΔT ≤ 100°F (tubes colder).

1) During emergency situations, Tube-to-Shell limit may be raised to ≤ 150°F.

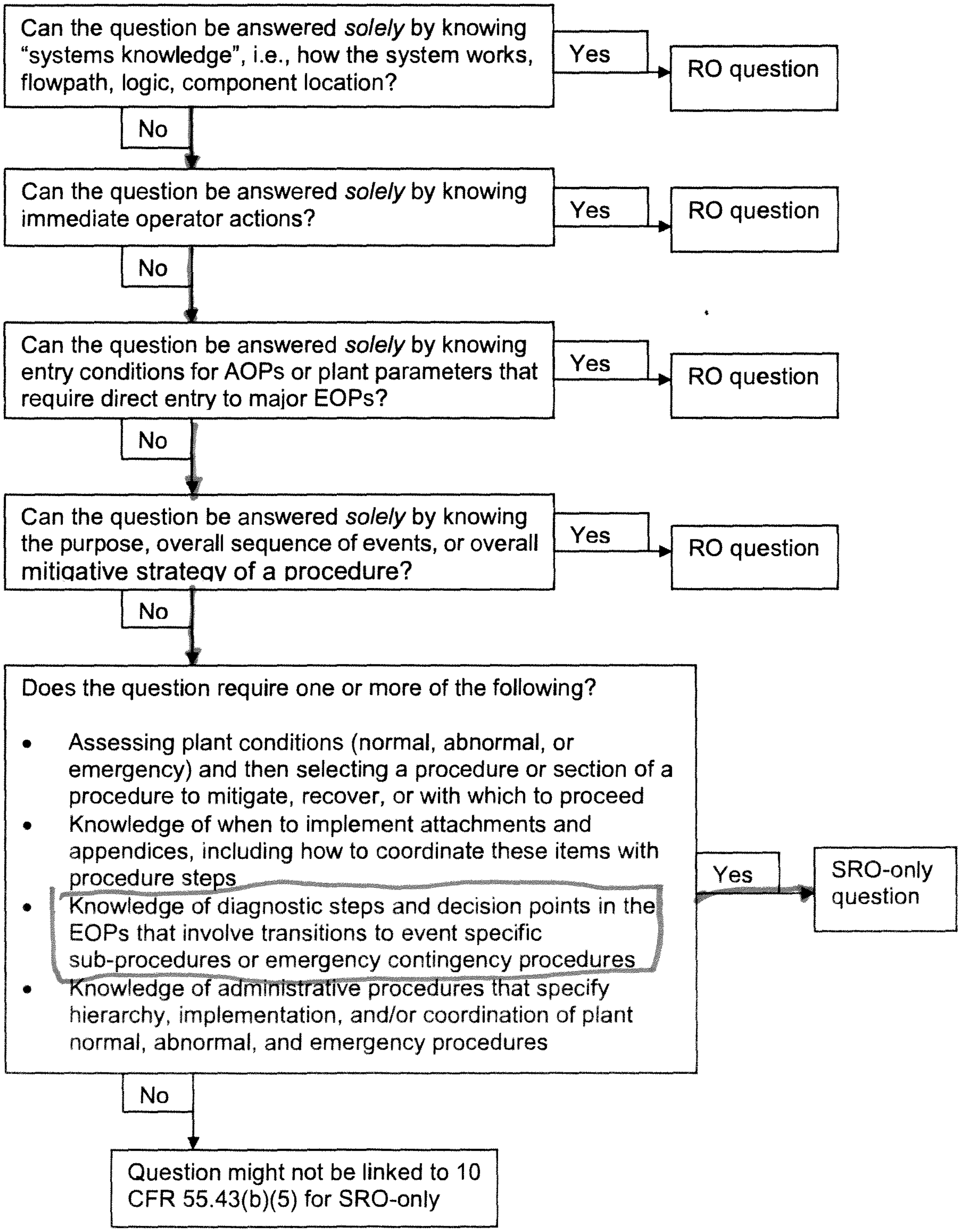
2) IF SCM is less than adequate, THEN cooldown rate limits do not apply.

B. IF a steam bubble is indicated in the idle loop
AND
ability to depressurize is inhibited,
THEN operate Loop High Point Vents as necessary to eliminate void.

**4. IF adequate SCM is lost while performing this procedure,
THEN GO TO Degraded Power (1202.007) section addressing loss of SCM, unless entry was from that section.**

5. IF ESAS actuates while performing this procedure, THEN GO TO ESAS (1202.010).

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



SRO
Tier 2
(all)

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0638 **Rev:** 2 **Rev Date:** 6/13/16 **Source:** Modified **Originator:** Possage/Cork

TUOI: A1LP-RO-ARCP **Objective:** 10 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 003 **System Title:** Reactor Coolant Pump

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off.

K/A Number: A2.01 **CFR Reference:** 43.5

Tier: 2 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 86

Given:
- Rx power is 100%
- HPI PUMP TRIP, K10-A6, in alarms
- RCP BLEEDOFF FLOW HI, K08-B7, in alarm

The CBO reports that RCP P-32B Seal Bleedoff Flow is 2.8 gpm.

Based on the above indications an RCP seal ____________ is occurring and as CRS you would direct ___________________ (actions) in response to the above conditions.

A. Degradation;
   Trip reactor, trip RCP, and go to 1202.001, Reactor Trip.

B. Failure;
   Trip reactor, trip RCP, and go to 1202.001, Reactor Trip.

C. Degradation;
   Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP.

D. Failure;
   Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP.

Answer:

C. Degradation;
   Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP.

Notes:

Answer "C" is correct, a transition is made from the alarm response procedure to the RCP AOP 1203.031. Since seal bleedoff is >2.5 gpm with a loss of seal injection, the action is to reduce power (another transition to a separate procedure) and stop the RCP as required by Section 1 of 1203.031 due to seal degradation vs. seal failure.
Section 2 of 1203.031 defines seal failure as:
>10 gpm rise in RCS leak AND change in seal cavity pressure behavior
RCP seal bleedoff or seal stage temp 200F AND no change in SI or ICW
DP across a single stage = RCS press, with seal BO established.

Answer "A" is incorrect but plausible, since the symptoms are those of seal degradation, however the actions to trip the reactor and then trip the RCP are incorrect for the given conditions.
Answer "B" is incorrect but plausible, since the symptoms for a seal failure are similar to those of seal degradation. The symptoms for a seal failure are not present and the RCP should be stopped, not tripped.
Answer "D" is incorrect but plausible, although the actions to reduce power and stop the RCP is correct, the symptoms are those of seal degradation vs. failure.

Modified original question since it no longer met SRO Only criteria. Revised all stem and all answers.

This question is SRO level since it meets 10CFR55.43(b)(5) due to assessment of condtions in the stem and selection of procedures.
The candidate must know which section of 1203.031 applies and what the subsequent procedural actions to direct are.

This question meets the K/A since it involves predicting the impact of a Reactor Coolant Pump malfunction (RCP seal degradation) and has the candidate select the procedure actions to use to mitigate the high RCP seal leakoff rate.

References:

1203.031, Reactor Coolant Pump and Motor Emergency

History:

New, created for 2007 SRO exam.
Modified for 2016 SRO exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0638 **Rev:** 0 **Rev Date:** 10/16/2006 **Source:** Direct **Originator:** Possage/Cork

TUOI: A1LP-RO-ARCP **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic Abnormal Plant Evolutions

System Number: 015 **System Title:** RCP Malfunctions

Description: Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection.

K/A Number: AA2.10 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** An

Question: **RO:** **SRO:**

Given:
- Rx power 100%
- HPI PUMP TRIP, K10-A6, in alarm
- RCP BLEEDOFF FLOW HI, K08-B7, in alarm

The CBO reports that RCP P-32B Seal Bleedoff Flow is 2.8 gpm.

Which of the following are the correct procedures and responses to the above conditions?

Prior to Modification

A. Trip RCP per 1203.012G, Annunciator K08 Corrective Action, and go to 1202.001, Reactor Trip.

B. Trip RCP per Section 2, Seal Failure, of 1203.031, Reactor Coolant Pump and Motor Emergency, and go to 1202.001, Reactor Trip.

C. Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP per Section 1, Seal Degradation, of 1203.031, Reactor Coolant Pump and Motor Emergency.

D. Reduce power using 1102.004, Power Operations, then stop RCP per 1103.006, Reactor Coolant Pump Operation, and go to 1203.022, Reactor Coolant Pump Trip.

Answer:

C. Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP per Section 1, Seal Degradation, of 1203.031, Reactor Coolant Pump and Motor Emergency.

Notes:

Answer "C" is correct since seal bleedoff is >2.5 gpm with a loss of seal injection as required by Section 1 of 1203.031.
Answer "A" is incorrect, although1203.012G contains actions for the annunciators listed, those actions direct the operator to go to 1203.031.
Answer "B" is incorrect, although 1203.031 is the correct procedure, the symptoms for a seal failure are not present and the RCP should be stopped, not tripped.
Answer "D" is incorrect, although the actions are similar to the correct actions in Section 1 of 1203.031, the RCP trip AOP should not be used unless the RCP trips without operator action.

References:

1203.031, Reactor Coolant Pump and Motor Emergency

History:

New, created for 2007 SRO exam.

SECTION 1
SEAL DEGRADATION

ENTRY CONDITIONS

One or more of the following:

- RCP seal bleed off flow high **OR** low
- RCP seal bleed off or seal stage temperature high **OR** low
- Drinking bird or T-111 system indicates high seal leakage
- RCP seal cavity pressure erratic
- RCP SEAL CAVITY PRESS HI/LO (K08-D7)

SECTION 1
SEAL DEGRADATION

NOTE

- RCP seal stage ΔP is determined as follows:
  - 1st stage ΔP = system pressure - lower seal cavity press.
  - 2nd stage ΔP = lower seal cavity pressure - upper seal cavity press.
  - 3rd stage ΔP = upper seal cavity pressure - RB atmospheric press.
- Third stage seal leakage by design is 0 to 0.08 gpm. Third stage leakage in excess of design will affect upper seal cavity pressure and seal bleed off flow.

4. Determine if any of the following conditions exist:

- RCP seal cavity pressure oscillations exceed 800 psi peak-to-peak
- ΔP across any stage exceeds 2/3 of system pressure
- A loss of seal injection
  AND ≥2.5 gpm total seal outflow, including seal bleedoff (excluding shaft sleeve leakage)
- RCP seal bleed off or seal stage temp reaches 180°F
  AND no interruption of seal injection **OR** ICW flow.

A. **IF** any of the above conditions exist,
THEN reduce reactor power to within the capacity of the unaffected RCP combination, using Rapid Plant Shutdown (1203.045)

B. **WHEN** power reduction is complete,
THEN stop the affected RCP(s) per Reactor Coolant Pump Operation (1103.006).

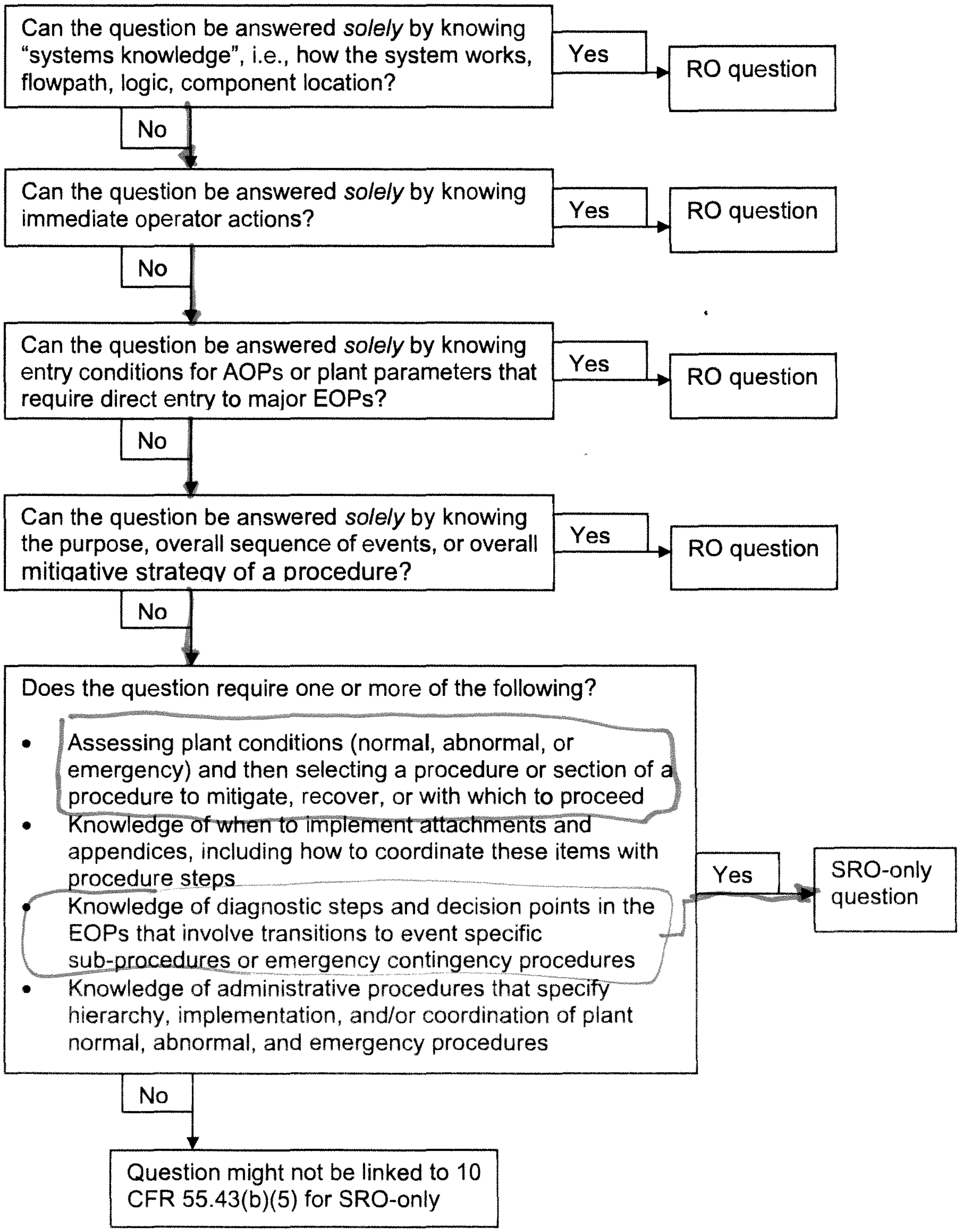
1) **IF** only 1 RCP in operation per loop,
THEN enter Tech Spec 3.4.4 Condition A (18-hour time clock).

**5. IF either of the following conditions exists on an idle RCP,
THEN plant shutdown to refurbish the seal should be considered:**

- ΔP across any stage exceeds 80% of system pressure
- ≥2.5 gpm total seal outflow, including seal bleedoff (excluding shaft sleeve leakage)

(continued)

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1052 **Rev:** 2 **Rev Date:** 7/22/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-AOP **Objective:** 5 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation System ESFAS

Description: Ability to (a) predict the impacts of the following malfunction or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc control power.

K/A Number: A2.05 **CFR Reference:** 43.2 / 43.5

Tier:	2	**RO Imp:**	3.7	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 87

REFERENCE PROVIDED

A reactor trip has occurred from 100% power with the following indications:

- Breaker position lights on the RIGHT side of C10 are off.
- Both OTSGs pressures are ~890 psig and slowly trending down.
- PZR level 75" and slowly trending down.
- Attempts to transfer the affected 125V DC panel to its emergency supply are unsuccessful.

Which of the following is the correct procedure to be implemented, in conjunction with EOPs, and which Technical Specification LCO action is applicable?

A. 1203.036, Loss of 125V DC
Tech.Spec 3.3.5.B

B. 1203.036, Loss of 125V DC
Tech Spec 3.3.7.A

C. 1203.053, Inadvertent ESAS Actuation
Tech.Spec 3.3.5.B

D. 1203.053, Inadvertent ESAS Actuation
Tech Spec 3.3.7.A

Answer:

B. 1203.036, Loss of 125V DC
Tech Spec 3.3.7.A

Notes:

"B" is the correct answer per Section 2, Loss of D02, in 1203.036. Loss of DC AOP should be performed in conjunction with entry into the Reactor Trip EOP 1202.001. Loss of DC bus D02 causes a reactor trip and loss of control power to even train breakers means they will not auto transfer during the trip. EDG 2 will not auto start due to loss of control power so the even train busses A2, A4, and H2 will be de-energized. A loss of DC and a loss of alternate AC will cause a loss of inverters Y22, Y24, Y25 which will cause a loss of power to 120V AC panels RS2 and RS4. This will cause ESAS Analog Channel 2 to trip and ESAS Digital Channel 2 to become inoperable, thus 3.3.7.B applies.
"A" is incorrect, but plausible as this choice contains the correct procedure 1203.036 but the incorrect Tech Spec. Tech Spec 3.3.5.A is applicable but not 3.3.5.B (3.3.5.B would be applicable if it was a loss of D01).
"C" is incorrect, but plausible since Tech Spec 3.3.5.A is applicable but not 3.3.5.B (3.3.5.B would be applicable if it was a loss of D01). 1203.053 would only be in use if this was a loss of D01.
"D" is incorrect, But plausble since it contains the correc Tech Spec but 1203.053 would only be in use if this

was a loss of D01.

This question is SRO only since it meets 10CFR55.43(b)(5): the question requires the candidate to assess the facility conditions given and to select the appropriate procedure, and determine the applicable Technical Specification LCO, 10CFR55.43(b)(2).

This question meets the K/A as it requires the candidate to predict the impact of the malfunction of ESFAS (a loss of DC control power - D02), and it requires the candidate to know which action (in the appropriate procedure) will mitigate the loss of D02.

Revised stem per NRC examiner commennt. JWC 7/15/16

References:

1203.036, Loss of 125V DC
Technical Specifications 3.3.5 and 3.3.7

BOTH Tech Specs must be in the SRO Handout!!!!

History:

New question for 2016 SRO exam.

Control power for the generator output breakers (5114 and 5118) is from the switchyard 125V DC system, however, the breakers do not auto trip on generator lockout due to loss of power to the generator lockout circuit. Switchyard protective relays may trip the generator output breakers sensing an electrical fault. If power is restored to D11, the generator lockout circuit will actuate to trip the output breakers. The generator output breakers should not be manually tripped at this time because a loss of AC power will result as the turbine coasts down carrying the A1/A2 buses and H1/H2 buses. DG2 only will auto start if power is lost.

Restoration of power to distribution panel D11 by manually transferring to the emergency power source will allow automatic actions to occur such as turbine trip and generator lockout. This results in generator output and Exciter Field breakers tripping, then automatic transfer to a SU Xfmr (SU 1 or SU 2).

If the generator output breakers were opened prior to re-energizing D11, an automatic transfer to offsite power could not take place and buses would remain energized from the turbine generator. If the turbine were tripped manually or electrically or mechanically due to overspeed or loss of vacuum, or the MSIVs are closed due to low pressure, the turbine would coast down and voltage and frequency control would be lost. The undervoltage condition would cause load shedding of even-train buses and starting of DG2. The odd train buses would become de-energized and DG1 could not start due to loss of DC control power.

If AC power is lost, inverters Y11, Y13, and Y15, cannot operate due to the loss of DC input power from D01. With a loss of alternate AC input power also, power will be lost to 120V AC panels RS1 and RS3. This causes a loss of power to two out of three ESAS analog channels, resulting in actuation of all ES even digital channels. Also, EFIC MSLI and EFW will actuate. Odd ES channels cannot actuate due to loss of power to the odd digital channels.

If AC power is <u>not</u> lost and a valid ES signal is received, the following valves will not reposition due to a loss of ES control (RA1 BKR 9).

- DH Cooler Bypass (CV-1433)
- Letdown Coolers Outlet (CV-1214 and CV-1216)
- RCP Seal Bleedoff Valves (CV-1270 thru CV-1273)

<u>Effects of Loss of D02</u>

A complete loss of bus D02 includes loss of 125V DC Station Battery Bank to Bus D02 (D06), loss of battery charger, and loss of distribution system. This results in the following conditions:

- If reactor power is >55%, reactor trip.
- Loss of power to even train distribution breaker control as well as other loads powered from bus D02.
- Loss of power to EOS Main Turbine Trip Solenoids (SV-6623 and SV-6624).
- Loss of power to EOS Channel B (SY-6650).

A reactor trip will result from loss of bus D02 if reactor power is >55% due to loss of power to the RCP under-power monitor circuit (RA2 BKR 16).

The reactor trip results in turbine trip and generator lockout relays (286 G1-1, 286 G1-2, and 286 G1-3) actuation, which causes automatic transfer to offsite power. With a loss of control power to the even train breakers, these breakers will not operate. This results in a loss of AC power to the even train buses. DG2 will not start due to a loss of control power.

With no AC or DC power, inverters Y22, Y24 and Y25 will be lost resulting in loss of power to 120V AC Panels RS2 and RS4. Inverters Y11, Y13, and Y15 remain in a normal mode.

Loss of power to RS2 and RS4 results in EFIC actuation of MSL and EFW. With a loss of D21, control power to EFW Pump (P-7A) is lost and the turbine will trip on overspeed. EFW control valves associated with P-7A are failed full open (loss of RA2).

Loss of power to Y02 results in closure of Purification Demineralizer Inlet and Makeup Filter Inlet valves causing letdown relief valve to lift. Letdown must be isolated by closing LD Cooler E-29A Outlet MOV (CV-1214) and LD Cooler E-29B Outlet MOV (CV-1216).

Loss of DC control power to Condenser Vacuum Pump (C-5B), if operating, causes the Seal Recirc Pump (P-31B) to stop and vacuum pump inlet valve to close. Upon restoration of DC control power, the condenser vacuum pump will trip and must be restarted or will auto start on low vacuum.

If a valid ES signal is received, DH Cooler Bypass (CV-1432) will not reposition due to a loss of ES control (RA2 BKR 11).

Effects of Loss of Both D01 and D02

A complete loss of both bus D01 and D02 includes loss of:

- 125V DC Station Battery Bank to Bus D01 (D07)
- 125V DC Station Battery Bank to Bus D02 (D06)
- Battery chargers to D07 and D06

- D01 distribution system
- D02 distribution system

This loss results in the following conditions:

- Reactor trip
- Loss of power to main turbine trip solenoids (SV-8524 and SV-8527 and XZ-8524).
- Loss of power to EOS Overspeed Trip Protection.
- Loss of EOS Main Turbine Trip Solenoids (SV-6623 and SV-6624).

- Loss of power to generator lockout relays (286 G1-1, 286 G1-2, and 286 G1-3).
- Loss of power to distribution breaker control power.

KEY

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

LCO 3.3.7 The ESAS digital actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each digital actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital actuation logic channels inoperable.	A.1 Place associated component(s) in engineered safeguards configuration.	1 hour
	<u>OR</u>	
	A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	31 days

3.3 INSTRUMENTATION

3.3.5 Engineered Safeguards Actuation System (ESAS) Instrumentation

LCO 3.3.5 Three ESAS analog instrument channels for each Parameter in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Parameter.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one analog instrument channel inoperable.	A.1 Place analog instrument channel in trip.	1 hour
B. One or more Parameters with more than one analog instrument channel inoperable. <u>OR</u> Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 --------------NOTE--------------- Only required for RCS Pressure - Low setpoint. ----------------------------------- Reduce RCS pressure < 1750 psig.	36 hours
	<u>AND</u>	
	B.3 --------------NOTES------------- 1. Only required for Reactor Building Pressure High setpoint and High High setpoint. 2. LCO 3.0.4.a is not applicable when entering Mode 4. ----------------------------------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.5.2	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.5-1
Engineered Safeguards Actuation System Instrumentation

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Reactor Coolant System Pressure – Low Setpoint	≥ 1750 psig	≥ 1585 psig
2. Reactor Building (RB) Pressure – High Setpoint	1,2,3,4	≤ 18.7 psia
3. RB Pressure – High High Setpoint	1,2,3,4	≤ 44.7 psia

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

LCO 3.3.7 The ESAS digital actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each digital actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital actuation logic channels inoperable.	A.1 Place associated component(s) in engineered safeguards configuration.	1 hour
	<u>OR</u>	
	A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	31 days

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1099 **Rev:** 0 **Rev Date:** 6/14/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EFIC **Objective:** 43 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 061 **System Title:** Auxiliary/Emergency Feedwater

Description: Ability to determine operability and/or availability of safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 43.2

Tier: 2 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** Ap

Question: RO: SRO: 88

REFERENCE PROVIDED

Given:
- Unit is at 10% power
- Surveillance has just been performed on P-7A EFW pump.
- CBOT notes that indication is lost on SG A steam admission valve to K3 EFW pump turbine, CV-2613.
- Investigation shows that the breaker for CV-2613 (D-2512) is open and will not reset.

Which of the following actions will comply with all Technical Specifications?

A. Declare P-7A inoperable and restore to operable status within 72 hours.

B. Declare CV-2613 inoperable, de-energize in the closed position, and restore to operable status within 48 hours.

C. Declare CV-2613 inoperable and restore to operable status within 7 days.

D. De-energize CV-2613 in the closed position to maintain operability of P-7A.

Answer:

A. Declare P-7A inoperable and restore to operable status within 72 hours.

Notes:

"A" is correct, P-7A is the "green" train of EFW and the green DC powered steam admission valve CV-2613 is required for operability of P-7A, so TS 3.7.5.B must be entered and P-7A declared inoperable.
"B" is incorrect but plausible if the candidate confuses steam admission valve CV-2613 with steam supply valve CV-2617 and applies LCO 3.6.3 since CV-2617's duty as a reactor building isolation valve.
"C" is incorrect but plausible if the candidate confuses steam admission valve CV-2613 with steam supply valve CV-2617 and applies LCO 3.7.5.A.
"D" is incorrect but plausible sounding as CV-2613 one of two steam admission valves so with CV-2613 closed CV-2663 is still available but CV-2663 is an "enhancement" and will not maintain operability per 1106.006. De-energizing CV-2663 in the closed position will maintain operabiilty of P-7A but the opposite is not true.

This question is SRO only because it requires the candidate to apply LCO action statements to the conditions given and determine the applicable action and completion time. This particular LCO requires knowledge of TS bases to recognize which action is applicable.

This question matches the K/A since it pertains to Emergency Feedwater and requires the candidate to determine operability of P-7A, a safety related EFW pump.

References:

Technical Specifications 3.7.5 and 3.6.3 (both must be in SRO handout)

1106.006, Emergency Feedwater Pump Operation.

History:

New for 2016 SRO exam

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

---NOTE---
Only one EFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

--NOTE--
LCO 3.0.4.b is not applicable when entering Mode 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven EFW pump inoperable. <u>OR</u> --------------NOTE------------ Only applicable if MODE 2 has not been entered following refueling. ----------------------------------- Turbine driven EFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One EFW train inoperable for reasons other than Condition A in MODE 1, 2, or 3.	B.1 Restore EFW train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 18 hours
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 -------------NOTE---------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ----------------------------------- Initiate action to restore one EFW train to OPERABLE status.	 Immediately
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	------------------------------NOTE---------------------------- Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching $\geq$ 750 psig in the steam generators. --- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	 In accordance with the Inservice Testing Program

18.0 OPERABILITY

18.1 Discussion — This section aids in determining system operability for consistency. This is NOT a listing of all requirements necessary for system operability.

18.2 EFW Pump Turbine K3 Steam from SG A/SG B Valves (CV-2617 and CV-2667)

If either CV-2617 OR CV-2667 becomes inoperable, de-energizing and locking the valve in the open position will maintain EFW Pump (P-7A) operable. The only analysis of concern is a steam generator tube rupture with P-7A in service. In this situation, the associated steam supply valve will have to be manually closed in order to prevent OR stop an offsite release. Further consideration should be given to the acceptability of this condition prior to long-term continuous operation.

CV-2617 and CV-2667 are GDC-57 valves for containment penetration concerns and the controls of the "Operation of Containment Penetration Valves and Components (ANO1), section of Conduct of Operations (1015.001) apply. If CV-2617 OR CV-2667 are de-energized and locked open to maintain EFW Pump (P-7A) operable, then appropriate containment specification must be entered.
(Refer to CR-ANO-C-2014-01799.)

During a steam line break between the steam generator and Main Steam Isolation Valves, the affected generator will be isolated and the unaffected steam generator will be used for decay heat removal. In this situation, if the steam supply valve from the unaffected steam generator were to be failed closed prior to the steam line break, a single failure of EFW Pump (P-7B) could result in a total loss of emergency feedwater. Therefore, neither CV-2617 OR CV-2667 can be de-energized and locked in the closed position to maintain P-7A operable.

Reference "MOV Operations" section of Conduct of Operations (1015.001) for valve operations.

With CV-2617 and CV-2667 closed and in AUTO, opening CV-2613 or CV-2663 renders P-7A inoperable per TS 3.7.5.

If CV-2617 and CV-2667 are closed for an extended period of time, with the desire to maintain P-7A available, then ONE of the following must be met to verify downstream piping does not contain excessive condensate:

- Piping temperature upstream of K-3 Steam Flow Orifice (FO-2603) greater than 250°F.
- K-3 Combined Steam Traps (ST-129 or ST-130) have an active discharge of water-steam mixture with a consistent flow volume over an extended time period.

Prior to returning EFW Pump (P-7A) to operable status or operating P-7A to determine operability after CV-2617 and CV-2667 have been closed for an extended period of time, the following conditions must be met to verify downstream piping does not contain excessive condensate:

- CV-2617 and CV-2667 must be open.
- Piping temperature upstream of K-3 Steam Flow Orifice (FO-2603) must be greater than 250°F.
- K-3 Combined Steam Traps (ST-129 or ST-130) have an active discharge of water-steam mixture with a consistent flow volume over an extended time period.

18.3 EFW Pump Turbine K3 Steam Admission Valves (CV-2613 and CV-2663)

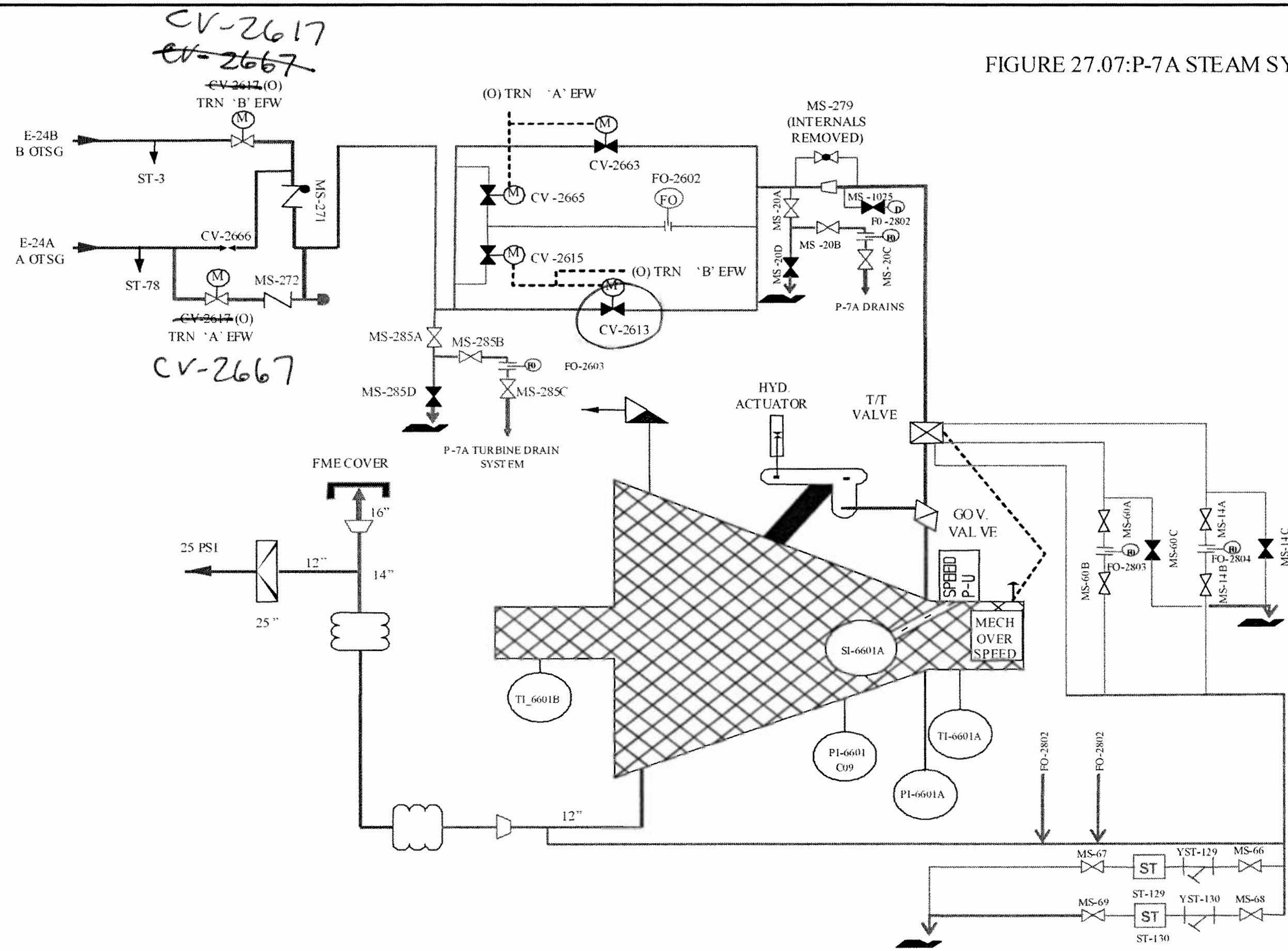
The required supply of steam to EFW Pump (P-7A) will be via the green DC powered CV-2613. If CV-2613 becomes inoperable then P-7A is also inoperable. The red DC powered CV-2663 provides enhanced EFW reliability, but this steam flow path is not required. If CV-2663 becomes inoperable, de-energizing CV-2663 in the closed position will maintain P-7A operable. Reference "MOV Operations" section of Conduct of Operations (1015.001) for valve operations.
Common to CV-2613 and CV-2663 is the ramp circuitry for P-7A. The ramp circuit is green powered and can be energized by opening either CV-2613 or CV-2663 electrically or manually. The ramp circuit is energized when CV-2613 or CV-2663 is >90% open. Energizing the ramp circuit changes the speed setpoint from ~910 RPM to ~3650 RPM. If the ramp circuit is energized when the steam admission valves are closed, then it is possible that P-7A will trip on overspeed due to a slow response from EFW Turbine K3 Gov Servo (CV-6601B) if the steam admission valves are subsequently opened.
Therefore, if maintenance is required on CV-2663 that affects the integrity of the valve operator then P-7A will be declared inoperable. This includes, but is not limited to, the removal of the limit deck cover, replacement of gears and removal of the motor.

With EFW Pump Turbine K3 Steam from SG A/SG B Valves (CV-2617 and CV-2667) closed and in AUTO, opening CV-2613 or CV-2663 renders P-7A inoperable.

18.4 EFW Pump Turbine K3 Steam Admission Valve Bypasses (CV-2615 and CV-2665)

CV-2615 and CV-2665 are designed to provide a smoother transient upon admission of steam to EFW Pump Turbine (K-3) and provide an initial small flow to prevent steam hammer forces caused by sudden opening of the steam admission valves. Therefore, green DC powered valve CV-2615 must be operable to maintain EFW Pump (P-7A) operable. The inoperability of CV-2665 does not affect the operability of P-7A. EFW Pump Turbine K3 Steam Admission Valve (CV-2663) should be de-energized in the closed position if CV-2665 fails. Reference "MOV Operations" section of Conduct of Operations (1015.001) for valve operations.

FIGURE 27.07:P-7A STEAM SYSTEM



INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1046 **Rev:** 1 **Rev Date:** 7/21/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-RO-EOP07 **Objective:** 12.3 **Point Value:** 1

Section: 3.6 **Type:** Plant Systems: Electrical

System Number: 062 **System Title:** AC Electrical Distribution System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Keeping the safeguards buses electrically separate.

K/A Number: A2.06 **CFR Reference:** 43.5

Tier:	2	**RO Imp:**	3.4	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: RO: SRO: 89

Given:

- Plant has lost all offsite power.
- 1202.007, Degraded Power, EOP is in use.
- #2 EDG is supplying A4.
- #1 EDG has tripped due to low lube oil pressure.
- AAC Generator is OOS for maintenance.

- P-7A EFW pump tripped and could not be reset.
- RCS pressure 2325 psig.
- CET average indicates 612°F.
- A3 and A4 buses were cross-tied and P-7B EFW pump started.
- CBOT reports SU1 voltage is now 22.5 KV.

Which procedure should you transition to and which is the procedurally required action for the above conditions?

A. Go to 1202.004, Overheating, while continuing with bus restoration section of 1202.007.

B. Go to 1202.005, Inadequate Core Cooling, while continuing with bus restoration section of 1202.007.

C. Go to 1202.011, HPI Cooldown, and dispatch an operator to perform Att. 2, "Recovery from Degraded Power Breaker Alignment and UV Relay Defeat".

D. Go to 1107.002, ES Electrical System Operation, and restore buses to normal using "Returning Paralleled Buses A3 and A4 to Normal" section, while continuing with 1202.007.

Answer:

D. Go to 1107.002, ES Electrical System Operation, and restore buses to normal using "Returning Paralleled Buses A3 and A4 to Normal" section, while continuing with 1202.007.

Notes:

"D" is correct, per step 57.D.2 of 1202.007, Degraded Power, once off-site power becomes available, then buses should be restored to normal using 1107.002 (transition). This will maintain separation of ES buses.
"A" is incorrect but plausible. The RCS temperature given indicates entry conditions are met for the Overheating EOP but an SRO candidate should know to stay in the Degraded Power EOP since it has a section for mitigating an overheating condition.
"B" is incorrect but plausible. The 1202.007 EOP does direct entry into the Inadequate Core Cooling EOP

(step 37) and while the RCS temperature given is quite high, the RCS is not in an ICC condition. If a candidate misreads the EOP figures, then this distracter is quite plausible.
"C" is incorrect but plausible. The 1202.007 EOP does direct entry into the HPI Cooldown EOP (step 47) and Attachment 2 is directed to be performed if A2 is energized from A4 in step 115 but the two are not performed together.

This is an SRO level question as it meets 10 CFR 55.43(b)(5), assessment of conditions and selection of procedures. It is not RO level since it requires in-depth knowledge of AOPs and EOPs.

This question meets the K/A as it requires the candidate to assess the conditions given and predict the impact, i.e., the ES buses A3 and A4 are cross-tied and now that SU1 is available as indicated by 22.5 KV, then the A3 and A4 buses should be electrically separated to protect them from common faults. The candidate also needs to know the procedure heirarchy, i.e., the EOP user's guide and for ANO that means staying in the Degraded Power EOP despite the indications of heat transfer upsets.

Modified D per NRC examiner suggestion. JWC 7/21/16

References:

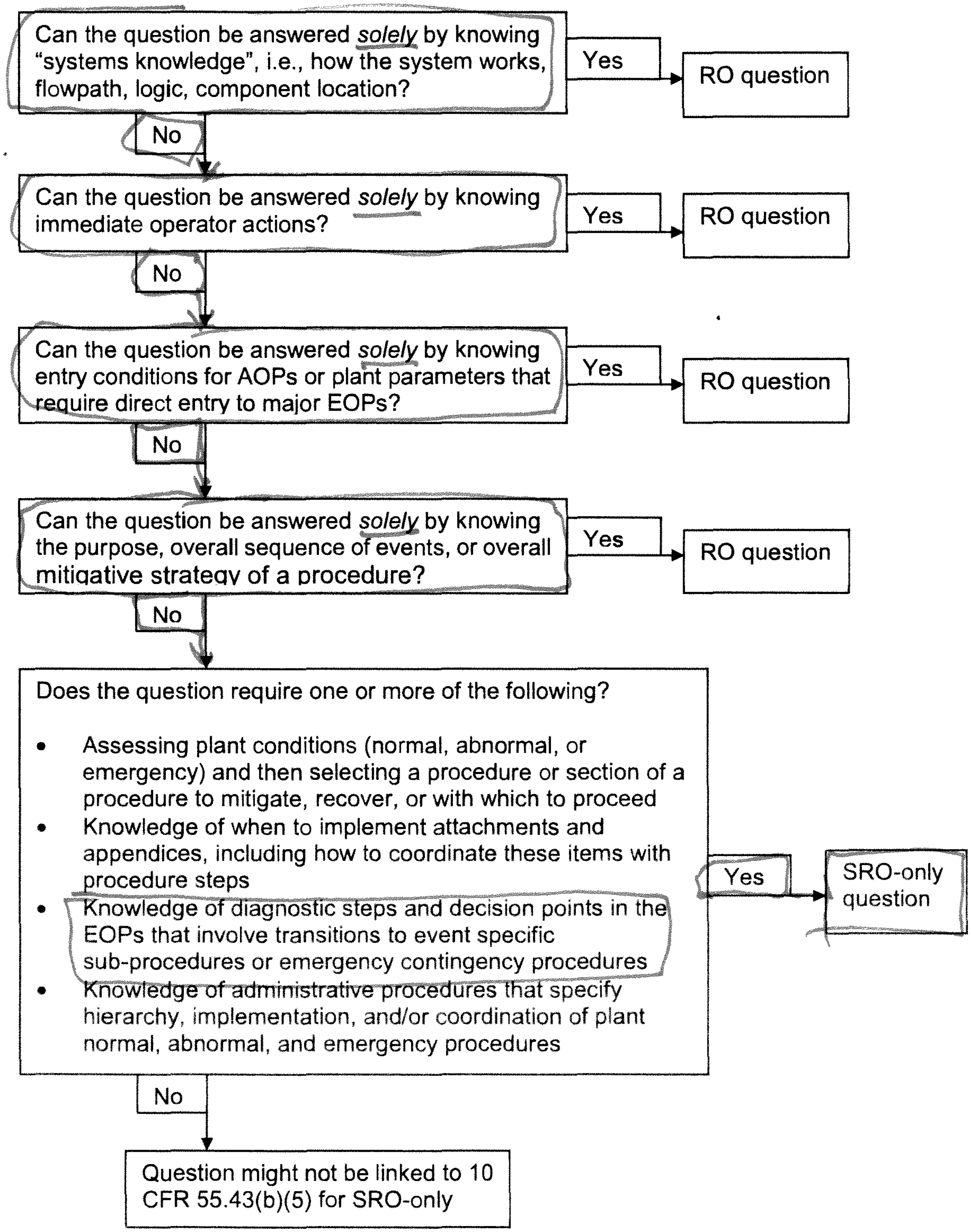
1202.007, Degraded Power
1202.013, EOP Figures

History:

New SRO question for 2016 exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
57. (Continued)	
D. **IF** A3 is de-energized **AND** P7A is unavailable, **THEN** restore power to P7B as follows:	
1) Energize A3 using ES Electrical System Operation (1107.002), "Bus A3 to A4 Crosstie to Energize Dead Bus" section.	
a) **IF** another DG **OR** off-site power becomes available, **THEN** restore buses to normal using ES Electrical System Operation (1107.002), "Returning Paralleled Buses A3 and A4 to Normal" section.	Correct Answer
2) Start P7B **AND** **GO TO step 65.**	
E. Restore EFW using Annunciator K12 Corrective Action (1203.012K), while continuing with this procedure.	

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1047 **Rev:** 0 **Rev Date:** 3/21/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 3.6 **Type:** Plant Systems: Electrical

System Number: 064 **System Title:** Emergency Diesel Generator

Description: Knowledge of abnormal condition procedures.

K/A Number: 2.4.11 **CFR Reference:** 43.5

Tier:	2	**RO Imp:**	4.0	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 90

Given:
- Unit 1 is in Mode 1.
- It is August and ambient outside temperature is 103°F.
- CBOT reports that both A3 and A4 bus voltages are ~3700 volts.
- CBOT also reports Startup #1 Transformer voltage is 22.1 KV and SU#2 Transformer is 160KV.
- After being contacted, Dispatcher reports voltage regulators are in-service and working properly but a major capacitor bank is out of service.
- This condition has not improved after several hours.
- No grid disturbances are occurring.

Procedure 1203.037, Abnormal ES Bus Voltage and Degraded Offsite Power, has been entered.

Which of the following procedure sections should be transitioned to and which procedurally required actions are warranted for the above conditions?

A. In accordance with Section 3, Offsite Voltage Abnormal, start one available DG, parallel the DG to the grid, and separate the associated ES bus from the grid by opening its feeder breaker.

B. In accordance with Section 2, ES Bus Voltage Low, start one available DG, parallel the DG to the grid, and separate the associated ES bus from the grid by opening its feeder breaker.

C. In accordance with Section 3, Offsite Voltage Abnormal, start one available DG, de-energize the associated ES bus by opening its feeder breaker, and verify DG output breaker closes.

D. In accordance withSection 2, ES Bus Voltage Low, start one available DG, de-energize the associated ES bus by opening its feeder breaker, and verify DG output breaker closes.

Answer:

B. In accordance with Section 2, ES Bus Voltage Low, start one available DG, parallel the DG to the grid, and separate the associated ES bus from the grid by opening its feeder breaker.

Notes:

"B" is correct, bus voltage is low but not low enough to autostart the DGs, no grid disturbance is expected, so per 1203.037, section 1, step 6.A one DG should be started, paralleled to the grid, and the associated ES bus separated from the grid.
"A" is incorrect since Section 3 does not contain this action, instead major loads are secured to reduce voltage but offsite voltages are not low enough to require this section to be used. Section 4, Offsite Frequency Low, contains this action. "A" contains the correct action but the wrong procedure section.
"C" is incorrect since Section 3 does not contain this action, instead major loads are secured to reduce voltage

but offsite voltages are not low enough to require this section to be used. Section 4, Offsite Frequency Low, contains this action. "C" contains the wrong procedure section and the wrong action but completes the 2x2 format.
"D" is incorrect since no grid disturbance is expected, therefore the ES bus should not be de-energized to allow the DG to automatically re-energize it, but plausible since it refers to the correct procedure.

This is an SRO level question as it meets 10 CFR 55.43(b)(5), assessment of conditions and selection of procedures. It is not RO level since it requires in-depth knowledge of AOPs and has an SRO specific learning objective.

This question matches the K/A since it concerns both Emergency Diesel Generators and abnormal condition procedures.

References:

1203.037, Abnormal ES Bus Voltage and Degraded Offsite Power

History:

New SRO question for 2016 exam.

SECTION 1 -- ES BUS VOLTAGE LOW (continued)

NOTE

- The EDG will be lightly loaded when the buses are separated from the grid. Running EDGs at low loads for long periods causes excessive engine wear.
- A bumpless transfer to the EDGs is preferred because of the possibility of inverters being on alternate source, or battery chargers supplying DC loads or other loads being used that might not restart when voltage is regained.
- Non-ES bus loads would be subjected to the harmful effects of low voltage (high motor currents, overheating, etc.) and are not protected by the transfer of the ES buses to the EDG. Therefore a tap change on the startup transformer or 480V load center transformer should be considered.
- Stripping of non-essential loads will prevent unnecessary exposure to the low voltage.
- During periods of known off-site electrical grid disturbances, neither diesel should be paralleled to the grid.

6. <u>IF</u> normal voltage levels are <u>NOT</u> regained,
<u>THEN</u> perform the following to make an orderly transfer of ES bus loads to the Emergency Diesel Generator to preclude automatic DG start and load shedding on 480 volt bus undervoltage (as high as 439 volts):

A. <u>IF</u> grid conditions are stable,
<u>THEN</u> perform the following:

1) Start one available DG per "DG1 (or DG2) Start From Control Room" section of Emergency Diesel Generator Operation (1104.036).

2) Parallel the associated DG to the grid per "DG1 (or DG2) Start From Control Room" section of Emergency Diesel Generator Operation (1104.036).

3) Separate the associated ES Bus from the grid by opening its feeder breaker:

Bus A3	Bus A4
A-309	A-409

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1056 **Rev:** 1 **Rev Date:** 7/21/16 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-NNI **Objective:** 35 **Point Value:** 1

Section: 3.2 **Type:** Reactor Coolant System Inventroy Control

System Number: 011 **System Title:** Pressurizer Level Control

Description: Ability to apply technical specifications for a system.

K/A Number: 2.2.40 **CFR Reference:** 43.2

Tier:	2	**RO Imp:**	3.4	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: RO: SRO: 91

It is 0900 on Sunday, July 31, 2016.
You are assuming the watch after being on assignment to Training.
The plant is operating at 100% power.
While reviewing the logs you notice PZR level transmitter LT-1001 failed LOW at 2100 on Thursday, June 30.

Technical Specification LCO 3.3.15 Action B.1 states to initiate action to prepare and submit aa special report to the NRC.

When is the special report to the NRC due?

A. 8 hours

B. 24 hours

C. 30 days

D. 60 days

Answer:

C. 30 days

Notes:

"C" is correct, LCO action 3.3.15.B.1 is required since LT-1001 has been inoperable for greater than 30 days. The bases for this LCO action states the report is due within 30 days.
"A" is incorrect but plausible if the candidate incorrectly concludes that the report due time is similar to some LCO action time requirements.
"B" is incorrect but plausible if the candidate incorrectly concludes that the report due time is similar to some LCO action time requirements.
"D" is incorrect but plausible if the candidate incorrectly concludes that the report due time is twice that of the actual time.

This is SRO level since it involves 10CFR55.43(b)(2), application of Technical Specifications.

This question meet the K/A since it involves a Pressurizer Level channel (LT-1001) used in the Pzr Level Control system and it involves applicatlion of Tehcnical Specifications.

References:

Technical Specifications 3.3.15 and bases

History:

New question for 2016 SRO exam

ACTIONS (continued)

A.1

When one or more Functions have one required channel inoperable, the inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience. This takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

Required Action B.1 specifies initiation of actions to prepare and submit a Special Report to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. The Special Report is to be submitted in accordance with 10 CFR 50.4 within 30 days of entering Condition B. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. The Completion Time of "Immediately" for Required Action B.1 identifies the start of the "clock" for submittal of the Special Report. Condition B is modified by a Note requiring Required Action B.1 to be completed whenever the Condition is entered. The Note ensures the requirement to prepare and submit the report is completed. Restoration alone per Required Action A.1 after the initial Completion Time of 30 days does not alleviate the need to report the extended inoperability to the NRC.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation action operation and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the probability that the PAM Function will be unavailable should an accident occur.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.15-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action and associated Completion Time of Condition C, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

3.3 INSTRUMENTATION

3.3.15 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.15 The PAM instrumentation for each Function in Table 3.3.15-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to prepare and submit a Special Report.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.15-1 for the channel.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------
These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.15.2	---------------NOTE--------------- Neutron detectors are excluded from CHANNEL CALIBRATION. ------------------------------------ Perform CHANNEL CALIBRATION.	 18 months

Table 3.3.15-1
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Wide Range Neutron Flux	2	E
2. RCS Hot Leg Temperature	2	E
3. RCS Hot Leg Level	2	F
4. RCS Pressure (Wide Range)	2	E
5. Reactor Vessel Water Level	2	F
6. Reactor Building Water Level (Wide Range)	2	E
7. Reactor Building Pressure (Wide Range)	2	E
8. Penetration Flow Path Automatic Reactor Building Isolation Valve Position	2 per penetration flow path[(a)(b)]	E
9. Reactor Building Area Radiation (High Range)	2	F
10. Deleted		
11. Pressurizer Level	2	E
12. a. SG "A" Water Level – Low Range	2	E
b. SG "B" Water Level – Low Range	2	E
c. SG "A" Water Level – High Range	2	E
d. SG "B" Water Level – High Range	2	E
13. a. SG "A" Pressure	2	E
b. SG "B" Pressure	2	E
14. Condensate Storage Tank Level	2	E
15. Borated Water Storage Tank Level	2	E
16. Core Exit Temperature (CETs per quadrant)	2	E
17. a. Emergency Feedwater Flow to SG "A"	2	E
b. Emergency Feedwater Flow to SG "B"	2	E
18. High Pressure Injection Flow	2	E
19. Low Pressure Injection Flow	2	E
20. Reactor Building Spray Flow	2	E

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1048 **Rev:** 0 **Rev Date:** 3/22/16 **Source:** New **Originator:** J. Cork

TUOI: A1LP-FUEL-FHPRO **Objective:** 1 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 034 **System Title:** Fuel Handling Equipment

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Mispositioned fuel element.

K/A Number: A2.03 **CFR Reference:** 43.7

Tier: 2	**RO Imp:** 3.3	**RO Select:** No	**Difficulty:** 3
Group: 2	**SRO Imp:** 4.0	**SRO Select:** Yes	**Taxonomy:** Ap

Question: **RO:** **SRO:** 92

Given:
- Unit 1 is in Mode 6 with core reload in progress
- You are the SRO In Charge of Fuel Handling
- You just reported to control room communicator a fuel assembly is inserted into core location G6.
- ATC reports to control room communicator that Source Range count rate is increasing continuously.

Which of the following actions are procedurally required for the above conditions?

A. Commence Attachment K "Setting Containment Closure" of 1015.002, Decay Heat Removal and LTOP System Control.

B. Notify Radiation Protection to monitor Main Fuel Bridge area with beta sensitive monitoring equipment.

C. Direct fuel handlers to remove the last assembly inserted into the core.

D. Make announcement over plant PA system to evacuate the Rx Bldg and activate RB evacuation alarm.

Answer:

C. Direct fuel handlers to remove the last assembly inserted into the core.

Notes:

"C" is correct per 1502.004, step 8.15.8, the last assembly inserted into the core or replacing the last removed control rod when source range count rate rises unexpectedly.
"A" is incorrect, this action is not required for this situation but is required for a damaged fuel assembly per 1203.042, Refueling Abnormal Operations, Section 1, Fuel Handling Accident.
"B" is incorrect, this action is not required for this situation but is required for a damaged fuel assembly per 1203.042, Refueling Abnormal Operations, Section 1, Fuel Handling Accident.
"D" is incorrect, this action is not required for this situation but is required for a damaged fuel assembly per 1203.042, Refueling Abnormal Operations, Section 1, Fuel Handling Accident.

This is an SRO level question as it meets 10 CFR 55.43(b)(7), fuel handling facilities and procedures. It is not RO level since this specific action is an SRO In Charge of Fuel Handling responsibilty iin 1502.004.

This question meets the K/A as it has the candidate assess the impact of a mispositioned fuel assembly and must choose one of the choices, all of which are procedural steps. The specific procedure references are NOT included in these choices since three of the four come from the same procedure and would reduce their plausibility.

References:

1502.004, Control of Unit 1 Refueling

History:

New SRO question for 2016 exam.

CAUTION

If the count rate on the source range detectors, or auxiliary incore detectors (if used), rises unexpectedly (sustained doubling or continuously increasing count rate following completion of a core geometry change), following shall be performed:

- The SRO in charge should consider removing the last inserted fuel assembly or replacing the last removed control rod, as applicable.
- Obtain a RCS boron sample and confirm refueling boron concentration requirements remain satisfied.
- Core alterations shall not proceed until the cause is found and resumption is approved by the SRO in Charge of Fuel Handling and Reactor Engineering.

8.15.8 IF the count rate on the source range detectors OR auxiliary incore detectors (if used) rises unexpectedly, THEN perform one of the following:

- IF count rate continues to rise unexpectedly, THEN the SRO in charge should consider one of the following as applicable:
  - Replacing the last removed control rod from the core
  - Removing the last inserted fuel assembly
- ~~IF count rate stabilizes,~~ THEN the SRO in Charge of Fuel Handling may give the Bridge Operator permission to continue with the following step.

A. Record neutron count rates in the approved fuel shuffle sequence.

8.15.9 IF a control component has been removed from the core by itself, THEN verify neutron count rates have been recorded in the approved fuel shuffle sequence.

8.16 Perform the following to insert a fuel assembly and/or control component:

8.16.1 Shoe horns may be moved and placed at the discretion of the SRO in Charge of Fuel Handling or the Reactor Engineer.

8.16.2 Use 3-part communication between the Bridge Operator and the Control Room Communicator to verify the proper location as specified in the "Nuclear Fuel Transfer Report" (form 1022.012C) of Storage, Control & Accountability of Nuclear Fuel (1022.012).

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1053 **Rev:** 3 **Rev Date:** 7/15/16 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-AFIRE **Objective:** 6 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 086 **System Title:** Fire Protection

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure to actuate the FPS when required, resulting in fire damage.

K/A Number: A2.04 **CFR Reference:** 43.5

Tier:	2	**RO Imp:**	3.3	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 93

The plant is at 100% power when the "FIRE" alarm comes in.

The CBOT checks the C463 panels and reports a fire alarm is indicated in the Lower South Electrical Penetration Room (LSEPR), Zone 105-T.

The investigating Inside AO reports the LSEPR deluge valve did NOT actuate and can NOT be manually actuated.

The Inside AO also reports the fire as severe.

Which of the following procedures should be transitioned to and will contain actions that will allow the control room staff to quickly mitigate the specific consequences of components damaged by fire in this area?

A. 1203.009, Fire Protection System Annunciator Corrective Action

B. 1203.049, Fires in Areas Affecting Safe Shutdown

C. 2203.034, Fire or Explosion

D. ANO Pre-Fire Plan for Zone 105-T

Answer:

B. 1203.049, Fires in Areas Affecting Safe Shutdown

Notes:

"B" is correct, starting with the annunciator corrective action (1203.009), the CRS will transition to 2203.034 after the fire is confirmed, and then transition to 1203.049 from section 2 of 2203.034. The LSEPR (105-T) will be listed in 1203.049 and contains specific actions for a fire in this area.
"A" is incorrect, this distracter is plausible in that it will be used to respond to the annunciator and will contain direction to actuate the deluge but it does not contain specific actions for affected components in this area.
"C" iis incorrect, this distracter is plausible in that it will be used to dispatch the fire brigade but it will direct the user to go to 1203.049.
"D" is incorrect, this distracter is plausible in that it will be used by the fire brigade to respond to the fire but it does not contain specific actions for affected components in this area since it is a 1203.049 area.

This question is SRO only since it meets 10CFR55.43(b)(5): the question requires the candidate to evaluate the conditions given and to select the appropriate procedure and action within that procedure which would assist in mitigating the event.

This question meets the K/A since it requires the candidate to assess the malfunction of the LSER deluge valve and select the procedure containing actions which will assist in mitigating the malfunction.

Added "specific" prior to "consequences" in stem at request of NRC examiner. JWC 7/15/16

References:

1203.049, Fires in Areas Affecting Safe Shutdown

History:

This is a modification of QID 014, last used on 2007 SRO exam.
Selected for 2016 SRO exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0014 **Rev:** 1 **Rev Date:** 12/4/06 **Source:** Direct **Originator:** Cork/Possage

TUOI: A1LP-RO-AFIRE **Objective:** 6 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 086 **System Title:** Fire Protection System

Description: Knowledge of fire in the plant procedure.

K/A Number: 2.4.27 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 2 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** K

Question: **RO:** **SRO:**

A fire watch reported a severe fire in the Lower South Electrical Equipment Room and the fire brigade has been dispatched.

What guidance will allow the control room staff to quickly mitigate the consequences of potentially affected components?

A. Go to procedure 1203.049, Fires in Areas Affecting Safe Shutdown.

PARENT

B. Go to procedure 1107.001, Electrical System Operations, breaker alignment attachments.

C. Determine affected components from the Fire Zone drawings maintained in the control room.

D. Go to the ANO Pre-Fire Plan for the affected fire zone for a listing of affected components.

Answer:

A. Go to procedure 1203.049, Fires in Areas Affecting Safe Shutdown.

Notes:

2203.034, Smoke Fire or Explosion, directs operators to use the ANO Pre-Fire Plan to determine "Affected Components of Interest", for areas NOT affecting safe shutdown but to go to 1203.049 for those that are, therefore (a) is the correct response.

Answers (b), [c] and (d) are incorrect because they describe various options that are available and could be used by operators in conjunction with the 2203.034, but do not provide a listing that operators could use to "quickly mitigate" the consequences affected components in safe shutdown areas.

References:

2203.034, Chg. 007-00-0
1203.049, Chg. 003-00-0

History:

Developed for 1998 RO/SRO Exam.
Selected for use in 2002 RO/SRO exam.
Modified for use in 2007 SRO Exam.

Distracter

Location: C19

Device and Setpoint:

Actuation in a zone monitored by Pyrotronics C463-1 thru C463-3 or Notifier C463

FIRE

Alarm: K12-A1

1.0 OPERATOR ACTIONS

1. Check AND acknowledge red alarm LEDs on both Pyrotronics and Notifier C463 panels per Exhibit A of this procedure.

CAUTION

In a post-LOCA condition, a charcoal filter heat-detector fire alarm from the following filters could be caused by the filter being exposed to iodine and cause high radiation:

- Control Room Return and Outside Air Supply Charcoal Filter (VFC-2)
- Outside Air Supply Charcoal Filter (VFC-2A)
- Pent Rm Vent Fan Charcoal Filters (VFC-5A or VFC-5B)

2. Dispatch operator to affected zone to investigate AND perform the following as applicable:

A. IF alarm is in the RB, THEN perform the following:

1) IF RB is closed, THEN perform the following:

a. Consider monitoring the following to attempt to validate the alarm:

- RCP oil levels for abnormal reduction
- RB temperatures for abnormal rise
- RCP stator winding temperatures for abnormal rise
- RCP bearing temperatures for abnormal rise
- Change in RCS leakrate
- RB sump level indicators
- Other coincidental event which could be attributed to a fire in the RB (e.g. breakers opening)

b. Inform RP that Reactor Bldg entry is required per "Job Coverage for Reactor Building Power Entries" Attachment 9 of Unit 1 Off-Normal Operations (1601.307).

(Alarm in the RB, continued next page)

NOTE

RB fire extinguishers are only installed during periods of shutdown maintenance.

c. Perform Lighting and Miscellaneous Electrical Distribution (1107.005), Restoration of Reactor Building Lighting for Building Entry, Attachment 3.

2) IF RB is open,
THEN perform one of the following:

- Attempt to contact RP, RB Coordinator or other personnel in RB to investigate AND report.
- Dispatch an operator to the RB to check for fire.

B. IF alarm from protectowire from Corridor 98 or Cable Spreading Room,
THEN perform the following:

NOTE

- Protectowire Panel (C465) is red cabinet North of X-3 Xfmr.
- Protectowire zones 1 thru 6 are located in cable spreading room. Zones 1 and 2 are cross-zoned and cover the same area within the room.
- Protectowire zones 7 thru 10 are located in corridor 98 fire zone.

1) Dispatch operator to panel C465 to check alarming zone (meter at ~45 milliamps).

2) IF it is desired to silence alarm,
THEN perform the following:

a. Open C465 with a large standard screwdriver.

b. Use appropriate personal electrical safety equipment for 120VAC.

c. Place the alarming zone's Alarm (left) silence toggle in UP position (trouble alarm will sound).

d. Place the alarming zone's Trouble (right) silence toggle in UP position.

e. Close C465.

C. IF alarm from Zone 67-U Hot Lab (B5-1U),
THEN check for smoke detector actuation in all of the following rooms:

- Radiation chemistry lab
- Count room
- Sample room

D. IF alarm from Zone 160-B Computer Room False Floor/Computer Cabinets (B5-1L),
THEN check for smoke detector actuation in the following areas:

- High Sensitivity Smoke Detector (HSSD) (receives signal from computer cabinets and computer room.)
- Computer room's false floor

NOTE
When smoke detector is installed, Control Room Cabinet C20 has no red lamp for actuation indication. (Ref EC47629)

E. IF alarm from Zone 129-F Control Room Cabinets (A4-8L),
THEN check back of each main control room panel for red lamp to determine which panel has the actuated detector.

3. IF fire is confirmed,
THEN perform the following:

A. IF fire is in the RB,
THEN perform the following:

N/A

1. IF in Modes 1 or 2
AND CRS/SM concurs,
THEN trip the reactor and perform Reactor Trip (1202.001) in conjunction with this procedure.

2. Open the following valves:

- Fire Water to RB Supply valve (CV-5611)
- Fire Water to RB Supply valve (CV-5612)

3. Notify Shift Manager to implement Emergency Action Level Classification (1903.010).

B. IF fire is in Corridor 98,
THEN actuate the Corridor 98 deluge.
IF fire is in the battery rooms or in D01/D02 rooms only,
THEN Do NOT actuate Corridor 98 deluge.

N/A

> **NOTE**
> If the sprinklers have fusible heads, fire water flow will not occur until the heads melt.

n/A

C. IF the affected area has a pre-action, deluge or halon suppression system,
THEN verify the associated UAV or halon system has tripped. For any system that is capable, may trip the system at C463.

1) WHEN halon system actuated from C463 has discharged,
THEN return the trip switch to NORMAL (up) position.

D. IF fire is in a charcoal filter,
THEN stop the associated vent fan:

1) IF Zone 160-B VSF-9 Charcoal Filters In Computer Room (VFC-2, VFC-2A),
THEN stop Control Room Emerg Recirc (VSF-9).

2) IF Zone 47-Y Pent Room Vent Fan Charcoal Filters VFC-5A or VFC-5B,
THEN stop Lead or Standby Pent Vent Fan (VEF-38A or VEF-38B) from C26.

E. IF fire is in Control Room or Cable Spreading room,
THEN refer to Alternate Shutdown (1203.002).

1) Make SCBAs available to Control Room personnel.

F. IF fire is confirmed to be a hydrogen fire in the H_2 seal oil system,
THEN trip the reactor and refer to Reactor Trip (1202.001).

1) Vent generator of hydrogen and purge with CO_2 using Generator Hydrogen System (1106.002), "Purging Hydrogen with CO_2 During Emergency Conditions" section.

G. GO TO Fire or Explosion (2203.034).

1. **Determine location of smoke/fire.**

____

NOTE
• Guidance for combating large transformer fires within the Protected Area is provided in Section 3, Large Transformer Fire or Explosion.
• If CONFIRMED fire in the Turbine, Auxiliary or Auxiliary Extension buildings is coincident with a loss of CCW to the RCPs and CCW can NOT be restored, then RCPs must be secured within 5 minutes of loss of CCW using RCP Emergencies, 2203.025.

2. **IF smoke/fire is WITHIN the Protected Area but NOT affecting any large transformers, THEN GO TO Section 2, Protected Area Fire or Explosion.**

NOTE
Large transformers are defined as the Unit Aux Transformer for both units, the main transformers for both units, and all three Startup Transformers.

3. **IF smoke/fire is in a large transformer as defined in the NOTE above, THEN GO TO Section 3, Large Transformer Fire or Explosion.**

4. **IF smoke/fire is OUTSIDE of the Protected Area but within the Owner Controlled Area, THEN GO TO Section 4, Owner Controlled Area Fire or Explosion.**

Distractor

NOTE

A Severe Fire exists in an area when ANY of the following conditions are present:

- Smoke in area prevents assessing status of fire
- Door/room is hot preventing access to the area
- Fire in cable tray affecting several cables
- Fire in 4160v bus or 480v load center

*13. **IF fire is or becomes severe on Unit 1**
AND in any of the following safety-related areas,

LOCATION	FIRE AREA
Steam Pipe Room (Penthouse)	**B-1@170-Z**
Boric Acid Add Tank/ Pump RM, Respirator Storage Room Controlled Access, UNEPR, Hot Mechanic Shop De-con Room	**B1@NAB** 120-E, 125-E, 128-E, 149-E, 79-U
Condensate Demineralizer Room	**B1@73-W** 73-W
Remainder of Zone 197-X (excluding ICW Cooler (E-28A/B/C) and FW Heaters (E-1A/B) Area) Boiler Room, Dirty/clean Lube Oil Storage Tank, Computer Room, Reactor Bldg Purge Rm, Vent Equipment Area 404', Spent Fuel Pool Area	**B1@Balance** 197-X, 157-B, 159-B, 160-B, 161-B, 163-B, 167-B, 168-B, 175-CC, 187-DD, 75-AA, 78-BB
Radwaste Processing Room, Purification Demineralizer Room, Pipe Room, Emergency Feedwater Pump Room, Penetration Ventilation Room, Lower North Piping Penetration Room	**C** 20-Y, 31-Y, 34-Y, 38-Y, 47-Y, 53-Y **(335' Aux Building)**
North Emergency Diesel Room	**D** 86-G
South Switchgear Room	**E** 100-N
South Battery & DC Equipment RM	**F** 110-L
South Emergency Diesel Room	**H** 87-H
Corridor 98	**I-1** 98-J
North Switchgear Room (A4)	**I-2**, 99-M

LOCATION	FIRE AREA
Pipe-way Room (under ICW coolers)	**B-1@40-Y**
Part of Zone 197-X - ICW Cooler (E-28A/B/C) and FW Heaters (E-1A/B) Area only.	**B-1@197-X**
Tendon Gallery Access Room, West Decay Heat Removal Pump, General Access Room	**B-7** 12-EE, 14-EE, 4-EE
Electrical Equipment Room, Lower South Electrical Penetration, Upper South Electrical Penetration, Lower South Piping Penetration, Compressor Room, Upper South Piping Penetration	**B-8** 104-S, 105-T, 144-D 46-Y, 76-W, 77-V
Lab & Demineralizer Access RM, Reactor Coolant Makeup Tank RM, Communications RM, General Access El. 354' & Stairway	**B-9** 67-U, 68-P, 88-Q, 89-P
Lower North Electrical Penetration Room	**I-3**, 112-I
North Side Containment Building	**J North**, 32-K
South Side Containment Building	**J South,** 33-K
Manhole 1MH-03 or 1MH-05, Between Aux Bldg and Intake Structure	**MH03 & MH05**
Manhole 1MH-06 and 1MH-04 Between Aux Bldg and Intake Structure	**MH04 & MH06**
North Battery Room	**O**, 95-O

THEN perform the following:

- **Perform FIRES IN AREAS AFFECTING SAFE SHUTDOWN (1203.049).**
- **IF Unit Shutdown required by 1203.049,**
  THEN consider turning over this procedure to Unit 2.

- **IF** Fire or Explosion (2203.034) has directed entry to this procedure
  AND severe fire confirmed in ANY of the following areas,
  THEN perform appropriate actions for that zone:

Fire Area	Fire Zone	Page #	TAB
B1@170-Z	170-Z	4	1
B1@40-Y	40-Y	16	2
B-1@NAB	120-E, 125-E, 128-E, 149-E, 79-U	24	3
B-1@73-W	73-W	41	4
B-1@197-X	Part of Zone 197-X - ICW Cooler (E-28A/B/C) and FW Heaters (E-1A/B) Area only.	56	5
B1@Balance	197-X (excluding ICW Cooler (E-28A/B/C) and FW Heaters (E-1A/B) Area), 157-B, 159-B, 160-B, 161-B, 163-B, 167-B, 168-B, 175-CC, 187-DD, 75-AA, 78-BB	69	6
B-7	12-EE, 14-EE, 4-EE	85	7
B-8	46-Y, 77-V	88	8
B-8	104-S, 105-T, 144-D, 76-W,	99	9
B-9	67-U, 68-P, 88-Q, 89-P	114	10
C	20-Y, 31-Y, 34-Y, 38-Y, 47-Y, 53-Y	124	11
D	86-G	141	12
E	100-N	143	13
F	110-L	158	14
H	87-H	173	15
I-1	98-J	175	16
I-2	99-M	191	17
I-3	112-I	207	18
J North	32-K	222	19
J South	33-K	230	20
MH03/ MH05	MH-03 or MH05	238	21
MH04/ MH06	MH-04 or MH06	249	22
O	95-O	250	23

> **NOTE**
> - Fire may cause spurious actuation of valves/motors and loss of status indication.
> - The following are sub-sections within the tabbed fire zone sections in the Pre-Fire Plan.
>   - A. "Affected Components of Interest" is a list of safe shutdown equipment and cabling exposed to the fire.
>   - B. "Available Safe Shutdown Instrumentation" is a list of instrumentation that should remain unaffected by a fire in that zone. The list does not exclude the use of other instrumentation, if available.

27. IF Fires in Areas Affecting Safe Shutdown (1203.049, 2203.049) does NOT apply, THEN using applicable tabbed section(s) of the Pre-Fire Plan, adapt plant control and fire fighting tactics for the situation.

Distracter

A. IF a CV operates spuriously as a result of fire,
THEN place in desired position, de-energize and operate manually.

B. IF fire is of sufficient magnitude to cause component failure,
THEN consider the following actions:

- Mark or flag instruments that are suspect due to cabling in the fire zone.

> **NOTE**
> De-energization to fail-safe position prevents component failure to least desired position.

- Place vital affected components in desired position and de-energize.
- De-energize affected switchgear, including control power and have Dispatcher open applicable off-site feeder breakers.

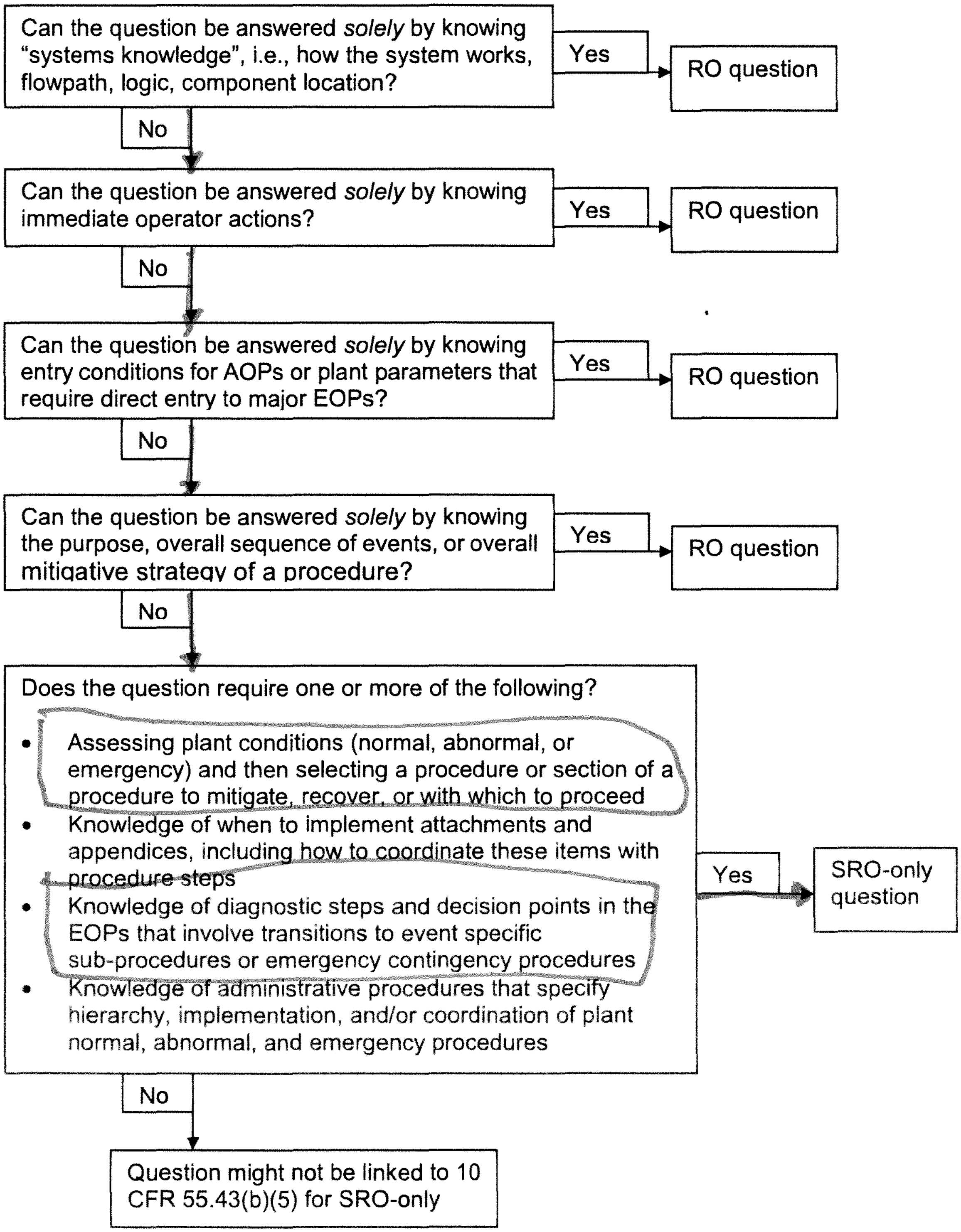
***28. IF fire affects ANY of the following,
THEN consider plant shutdown and cooldown:**

- Offsite power availability
- DG, DG rooms or DG support equipment
- Actual or potential loss of a train of ESF equipment
- Ability of secondary systems to support Reactor operations

29. Contact Emergency Medical Team and notify of the status of the following, as applicable:

- Fire
- Injured person(s)

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



SRO

Tier 3

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1055 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** Bank **Originator:** Cork

TUOI: ASLP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

K/A Number: 2.1.5 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	C

Question: RO: SRO: 94

The plant is at 100% power on New Year's Eve night shift.
The on-duty CRS has a heart attack and must be transported to St. Mary's at 0530.

What is the LATEST time at which a replacement CRS must be in the Control Room to preclude a violation of Technical Specifications?

A. 0830

B. 0730

C. 0630

D. 0530

Answer:

B. 0730

Notes:

Answer "B" is the correct answer since the maximum time the shift can be below the minimum complement is two hours.
Answer "A" is incorrect but plausible if the candidate doesn't recall the time requirement and believes the replacement must be in the control room within 3 hours.
Answer "C" is incorrect but plausible if the candidate doesn't recall the time requirement and believes the replacement must be in the control room within 1 hour.
Answer "D" is incorrect but plausible if the candidate doesn't recall the time requirement and believes the replacement must be immediate.

This is an SRO level question as it meets 10CFR55.43(b)(2), it requires knowledge of Technical Specification staffing requirements not expected of Ros and has an SRO specific learning objective.

This question matches the K/A as the candidate must be able to recall shift staffing requirements.

References:

Technical Specification 5.2.c

History:

Revised QID 885 for 2016 SRO exam (still bank).
Revised question QID 885 (last used in the 2014 SRO exam) by changing time in stem from 0430 to 0530 thereby making 0730 the correct answer (vs. 0600 in QID 885"A"). Revised all answer choices. Made choices highest to lowest vs. lowest to highest.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0885 **Rev:** 0 **Rev Date:** 9/4/14 **Source:** Modified **Originator:** J. Cork

TUOI: ASLP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

K/A Number: 2.1.5 **CFR Reference:** 41.10 / 43.5 / 45.12

Tier:	3	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** **SRO:**

The plant is at 100% power on New Year's Eve night shift.
The on-duty CRS has a heart attack and must be transported to St. Mary's at 0430.

What is the LATEST time at which a replacement CRS must be in the Control Room BEFORE Technical Specifications are violated?

PARENT

A. 0400

B. 0500

C. 0600

D. 0700

Answer:

C. 0600

Notes:

Answer [C] is the correct answer since the maximum time the shift can be below the minimum complement is two hours.
Answers [A], [B], [D] are one hour increments around the correct answer.

Modified question #407 by changing time in stem from 0210 to 0310 thereby making "B" the correct answer (vs. "A").

References:

Technical Specifications 5.2.2 c

History:

Modified QID 407 for 2014 SRO Exam.

5.0 ADMINSTRATIVE CONTROLS

5.2 Organization

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) for one unit, one control room, and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. The operations manager or assistant operations manager shall hold an SRO license.

f. When in MODES 1, 2, 3, or 4, an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operations of the unit. This individual shall meet the qualifications specified by ANSI/ANS 3.1-1993 as endorsed by RG 1.8, Rev. 3, 2000.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0846 **Rev:** 1 **Rev Date:** 7/22/16 **Source:** Modified **Originator:** D. Thompson

TUOI: A1LP-SRO-FH **Objective:** 6 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledge and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of the fuel-handling responsibilities of the SRO.

K/A Number: 2.1.35 **CFR Reference:** 43.7

Tier:	3	**RO Imp:**	2.2	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	K

Question: RO: SRO: 95

Given:
- Unit 1 refueling is in progress.
- Due to difficulties inserting a fuel assembly into the core, the Bridge Operator requests an alteration in the fuel load sequence.

In order to change the fuel shuffle sequence the SRO in Charge of Fuel Handling and ___________ must approve the change per 1502.004, Control of Unit 1 Refueling.

A. Control Room Supervisor

B. Shift Manager

C. Reactor Engineer

D. Refueling Project Manager

Answer:

C. Reactor Engineer

Notes:

"C" is correct per 1502.004, step 8.11.
"A" is incorrect but plausible as the CRS does have an SRO license and is directly in charge of control room operations.
"B" is incorrect but plausible as the Shift Manager is the the person responsible for shift operations.
"D" is incorrect but plausible as this person is the the most senior individual in the Refueling Project.

This questions is SRO level because it involves fuel handling facilities and procedures, i.e., meets 10CFR55.43(b)(7).

This question matches the K/A since it requires knowledge of the fuel handling responsibility of who is requires to alter the fuel load sequence.

References:

1502.004, Control of Unit 1 Refueling

History:

Modified QID 250 for 2011 SRO Exam.
Selected for 2016 SRO exam.

8.10 Concurrently with refueling operations, perform the following attachments:

- Attachment A -- Dilution Prevention Valve Check
- Attachment B -- Refueling Boron, Temperature, and Level Check
- Attachment D -- Refueling Housekeeping and Access Control
- "Refueling Integrity Control" section of Attachment I of Decay Heat Removal And LTOP System Control (1015.002)

{4.3.5}

8.11 IF deviations from the fuel shuffle sequence are needed, THEN obtain approval from both of the following and make changes per the requirements of Storage, Control and Accountability of Nuclear Fuel (1022.012). Approval may be verbal if the change does not affect the destination.

- **The SRO in Charge of Fuel Handling**
- **Reactor Engineer**

8.11.1 The sequence of loading/unloading/shuffling assemblies may be altered for the following reasons:

- To facilitate use of inspection and handling equipment
- To allow incore shuffles to be performed while waiting for fuel assemblies from the SF Pool

8.12 Perform the following;

- Read stable count rate for future reference from at least two indications.
- Record stable count rate for future reference from at least two indications.

8.12.1 IF alternate Source Range NI(s) used, THEN identify alternate instrument(s). N/A the NI(s) not used.

NI-501_______cps NI-502_______cps

_________ ______ cps _________ ______ cps

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0486 **Rev:** 2 **Rev Date:** 7/21/16 **Source:** Modified **Originator:** Cork

TUOI: ASLP-SRO-OPSPR **Objective:** 6 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for conducting special or infrequent tests.

K/A Number: 2.2.7 **CFR Reference:** 43.3

Tier:	3	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	C

Question: RO: SRO: 96

The system engineer for the Makeup and Purification System gives you a test procedure he wants interim approval of.
The test involves entering the Makeup Tank pressure curve's Restricted Operating Region of Exhibit A of 1104.002.
The system engineer believes the curve is too conservative.

Which of the following options are required to ensure the test procedure is in compliance with the licensing basis?

A. Get Asst. Ops Manager approval and implement the test procedure per EN-OP-112, Night and Standing Orders.

B. Approve the procedure after it has a Technical Review per 1000.006, Procedure Control.

C. Use the Procedure Modification process in 1000.006, Procedure Control, to implement the test procedure.

D. Ensure a PAD review per EN-LI-100, Process Applicability Determination, is completed to support the test procedure.

Answer:

D. Ensure a PAD review per EN-LI-100, Process Applicability Determination, is completed to support the test procedure.

Notes:

"D" is correct, any test or experiment that is not described in the SAR is required to have a PAD review which will then intitiate a 50.59 evaluation which in turn will determine if the test procedure conflicts with our licensing basis.
"A" is incorrect, a night order may not be used for a test procedure but plausible since night orders may be used for additional instructions which do NOT conflict with the SAR or existing procedural guidance.
"B" is incorrect, but plausible since all procedure changes require a technical review but this is insufficient scrutiny for a test procedure.
"C" is incorrect, but plausible since the procedure modification process has replaced the old procedure "deviation" section allowing operations to continue when the existing procedure guidance won't allow it but this process may not be used for something like a test procedure.

This question is SRO level since it tests the knowledge identified in 10CFR55.43(b)(3): what is needed to make operating changes to the facility. This question is based upon OE.

This question matches the K/A as it tests the knowledge of performing special or infrequent tests, in this case, the SRO must recognize that any new test must have a PAD review.

References:

1000.006, Procedure Control

History:

New question for 2004 SRO exam.
Used on 2004 SRO Exam.
This question was heavily modified to reflect current procedural guidance for the 2016 SRO exam. The initial conditions are the sam but ALL answer choices were revised. The question stem was revised to be more specific.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0486 **Rev:** 0 **Rev Date:** 10/8/2003 **Source:** Direct **Originator:** Cork/Pullin
TUOI: ASLP-SRO-OPSPR **Objective:** 6 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for conducting tests or experiments not described in the Safety Analysis Report.

K/A Number: 2.2.7 **CFR Reference:** 43.3 / 45.13

Tier:	3	**RO Imp:**	2.0	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	An

Question: **RO:** **SRO:**

The system engineer for the Makeup and Purification System approaches you, the Shift Manager, with a test procedure he wants interim approval of.
The test involves entering the Restricted Operating Region of Exhibit A of 1104.002.
The system engineer believes the curve is too conservative.

Prior to Revision

Which of the following options should you take?

A. Follow the guidance in 1000.006 and approve the procedure using the Interim Approval process.

B. Approve the procedure in accordance with 1000.006 as a Temporary Instruction.

C. Use the procedure as written using the Procedure Deviation process in 1000.006.

D. Approve the procedure using Standard approval process and a 50.59 Evaluation.

Answer:

D. Approve the procedure using Standard approval process and a 50.59 Evaluation.

Notes:

Any test or experiment that may or may not be described in the FSAR is required under the Standard Approval process in 1000.006 to undergo a 50.59 review.
Thus Answer "D" is the only correct choice, all other processes listed cannot be used with the test described.

References:

1000.006, ~~Rev. 050-03-0~~

History:

New developed for 2004 SRO exam.
Used on 2004 SRO Exam.

7.3.9 **COMPLETE** Forms 1000.006A, Procedure Cover Sheet, 1000.006B, Procedure Approval Request, and 1000.006C, Description of Change for inclusion with the draft procedure. Include information on how the change impacts the site and its processes. For example what is the impact on equipment, supplies personnel, training and other change management aspects. (**REFER TO** EN-PL-155, Entergy Nuclear Change Management)

7.3.10 **DETERMINE** if a PAD Review is required. REFER TO Section 8.3 and EN-LI-100, Process Applicability Determination.

7.3.11 **WHEN** the Originator/Procedure Writer completes the draft, **THEN CONSULT** with the Responsible Supervisor/Section leader to determine validation requirements as follows:

NOTE

PM Requirements (PMRQ) Procedures may be validated upon initial use, by observation, provided that the requirement for validation is tracked until completion, and prior approval has been obtained by the Supervisor/Section responsible for PMRQ Procedure development. Any procedure previously issued that has **NOT** been validated, may be validated during performance by observation.

A. Validations shall be performed on any procedure change that adds, removes, or changes a component's position, modifies or changes state of a component, OR alters the sequence of steps or actions.

B. **WHEN** the revision number ends on 0 or 5, **THEN** the entire procedure shall be validated to ensure procedures are reviewed periodically and remain up to date on current standards and expectations with the following exceptions:

- Emergent procedure change to support plant shutdown, startup, or operation.
- Emergent procedure change needed to support maintenance when the plant is in an LCO.
- If the procedure has been validated within the last 6 months.
- If approved by the General Manager, Department Head or Assistant Operations Manager.
- Editorial Change being performed.
- **IF** a validation is **NOT** performed, **THEN** initiate a condition report to determine if a validation is required prior to the next 0 or 5 revision.

8.3 PROCESS APPLICABILITY DETERMINATION (PAD) REVIEWS

8.3.1 PAD Reviews are required in accordance with EN-LI-100 for all procedure revisions, new procedures and procedure deletions except as follows:

- The procedure has been Programmatically Excluded from further PAD Review. If the procedure scope has changed, the procedure will require another PAD Review for Programmatic Exclusion. Procedure deletions will still require a PAD Review.

- The entire scope of the procedure revision meets one or more of the criteria contained in Form 1000.006S, Procedure Changes Not Requiring a Process Applicability Determination (PAD) Review.

8.3.2 **IF** a procedure has been Programmatically Excluded, **THEN INDICATE** on Form 1000.006A, Procedure Cover Sheet that the procedure has been Programmatically Excluded. No future PAD Review documentation is required as long as the scope of the procedure has not changed.

8.3.3 **IF** the entire scope of the procedure revision meets one or more of the criteria of Form 1000.006S, Procedure Changes Not Requiring a Process Applicability Determination (PAD) Review, **THEN COMPLETE** Form 1000.006S and **ATTACH** the form to the procedure change package. No PAD Review is required.

ENTERGY OPERATIONS INCORPORATED
ARKANSAS NUCLEAR ONE

TITLE	DOCUMENT NO.	CHANGE NO.

AFFECTED UNIT		SAFETY-RELATED
☐ UNIT 1 ☐ UNIT 2	☐ PROCEDURE ☐ ELECTRONIC DOCUMENT	☐ YES ☐ NO

TYPE OF CHANGE
☐ NEW ☐ DELETION ☐ REVISION

DOES THIS DOCUMENT:

1. Supersede or replace another procedure? (If YES, complete Form 1000.006B for deleted procedure)	☐ YES	☐ NO
2. Alter or delete an existing regulatory commitment? (If YES, coordinate with Licensing before implementing)	☐ YES	☐ NO
3. Require a PAD Review per Form 1000.006S? (If NO, attach completed Form 1000.006S)	☐ YES	☐ NO
4. Changes Surveillance Test Program (i.e.; Technical Specifications, NRC Commitment, surveillance activity)? (If YES, complete Form 1000.009A)	☐ YES	☐ NO
5. Create an Intent Change? (If YES, Standard Approval Process required)	☐ YES	☐ NO
6. Implement or change IPTE requirements? (If YES, OSRC review required)	☐ YES	☐ NO
7. Implement a Temporary Modification? (If YES, OSRC review required)	☐ YES	☐ NO
8. Implement a change that could affect reactivity per COPD-030, Att 9.3, Section 1? (If YES, then perform required actions per 1000.006)	☐ YES	☐ NO
Was the Master Electronic File used as the source document?	☐ YES	☐ NO

Entergy

STANDARD APPROVAL PROCESS

ORIGINATOR SIGNATURE: (Includes review of Att 6)	DATE:
Print and Sign Name:	PHONE #:
INDEPENDENT REVIEWER: Print and Sign Name:	DATE:
ENGINEERING: Print and Sign Name:	DATE:
CODE PROGRAMS - NDE: Print and Sign Name:	DATE:
UNIT SURVEILLANCE COORDINATOR: Print and Sign Name:	DATE:
SECTION LEADER: Print and Sign Name:	DATE:
QUALITY ASSURANCE: Print and Sign Name:	DATE:
OTHER SECTION LEADERS: Print and Sign Name:	DATE:
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OTHER SECTION LEADERS: Print and Sign Name:	DATE:

OSRC CHAIRMAN / TECHNICAL REVIEWER: Print and Sign Name:	DATE:
FINAL APPROVAL: Print and Sign Name:	DATE:
REQUIRED EFFECTIVE DATE:	

FORM TITLE:	FORM NO.	CHANGE NO.
PROCEDURE APPROVAL REQUEST	1000.006B	064

E-DOC TITLE:		E-DOC NO.	CHANGE NO.
Entergy	PROCEDURE CHANGES NOT REQUIRING A PROCESS APPLICABILITY DETERMINATION (PAD) REVIEW	1000.006S	058

Procedure No. ____________________ Revision No. ________________

Title ____

Originator ______________________________ Date _____________

The following are types of <u>procedure</u> changes that do not require a PAD Review. Documentation is established by indicating on the appropriate 1000.006 form that a PAD Review is not required and this form will be attached to the procedure change package. It is not necessary to complete any EN-LI-100 forms if a PAD Review is not required.

NOTE

All other changes, not programmatically excluded, <u>require</u> a PAD Review per EN-LI-100.

☐	1	Correcting grammar or spelling errors.
☐	2	Corrections to the numbering of steps, sections, forms, attachments, exhibits or pages without changing sequence.
☐	3	Addition/modification of text to improve clarity without changing process or intent.
☐	4	Correcting reference to step or section numbering (alpha/numeric) within the procedure.
☐	5	Correcting references to procedure titles, numbers, sections or steps of another procedure.
☐	6	Correcting <u>obvious</u> clerical/typographical errors that reference incorrect equipment/component designations/stock numbers (letters or numbers).
☐	7	Correcting references to equipment location, room number, general direction (north, south, etc.), elevation, or cabinet number.
☐	8	Cosmetic changes (i.e., affecting appearance only) that do not affect process or intent.
☐	9	Changing previously approved organization or individual titles.
☐	10	Adding/correcting references in the reference section or in the body of the procedure <u>or</u> adding a procedural step that references the use of another procedure.
☐	11	Incorporating information from approved Engineering Processes as long as the process has received a PAD Review in accordance with EN-LI-100, and the PAD/50.59 review(s) encompasses the changes being made. Reference and attach PAD/50.59 Review(s) for Engineering Process used: ____________________.
☐	12	Adding steps for gathering or disseminating information, e.g., recording data, making plant announcements, making calls to get information, etc.
☐	13	Adding steps to close configuration control loops where steps were obviously intended to exist.
☐	14	Adding, modifying or deleting steps or information in a procedure that have been evaluated or incorporated into another procedure.
☐	15	Adding the initial level of use designator to a procedure, changing the format or location of the level of use designator in accordance with approved procedures or changing the level of use designation.
☐	16	Adding, modifying or deleting IPTE requirements.
☐	17	Administrative changes made as part of the reactivity impact program.
☐	18	Adding, modifying or deleting portions of the Inservice Inspection (ISI) and Inservice Testing (IST) Programs that are controlled in accordance with 10CFR50.55a (e.g. changing acceptance criteria values for surveillances, etc.) Engineering process used (ECN, SEP, etc.):___________

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0879 **Rev:** 1 **Rev Date:** 5/16/16 **Source:** Bank **Originator:** NRC Exam Bank

TUOI: ASLP-SRO-MNTC **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

K/A Number: 2.2.17 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	2.6	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

Question: RO: SRO: 97

In accordance with EN-WM-100, Work Request (WR) Generation, Screening and Classification, an approved Work Order Package ______ required for Emergency maintenance. Prior to performing work, authorization to begin the work must be approved at a MINIMUM by the ______ .

A. Is
Shift Manager

B. Is NOT
Shift Manager

C. Is
Work Week Manager

D. Is NOT
Work Week Manager

Answer:

B. is NOT
Shift Manager

Notes:

Answer B is correct per EN-WM-100. Emergency maintenance can be approved by the Shift Manager and a work order is used to document the work performed as soon as practical afterwards.
Answer A is incorrect, this answer is plausible in that it has the proper authority but a work order package is not required prior to the work, however this is the normal (non-emergency) sequence.
Answer C is incorrect, this answer is plausible in that it has the correct sequence for work order preparation but the incorrect approval authority although the Work Week Manager is the ultimate authority for executing work per WN-WM-101 for non-emergency situations.
Answer D is incorrect, this answer has the incorrect authority (although plausible as in the explanation for C) and the incorrect sequence (although plausible in the explanation for A).

Revised question by removing "Priority 1" to avoid the possibility of having two correct answers. Also made the stem two separate sentences.

This is SRO level since it involves work package approval and authorization for plant maintenance, this is part of 10CR55.43(b)(5). Training is NOT given to ROs for this administrative duty.

This meets the K/A since it involves knowledge of the managing maintenance activities, i.e., emergency maintenance, and work prioritization.

References:

EN-WM-100, Work Request Generation, Screening and Classification

History:

Selected for 2014 SRO Exam. (Direct from Crystal River Exam 2011 SRO Question #21, slightly changed to align with ANO)
Selected for 2016 SRO exam

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0879 **Rev:** 0 **Rev Date:** 6/3/14 **Source:** Bank **Originator:** NRC Exam Bank

TUOI: ASLP-SRO-MNTC **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

K/A Number: 2.2.17 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	2.6	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 97

In accordance with EN-WM-100, Work Request (WR) Generation, Screening and Classification, an approved Work Order Package ______ required for Priority 1 (Emergency) maintenance, prior to performing work and authorization to begin the work must be approved at a MINIMUM by the ______ .

A. Is
Shift Manager

B. Is NOT
Shift Manager

C. Is
Work Week Manager

D. Is NOT
Work Week Manager

Prior to Revision

Answer:

B. is NOT
Shift Manager

Notes:

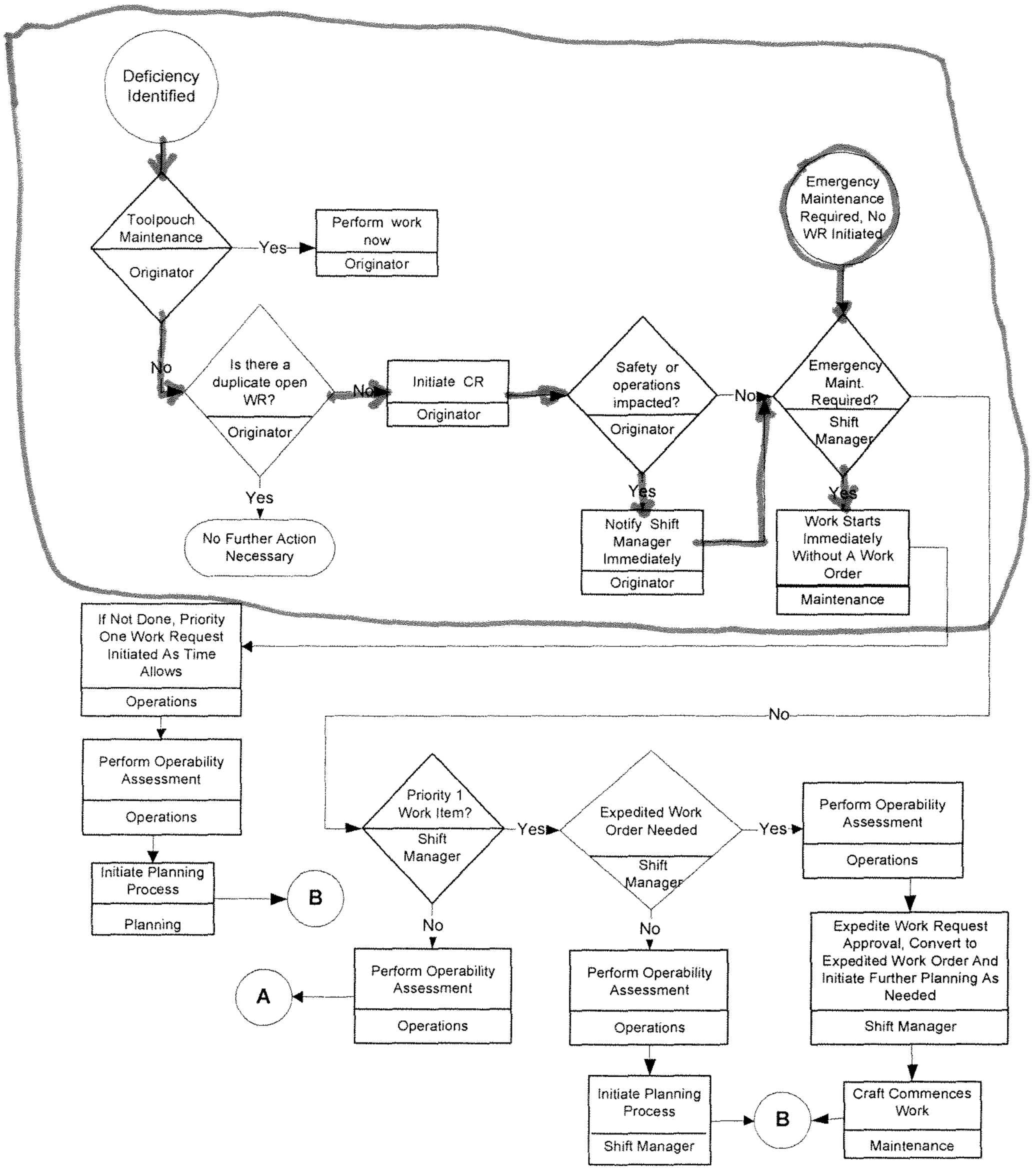
Answer B is correct per EN-WM-100. Emergency maintenance can be approved by the Shift Manager and a work order is used to document the work performed as soon as practical afterwards.
Answer A is incorrect, this answer is plausible in that it has the proper authority but a work order package is not required prior to the work, however this is the normal (non-emergency) sequence.
Answer C is incorrect, this answer is plausible in that it has the correct sequence for work order preparation but the incorrect approval authority although the Work Week Manager is the ultimate authority for executing work per WN-WM-101 for non-emergency situations.
Answer D is incorrect, this answer has the incorrect authority (although plausible as in the explanation for C) and the incorrect sequence (although plausible in the explanation for A).

References:

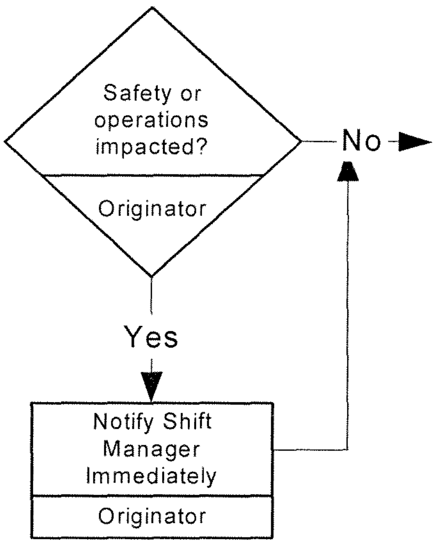
EN-WM-100, Work Request Generation, Screening and Classification

History:

Selected for 2014 SRO Exam. (Direct from Crystal River Exam 2011 SRO Question #21, slightly changed to align with ANO)
Selected for 2016 SRO exam



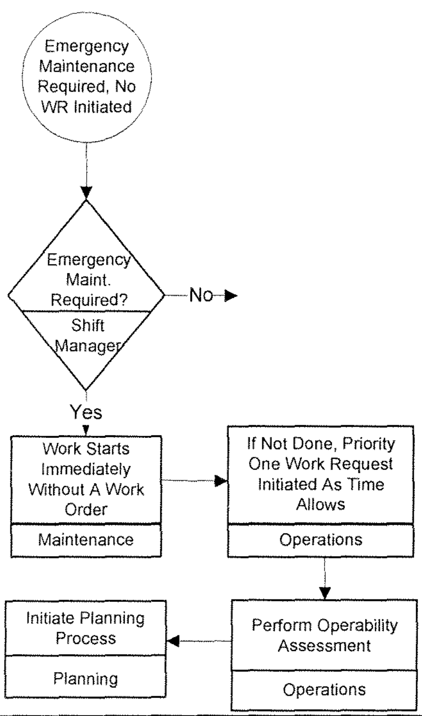
Deficiency Identified
Toolpouch Maintenance
Originator
Yes
Perform work now
Originator
No
Is there a duplicate open WR?
Originator
No
Initiate CR
Originator
Yes
No Further Action Necessary
Safety or operations impacted?
Originator
No
Yes
Notify Shift Manager Immediately
Originator
Emergency Maintenance Required, No WR Initiated
Emergency Maint. Required?
Shift Manager
Yes
Work Starts Immediately Without A Work Order
Maintenance
If Not Done, Priority One Work Request Initiated As Time Allows
Operations
Perform Operability Assessment
Operations
Initiate Planning Process
Planning
B
No
Priority 1 Work Item?
Shift Manager
Yes
Expedited Work Order Needed
Shift Manager
Yes
Perform Operability Assessment
Operations
No
Perform Operability Assessment
Operations
A
No
Perform Operability Assessment
Operations
Expedite Work Request Approval, Convert to Expedited Work Order And Initiate Further Planning As Needed
Shift Manager
Initiate Planning Process
Shift Manager
B
Craft Commences Work
Maintenance



If the deficiency meets any of the following criteria, the Shift Manager shall be notified immediately.

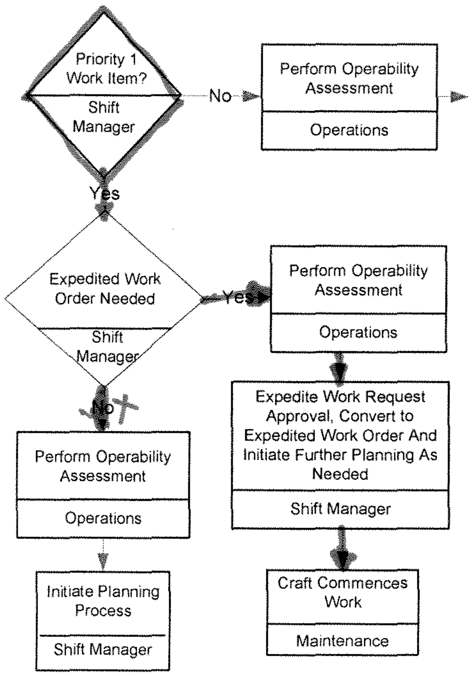
- it is potentially reportable
- it affects the operability or functionality of a safety related system
- it could cause a plant trip or transient
- it presents an immediate threat to personnel or equipment safety

The Shift Manager uses the information provided by the identifier to determine whether work should be started prior to the next Screening meeting.



If plant conditions meet the requirements of Emergency Maintenance, the Shift Manager may direct work activities to begin immediately, prior to the generation of a CR & WR. When time permits an Urgent – Priority 1 WR should be initiated. For Emergency Maintenance, any or all steps in the work control process can be bypassed at the discretion of Shift Manager. The work performed shall be documented at the completion of the work.

The SM ensures Operational Screening is performed for new work requests each shift.



If the Shift Manager (SM) determines that the work process should start immediately, SM prioritizes the WR as Priority 1 and assigns the WR to an implementing department, and performs the screening and classification. The SM contacts the appropriate personnel to initiate the planning of the work package and to begin repairs.

For Priority 1 work, the SM may elect to initiate an expedited work order to allow work to commence in the field prior to completion of detailed work package planning. The work performed under an expedited work order must be that which can be characterized as skill-of-the-craft. An expedited work order is not a routine activity, but may be utilized in situations where an immediate threat is present to personnel or equipment safety, or station availability. The Shift Manager is responsible for assessing the risk of performing work under an expedited work order.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1049 **Rev:** 1 **Rev Date:** 7/8/16 **Source:** Modified **Originator:** J. Cork

TUOI: A1QC-SRO-QUAL **Objective:** 3.23 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 43.4

Tier:	3	**RO Imp:**	3.2	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: RO: SRO: 98

Given:

- A Site Area Emergency has been declared on Unit 1.
- An Emergency Medical Team must enter a 500 REM/hr area to rescue a critically injured employee (they are directed, i.e., not volunteers).

Which of the following is the MAXIMUM time each individual team member can stay in this area and who can authorize a team member to extend this maximum time if they volunteer to do so?

A. 1 minute
Shift Manager

B. 3 minutes
Shift Manager

C. 1 minute
OSC Manager

D. 3 minutes
OSC Manager

Answer:

B. 3 minutes
Shift Manager

Notes:

"B" is correct since the exposure limit to save a life is 25 Rem in 1903.033 and in a 500 REM/hr area this equates to 8.33 REM/minute so the total stay time would be 3 minutes. Authorization to exceed 10CFR20 limits is given by Shift Manager or Emergency Director or Emergency Plant Manger per 1903.033.
"A" is incorrect, but plausible in case the examinee uses the limit of 10 Rem to save valuable equipment, and the responsible person is correct.
"C" is incorrect, but plausible in case the examinee uses the limit of 10 Rem to save valuable equipment, and the responsible person is incorrect.
"D" is incorrect, but plausible since the stay time is correct but the responsible person is incorrect.

This is an SRO level question as it meets 10 CFR 55.43(b)(4), radiation hazards. It is not RO level since this specific knowledge is not part of the initial RO curriculum and has an SRO specific objective.

This question meets the K/A since it is specifically about emergency radiation exposure limits.

The dose rate for the area was changed from 100 R/hr to 500 R/hr which changes the correct answer from 15 minutes to 3 minutes. Changing a condition to make a different choice correct meets the critieria of a modified question per ES-401, D.2.f. Modified other distracters to make them more plausible.

Revised per recommendation of NRC examiner. JWC 7/21/16

References:

1903.033, Protective Action Guidelines for Rescue/Repair & Damage Control Teams

History:

This is a Modified version of QID 120 for use in the 2016 SRO exam.

6.2.4 Immediate Actions

A. **IF** exposure to significant radioiodine concentrations is possible or a General Emergency has been declared, **THEN** refer to procedure 1903.035, "Administration of Potassium Iodide" for guidance.

B. Rescue/repair and damage control teams shall be briefed using Form 1903.033B, "OSC Team Briefing Form". This form serves as an emergency RWP and Work Order. Instructions for conducting reentry team briefings are contained in Attachment 3.

C. Rescue/repair and damage control team members who are not qualified as Advanced Radiation Workers shall be accompanied by a member of the Emergency Radiation Team during initial entry and subsequent reentries into plant areas until radiation areas have been marked. Advanced Radiation Workers must take a radiation detection instrument with them upon reentry.

D. **IF** the situation requires reentry for the purpose of search and rescue,
THEN personnel from the Emergency Medical Team and Emergency Radiation Team shall be assigned to the rescue team.

E. The Shift Manager or OSC Manager shall ensure that briefings are conducted, per Section 6.2.4.B or 6.2.4.F as appropriate, and authorization for exceeding 10CFR20 exposure limits is granted and documented on Form 1903.033A.

F. In the event that the time required for a formal briefing jeopardizes plant equipment or personnel safety, the briefing may be accomplished as the entry is being made subject to the following:

1. The specific exposure limit being authorized is specified.
2. Expected dose rates and stay times are specified.
3. The Shift Manager, Emergency Plant Manager, or Emergency Director has given verbal approval for the activity and authorized exposures above 10CFR20 limits.
4. Forms 1903.033A and B are completed as a follow-up to the emergency response activities.

G. For reentry team electronic dosimeter settings, refer to Attachment 2 of this procedure.

H. Reentry teams should be provided with a copy of form 1903.033D, "OSC Team Observation Report", to record their observations during reentry.

I. A Rescue/Repair and Damage Control Team has been formed. A reentry must be made for: (check one)

- ☐ 1. Protecting valuable property (lower dose not practicable). Planned dose shall not exceed 10 Rem TEDE.
- ☐ 2. Lifesaving or protection of large populations (lower dose not practicable). Planned dose shall not exceed 25 Rem TEDE.
- ☐ 3. >25 Rem TEDE:
  - a. Lifesaving or protection of large populations.
  - b. Only on a voluntary basis to persons fully aware of the risks involved.

II. The individuals listed below have been briefed on the requirements of the task and the guidelines in section 6.1.3. They have been authorized to exceed the dose limits of 10CFR20 if necessary to accomplish this task within the guidelines listed in Section 6.1.3.

NAME (PRINTED)	SIGNATURE **	BADGE NUMBER

III. AUTHORIZATION *
Print & Sign
SM/EPM/ED ______________________ / ______________________ (signed) ______________ (date)

* May be given verbally via telephone, radio, or other means.

** Signifies person has been briefed concerning guidelines for exceeding 10CFR20 dose limits (1903.033A).

cc: Personnel File
Personal Dosimetry Record

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1049 **Rev:** 0 **Rev Date:** 3/23/16 **Source:** Modified **Originator:** J. Cork

TUOI: ASLP-SRO-RADP **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 43.4

Tier:	3	**RO Imp:**	3.2	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** **SRO:** 98

Given:

- A Site Area Emergency has been declared on Unit 1.
- An Emergency Medical Team must enter a 500 REM/hr area to rescue a critically injured employee (they are directed, i.e., not volunteers).

Which of the following is the MAXIMUM time each individual team member can stay in this area?

Revised

A. 1 minutes

B. 3 minutes

C. 15 minutes

D. 30 minutes

Answer:

B. 3 minutes

Notes:

"B" is correct since the exposure limit to save a life is 25 Rem in 1903.033 and in a 500 REM/hr area this equates to 8.33 REM/minute so the total stay time would be 3 minutes.
"A" is incorrect, but plausible in case the examinee uses the limit of 10 Rem to save valuable equipment.
"C" is incorrect, this is the previous correct answer and is left unchanged to eliminate those that simply study old exams.
"D" is incorrect, but plausible in case the examinee made the mathematical error of using 50 REM/hr vs. 500.

This is an SRO level question as it meets 10 CFR 55.43(b)(4), radiation hazards. It is not RO level since this specific knowledge is not part of the initial RO curriculum and this has an SRO specific lesson plan and objective.

This question meets the K/A since it is specifically about emergency radiation exposure limits.

The dose rate for the area was changed from 100 R/hr to 500 R/hr which changes the correct answer from 15 minutes to 3 minutes. Changing a condition to make a different choice correct meets the critieria of a modified question per ES-401, D.2.f. Modified other distracters to make them more plausible.

References:

1903.033

History:

This is a Modified version of QID 120 for use in the 2016 SRO exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0120 **Rev:** 1 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** JCork

TUOI: ASLP-SRO-RADP **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier: 3 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3

Group: G **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** Ap

Question: **RO:** **SRO:**

Given:

- A Site Area Emergency has been declared on Unit 1.
- An Emergency Medical Team member must enter a 100 REM/hr area to rescue a critically injured employee (he is directed, i.e., not a volunteer).

Which of the following is the MAXIMUM time an individual team member can stay in this area?

PARENT

A. 3 minutes

B. 6 minutes

C. 15 minutes

D. 30 minutes

Answer:

C. 15 minutes

Notes:

The limit for life saving is 25 rem TEDE. 100R/hr means a 15 minute stay time, therefore "C" is correct. "A" and "B" are the limits for all acitivities and protecting valuable property, respectively. "D" is just double the correct answer.

References:

1903.033, Chg. 018-01-0

History:

Modified for use in 1998 SRO exam
Modified question from NRC developed SRO exam 2/6/95, no. 94
Modified for use in 2005 SRO exam.

Dose limit* (Rem TEDE)	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Life saving or protection of large populations	Lower dose not practicable
>25	Life saving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved (refer to Attachment 1 of this procedure for health risks).

* Workers performing services during emergencies should limit dose to the lens of the eye to <u>three times</u> the listed value and doses to any other organ (including skin and body extremities) to <u>ten times</u> the listed value.

6.1.4 Rescue/repair and damage control personnel shall perform their duties in the most safe and efficient manner possible. Once their operations have been completed, they shall follow self-monitoring and personnel decontamination procedures as specified by the RAD Coordinator.

6.2 ACTIONS

NOTE

Prompt medical attention shall take precedence over RP procedures for a seriously injured individual.

6.2.1 Emergency Medical Team may enter Radiologically Controlled Areas without SRDs or Alarming Dosimeters during a "Personnel Emergency" as long as an RP Technician is providing radiological instructions and is monitoring dose rates and time in the area.

6.2.2 Personnel selected for the rescue/repair and damage control teams should report to the OSC (unless otherwise instructed) for their briefing.

6.2.3 The rescue/repair and damage control team leader shall function under the direction of the Shift Manager/OSC Manager.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 1088 **Rev:** 1 **Rev Date:** 7/15/16 **Source:** New **Originator:** Cork

TUOI: A1QC-SRO-QUAL **Objective:** EP3.21 **Point Value:** 1

Section: 2.0 **Type:** Generic Kas

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of the emergency plan.

K/A Number: 2.4.29 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	3.1	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	K

Question: RO: SRO: 99

Which of the following MUST be performed if a General Emergency is declared?

A. Offsite evacuation

B. Localized evacuation

C. TSC/OSC evacuation

D. Exclusion area evacuation

Answer:

D. Exclusion area evacuation

Notes:

"D" is correct, per step 6.1.3.A of 1903.030 an exclusion area evacuation shall be initiated if a GE is declared.
"A" is incorrect but plausible since, in most cases, an offiste evacuation will be recommended by the Emergency Director but there are situations where sheltering vs. evacuation will be recommended.
"B" is incorrect but plausible since a localized evacuation is quite possible during a GE event but is not always required.
"C" is incorrect but plausible since a GE could cause an TSC/OSC evacuation but it is not a certainty, IAW 1903.030 these are not normally evacuated in the event of a plant evacuation. They are only evacuated if the radiation levels exceed exceed 2.5 mR/hr or airborne activity >9E-10 µCi/cc.

Knowledge of evacuations during a GE is taught exclusively to the SRO candidates, this is therefore an SRO level question that is linked to 10CFR55.43(b)(5).

This question matches the K/A since it requires knowledge of the emergency plan.

Changed "C" to TSC/OSC per request of NRC examiner. JWC 7/15/16

References:

1903.030, Evacuation

History:

New question for 2016 SRO exam.

5. An uncontrolled toxic gas leak exists (originating either on-site or off-site) and the hazard is not confined to a well-defined area.

B. The decision to evacuate non-essential personnel (including the general public) or retain them on-site should be based on the course of action which presents the minimum risk to personnel. Examples of extenuating conditions that may result in deciding against a plant evacuation are:

1. An ongoing security threat within the protected area. Refer to OP-1203.048 and consult with the Security Shift Supervisor to aid in determining the safest course of action.

2. Inclement weather (e.g., Tornado, high winds, hazardous road conditions may preclude a safe evacuation of plant personnel).

3. Radiological hazards exist. (Determine which action would result in lower dose to nonessential personnel).

4. Other impediments to a plant evacuation (e.g. large fire, damaged access points, debris, etc.)

6.1.3 Exclusion Area Evacuation

A. An exclusion area evacuation shall be initiated if a General Emergency is declared. If a Site Area Emergency is declared, an exclusion area evacuation should be considered (Use 1903.011).

B. An exclusion area evacuation shall be initiated if RP survey results indicate that general area radiation levels exceed 2.5 mR/hr within the Exclusion Area.

C. An exclusion area evacuation shall be initiated if personnel (including the general public) within the Exclusion Area could receive an exposure to a toxic gas (e.g., transportation accident involving truck, rail, or barge).

6.1.4 Offsite Evacuation

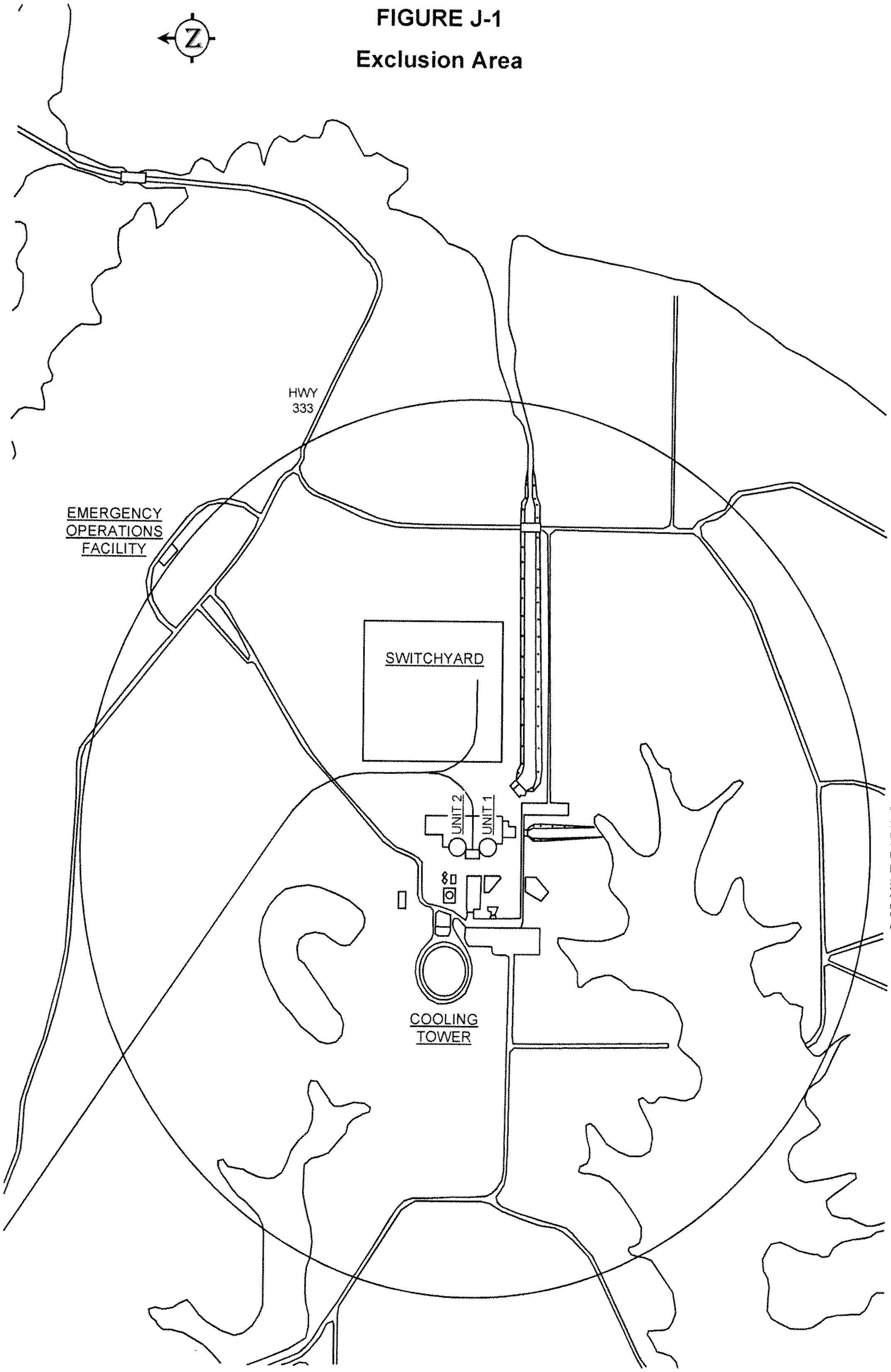
An Offsite Evacuation shall be recommended in accordance with Procedure 1903.011, "Emergency Response/Notifications".

6.1.5 EOF Evacuation

See Procedure 1903.067, "Emergency Response Facility Emergency Operations Facility (EOF)".

FIGURE J-1

Exclusion Area



QID: 0411 **Rev:** 1 **Rev Date:** 5/16/2016 **Source:** Bank **Originator:** E-Plan

TUOI: ASLP-RO EPLAN **Objective:** 7 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

K/A Number: 2.4.30 **CFR Reference:** 43.5

Tier:	3	**RO Imp:**	2.7	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 100

Unit 1 is shutdown for a refueling outage.
A fire was reported at 0844 in the Reactor Building.
It is now 0920 and the fire is still burning.

Based on the above conditions what is the time requirement per 1903.011, Emergency Response/Notifications for notification to the NRC?

A. Notification to the NRC is required within 15 minutes of the declaration of an emergency class and notify the Arkansas Department of Health within 1 hour.

B. Notification to the NRC is required within 30 minutes of the declaration of an emergency class, after notifying the Arkansas Department of Health.

C. Notification to the NRC is required immediately following declaration of an emergency class and notify the Arkansas Department of Health within 1 hour.

D. Notification to the NRC is required immediately following notification of the Arkansas Department of Health and within 1 hour of the declaration of an emergency class.

Answer:

D. Notification to the NRC is required immediately following notification of the Arkansas Department of Health and within 1 hour of the declaration of an emergency class.

Notes:

"D" is correct since this is the procedural requirement.
"A" is incorrect but plausible as it is the reverse of the correct requirement.
"C" is incorrect but plausible since the NRC is notified immediately and it has a one hour requirement, but the sequence is incorrect.
"B" is incorrect but plausible since the sequence is correct and 30 minutes is close to the notification time, but this is not in accordance with 1903.011.

References:

1903.011Y, Emergency Class Initial Notification Message

History:

Modified E-Plan exam bank QID#61 for use in 2001 SRO Exam.
Selected for use in 2002 SRO exam.
Selected for 2010 SRO exam
Repeated for 2011 SRO Exam.

Selected for 2016 SRO exam.

INITIAL RO/SRO EXAM BANK QUESTION DATA
ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0411 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** Repeat **Originator:** E-Plan

TUOI: ASLP-RO EPLAN **Objective:** 7 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

K/A Number: 2.4.30 **CFR Reference:** 41.10 / 43.5 / 45.11

Tier:	3	**RO Imp:**	2.7	**RO Select:**	No	**Difficulty:**	2
Group:	G	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 100

A fire was reported at 0844 in the vicinity of the Old Radwaste Building.
It is now 0920 and the fire is still burning.

Based on the above conditions what is the time requirement for notification to the NRC?

Prior to Revision

A. Notification to the NRC is required within 15 minutes of the declaration of an emergency class.

B. Notification to the NRC is required immediately following notification of the ADH and within 1 hour of the declaration of an emergency class.

C. Notification to the NRC is required immediately following declaration of an emergency class and notify the ADH within 1 hour.

D. Notification to the NRC is required within 4 hours of the declaration of an emergency class.

Answer:

B. Notification to the NRC is required immediately following notification of the ADH and within 1 hour of the declaration of an emergency class.

Notes:

Answer [B] is correct since this is the procedural requirement.
Answer [A], [C], [D] are incorrect, these are not in accordance with 1903.011.

References:

1903.011Y, Emergency Initial Notification Message, Chg 037

History:

Modified E-Plan exam bank QID#61 for use in 2001 SRO Exam.
Selected for use in 2002 SRO exam.
Selected for 2010 SRO exam
Repeated for 2011 SRO Exam.

ACTIONS FOR INITIAL NOTIFICATION

This form is used to notify the NRC, State and local governments of the following:

- Emergency Class Declaration
- Emergency Class Change (Upgrade or Downgrade)
- PAR Change

The Arkansas Department of Health (ADH) and other offsite response organizations **SHALL** be notified within **15 minutes** of any of the above events.

The Nuclear Regulatory Commission (NRC) **SHALL** be notified **immediately** following notification of the ADH and **SHALL NOT** exceed **1 hour** following the declaration of an emergency class.

ERDS must be initiated within 1 hour of the declaration of an **ALERT or higher** emergency class.

NOTE

The material contained within the symbols (*) throughout this form is proprietary or private information.

The Emergency Telephone Directory contains the emergency telephone numbers that you may need to complete this notification.

Computer generated Form 1903.011-Y may be used for notifications. The computer generated form is not an identical copy to the hard copy form, but contains all necessary information.

INSTRUCTIONS

1.0 Complete Initial Notification Message in accordance with Step 1.1 Computerized Notification Method **OR** Step 1.2 Manual Notification Method. (Computerized Notification Method preferred)

1.1 Computerized Notification Method

1.1.1 **IF** the Computerized Notification Method fails while performing notifications,
THEN go to the "Manual Notification Method" Step 1.2.

1.1.2 **IF** not already logged onto the notifications computer,
THEN perform the following:

A. Sign onto the computerized notification system computer using your Entergy logon ID and password. Control Room may use a generic ID and password.

B. Verify your computer is connected to a local or network printer in your area.
[Start]→[Settings]→[Printers and Faxes]

1.1.3 On the desktop double click the "EP Notification" icon **OR** select [Start], [(All) Programs], [EP Notifications], [EP Notifications Version XXXX] to start notification program.

1.1.4 Enter the appropriate data into the data fields for the Initial Notification Message. Use the [Tab] key (preferred) or mouse to navigate through the form. Refer to Emergency Class Notification Instructions page 8 of this form as needed.

1.1.5 **WHEN** the data fields are populated,
THEN press the [Create PDF only] button.

1.1.6 **IF** you receive an error message (i.e. "You have not correctly entered all the required data on Tab..."),
THEN review the form and make corrections.
Go to Step 1.1.5 above.

Question No. 1 **QID:** 0324

QUESTION 1 DELETED PER POST-EXAM COMMENTS. POINT VALUE = 0.

Answer:

A. Cycle ERV as required, this prevents challenges to the PZR safeties.

Question No. 2 **QID:** 1089 **Point Value:** 1

Answer:

C. To prevent possible core uncovery if the RCPs were tripped later.

Question No. 3 **QID:** 0684 **Point Value:** 1

Answer:

C. Override and secure all HPI pumps.

Question No. 4 **QID:** 0183 **Point Value:** 1

Answer:

B. Seal injection control valve (CV-1207) is slowly opened to minimize thermal shock to the RCP seals and prevent damage to seals, independent of normal makeup restoration.

Question No. 5 **QID:** 1091 **Point Value:** 1

Answer:

C. 3500 gpm

Question No. 6 **QID:** 0008 **Point Value:** 1

Answer:

C. Letdown Coolers

Question No. 7 **QID:** 1101 **Point Value:** 1

Answer:

D. cause the associated channel to trip

Question No. 8 **QID:** 0332 **Point Value:** 1

Answer:

A. Controlling reactor coolant system pressure low within the limits of Figure 3.

Question No. 9 **QID:** 0686 **Point Value:** 1

Answer:

A. Verify EFW isolation and control valves to "A" OTSG closed.

Question No. 10 **QID:** 0146 **Point Value:** 1

Answer:

A. 1203.027, Loss of Steam Generator Feed

Question No. 11 **QID:** 1097 **Point Value:** 1

Answer:

D. Energize either 4160v AC bus A3 OR A4 from the AAC Diesel Generator

Question No. 12 **QID:** 1057 **Point Value:** 1

Answer:

B. RB Spray Pumps will start followed by the RB Cooling Fans due to the time delay relays which prevent EDG over-loading.

Question No. 13 **QID:** 1095 **Point Value:** 1

Answer:

C. MFW Pumps P-1A and P-1B

Question No. 14 **QID:** 0513 **Point Value:** 1

Answer:

C. EDG #1 will NOT start automatically but may be started manually by overriding the governor run solenoid.

Question No. 15 **QID:** 1058 **Point Value:** 1

Answer:

A. Close SW Inlet (CV-3822) to E-35A and verify SW Outlet (SW-22A) throttled to prevent DH cooler water hammer.

Question No. 16 **QID:** 1102 **Point Value:** 1

Answer:

A. Establish SG Pressure control using Atmospheric Dump Isolation valves

Question No. 17 **QID:** 0626 **Point Value:** 1

Answer:

B. CET temperatures stable or dropping.

Question No. 18 **QID:** 0891 **Point Value:** 1

Answer:

B. Place the EHC controls in Turbine Manual

Question No. 19 **QID:** 0320 **Point Value:** 1

Answer:

C. Take manual control of SG/RX master.

Question No. 20 **QID:** 0184 **Point Value:** 1

Answer:

C. Trip the reactor due to no on-scale indication of neutron flux.

Question No. 21 **QID:** 1061 **Point Value:** 1

Answer:

B. 1000 to 1020 psig

Question No. 22 **QID:** 1062 **Point Value:** 1

Answer:

C. Trip only the turbine when vacuum drops below 26.5 inches Hg

Question No. 23 **QID:** 1096 **Point Value:** 1

Answer:

C. Isolate letdown to reduce dose rates in the aux building.

Question No. 24 **QID:** 0162 **Point Value:** 1

Answer:

B. 50%/min, 360 MWe

Question No. 25   **QID:** 0276   **Point Value:** 1

Answer:

A. Place EDG #1 output breaker in PULL-TO-LOCK and release.

Question No. 26   **QID:** 1064   **Point Value:** 1

Answer:

D. From Startup Transformer #2 via A1/A2 due to installlation of overhead links.

Question No. 27   **QID:** 1105   **Point Value:** 1

Answer:

C. Initiate boration to restore SDM to within COLR limits within 15 minutes

---

Question No. 28   **QID:** 0326   **Point Value:** 1

Answer:

A. Rise to potentially seal damaging temperature >200 °F due to bleedoff in excess of seal cooler capacity.

Question No. 29   **QID:** 0258   **Point Value:** 1

Answer:

a. "A" HPI pump will be damaged due to loss of suction.

Question No. 30   **QID:** 0654   **Point Value:** 1

Answer:

A. Loss of power to the letdown demineralizer inlet valves.

Question No. 31   **QID:** 1068   **Point Value:** 1

Answer:

B. Pressurizer level will drop

Question No. 32   **QID:** 0611   **Point Value:** 1

Answer:

B. Stop P-34A Decay Heat pump and close CV-1404, Decay Heat Suction.

Question No. 33 **QID:** 1090 **Point Value:** 1

Answer:
B. B-6

Question No. 34 **QID:** 0561 **Point Value:** 1

Answer:
D. Quench Tank pressure 3.5 psig after a 3 minute blow of the ERV.

Question No. 35 **QID:** 0627 **Point Value:** 1

Answer:
C. ICW pump P-33B would shift to Non-Nuclear loop, P-33C would auto-start.

Question No. 36 **QID:** 1070 **Point Value:** 1

Answer:
B. RCP Seal Cooling Pump Bypass CV-2287 will open

Question No. 37 **QID:** 1071 **Point Value:** 1

Answer:
C. Heater Bank 4 OFF

Question No. 38 **QID:** 0085 **Point Value:** 1

Answer:
D. Inverter Y24 from D02

Question No. 39 **QID:** 1093 **Point Value:** 1

Answer:
B. 2 out of 3

Question No. 40 **QID:** 1073 **Point Value:** 1

Answer:
C. Containment atmposphere iodine concentration would be higher.

Question No. 41   **QID:** 1103   **Point Value:** 1

Answer:

A. Provide for conversion of analog signals to digital output signals

Question No. 42   **QID:** 0909   **Point Value:** 1

Answer:

C. 5 & 6
CRD Cooling, Chilled Water, RCP Motor Cooling

Question No. 43   **QID:** 1075   **Point Value:** 1

Answer:

D. Reactor Building sump level dropping

Question No. 44   **QID:** 1074   **Point Value:** 1

Answer:

D. 100 °F/hr, prevent brittle fracture of the Rx Vessel due to neutron embrittlement

Question No. 45   **QID:** 0565   **Point Value:** 1

Answer:

A. Feedwater loop B demand is greater than feedwater loop A demand.

Question No. 46   **QID:** 0269   **Point Value:** 1

Answer:

C. Service Water System Loops I and II

Question No. 47   **QID:** 1076   **Point Value:** 1

Answer:

C. 7 to 7.5 "/min

Question No. 48   **QID:** 1077   **Point Value:** 1

Answer:

C. Main Generator Lockout

Question No. 49 **QID:** 0140 **Point Value:** 1

Answer:

A. Loss of Seal Injection, verify seal cooling is maintained

Question No. 50 **QID:** 1078 **Point Value:** 1

Answer:

B. Local annunciator for Charger D03A, "DC OUTPUT BREAKER OPEN"

Question No. 51 **QID:** 0792 **Point Value:** 1

Answer:

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

Question No. 52 **QID:** 1065 **Point Value:** 1

Answer:

D. RCS activity due to release of fission products is rising, a reactor startup may not commence.

Question No. 53 **QID:** 1079 **Point Value:** 1

Answer:

D. P-4A to P-4B crosstie valves CV-3644 OPEN & CV-3646 CLOSED;
P-4C to P-4B crosstie valves CV-3640 CLOSED & CV-3642 OPEN;
ACW isolation CV-3643 CLOSED.

Question No. 54 **QID:** 0227 **Point Value:** 1

Answer:

C. Close Unit 1 to Unit 2 Instrument Air cross-connect.

Question No. 55 **QID:** 0104 **Point Value:** 1

Answer:

D. Verify containment isolation valves are in position marked with black tape background.

Question No. 56 **QID:** 0674 **Point Value:** 1

Answer:

C. Insert Group 5 and Group 6 rods in sequence.

Question No. **57** **QID:** **0193** **Point Value:** **1**

Answer:

C. Thot 598 degrees, Tcold 559 degrees

Question No. **58** **QID:** **1066** **Point Value:** **1**

Answer:

C. RS-3

Question No. **59** **QID:** **1067** **Point Value:** **1**

Answer:

B. CETS are failing due to short circuits, trip all running RCPs.

Question No. **60** **QID:** **0200** **Point Value:** **1**

Answer:

C. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

Question No. **61** **QID:** **1094** **Point Value:** **1**

Answer:

C. Feed both SGs at ≤ 0.2 x 10e6 lbm/hr to establish 300 to 340" level while maintaining Tube-to-Shell DT limits.

Question No. **62** **QID:** **1080** **Point Value:** **1**

Answer:

B. 905

Question No. **63** **QID:** **0470** **Point Value:** **1**

Answer:

D. 2 and 4

Question No. **64** **QID:** **0379** **Point Value:** **1**

Answer:

B. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

Question No. 65 **QID:** 0309 **Point Value:** 1

Answer:

A. Both Feedwater Loop Demands, Reactor Demand and Diamond Panel.

Question No. 66 **QID:** 1083 **Point Value:** 1

Answer:

C. Control Board Operator or STA

Question No. 67 **QID:** 0838 **Point Value:** 1

Answer:

C. A new diagnosis for high blood pressure.

Question No. 68 **QID:** 1084 **Point Value:** 1

Answer:

B. Bypassing the E-3/4A Feedwater Heaters.

Question No. 69 **QID:** 0231 **Point Value:** 1

Answer:

D. Preparer and reviewer must include a licensed operator from each unit.

Question No. 70 **QID:** 1082 **Point Value:** 1

Answer:

C. Deleting a QC hold point in a procedure section for a filter change.

Question No. 71 **QID:** 1081 **Point Value:** 1

Answer:

B. Continuous RP coverage

Question No. 72 **QID:** 0751 **Point Value:** 1

Answer:

A. TEDE 2000 mrem per year; SDE, WB= 40 rem; and LDE= 12 rem

ANO Unit 1 - 2016 RO NRC Written Exam   KEY

Question No. 73   **QID:** 0242   **Point Value:** 1

Answer:

B. Instruments that should be reliable during accident conditions.

Question No. 74   **QID:** 0051   **Point Value:** 1

Answer:

A. All Operations personnel on watch should report to the Control Room.

Question No. 75   **QID:** 0848   **Point Value:** 1

Answer:

A. P-6A Electric Fire Pump non-functional due to on-going surveillance at 0600.

Question No. 76 **QID: 1100** **Point Value: 1**

Answer:

B. Reactor Trip, 1202.001

Question No. 77 **QID: 0639** **Point Value: 1**

Answer:

A. Stop P-34A DH pump and close P-34A suction valve from RCS (CV-1434) per Section 3, DH Removal System Leak >20 GPM

Question No. 78 **QID: 1085** **Point Value: 1**

Answer:

B. Alert due to failure of RPS

Question No. 79 **QID: 0584** **Point Value: 1**

Answer:

B. No, this could overstress reactor vessel.

Question No. 80 **QID: 0586** **Point Value: 1**

Answer:

C. 78 hours

Question No. 81 **QID: 1050** **Point Value: 1**

Answer:

C. Trip P-32A and P-32C RCPs in accordance with Overheating (1202.004).

Question No. 82 **QID: 0347** **Point Value: 1**

Answer:

D. Return the assembly to an available location in the reactor vessel.

Question No. 83 **QID: 1086** **Point Value: 1**

Answer:

D. Suspend the release and initiate a condition report per ODCM L2.3.1.

Question No. 84 **QID: 1045** **Point Value: 1**

Answer:

D. Establish a continuous fire watch within one hour for the UNEPR.

Question No. 85 **QID: 0737** **Point Value: 1**

Answer:

C. Reduce cooldown rate per 1203.013, Natural Circulation Cooldown.

Question No. 86 **QID: 0638** **Point Value: 1**

Answer:

C. Degradation;
Reduce power using 1203.045, Rapid Plant Shutdown, then stop RCP.

Question No. 87 **QID: 1052** **Point Value: 1**

Answer:

B. 1203.036, Loss of 125V DC
Tech Spec 3.3.7.A

Question No. 88 **QID: 1099** **Point Value: 1**

Answer:

A. Declare P-7A inoperable and restore to operable status within 72 hours.

Question No. 89 **QID: 1046** **Point Value: 1**

Answer:

D. Go to 1107.002, ES Electrical System Operation, and restore buses to normal using "Returning Paralleled Buses A3 and A4 to Normal" section, while continuing with 1202.007.

Question No. 90 **QID: 1047** **Point Value: 1**

Answer:

B. In accordance with Section 2, ES Bus Voltage Low, start one available DG, parallel the DG to the grid, and separate the associated ES bus from the grid by opening its feeder breaker.

Question No. **91** **QID:** **1056** **Point Value:** **1**

Answer:

C. 30 days

Question No. **92** **QID:** **1048** **Point Value:** **1**

Answer:

C. Direct fuel handlers to remove the last assembly inserted into the core.

Question No. **93** **QID:** **1053** **Point Value:** **1**

Answer:

B. 1203.049, Fires in Areas Affecting Safe Shutdown

Question No. **94** **QID:** **1055** **Point Value:** **1**

Answer:

B. 0730

Question No. **95** **QID:** **0846** **Point Value:** **1**

Answer:

C. Reactor Engineer

Question No. **96** **QID:** **0486** **Point Value:** **1**

Answer:

D. Ensure a PAD review per EN-LI-100, Process Applicability Determination, is completed to support the test procedure.

Question No. **97** **QID:** **0879** **Point Value:** **1**

Answer:

B. is NOT
Shift Manager

Question No. **98** **QID:** **1049** **Point Value:** **1**

Answer:

B. 3 minutes
Shift Manager

Question No. **99** **QID: 1088** **Point Value: 1**

Answer:

D. Exclusion area evacuation

Question No. **100** **QID: 0411** **Point Value: 1**

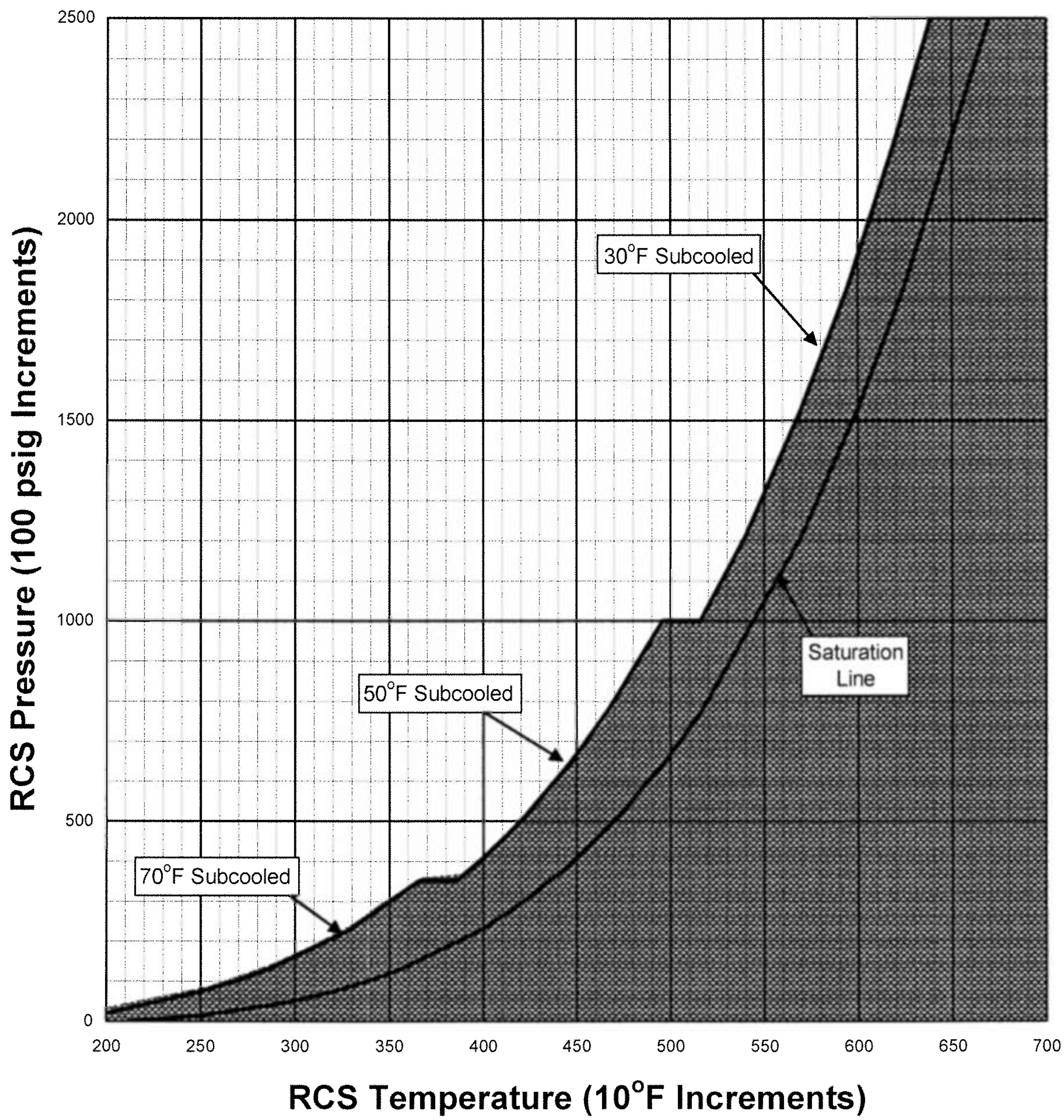
Answer:

D. Notification to the NRC is required immediately following notification of the Arkansas Department of Health and within 1 hour of the declaration of an emergency class.

2016
ANO UNIT 1
NRC INITIAL LICENSE EXAMINATION REFERENCE MATERIAL

RO

FIGURE 1
Saturation and Adequate SCM



RCS Pressure	Adequate SCM
>1000 psig	≥30°F
350 to 1000 psig	≥50°F
<350 psig	≥70°F

FIGURE 2
SG Pressure vs T-sat

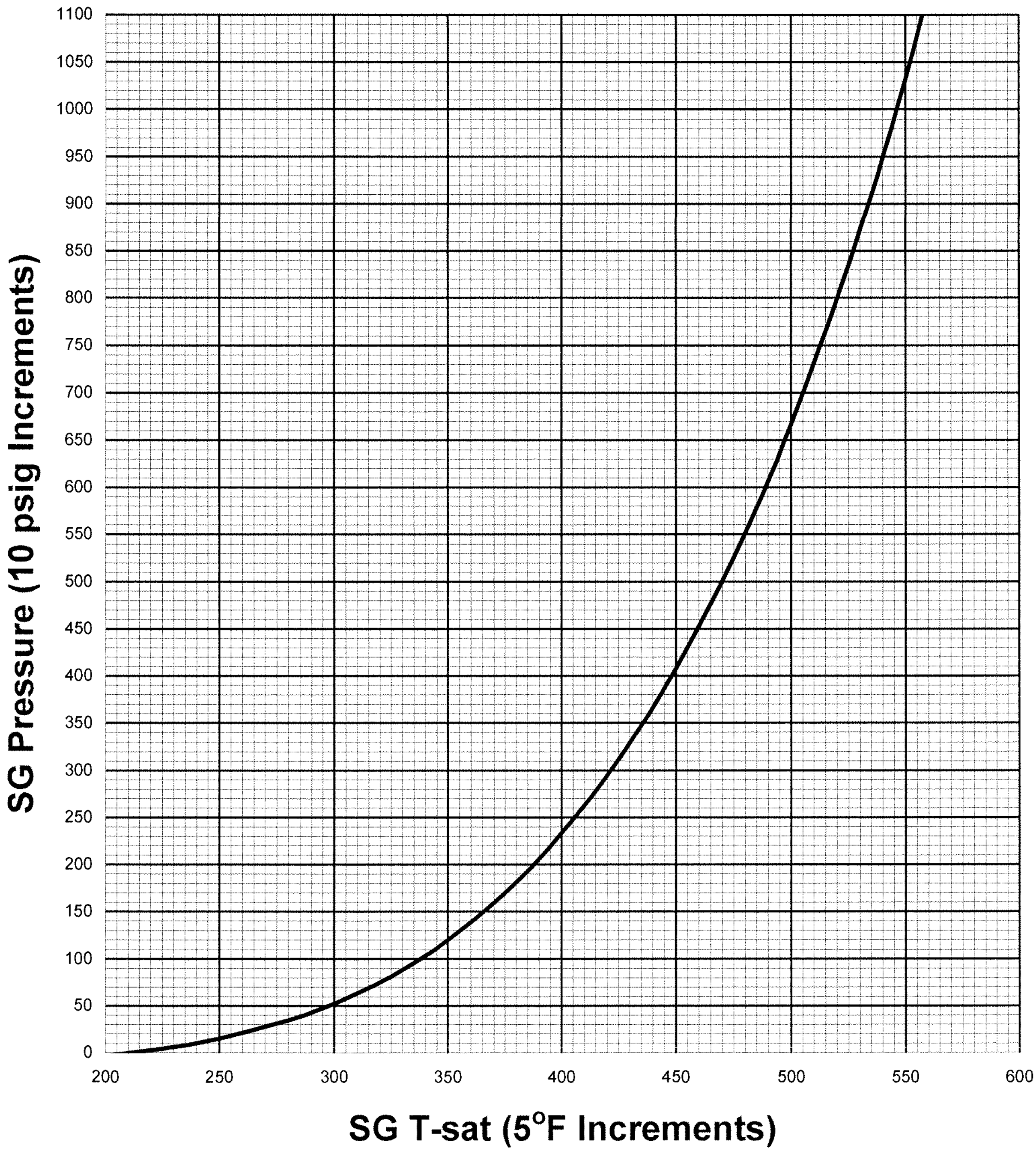


FIGURE 3
RCS Pressure vs Temperature Limits

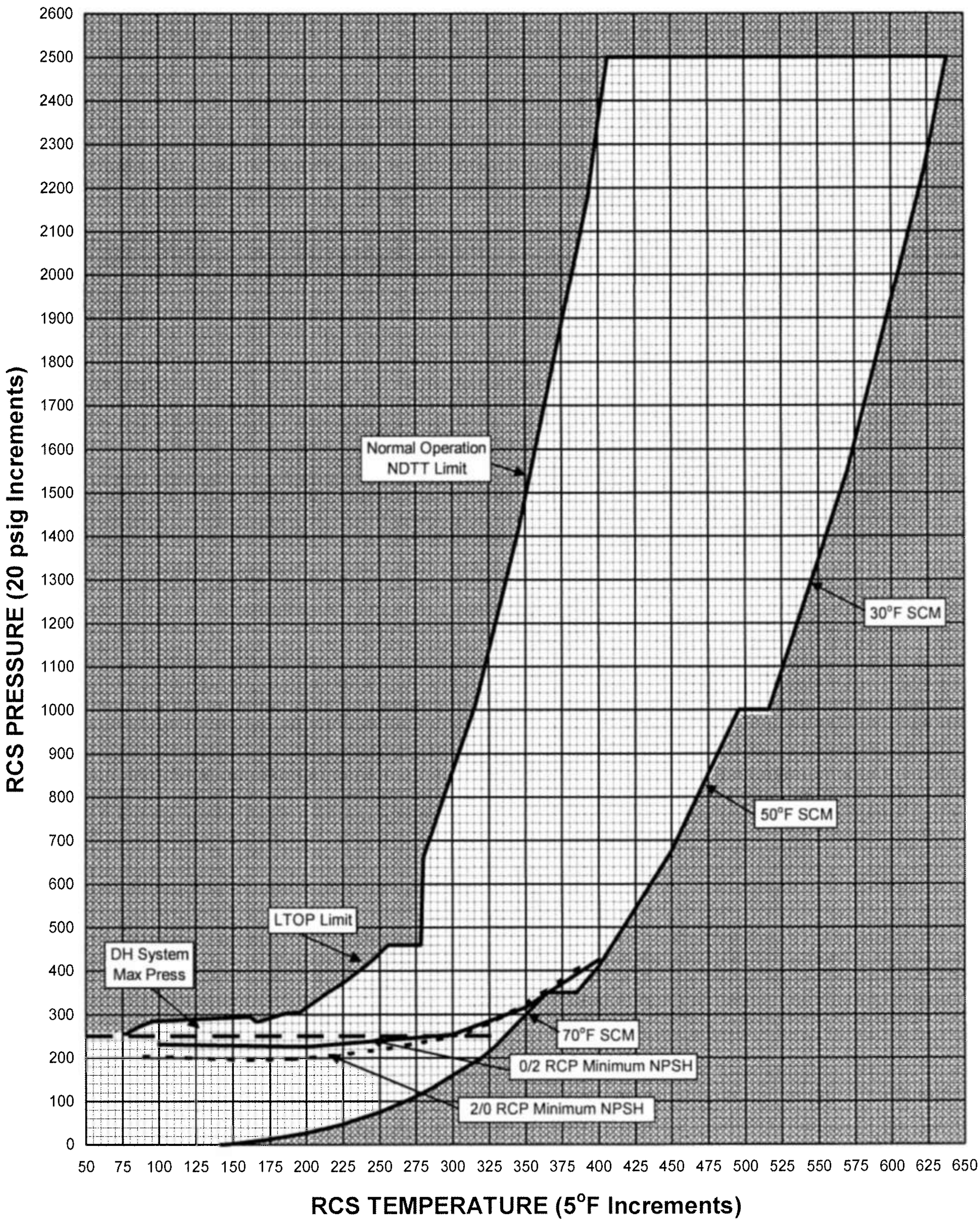


FIGURE 4
Core Exit Thermocouple for Inadequate Core Cooling

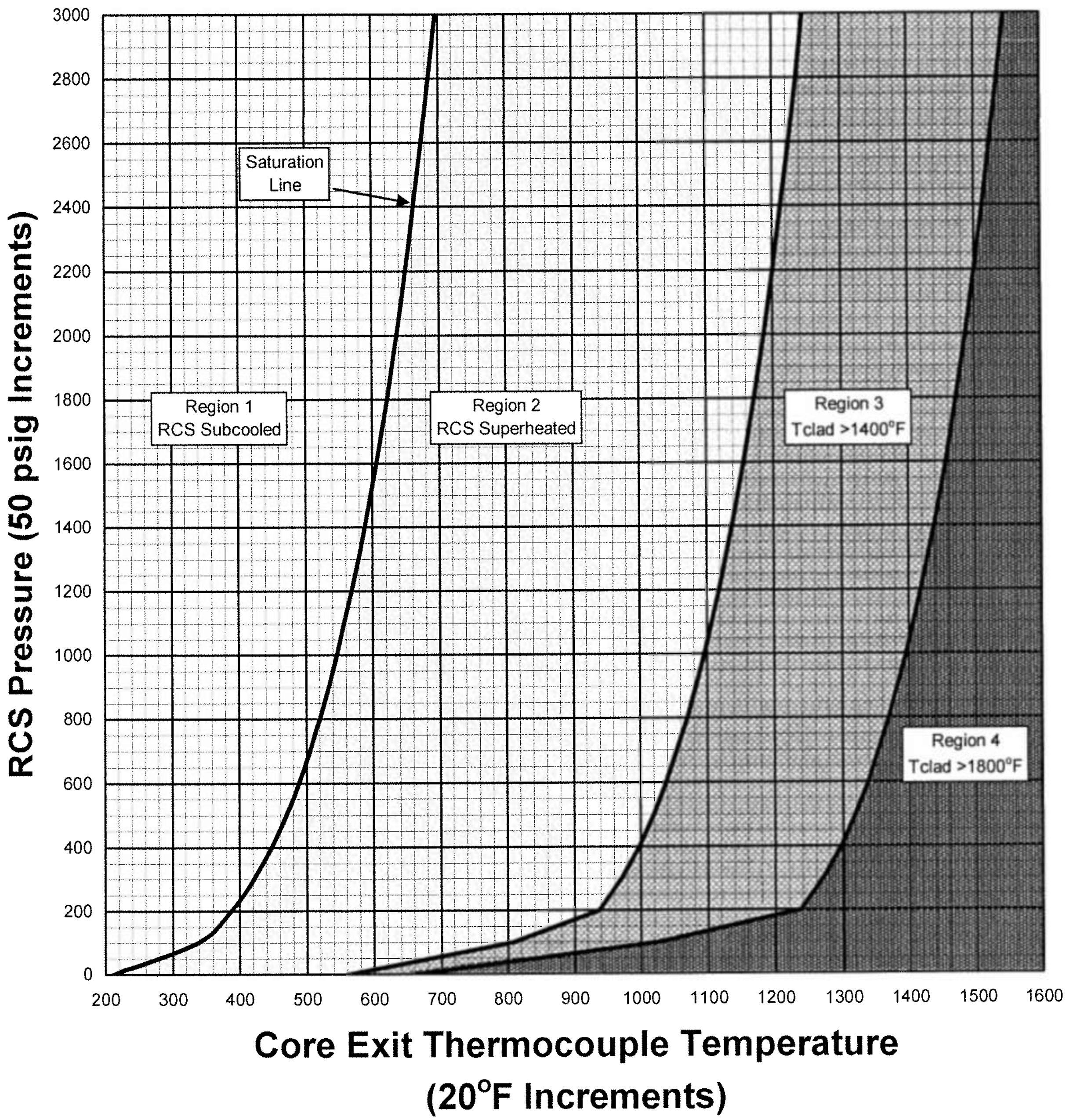


FIGURE 5

SG Pressure to Establish 40° to 60°F Primary to Secondary ΔT

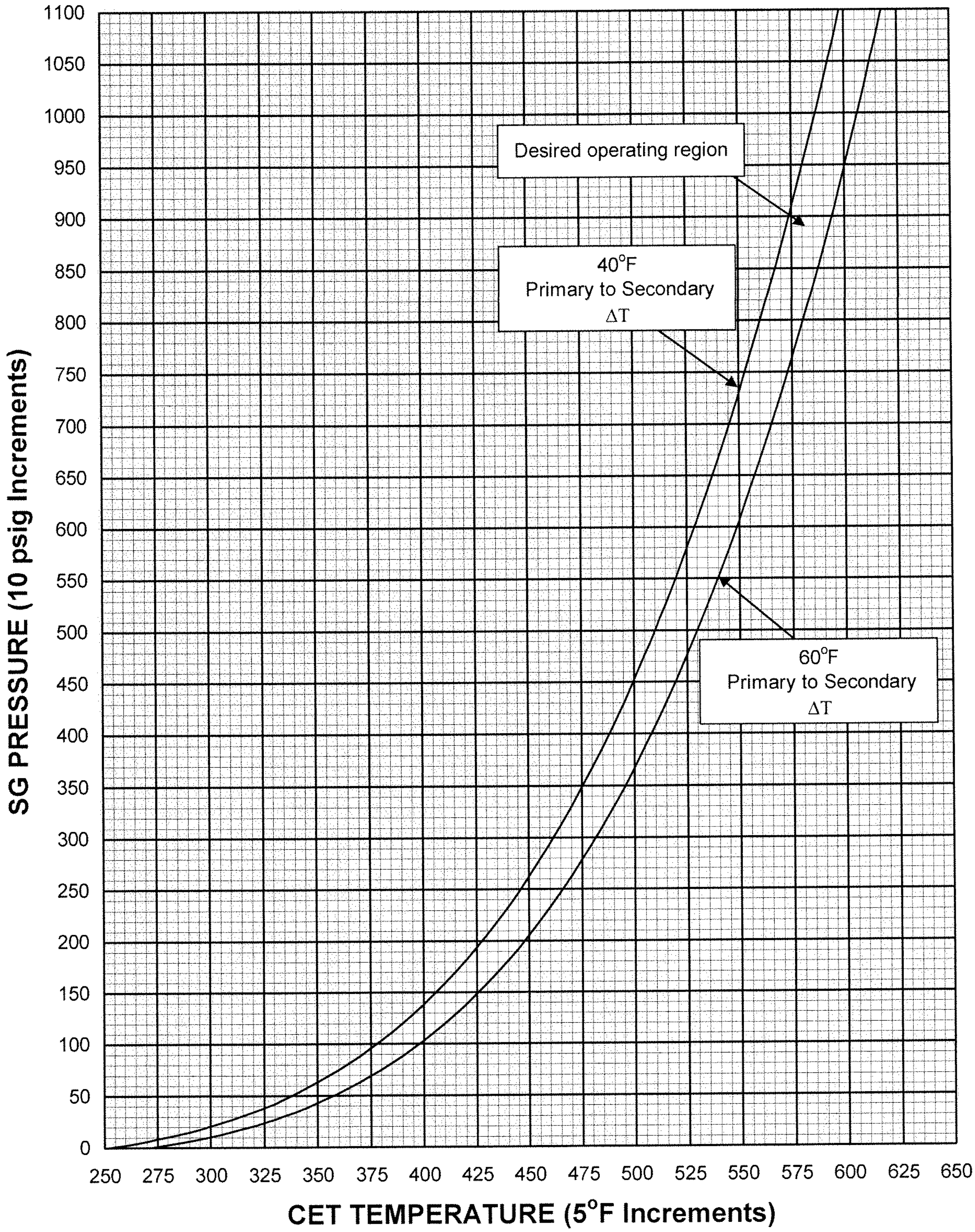
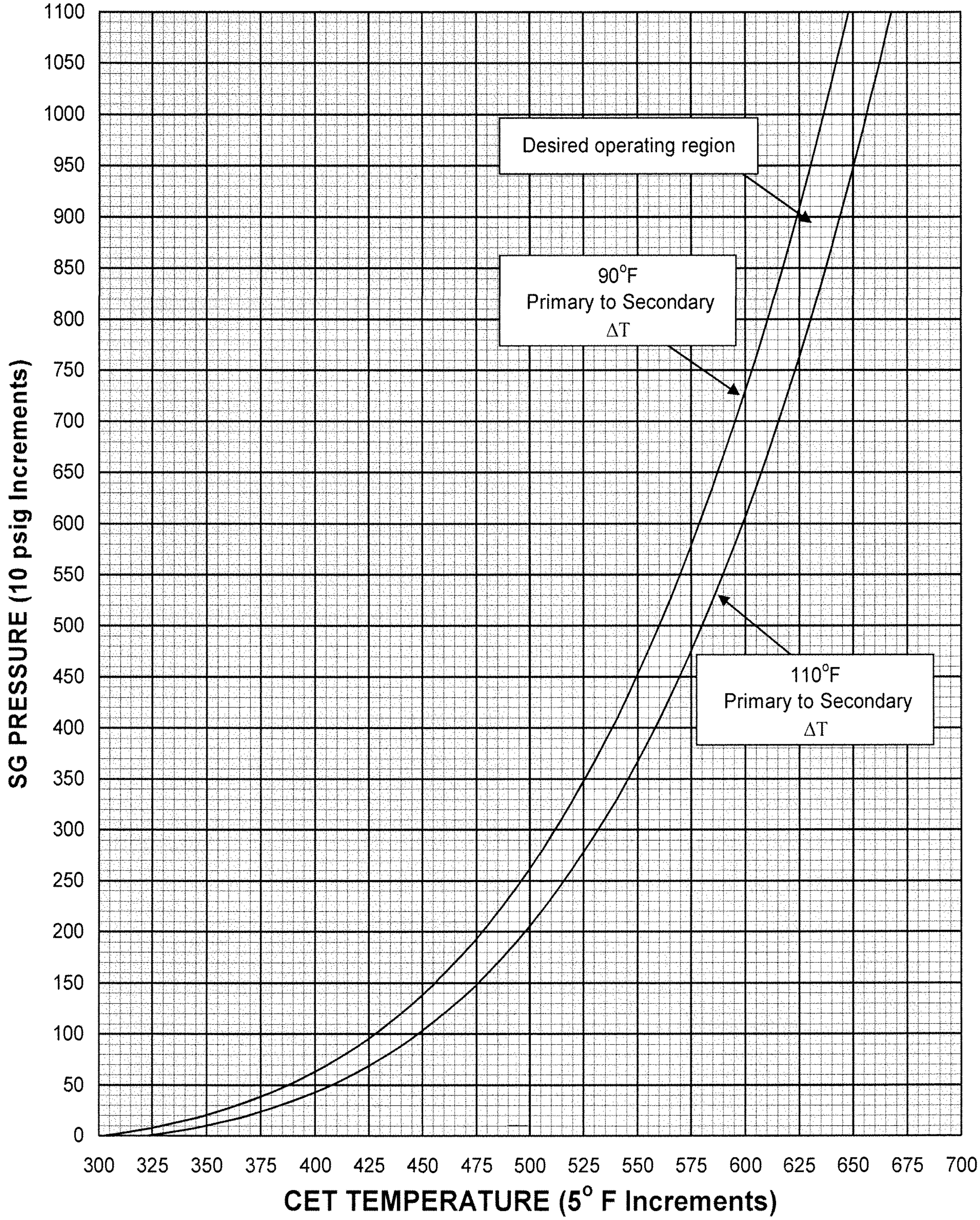


FIGURE 6

SG Pressure to Establish 90° to 110°F Primary to Secondary ΔT



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<u>SRO</u>

TRM 3.3 INSTRUMENTATION

TRM 3.3.6 Fire Detection System Instrumentation

TRO 3.3.6

--NOTE--

1. Reactor Building smoke detectors are not required to be FUNCTIONAL during Type A Integrated Leak Rate Testing.
2. All non-functional detectors specified in TRM Table 3.3.6-1 will be tracked.
3. TRO entry not required solely due to maintenance or testing activities where FUNCTIONALITY is expected to be restored within one hour.

The following heat/smoke detectors in the locations specified in TRM Table 3.3.6-1 shall be FUNCTIONAL:

1. A minimum of 50% of the heat/smoke detectors in locations outside the Reactor Building, and,

2. All heat/smoke detectors located inside the Reactor Building.

APPLICABILITY: At all times

ACTIONS

---NOTE---

1. Separate Condition entry is allowed for each location specified in TRM Table 3.3.6-1.
2. In lieu of Required Actions establishing a fire watch or requiring equipment restoration, the licensee may choose to establish compensatory measures commensurate with the evaluated risk for continued operation with non-functional detectors. All other Required Actions are applicable regardless of compensatory measures established.
3. Entry into Condition A or C requires documentation of a Fire System Impairment, except when the non-functional detector is a result of maintenance or testing lasting less than 12 hours.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Not applicable to Reactor Building fire detectors. ----------------------------------- Less than 50% of the detectors in the locations specified in TRM Table 3.3.6-1 FUNCTIONAL.	A.1 Establish a 1-hour roving fire watch. <u>AND</u>	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Condition A (continued)	A.2	Restore at least 50% of the detectors in the locations specified in TRM Table 3.3.6-1 to FUNCTIONAL status.	14 days
B. One or more detectors in the locations specified in TRM Table 3.3.6-1 non-functional that result in complete loss of automatic actuation function of a fire suppression system.	B.1	Declare the associated Fire Suppression Sprinkler/Halon System non-functional and enter applicable Conditions and Required Actions of TRO 3.7.9 and/or 3.7.10.	Immediately
C. One or more Reactor Building fire detectors non-functional.	C.1	-------------NOTE-------------- Only required in Mode 1 and 2, or when Required Action C.2 cannot be performed. -----------------------------------	
		Monitor and record Reactor Building temperature.	Once per hour
	<u>AND</u>		
	C.2	-------------NOTE-------------- Only required in Modes 3, 4, 5, 6 and defueled when environmental and radiological conditions permit unescorted entry. -----------------------------------	
		Verify fire watch patrol of the affected area.	Once per 8 hours
D. Required Actions and associated Completion Time for Condition A, B, or C not met.	D.1	Initiate a condition report.	Immediately
	<u>AND</u>		
	D.2	Determine any limitations for continued operation of the plant.	24 hours

TRM Table 3.3.6-1

SAFETY-RELATED AREAS PROTECTED BY HEAT/SMOKE DETECTORS

Protected Area Description	Fire Zone	Elevation	Controls Suppression System
Spent Fuel Area	159-B	404'	N/A
Computer Transformer Room	167-B	404'	N/A
Upper North Reactor Building Cable Spreading Area	32-K	401'	FS-5643
Upper South Reactor Building Cable Spreading Area	33-K	401'	FS-5644
Controlled Access Area	128-E	386'	N/A
Main Control Room Ceiling	129-F	386'	Halon System #3
Auxiliary Control Room Ceiling	129-F	386'	Halon System #2
Auxiliary Control Room Floor	129-F	386'	Halon System #1
Upper South Electrical Penetration Room	144-D	386'	UAV-5616
Upper North Electrical Penetration Room	149-E	386'	UAV-5615
Lower South Electrical Penetration Room	105-T	374'	UAV-5626
Lower North Electrical Penetration Room	112-I	373'	UAV-5625
Lower North Reactor Building Cable Spreading Area	32-K	373'	FS-5642
Lower South Reactor Building Cable Spreading Area	33-K	373'	FS-5645
South Switchgear Room	100-N	372'	N/A
South Inverter Room	110-L	372'	N/A
South Battery Room	110-L	372'	N/A
Cable Spreading Room	97-R	372'	UAV-5638
Hallway	98-J	372'	UAV-5639
North Switchgear Room	99-M	372'	N/A
4160 VAC Switchgear Area	197-X	372	N/A
West Heater Deck Area	197-X	372	N/A
North Emergency Diesel Generator Room	86-G	369'	UAV-5602
South Emergency Diesel Generator Room	87-H	369'	UAV-5601
Electrical Equipment Room.	104-S	368'	N/A
North Upper Piping Penetration Room	79-U	360'	UAV-5654
South Upper Piping Penetration Room	77-V	356'	N/A
Tank Room	68-P	354'/374'	N/A
Intake Structure	INTAKE	354'/366'	N/A
Laboratory And Demineralizer Access Area	67-U	354'	N/A
Condensate Demineralizer Area	73-W	354'	N/A
Compressor Room.	76-W	354'	N/A
Bowling Alley (Near Train Bay)	197-X	354	N/A
Pipe Area	40-Y	341'	N/A

TRM 3.7 PLANT SYSTEMS

TRM 3.7.9 Fire Suppression Sprinkler System

TRO 3.7.9

-----NOTE-----

Fire Suppression Water System sectionalized, loop, or sprinkler system valves may be closed to support system testing provided an individual is stationed at the valve with direct communication with the control room, such that the valve can be re-opened without delay if needed.

The Fire Suppression Sprinkler Systems specified in TRM Table 3.7.9-1 shall be FUNCTIONAL.

APPLICABILITY: At all times

ACTIONS

-----NOTE-----

1. Separate Condition entry is allowed for each sprinkler system specified in TRM Table 3.7.9-1.
2. In lieu of Required Actions establishing a fire watch, verifying FUNCTIONAL smoke and/or heat detection for the affected areas, establishing backup suppression equipment, or returning non-functional fire suppression sprinkler systems to FUNCTIONAL status, the licensee may choose to establish compensatory measures commensurate with the evaluated risk for continued operation with non-functional Fire Suppression Sprinkler Systems. All other Required Actions are applicable regardless of compensatory measures established.
3. Entry into Condition A requires documentation of a Fire System Impairment, except when non-functionality is a result of maintenance or testing lasting less than 12 hours.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Fire Suppression Sprinkler Systems specified in TRM Table 3.7.9-1 non-functional.	A.1.1 Establish a continuous fire watch in the affected area.	1 hour
	<u>OR</u>	
	A.1.2 Verify FUNCTIONAL smoke and/or heat detection for the affected area with control room alarm.	1 hour
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Establish backup fire suppression equipment for the affected area. AND A.3 Restore the non-functional Fire Suppression Sprinkler System to FUNCTIONAL status.	1 hour 14 days
B. Required Actions and associated Completion Time for Condition A not met.	B.1 Initiate a condition report. AND B.2 Determine any limitations for continued operation of the plant.	Immediately 24 hours

TEST REQUIREMENTS

TEST		FREQUENCY
TR 3.7.9.1	-------------------------------NOTE-------------------------------- Not required for sprinkler system components located in the Reactor Building. --- Verify each Fire Suppression Sprinkler System manual, power operated, or automatic valve in the flow paths specified in TRM Table 3.7.9-1 that is not locked, sealed, or otherwise secured in position, is correctly aligned and capable of transporting water from the system main to the sprinkler heads.	31 days

TRM Table 3.7.9-1

SAFETY-RELATED AREAS PROTECTED BY SPRINKLER SYSTEMS

Suppression Sprinkler Systems	Fire Zone	Elevation	Control Valve / Flow Switch
Upper North Reactor Building Cable Spreading Area	32-K	401'	FS-5643
Upper South Reactor Building Cable Spreading Area	33-K	401'	FS-5644
Decon Room and Hot Mechanic Shop*	149-E	386'	FS-5630
Upper South Electrical Penetration Room	144-D	386'	UAV-5616
Upper North Electrical Penetration Room	149-E	386'	UAV-5615
Lower South Electrical Penetration Room	105-T	374'	UAV-5626
Lower North Electrical Penetration Room	112-I	373'	UAV-5625
Lower North Reactor Building Cable Spreading Area	32-K	373'	FS-5642
Lower South Reactor Building Cable Spreading Area	33-K	373'	FS-5645
Cable Spreading Room	97-R	372'	UAV-5638
Hallway	98-J	372'	UAV-5639
North Emergency Diesel Generator Room	86-G	369'	UAV-5602
South Emergency Diesel Generator Room	87-H	369'	UAV-5601
Laboratory and Demineralizer Access Area*	67-U	354'	UAV-5628
Condensate Demineralizer Area	73-W	354'	UAV-5627
Intake Structure	INTAKE	354'	FS-5600
EFW Pump Room, P7A	38-Y	335'	UAV-5607
T-57A Diesel Generator Fuel Vault	251	328'	UAV-5609
T-57B Diesel Generator Fuel Vault	252	328'	UAV-5610

* Area is covered by a Sprinkler system without a corresponding Detection System.

3.3 INSTRUMENTATION

3.3.5 Engineered Safeguards Actuation System (ESAS) Instrumentation

LCO 3.3.5 Three ESAS analog instrument channels for each Parameter in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Parameter.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one analog instrument channel inoperable.	A.1 Place analog instrument channel in trip.	1 hour
B. One or more Parameters with more than one analog instrument channel inoperable. OR Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 --------------NOTE--------------- Only required for RCS Pressure - Low setpoint. ------------------------------------	
	Reduce RCS pressure < 1750 psig.	36 hours
	AND	
	B.3 --------------NOTES------------- 1. Only required for Reactor Building Pressure High setpoint and High High setpoint. 2. LCO 3.0.4.a is not applicable when entering Mode 4. ------------------------------------	
	Be in MODE 4.	12 hours

Table 3.3.5-1
Engineered Safeguards Actuation System Instrumentation

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Reactor Coolant System Pressure – Low Setpoint	≥ 1750 psig	≥ 1585 psig
2. Reactor Building (RB) Pressure – High Setpoint	1,2,3,4	≤ 18.7 psia
3. RB Pressure – High High Setpoint	1,2,3,4	≤ 44.7 psia

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

LCO 3.3.7 The ESAS digital actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each digital actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital actuation logic channels inoperable.	A.1 Place associated component(s) in engineered safeguards configuration.	1 hour
	<u>OR</u>	
	A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	31 days

3.6 REACTOR BUILDING SYSTEMS

3.6.3 Reactor Building Isolation Valves

LCO 3.6.3 Each reactor building isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Penetration flow paths, except for purge valve penetration flow paths, may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for system(s) made inoperable by reactor building isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building," when isolation valve leakage results in exceeding the overall reactor building leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE--------------- Only applicable to penetration flow paths with two reactor building isolation valves. ----------------------------------- One or more penetration flow paths with one reactor building isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	48 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside the reactor building <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside the reactor building
B. ------------NOTE---------------- Only applicable to penetration flow paths with two reactor building isolation valves. ------------------------------------- One or more penetration flow paths with two reactor building isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE---------------- Only applicable to penetration flow paths with only one reactor building isolation valve and a closed system. ------------------------------------ One or more penetration flow paths with one reactor building isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	<u>AND</u>	
	C.2 --------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	Once per 31 days
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 ---------------NOTE--------------- LCO 3.0.4.a is not applicable when entering Mode 4. ------------------------------------- Be in MODE 4.	12 hours

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

----------NOTE----------
Only one EFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable when entering Mode 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven EFW pump inoperable. <u>OR</u> -----NOTE----- Only applicable if MODE 2 has not been entered following refueling. -------------- Turbine driven EFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One EFW train inoperable for reasons other than Condition A in MODE 1, 2, or 3.	B.1 Restore EFW train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 18 hours
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 -------------NOTE---------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ---------------------------------- Initiate action to restore one EFW train to OPERABLE status.	Immediately
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	------------------------------NOTE----------------------------- Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching ≥ 750 psig in the steam generators. --- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTE--

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour AND Once per 12 hours thereafter
	AND	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 ------------NOTE---------------- Startup Transformer No. 2 may be removed from service for up to 30 days for preplanned preventative maintenance. This 30 day Completion Time may be applied not more than once in any 10 year period. ------------------------------------- Restore required offsite circuit to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>	
	B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure. <u>OR</u>	24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG. AND B.4 Restore DG to OPERABLE status.	24 hours 7 days AND 10 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND C.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours
D. One required offsite circuit inoperable. AND One DG inoperable.	-------------------NOTE------------------- Enter applicable Conditions and Required Actions of LCO 3.8.6, "Distribution Systems – Operating," when Condition D is entered with no AC power source to any train. --- D.1 Restore required offsite circuit to OPERABLE status. OR D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 ---------------NOTE--------------- LCO 3.0.4.a is not applicable when entering Mode 4. -------------------------------------- Be in MODE 4.	6 hours 12 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	-----------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. --- Verify each DG starts from standby conditions and, in $\leq$ 15 seconds achieves “ready-to-load” conditions.	31 days