ANO UNIT 1 – 2014

TIER 1

GROUP 1

Questions 1-18

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

QID: 0869 **Rev:** 0 **Rev Date:** 8/20/14 **Source:** New **Originator:** Possage

TUOI: A1LP-RO-ARCP **Objective:** 34 **Point Value:** 1

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

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

Description: Ability to operate and monitor the following as they apply to a reactor trip: RCP operation and flow rates

K/A Number: EA1.04 **CFR Reference:** 41.7 / 45.5 / 45.6

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

Group: 1 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** Ap

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

Given:
Reactor Power - 100%
RCS Loop A Flow - 35 X 10E6 lbm/hr
RCS Loop B Flow - 72 X 10E6 lbm/hr

What condition has caused the given conditions and what actions should be taken in accordance with 1203.031, Reactor Coolant Pump and Motor Emergency, after Reactor Trip Immediate Actions are completed?

A. "A" RCP Sheared Shaft / Trip ONLY "A" RCP

B. "A" RCP Sheared Shaft / Trip ALL RCPs

C. "C" RCP Sheared Shaft / Trip ONLY "C" RCP

D. "C" RCP Sheared Shaft / Trip ALL RCPs

Answer:

C. "C" RCP Sheared Shaft / Trip ONLY "C" RCP

Notes:

Correct Answer:
C. "C" RCP Sheared Shaft / Trip "C" RCP

A. Incorrect because "A" RCP is in the wrong Loop but plausible because the loop is A.
B. Incorrect because "A" RCP is in the wrong Loop but plausible because the loop is A.
D. Incorrect, for a sheared shaft only the affected pump is trippped, for reverse rotation ALL RCPs are tripped. This is plausible because the C RCP is in the A loop.

References:

STM 1-03 Reactor Coolant System
1203.031, Reactor Coolant Pump and Motor Emergency - Section 7, Sheared Shaft

History:

New for 2014 RO/SRO Exam

SECTION 7
SHEARED SHAFT

INSTRUCTIONS

1. **Trip reactor AND perform immediate actions of Reactor Trip (1202.001).**

NOTE

RCP vibration monitoring system and motor current indicators are useful in determining affected RCP. Motor current is indicated locally at the breakers and at the following SPDS points: A1RCPA, A1RCPB, A1RCPC and A1RCPD.

2. **Trip affected RCP.**

3. **While continuing with this section, refer to Emergency Operating Procedure (1202.XXX).**

4. **Commence RCS cooldown per Plant Shutdown and Cooldown (1102.010), "Depressurization and Cooldown of the RCS for Refueling/Maintenance" section.**

5. **Notify Predictive Maintenance to collect and analyze RCP vibration data.**

NOTE

RCP vibration monitor and loose parts monitoring system are useful in determining whether affected RCP impeller is rotating in pump casing.

6. **IF impeller rotation in pump casing is indicated for affected RCP,**
   THEN stop remaining RCP in the same loop per Reactor Coolant Pump Operation (1103.006).

7. **Monitor affected RCP seal performance.**

END

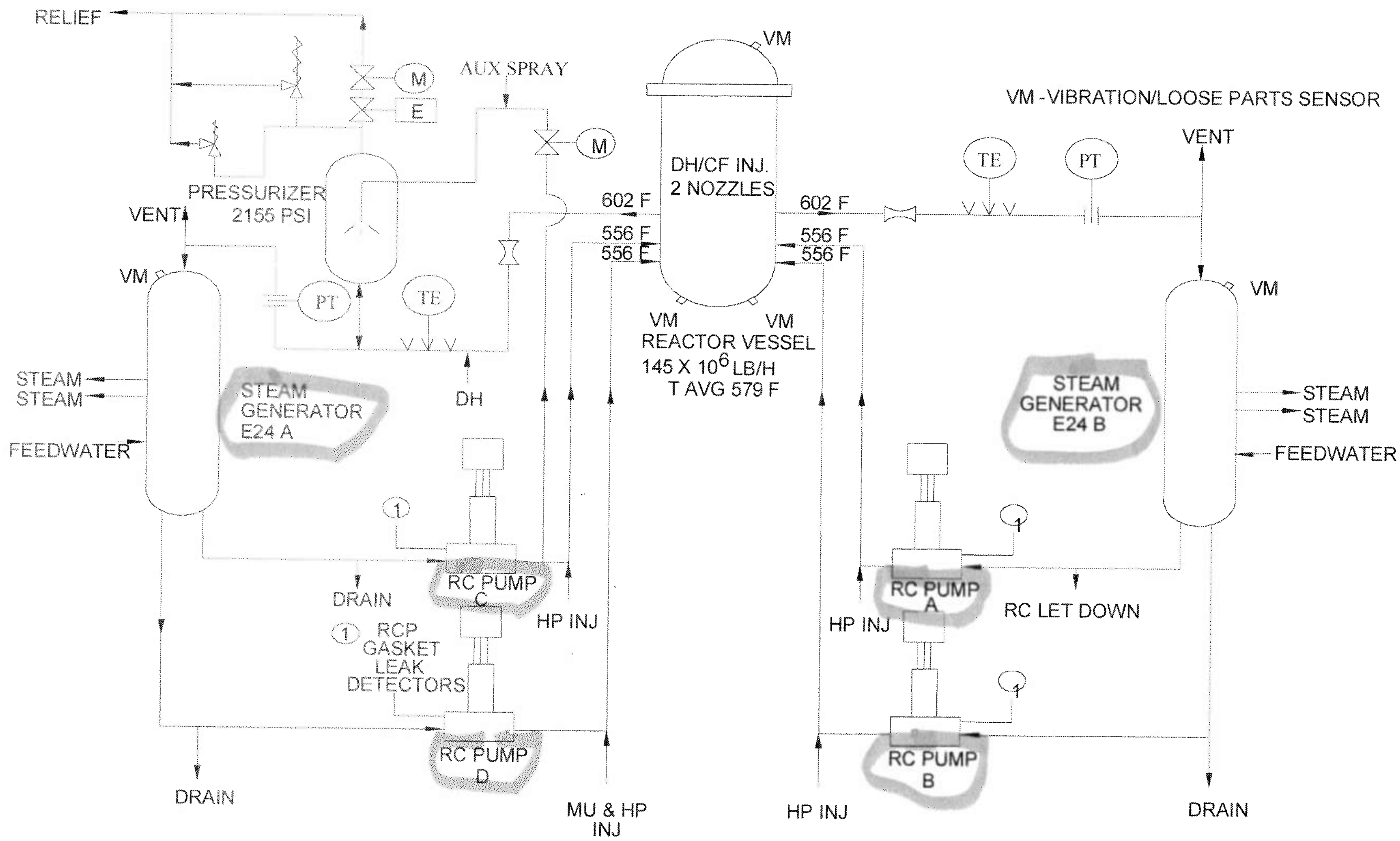


FIGURE 03.03: REACTOR COOLANT SYSTEM - SIMPLIFIED

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

QID: 0870 **Rev:** 0 **Rev Date:** 8/20/14 **Source:** New **Originator:** Possage

TUOI: A1LP-RO-EOP01 **Objective:** 12 **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: ECCS termination or throttling

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

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

Group: 1 **SRO Imp:** 4.5 **SRO Select:** Yes **Taxonomy:** Ap

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

Given:
- Reactor manually tripped due to excessive Pressurizer Code Safety Valve Leakage
- RCS pressure is 1650 psig
- HPI has been initiated with the following flowrates:
  - CV-1220 flow 135 gpm
  - CV-1221 flow 135 gpm
- Pressurizer level is 317" and rising rapidly
- CET temperatures 555 °F and dropping

What procedurally required actions should be taken as pressurizer level reaches 320 inches and why?

A. Raise HPI flow to maintain minimum SCM

B. Raise HPI flow to raise RCS pressure to >1700 psig

C. Reduce HPI flow due to PTS concerns

D. Reduce HPI flow to limit RCS pressure rise

Answer:

D. Reduce HPI flow to limit RCS pressure rise

Notes:

D is correct, as the pressurizer goes solid the non-compressible nature of water will cause RCS pressure to rise rapidly requiring the operator to reduce HPI flow to limit the RCS pressure rise. Throttling of HPI is allowed per RT-14 as long as SCM is adequate (per the given conditions) and CET temps are dropping.
A is incorrect, SCM is currently adequate and minimized so no adjustment of HPI is necessary based soley on SCM.
B is incorrect, while RCS pressure <1700 psig is a floating step to initiate HPI, with the pressurizer full the operator will have to lower flow for pressure control.
C is incorrect, with the given RCS conditions SCM is adequate so there is no reason to believe the RCPs are secured and there is nothing that implies an excessive cooldown therefore PTS limts are not in effect.

References:

1202.001, Reactor Trip
1202.013, Repetitive Tasks, RT-14
1202.013, EOP Figures, Figure 3

History:

New for 2014 RO/SRO Exam

Floating Steps

RCS Inventory/Press

- **<u>IF</u>** SCM is less than adequate
  <u>AND</u> 4160V bus A1 or A2 is energized,
  <u>THEN</u> GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN" procedure.

- **<u>IF</u>** SCM is less than adequate
  <u>AND</u> only EDG power is supplying 4160V buses,
  <u>THEN</u> GO TO 1202.007, "DEGRADED POWER" procedure.

- Control RCS press within limits of Figure 3 (RT-14).

- **<u>IF</u>** PZR level approaches 55",
  <u>THEN</u> perform step 27.

- **<u>IF</u>** PZR level drops below 30" **<u>OR</u>** RCS press drops below 1700 psig,
  <u>THEN</u> initiate HPI (RT-2).

- **<u>WHEN</u>** PZR level is 90 to 110" **<u>OR</u>** 200 to 220" (with Emergency Boration in progress)
  <u>AND</u> RCS temp is stable,
  <u>THEN</u> check MU Tank and BWST level stabilize.

RCS TEMP

- **<u>IF</u>** RCS temp is rising above:

  580°F T-hot with any RCP on
  <u>OR</u>
  610°F CET temp with all RCPs off,

  <u>THEN</u> GO TO 1202.004, "OVERHEATING" procedure.

- **<u>IF</u>** RCS T-cold is < 540°F **<u>AND</u>** dropping,
  <u>THEN</u> GO TO 1202.003, "OVERCOOLING" procedure.

ESAS

- **<u>IF</u>** ESAS actuates
  <u>OR</u>
  RCS press drops below 1550 psig
  <u>OR</u>
  RB press is ≥ 18.7 psia,
  <u>THEN</u> GO TO 1202.010, "ESAS" procedure.

CONTROL RCS PRESS

4. <u>IF</u> RCS press is high,
<u>THEN</u> limit press using one or more of the following:

A. Throttle makeup flow.

B. Throttle HPI flow by performing the following:

1) Check adequate SCM **<u>AND</u>** any of the following conditions met:

- HPI Cooling (RT-4) **<u>not</u>** in progress
- CET temps dropping
- RCS press rising with Electromatic Relief ERV (PSV-1000) open

2) Verify both HPI Recirc Blocks open:

- CV-1300
- CV-1301

3) Throttle HPI.

C. **<u>IF</u>** RCP is running,
<u>THEN</u> operate Pressurizer Spray Control (CV-1008) in HAND.

D. **<u>IF</u>** PZR AUX Spray is in service,
<u>THEN</u> throttle Pressurizer AUX Spray (CV-1416) open.

E. Place Pressurizer Heaters in OFF.

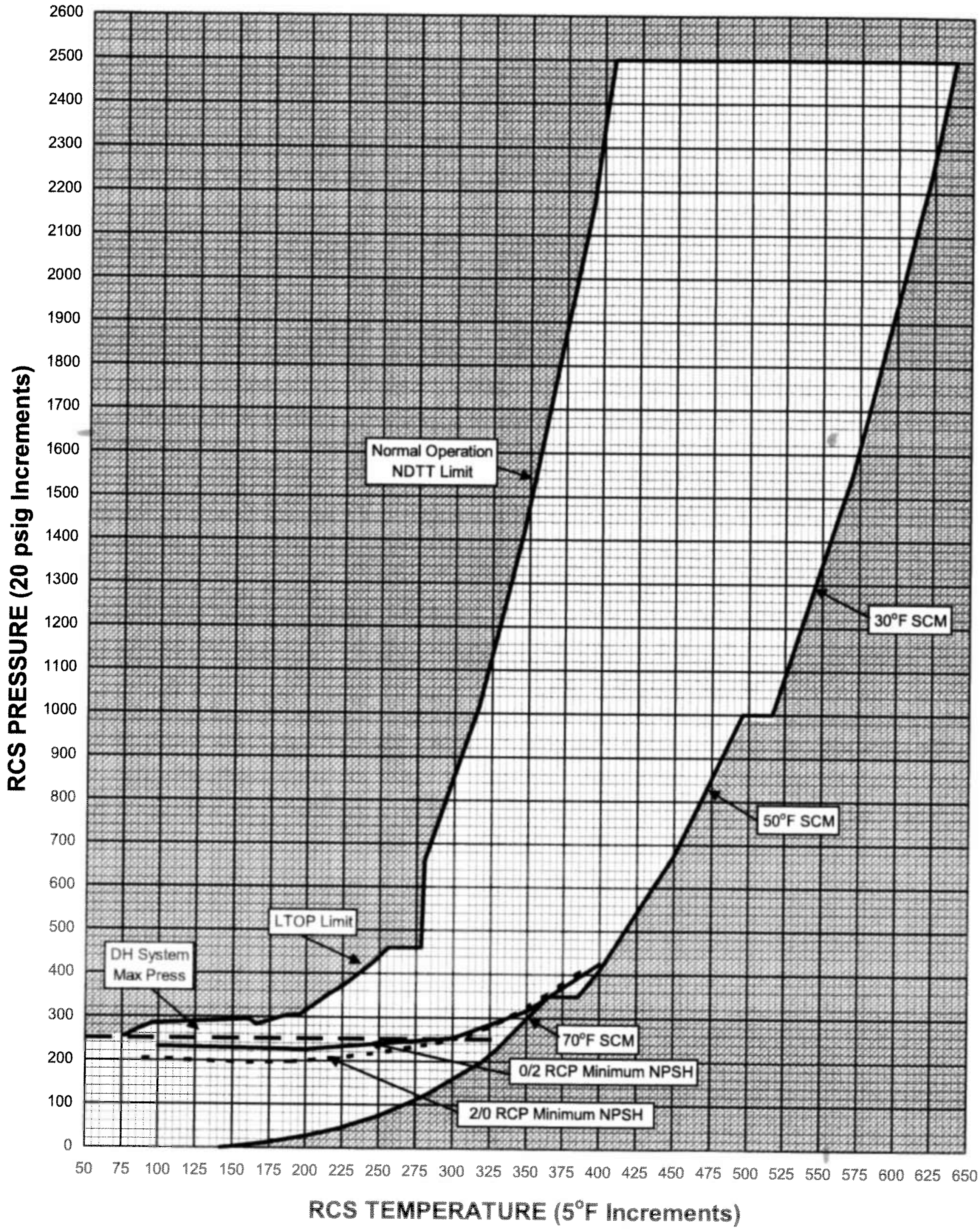
F. Raise Letdown flow.

1) **<u>IF</u>** ESAS has actuated,
<u>THEN</u> unless fuel damage or RCS to ICW leak is suspected restore Letdown per RT-13.

G. Verify Electromatic Relief ERV Isolation open (CV-1000)
<u>AND</u> cycle Electromatic Relief ERV (PSV-1000).

(4. CONTINUED ON NEXT PAGE)

FIGURE 3
RCS Pressure vs Temperature Limits



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

QID: 0881 **Rev:** 0 **Rev Date:** 9/3/14 **Source:** New **Originator:** Possage

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

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

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

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

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

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

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** K

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

Which of the following is an entry condition for 1203.039, Excess RCS Leakage?

A. Unexplained drop in Makeup Tank level

B. Pressurizer level indications differ by > 16"

C. Erratic Makeup flow and Seal Injection flow

D. Abnormal change in RCS pressure without a change in Pressurizer level

Answer:

A. Unexplained drop in Makeup Tank level

Notes:

Answer A is correct per 1203.039.
Answer B is an entry condition for 1203.015, Pressurizer Systems Failure, section 4, but could be construed as an RCS leak.
Answer C is an entry condition for 1203.026, Loss of Reactor Coolant Makeupm, section 2, but could be construed as an RCS leak.
Answer D is another entry condition for 1203.015, Pressurizer Systems Failure, section 6, but could be construed as an RCS leak..

References:

1203.039, Excess RCS Leakage

History:

New for 2014 Exam.

ENTRY CONDITION

NOTE

This procedure is intended for RCS leak rates which pose a threat to plant operations but do not require use of Emergency Operating Procedures. Small RCS leaks which are not an immediate threat to plant operations are addressed in RCS Leak Detection (1103.013).

One or more of the following:

- Unexplained drop in makeup tank level
- Unexplained mismatch between RCS in-flow and out-flow
- Unexplained rise in reactor building temperature, pressure, or sump level
- Unexplained rise in reactor building dew point (PMS/PDS M6278, M6278RTD, M6279, M6279RTD)
- RB Leak Detector (RX-7460) high alarm(s) or rising activity

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

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

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

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

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** C

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

Given the following plant conditions:

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

All EOP actions have been performed for these conditions.

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

A. Reflux Boiling

B. Forced Convection

C. Natural Circulation

D. Natural Conduction

Answer:

A. Reflux Boiling

Notes:

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

References:

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

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO re-exam.
Selected for 2011 RO exam.
Selected for 2014 Exam.

- When using MFW with the MFW nozzles, the [loss of SCM setpoint] must be achieved within 25 minutes of loss of SCM.

Heat removal from the reactor coolant by the SGs is required for a range of LOCAs to satisfy the acceptance criteria of 10 CFR 50.46. For this range of LOCAs, the RCS inventory will decrease causing a loss of natural circulation, (i.e. during the transition from saturated natural circulation to boiler condenser cooling when the RCS water level is between the bottom of the hot leg bend and the EFW spray nozzles) resulting in a period of little primary to secondary heat transfer. This can cause the RCS to heat up and repressurize causing a decrease in HPI flow rate such that the HPI flow by itself may not be sufficient for keeping the core covered and adequately cooled. However, for this range of LOCAs, enough reactor coolant will be lost out the break, prior to any core uncovering, to provide a sufficient steam volume in the primary side of the SG tubes for boiler condenser cooling. This will provide a condensation surface for condensing the steam in the RCS and reducing the RCS pressure so that the HPI flow rate can be increased to a value where its heat removal rate will match the decay heat generation rate to assure peak clad temperatures (PCTs) remain within acceptable limits.

For smaller breaks, the inventory loss is compensated for by the MU/HPI systems, if actuated, with no loss of natural circulation. For larger breaks, the RCS will depressurize low enough with SG cooling to allow the HPI system to maintain sufficient liquid inventory in the reactor vessel to keep the core adequately cooled.

The required heat removal by the SGs can be induced by either spraying sufficient EFW into the SG(s) or by maintaining a sufficient volume of water in the SG(s) using either EFW or MFW.

<u>Establishing the [loss of SCM setpoint] With EFW</u>

When establishing the [loss of SCM setpoint], EFW should not be throttled unless it is necessary to maintain appropriate SG operations. Automatic EFW control systems should be allowed to function as designed to raise SG level to the [loss of SCM setpoint]. These systems should provide either level increases at prescribed rates or adequate flow rates until the required level is achieved. EFW manual flow control should only occur if either the automatic EFW control system is not functioning properly or if the SG becomes uncoupled (loss of heat transfer). For level rate control systems, among other possible failures, if the system does not initially feed due to a level error (actual level higher than target level), this is considered as not functioning properly. Even under manual flow control, full EFW flow should be provided unless throttling becomes necessary. Throttling could become necessary to control SG level at setpoint, to prevent exceeding pump runout limits, or to reduce the depressurization in an uncoupled SG. An uncoupled SG is

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

QID: 0883 **Rev:** 0 **Rev Date:** 9/3/14 **Source:** New **Originator:** Possage

TUOI: A1LP-RO-ARCP **Objective:** 30 **Point Value:** 1

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

System Number: 017 **System Title:** Reactor Coolant Pump Malfunctions (Loss of RC Flow

Description: Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP indicators and controls.

K/A Number: AK2.10 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	2.8	**SRO Select:**	Yes	**Taxonomy:**	K

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

Given:
- Plant Power 80%
- P-32A, Reactor Coolant Pump Trips

The CRS directs you to check for reverse rotation of P-32A.

Which of the following is an indication of reverse rotation per 1203.031, Reactor Coolant Pump and Motor Emergency?

A. RCS loop "A" flow Higher than expected

B. RCP Motor Bearing high temperature

C. RCP Motor Winding high temperature

D. RCP Speed indication on PMS is negative

Answer:

B. RCP Motor Bearing high temperature

Notes:

Answer B is correct per section 6 of 1203.031.
Answer A is incorrect but plausible as RCS loop flow is affected when an RCP rotates backwards. Eventhough the dP through the idle pump would be less than the dP through the core overall Loop A flow would be lower, not higher.
Answer C is incorrect but plausible since it involves a problem with the RCP but is not an indication of reverse rotation.
Answer D is incorrect but plausibe since the indication is for speed but PMS would not indicate a negative value.

References:

1203.031, Reactor Coolant Pump and Motor Emergency,

History:

New for 2014 Exam.

SECTION 6
RCP REVERSE ROTATION

ENTRY CONDITIONS

- **Associated RCS loop flow indicates lower than expected**
- **Plant computer reverse rotation alarm on idle RCP. (not applicable for P-32B)**
  - RCP P32-A REVERSE ROTATION (FS6510)
  - RCP P32-C REVERSE ROTATION (FS6512)
  - RCP P32-D REVERSE ROTATION (FS6513)
- **RCP high vibration**
- **RCP motor bearing high temperature**
- **Loss of zero speed indication on idle RCP (Indicated by portable instrumentation).**

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

QID: 0449 **Rev:** 0 **Rev Date:** 5/1/2002 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-AOP **Objective:** 3 **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: Location and isolability of leaks.

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

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

Group: 1 **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** K

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

Given:

- The plant is in Cold Shutdown at 100 degrees F.
- The "A" Decay Heat (DH) train has just been placed in service.
- The RCS level is decreasing and the Auxiliary Building sump HI LEVEL alarm is actuated.

What operator action is required?

A. Start makeup pump(s) to maintain RCS level.

B. Open the BWST outlet valve(s) to maintain NPSH to the DH pump.

C. Stop running DH pump and start other DH pump.

D. Stop the DH pump and isolate the DH system from the RCS.

Answer:

D. Stop the DH pump and isolate the DH system from the RCS.

Notes:

A is incorrect. Use of the makeup pump presents a Low Temperature Overpressure concern.
B is incorrect. It does not address isolation of the RCS leak.
C is incorrect. It does not address isolation of the RCS leak and poses a common mode failure concern.
D is correct. It protects the remaining train of Decay Heat and isolates the RCS leakage.

References:

1203.028, Loss of Decay Heat Removal

History:

Direct from regular exambank QID 2865.
Selected for use in 2002 RO/SRO exam. KA 005 2.4.10
Selected for 2014 Exam

SECTION 1 – LOSS OF INVENTORY OR DH REMOVAL SYSTEM LEAK <20 GPM

7. **IF additional makeup is required,**
   THEN refer to "RCS Makeup Methods" Attachment H of this procedure.

8. **IF the leak is into the ICW system,**
   THEN perform the following:

   A. **IF** RCS level is stabilized,
      THEN GO TO step for known "RCS Leakage into ICW System" of Excess RCS Leakage (1203.039).

   B. **IF** RCS level is **NOT** stabilized,
      THEN GO TO Section 3 – DH Removal System Leak >20 gpm.

9. **IF the leak is on the in-service Decay Heat Loop,**
   THEN perform the following to shift DH loops and isolate the leaking DH loop:

> **CAUTION**
> The RCS must have flow from one DH loop or RCPs when adding DI water or when adding any boron concentration less than that required to meet shutdown margin requirements of LCO 3.1.1. Ref. Tech Spec 3.4.6, 3.4.7, 3.4.8

   A. **IF** "A" DH system is to be placed into service,
      THEN perform the following:

      1) Close P-34A Suction from BWST (CV-1436).

      2) Open P-34A Suction from RCS (CV-1434).

   B. **IF** "B" DH system is to be placed into service,
      THEN perform the following:

      1) Close P-34B Suction from BWST (CV-1437).

      2) Open P-34B Suction from RCS (CV-1435).

(continued)

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

QID: 0886 **Rev:** 0 **Rev Date:** 9/4/14 **Source:** New **Originator:** Possage

TUOI: A1LPO-RO-ICW **Objective:** 7 **Point Value:** 1

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

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: The cause of possible CCW loss.

K/A Number: AA2.02 **CFR Reference:** 43.5 / 45.13

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

Group: 1 **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** C

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

Given:
- Unit 1 is at 100% power
- CRD Cooling Pump, P-79A, is in service
- Reactor Building Pressure rises to 19 psia

What is the impact to the CRD Cooling System?

A. P-79A in a run out condition

B. P-79A in a shutoff head condition

C. P-79A on minimum recirc

D. P-79A will stabilize with a slightly lower flow

Answer:

B. P-79A in a shutoff head condition

Notes:

The CRD cooling system is a branch of the Intermediate Cooling Water System, ANO-1's equivalent to the CCW system.

Answer B is correct since ESAS channel 5 or 6 actuate at 18.7 psia and close containment isolations to CRD cooling. CRD pump P-79A will be in a shutoff head condition.

Answer A is incorrect, ESAS channel 5 or 6 will actuate at 18.7 psia but this will isolate CRD cooling and not open additional flowpaths as in cooling to RB coolers, so no run out condition will occur.

Answer C is incorrect, ESAS channel 3 and 4 will actuate at 18.7 psia but they do not cause isolation of the CRD cooling system, and thus will not place P-79A on minimum recirc..

Answer D is incorrect, ESAS channel 3 and 4 will actuate at 18.7 psia but they do not cause isolation of the CRD cooling system which does occur when ESAS channels 5 or 6 actuate.

References:

1202.012, Repetitive Tasks, RT-10 Verify Proper ESAS Actuation
STM 1-65, Engineered Safeguards Acutation System

History:

New for 2014 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.

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

QID: 0933 **Rev:** 0 **Rev Date:** 9/18/14 **Source:** New **Originator:** Cork
TUOI: A1LP-RO-NNI **Objective:** 14 **Point Value:** 1

Section: 4.2 **Type:** Generic Abnormal Plant Evolutions
System Number: 027 **System Title:** Pressurizer Pressure Control Malfunction
Description: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners.

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

Tier:	1	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.8	**SRO Select:**	Yes	**Taxonomy:**	An

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

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

Current plant conditions are
Unit 1 is at 90% power
RCS pressure has now dropped to 2110 psig.

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

A. Pressurizer spray valve OPEN

B. Heater bank 3 ON

C. Heater bank 4 ON

D. Heater bank 5 ON

Answer:

D. Heater bank 5 ON

Notes:

D is correct, Bank 5 should not be on until 2105 psig
A is incorrect but plausible, the spray valve should normally close at 2155 psig. However a MFW Pump trip above 80% power reduces the 2205 open setpoint for the spray valve to 2080 psig and the close setpoint drops to 2030 psig so it should be open
B is incorrect but plausible since this is another heater bank.
C is incorrect but plausible since this is another heater bank.

References:

1103.005, Pressurizer Operation

History:

New for 2014 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: 0887 **Rev:** 0 **Rev Date:** 9/4/14 **Source:** New **Originator:** Possage

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

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermo-hydraulics behavior.

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

Tier: 1 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** K

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

What does AMSAC monitor as a backup to the Reactor Protection System in the event of an Anticipated Transient Without Scram (ATWS)?

A. RCS pressure and Power Range reactor power (Bailey)

B. RCS pressure and Linear Range Gamma-Metrics reactor power

C. Total FW flow and Power Range reactor power (Bailey)

D. Total FW flow and Linear Range Gamma-Metrics reactor power

Answer:

D. Total FW flow and Linear Range Gamma-Metrics reactor power

Notes:

Answer D is correct, AMSAC monitors total FW flow and Linear Range G-M power to trip CRDM's if a thermal imbalance exists as exhibited by total FW flow <15% when Rx Pwr is >45%.
Answer A is incorrect, the other component of ANO's ATWS system is DROPS which monitors Reactor pressure but this is separate from AMSAC.
Answers B and C are combinations of correct and incorrect parameters.

References:

STM 1-59, Diverse Reactor Overpressure Protection System

History:

New for 2014 Exam.

The Loop A and B Main Feedwater Flow signals from the flow transmitters are provided to DROPS through non-1E Bailey voltage buffers in NNI cabinets C47-4 and C48-7.

Refer to table below for MFW flow transmitters associated with each DROPS channels.

Channel 1	Channel 2
Loop "A" MFW flow PDT-2627	Loop "A" MFW flow PDT-2628
Loop "B" MFW flow PDT-2677	Loop "B" MFW flow PDT-2678

The AMSAC turbine trip and EFW initiation signals are generated when **MFW flow is less than 15% of 6.0 x 10^6 LB/hr rated flow in both loops and when reactor power is greater than 45%.**

(Refer to Figure 59.04)

The turbine trip signals are summed in the existing turbine trip circuitry and upon receipt of both DROPS channels AMSAC signals, the auto-stop oil trip solenoid and the auto-stop back-up oil trip solenoid will be energized. Energizing either of the auto-stop oil trip solenoids will trip the turbine.

The DROPS AMSAC signal to trip the main turbine is accomplished by two relays in the turbine trip circuitry. The relay contacts are wired in series to form the 2 out of 2 coincidence logic to actuate the Auto-Stop oil trip and backup trip solenoids which trip the main turbine. The power for the coil and contacts on these relays is supplied from the 125 vdc bus in the turbine trip circuitry. DROPS Turbine trip confirmation is provided by the two trip contacts wired in series which provide a 125 vdc trip confirm signal back to DROPS. For additional information on the turbine trip circuitry refer to STM 1-24 Main Turb & Controls.

The DROPS AMSAC subsystem provides an energize to trip signal to initiate Emergency Feedwater. EFW actuation signals from DROPS inputs to EFIC channels "A" and "D" Initiate modules. DROPS channel 1 trip signal inputs to EFIC channel "A" and channel 2 trip signal inputs to EFIC channel "D". Initiation of EFIC Channels A and D will result in full EFW actuation.. Since EFIC trips actuate on loss of input signal or loss of EFIC power to the initiate modules the signals from DROPS are inverted by a Anticipatory Trip Initiation relay in the EFIC cabinets. The 1E relay coil and contacts are powered by the associated EFIC cabinet 28 vdc power supply. This relay is normally de-energized and its associated contacts closed. The AMSAC trip signal will energize the anticipatory trip initiation relay causing it contacts to open actuating EFW utilizing the normal initiation process. The non-1E AMSAC signal interfaces with the 1E portion of EFIC through photo-optic isolators installed in EFIC cabinets C37-1 "A" channel and C37-4 "D" channel..

For normal EFIC operation the trip signal from each of the four EFIC channels are combined in the channel "A" and channel "B" trip

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

QID: 0333 **Rev:** 0 **Rev Date:** 9-6-99 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-EOP06 **Objective:** 4 **Point Value:** 1

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

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

Description: Ability to operate and monitor the following as they apply to the SGTR: Steam and feedwater flow for mismatched condition.

K/A Number: EA1.02 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** An

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

After a reactor trip, which of the following would indicate a ruptured tube in the "A" Steam Generator?

A. "A" EFIC level is 31 stable "A" MFW Flow is .3 X E6 lb/hr
"B" EFIC level is 35 rising "B" MFW Flow is .2 X E6 lb/hr

B. "A" EFIC level is 31 stable "A" MFW Flow is .3 X E6 lb/hr
"B" EFIC level is 29 rising "B" MFW Flow is .4 X E6 lb/hr

C. "A" EFIC level is 28 rising "A" MFW Flow is .5 X E6 lb/hr
"B" EFIC level is 31 stable "B" MFW Flow is .3 X E6 lb/hr

D. "A" EFIC level is 35 rising "A" MFW Flow is .1 X E6 lb/hr
"B" EFIC level is 31 stable "B" MFW Flow is .3 X E6 lb/hr

Answer:

D. "A" EFIC level is 35 rising "A" MFW Flow is .1 X E6 lb/hr
"B" EFIC level is 31 stable "B" MFW Flow is .3 X E6 lb/hr

Notes:

ANO-1 does not have a steam flow instrument, a mismatch between steam flow and feedwater flow would be indicated by a level change or pressure change in the SG.

D indicates a tube rupture in the "A" OTSG, i.e., level rising and low feedwater flow rates in "A" OTSG.
A would indicate a tube rupture in the "B" OTSG and is thus incorrect.
B & C choices are possibiliities which indicate normal conditions for post trip.

References:

1202.006, Tube Rupture
1202.001, Reactor Trip
1202.012, Repetitive Tasks, RT-18

History:

Developed for 1999 exam.
Previously used under K/A 038 EA2.03 (Ability to determine and interpret the following as they apply to the SGTR: Which S/G is ruptured.) CFR 43.5 / 45.13 RO 4.4 SRO 4.6.
Used on 2004 RO/SRO Exam (T1 G2 038 EA1.02)
Selected for 2014 Exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
31. Check adequate SCM.	31. **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure.
32. Check RCS T-cold remains ≥ 540°F.	32. **IF** RCS T-cold is < 540°F **AND** dropping, **THEN GO TO 1202.003, "OVERCOOLING"** procedure.
33. Check RCS temp remains either: < 580°F T-hot with any RCP on **OR** < 610°F CET temp with all RCPs off	33. **GO TO 1202.004, "OVERHEATING"** procedure.
34. Check SG tube integrity (RT-18).	34. **GO TO 1202.006, "TUBE RUPTURE"** procedure.

CHECK SG TUBE INTEGRITY

1. Check the following indications:

   - None of the following radiation monitor indications rising OR in alarm:
     - Main Condenser process monitor (RI-3632)
     - Either OTSG N-16 Gross Detector:
       * RI-2691
       * RI-2692
     - Either Steam Line High Range Radiation Monitor:
       * RI-2681
       * RI-2682
   - No report from Nuclear Chemistry that SG tube leak exists.
   - No rise in unidentified RCS leakage accompanied by:
     - Higher than expected SG level
     - Lower than expected FW flow rate

END

INSTRUCTIONS | **CONTINGENCY ACTIONS**

5. IF Reactor power is > 20%,
   THEN begin controlled plant shutdown at ≥ 5% per minute.

6. Determine bad SG using one or more of the following:

   - OTSG N-16 Gross Detectors:

SG A	SG B
RI-2691	RI-2692

   - SGTR display on SPDS
   - Plant Monitoring System Alarms
   - Steam Line High Range Radiation Monitors:

SG A	SG B
RI-2682	RI-2681

   - Local steam line radiation survey
   - Nuclear Chemistry sample
   - At low FW flow rates:
     * Higher than expected SG level
     * Lower than expected FW flow rate
     * Lower than expected MFW pump speed

7. Verify Control of Secondary System Contamination (1203.014) being performed in conjunction with this procedure.

8. WHEN bad SG is known,
   THEN place bad SG EFW Pump Turbine K3 Steam Supply valve in MANUAL AND close:

SG A	SG B
CV-2667	CV-2617

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

QID: 0930 **Rev:** 0 **Rev Date:** 9/16/2014 **Source:** Modified **Originator:** Possage
TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

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

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

Description: Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: Actions contained in EOPs for steam line rupture.

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

Tier:	1	**RO Imp:**	4.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	Ap

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

Given:
-Reactor is tripped due to a steam line rupture.
-"A" SG pressure 590 psig
-"B" SG pressure 430 psig

Which valves are positioned correctly per RT-6?

A. "B" MSIV CV-2692 closed to isolate potential steam leaks and
"B" EFW Isolation valve CV-2620 closed to prevent EFW flow to the SG

B. "A" MSIV CV-2691 closed to isolate potential steam leaks and
"A" EFW Isolation valve CV-2627 closed to prevent EFW flow to the SG

C. "A" MFW Isolation valve CV-2680 closed and "A" EFW Control valve CV-2645 closed
to isolate all MFW and EFW flow to the SG to stop the over cooling affects from "A" SG

D. "B" MFW Isolation valve CV-2630 closed to prevent MFW flow to the SG and
"B" EFW Control valve CV-2647 throttled open to allow trickle feed to the SG

Answer:

A. "B" MSIV CV-2692 closed to isolate potential steam leaks and
"B" EFW Isolation valve CV-2620 closed to prevent EFW flow to the SG

Notes:

"A" is correct since it has the correct SG valves in the correct poition for a faulted B SG and the correct reason for their position. Both SG pressures are less than 600 psig but B SG pressure is more than 150 psig less than A so it will be isolated by MSLI and EFIC Vector.
The other responses are incorrect valve positions / reason for the given conditions.
D. distractor mentions "trickle feed", this concept is plausible because if both SG are available the direction is to feed the bad SG only as necessary to maintain primary to secindary heat transfer in associated loop. This guidance is contained in the Tube Rupture EOP.

Modified QID 0826 by changing which generator would be isolated (swapped SG pressures) so that the correct answer has changed from B to A, also added a reason for the valve position to more closely match the KA
This change required modification of distractors C and D so that there would not be multiple correct answers.

References:

1202.012, Repetitive Tasks, RT-6

History:

Modified for 2014 Exam QID #0826

VERIFY PROPER MSLI AND EFW ACTUATION AND CONTROL

1. **Verify MSLI actuation indicated for affected SG(s) on C09:**

   Train A:
   - Bus 1
   - Bus 2

   Train B:
   - Bus 1
   - Bus 2

2. **Verify EFW actuation indicated on C09:**

   Train A:
   - Bus 1
   - Bus 2

   Train B:
   - Bus 1
   - Bus 2

3. **Verify affected SG(s) MSIV, Main Feedwater Isolation, Main Feedwater Block, Low Load, and Startup valves closed:**

SG A		SG B
CV-2691	**MSIV**	CV-2692
CV-2680	**Main Feedwater Isolation**	CV-2630
CV-2625	**Main Feedwater Block**	CV-2675
CV-2622	**Low Load**	CV-2672
CV-2623	**Startup**	CV-2673

4. **Verify affected SG(s) ATM Dump Control System operating to maintain SG press 1000 to 1040 psig, unless SG depressurizes:**

SG A		SG B
CV-2676	**ATM Dump ISOL**	CV-2619
CV-2668	**ATM Dump CNTRL**	CV-2618

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:

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: 0826 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** New **Originator:** S. Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the interrelations between the steam line rupture and the following: Valves.

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

Tier: 1 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 2

Group: 1 **SRO Imp:** 2.5 **SRO Select:** No **Taxonomy:** C

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

Given:
-Reactor is tripped due to a steam line rupture.
-"A" SG pressure 430 psig
-"B" SG pressure 590 psig

Which valves are positioned correctly per RT-6?

A. "B" MSIV CV-2692 closed and "B" EFW Isolation valve CV-2620 closed

B. "A" MSIV CV-2691 closed and "A" EFW Isolation valve CV-2627 closed

C. "A" MFW Isolation valve CV-2680 closed and "A" EFW Control valve CV-2645 open

D. "B" MFW Isolation valve CV-2692 closed and "B" EFW Control valve CV-2647 closed

Answer:

B. "A" MSIV CV-2691 closed and "A" EFW Isolation valve CV-2627 closed

Notes:

"B" is correct since it has the correct SG valves in the correct poition.
The other responses are either the incorrect SG or incorrect valve positions.

References:

1202.012, Chg. 009, RT-6

History:

New for 2011 RO Exam.

PARENT

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

QID: 0888 **Rev:** 0 **Rev Date:** 9/4/14 **Source:** New **Originator:** Possage

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

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

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

Description: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A Number: 2.1.23 **CFR Reference:** 41.10 / 43.5 / 45.2 / 45.6

Tier: 1 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** An

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

Given:
- Plant Power 72%
- 'A' MFWP Speed 4300 rpm and steady
- 'B' MFWP Speed 4300 rpm and steady

Subsequently
- ATC reports 'B' MFWP Speed 3800 rpm and lowering
- There is no flow indicated to 'B' Steam Generator
- 'B' Operating Range level is reading 24%

Which INITIAL procedural action is taken per 1203.027, Loss of Steam Generator Feed?

A. Open Feedwater Pump Discharge Crosstie CV-2827

B. Trip 'B' MFW Pump

C. Manually actuate EFW

D. Take manual control of 'B' MFW Pump and raise speed

Answer:

B. Trip 'B' MFW Pump

Notes:

B is the correct action per the Loss of SG Feed AOP to take for a loss of flow to B SG with MFWP speed lowering but level still close to normal.

A is incorrect but plausible since this action could be taken manually and would occur automatically on a MFWP trips.
C is incorrect but plausible since this action is taken if feed is lost to a SG and cannot be recovered and a Rx trip is required.
D is incorrect but plausible since it seems feasible but this is not procedurally addressed in 1203.027.

References:

1203.027, Loss of Steam Generator Feed

History:

New for 2014 Exam.

INSTRUCTIONS

1. **IF either of the following conditions apply:**

   - SG level is <15" with no indication of recovery, or
   - Main feedwater flow is lost to either SG with no indication of recovery and power is >7%.

   THEN trip the reactor AND follow Emergency Operating Procedure (1202.001).

2. **IF both MFWPs are running**
   AND 1 MFWP has failed without tripping,
   THEN manually trip the bad MFWP.

3. **IF only one MFWP is operating,**
   THEN verify that Feedwater Pumps Disch Crosstie (CV-2827) is open.

4. **Verify ICS reduces power**
   OR manually reduce power to within capacity of available feedwater.

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

5. **Open Pressurizer Spray (CV-1008) in MAN as necessary.**

6. **WHEN RCS pressure starts to drop,**
   THEN verify Pressurizer Spray Control Mode switch in AUTO.

7. **Verify CV-1008 closes per one of the following setpoints OR isolate it, as necessary, by closing the Spray Isolation (CV-1009).**

   - Normal operation: Closes - 2155 psig
   - Power >80% AND MFWP trip: Closes - 2030 psig

8. **Attempt to determine cause of loss of feed and correct it.**

(continued)

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Effect of battery discharge rates on capacity.

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

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	C

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

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

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

A. The battery amperage will be fairly constant until the design battery capacity is exhausted.

B. The battery amperage will drop steadily until the design battery capacity is exhausted.

C. The battery amperage will rise steadily until the design battery capacity is exhausted.

D. The battery amperage will be fairly constant until the design battery capacity is exhausted and then will rapidly drop.

Answer:

C. The battery amperage will rise steadily until the design battery capacity is exhausted.

Notes:

P=IE; As the battery discharges under a constant load, battery voltage will drop and current (battery amperage) will rise, therefore C is the correct answer.
The other answers are plausible if the examinee doesn't understand electrical concepts.

References:

ELP-NLO-ELEC1

History:

Developed by NRC. KA 055 EA1.05
Used on 2004 RO/SRO Exam.
Selected for 2005 RO re-exam.
Selected for the RO/SRO 2010 exam.
Selected for the 2014 Exam.

METHODOLOGY

CONTENTS

DESCRIBE THE FOLLOWING ELECTRICAL PARAMETERS, INCLUDING THE UNIT OF MEASUREMENT AND THE RELATIONSHIP TO OTHER PARAMETERS: (3) • **VOLTAGE** • **CURRENT** • **RESISTANCE** • **RESISTIVITY** • **POWER** • **INDUCTANCE** • **CAPACITANCE**

POWER

- Electricity is generally used to do some sort of work, such as turning a motor or generating heat. Specifically, *power* is the rate at which work is done, or the rate at which heat is generated. The unit commonly used to specify electric power is the watt.
- In equations, you will find power abbreviated with the capital letter P, and watts, the units of measure for power, are abbreviated with the capital letter W. Power is also described as the current (I) in a circuit times the voltage (E) across the circuit.
  $P = I \times E$ or $P = IE$
- Using Ohm's Law for the value of voltage (E),
  $E = I \times R$
  And using substitution laws,
  $P = I \times (I \times R)$
- Power can be described as the current (I) in a circuit squared times the resistance (R) of the circuit.

  $P = I^2 R$

BASIC DC THEORY:

STATE THE PURPOSE OF A RECTIFIER (11)

RECTIFIERS

- Most electrical power generating stations produce alternating current. The major reason for generating AC is that it can be

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

QID: 0339 **Rev:** 0 **Rev Date:** 9/7/99 **Source:** Direct **Originator:** D Slusher

TUOI: A1LP-RO-ELECD **Objective:** 13C **Point Value:** 1

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

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 inverter swapping.

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

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

Group: 1 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** C

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

Inverters are aligned with Y-25 supplying RS-4 and Y-22 supplying RS-2.
Shifting the manual output transfer switch (S-2) on the Y-25 inverter to the "System Output To Y-22" position would:

A. Power RS-2 from Y-25.

B. De-energize RS-4.

C. Parallel RS-2 and RS-4.

D. Damage the Y-25 inverter.

Answer:

B. De-energize RS-4.

Notes:

"b" is the only correct answer since Y25 is a "swing" inverter with the capability of supplying either RS-2 or RS-4 but not both. Shifting the manual output transfer switch without another inverter available to power RS-4 will de-energize RS-4.
"a" and "c" are incorrect, Y-25 will not power RS-2 until the manual output transfer switch on Y22 is placed in the "Y25" position.
"d" is incorrect, the load on Y-25 will be lost.

References:

STM 1-32, Rev. 41

History:

Used in 1999 exam.
Direct from ExamBank, QID# 4510
Used in 2005 RO exam
Selected for 2014 RO exam

4.3.1.2 Simplified Inverter Operation

(SEE FIGURE 32.54)

The inverter can be considered as four blocks: input, output, oscillator, and power switching circuit. Inverter operation of the various inverters is essentially identical

The input to the inverter is DC from either D01 or D02. The DC input is filtered to maintain a smooth DC. The filtered DC is supplied to an SCR type static inverter.

The SCR, when gated on, will supply full power to the output. Both positive and negative SCRs are used to produce a square wave output. The oscillator block controls the frequency of the SCR output.

The oscillator generates gating pulses to control the switching of the SCR's. The oscillator frequency is controlled such that the inverter output frequency is maintained the same as that of the alternate AC source. Gating pulses are alternately applied to the positive and negative SCRs to reverse to generate a square wave.

The square wave is regulated and filtered by a constant voltage transformer (CVT) in the output block. The CVT maintains a steady output voltage. The output of the CVT is also a sine wave with very little noise.

Inverter Y-11 is the normal supply to RS-1 and inverter Y13 is the normal supply to RS-3. Inverter Y-22 is the normal supply to RS-2 and C540 and inverter Y-24 is the normal supply to RS-4. Inverters Y-15 and Y-25 are swing inverters. Inverter Y-15 can supply power to either RS-1 or RS-3. Inverter Y-25 can supply power to either RS-2 and C540 or RS-4. To shift RS power from the normal inverter to the swing inverter, the inverters must be placed on the alternate AC source. The inverters are verified to be in sync using the sync indicating lights on Y-11, Y-13, Y-22 or Y-24 (whichever is being transferred to the swing inverter). Then manual transfer switches, at the top of the inverters, are aligned to supply RS from Y-15 or Y-25. Y-15 and Y-25 may supply only one of the vital panels at one time.

K

The inverters are equipped with relaying which provides an inverter trouble alarm on Control Room annunciator should one of the following conditions occur:

- DC Input Undervoltage
- DC Input Overvoltage
- Inverter Output Undervoltage
- Inverter Output Overvoltage
- Inverter Failure
- Out of Sync
- Fan Failure
- Static Switch Transferred
- System Overtemperature
- Bypass Failure

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

QID: 0890 **Rev:** 0 **Rev Date:** 9/4/14 **Source:** New **Originator:** Possage

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

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

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

Description: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The length of time after the loss of SWS flow to a component before that component may be damaged.

K/A Number: AA2.06 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** K

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

Given:
- RCS Pressure 1500 psig
- ESAS Channels 1-4 actuated
- CV-3822, Decay Heat Removal Cooler Service Water E-35A Inlet has failed closed

Per 1104.004, Decay Heat Removal Operating Procedure, what is the MAXIMUM length of time P-34A, Decay Heat Removal Pump, can operate in this condition?

A. 5 minutes

B. 10 minutes

C. 15 minutes

D. 20 minutes

Answer:

D. 20 minutes

Notes:

Per 1104.004, Decay Heat Removal Operating Procedure, limit and precaution 5.8 states that operation in recirculation mode without Service Water shall not exceed 20 minutes. This makes only answer D correct. Answers A, B, and C are in 5 minute increments leading up to 20 minutes.

References:

1104.004, Decay Heat Removal Operating Procedure

History:

New for 2014 Exam.

4.3.5 P 11697, Emergency Procedures Revised to Require Isolation of ECCS Pump Rooms During Initiation of Recirculation Phase of LOCA Cooling. Contained in "DH System Aux Spray Alignment Prior to RB Sump Recirc" section.

4.3.6 P 227, GL 88-17 implement recommendations concerning loss of decay heat removal in the normal operating procedure including entry into and operation in reduced inventory.

5.0 LIMITS AND PRECAUTIONS

5.1 See Tech Spec for limits on system operability, testing, shutdown of equipment for maintenance during power operation and allowed system leakage.

5.2 The Decay Heat system shall be isolated from the RCS when RC temperature is >280°F.

5.3 Decay Heat pump discharge relief setpoint is 445 +22.5/-13.35 psig. Discharge pressure should be maintained <400 psig to prevent challenging the relief.

5.4 Maximum pressure at the Decay Heat pump suction is:

- 300 psig with suction from RCS
- 75 psig with suction from BWST or RB sump

5.5 If the RCS is opened for refueling or maintenance, an inadvertent injection from the BWST will cause the reactor vessel or refueling canal level to rise, endangering personnel and equipment.

5.5.1 When the RCS has been cooled and depressurized, the BWST shall be isolated from the Decay Heat system by closing the following valves associated with the operating pump. Single valve isolation is permissible for brief periods such as valve stroke testing.

- Decay Heat P-34A Suction from BWST (CV-1436)
- BWST T-3 Outlet (CV-1407)
- Decay Heat P-34B Suction from BWST (CV-1437)
- BWST T-3 Outlet (CV-1408)

5.6 Maximum allowable flow per pump shall not exceed 4000 gpm.

5.7 To prevent pump cavitation during ES mode, flow should be throttled only as necessary to reduce flows to within normal indicated flow bands.

5.8 Operation in recirculation mode (flow from cooler outlet back to pump suction only) without Service Water cooling shall not exceed 20 minutes.

QID: 0346 **Rev:** 1 **Rev Date:** 3/11/04 **Source:** Direct **Originator:** D. Slusher

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

Section: 4.2 **Type:** Generic APE

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

Description: Knowledge of abnormal condition procedures.

K/A Number: 2.4.11 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 2.5

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** K

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

In accordance with 1203.024, Loss of Instrument Air, what is the lowest instrument air pressure that Unit 1 and Unit 2 instrument air systems should remain crossconnected?

A. 80 psig

B. 60 psig

C. 55 psig

D. 35 psig

Answer:

B. 60 psig

Notes:

B is the correct value per 1203.024.
A is incorrect but plausible as this is the value at which Breathing Air can be cross-connected with IA.
C is incorrect but plausible as this is the value at which the TBV's will fail closed.
D is incorrect but plausible since this is the value at which IA is considered lost and the Rx should be tripped.

References:

1203.024, Loss of Instrument Air

History:

Used in 1999 exam.
Direct from ExamBank, QID# 763 used in class exam
Used on 2004 RO/SRO Exam. KA 065 AK3.04
Selected for 2014 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 RP 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**.	

QID: 0896 **Rev:** 0 **Rev Date:** 9/10/14 **Source:** New **Originator:** 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: Knowledge of the reasons for the following responses as they apply to the (Inadequate Heat Transfer): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

K/A Number: EK3.1 **CFR Reference:** 41.5 / 41.10, 45.6, 45.13

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

Group: 2 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** K

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

Given:
- 1202.004, Overheating EOP has been entered due to loss of all feedwater.
- ALL four RCPs are running

What is the PRIMARY reason for reducing the number of running RCPs to one in each loop in an overheating condition?

A. To ensure Pressurizer spray is available to aid in RCS pressure reduction.

B. To limit heat input into the RCS while maintaining forced flow.

C. To ensure even mixing of RCS due to rising boron concentration in RCS.

D. To maintain flow to limit stresses on SGs from high tube to shell DT.

Answer:

B. To limit heat input into the RCS while maintaining forced flow.

Notes:

B is correct per AREVA EOP technical bases document.
A is incorrect but plausible since RCS pressure will be high during an Overheating event. This is NOT the reason for reducing the number of RCPs to one per loop but is the reason it is preferred to leave "C" RCP running.
C is incorrect but plausible since boron concentration will be rising due to HPI. This is an excellent reason for running RCPs but boron concentration can be changed in the RCS with just Decay Heat pumps running AND this is NOT the reason RCPs are reduced to one per loop.
D is incorrect but plausible since tube to shell DT is a concern but ALL RCPs are tripped if the limit is reached.

References:

1202.004, Overheating
B&W GEOG Bases, Vol. 2, Part III.C

History:

New for 2014 Exam.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

P-32A and P-32C should be left running if available.

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 ≤ 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 ≥ 2450 psig
- RCS press approaches NDTT Limit (Figure 3)
- Secondary feed **not** expected to become available

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)

4.0 REDUCE RCP OPERATION TO A 1/1 CONFIGURATION AND RUN AS LONG AS SCM EXISTS AND SG TUBE – SHELL ΔT LIMITS ARE NOT EXCEEDED.

Indicators and Controls

Indicators:
- RCP status
- RCS pressure
- RCS temperature (T_{hot}, T_{cold}, incore thermocouples)
- SCM monitor
- SG shell temperatures
- P-T display
- SPDS

Controls:
- RCP motor controls

Purpose of Step

The purpose of this step is to limit the heat input to the RCS while maintaining forced flow in both loops.

Bases

FW is not yet restored, but RCS conditions do not yet require HPI cooling. Two RCPs are left running to reduce total heat input to the RCS. The preferred configuration is one RCP per loop so that forced flow exists in both SGs when feedwater is restored. Since it is not known in which SG(s) feedwater will be restored, or if it will be restored, one RCP should be left running in each loop if possible. The selection of RCPs to run should consider pressurizer spray flow capacity. While no heat transfer exists, SG tube-shell ΔT limits could be reached. If this occurs, the RCPs should be tripped to slow the temperature changes in the SG tubes.

The GEOG values for tube-shell ΔT limits (provided in Volume 3) are control parameters and therefore do not require error correction.

Sequence

There is no specific sequence requirement.

TBD Volume 3 References

III.C.2.4 and III.G.3.6

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

QID: 0891 **Rev:** 0 **Rev Date:** 9/4/14 **Source:** New **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:** Yes	**Taxonomy:** An

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

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 should be recommended to the CRS to alleviate these oscillations?

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

B. Place the EHC controls in Turbine Manual

C. Place the SG/Rx Master back in Automatic

D. Place the Diamond panel 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 might recall that a grid disturbance is the cause of the problem but changing the generator field voltage will not remove 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.
Answer D is incorrect but plausible if the examinee believes placing rod control in automatic will allow the ICS to counteract the perturbations.

References:

STM 1-24, Turbine Controls and Auxiliaries

History:

New for 2014 Exam.

computation on reference signals from the Operator's Panel, ICS, and turbine instrumentation feedback signals. It then generates an amplified output signal to the steam valve actuators. The cabinets also contain power supplies, blowers and terminal strips for interconnecting system wiring.

7.3 Basic EH Control Functions

(Fig. 17 and 18) The Electro-Hydraulic Control System (EHC) operates in one of two basic modes, Speed Control or Load Control. The position of the generator output breakers determines the mode of EHC operation. When both generator output breakers are open the EHC operates in Speed Control. When either generator output breaker is closed the EHC operates in Load Control.

A simplified speed control diagram is shown in Figure 24.17. When the turbine generator is initially latched the generator output breakers are open and the EHC is in speed control. Further, turbine generator speed can be controlled by either the throttle valves or the governor valves. In order to control speed on both the throttle valves and governor valves the EHC contains a Throttle Valve Controller and a Governor Valve Controller.

The Throttle Valve Controller and the Governor Valve Controller both consist of an automatic controller and a manual controller. During automatic control the operator inputs the desired speed setpoint into the summing (Σ) junction. Turbine generator speed is sensed by a speed probe at the turbine front standard and processed by the Speed Channel. The resulting turbine generator speed signal is input into the summing (Σ) junction. The desired speed and the turbine generator speed signal are opposite in polarity. Therefore, the summing junction output will be the difference between the desired speed and the turbine generator speed signal or Speed Error. The speed error is amplified by the speed error amp.

The speed error amp output is supplied to the Percent Regulation Amplifier and the Throttle Controller Gain circuit. The EHC is a proportional controller. Recall the operation of a proportional controller. If turbine generator speed is equal to the speed setpoint then the speed error is zero. The corresponding throttle/governor valve position would be closed. As speed error increases the throttle/governor valve open to return turbine generator speed to speed setpoint but a speed error must exist to maintain the throttle/governor valves open to control turbine generator speed. The Throttle Controller Gain and the Percent Regulation Amplifier both adjust how closely the turbine generator speed is maintained to the speed setpoint. The higher the gain the closer the turbine generator speed will be to the desired speed setpoint.

The Percent Regulation Amplifier output is supplied to the Automatic Governor Valve Controller and the Throttle Controller Gain circuit output is supplied to the Automatic Throttle Valve Controller. The automatic governor and throttle valve controllers convert the speed error to a corresponding Governor/Throttle valve position signal. The Valve Position Demand signal is input through an Auto/Manual Transfer Switch to the Governor/Throttle Valve Servos. Once the operator sets the desired turbine generator speed the Automatic Governor/Throttle Valve Controllers will maintain the

feedback. Failures of LVDT electrical components and mechanical linkages have caused large, rapid load swings and plant trips in the past.

* Gov. Control Signal: Indicates the total signal being sent to the turbine governor valves. At 100% all governor valves should be at their full load position.
* Valve Position Limit: Setpoint for the governor valve total open signal limit. When the total governor valve signal is equal to or greater than the valve position limit signal, the limiter will not allow the governor valves to open any further.
* Turbine Shaft Speed: Indicates the signal output of the Auxiliary Speed Channel.
* Reference Display: Indicates the signal being sent to the turbine throttle or governor valves. With the generator output breaker open, this value is turbine speed. With the breaker closed, the signal is the voltage signal to the servo valve. In all modes the setter is changed directly by either the operator or ICS. In Operator Auto, the reference display is used in conjunction with the setter display and the reference control pushbuttons.
* Setter Display: This indicator in all turbine modes except Operator Auto tracks the reference display. In Operator Auto, the operator sets the desired value in the setter using the reference control pushbuttons. When the GO pushbutton is depressed, the reference starts counting up to match the setter as the signal is sent to the turbine valves.
* Generator Electric Load: Indicates the generator output as sensed by the Megawatt Transducers. In addition to indication, this also represents the signal being supplied to the overspeed protection controller.

7.8.2 Operator's Panel Lower Section

(Fig 10)

* Throttle Pressure Controller: The TPC OUT or TPC IN pushbuttons are used to place the Throttle Pressure Controller in or out of service. With TPC IN, if throttle pressure drops to 835 psig the governor valves will run back to <20% open or until throttle pressure is above the 835 psig setpoint. Normally, the system is operated "TPC Out".
* Turbine Mode Control Pushbuttons:
  - Turbine Manual: Sends direct signals to the throttle or governor valves. The Turbine Manual pushbuttons for the throttle valves or governor valves will be back lighted depending on the turbine valve control mode in service. In this mode, the reference and setter displays are matched. No reference or feedback signals are used in this mode.
  - Operator Auto: In Operator Auto the reference control pushbuttons are illuminated. The EHC automatic controller is operable, controlling the turbine throttle and governor valves. The reference feedback signals are active in Operator Auto. Load changes are made by selecting the desired value in the setter display using the reference
  - Control Raise/Lower pushbuttons. When the desired value is entered in the setter, the operator depresses the GO pushbutton to start the change. The reference display starts

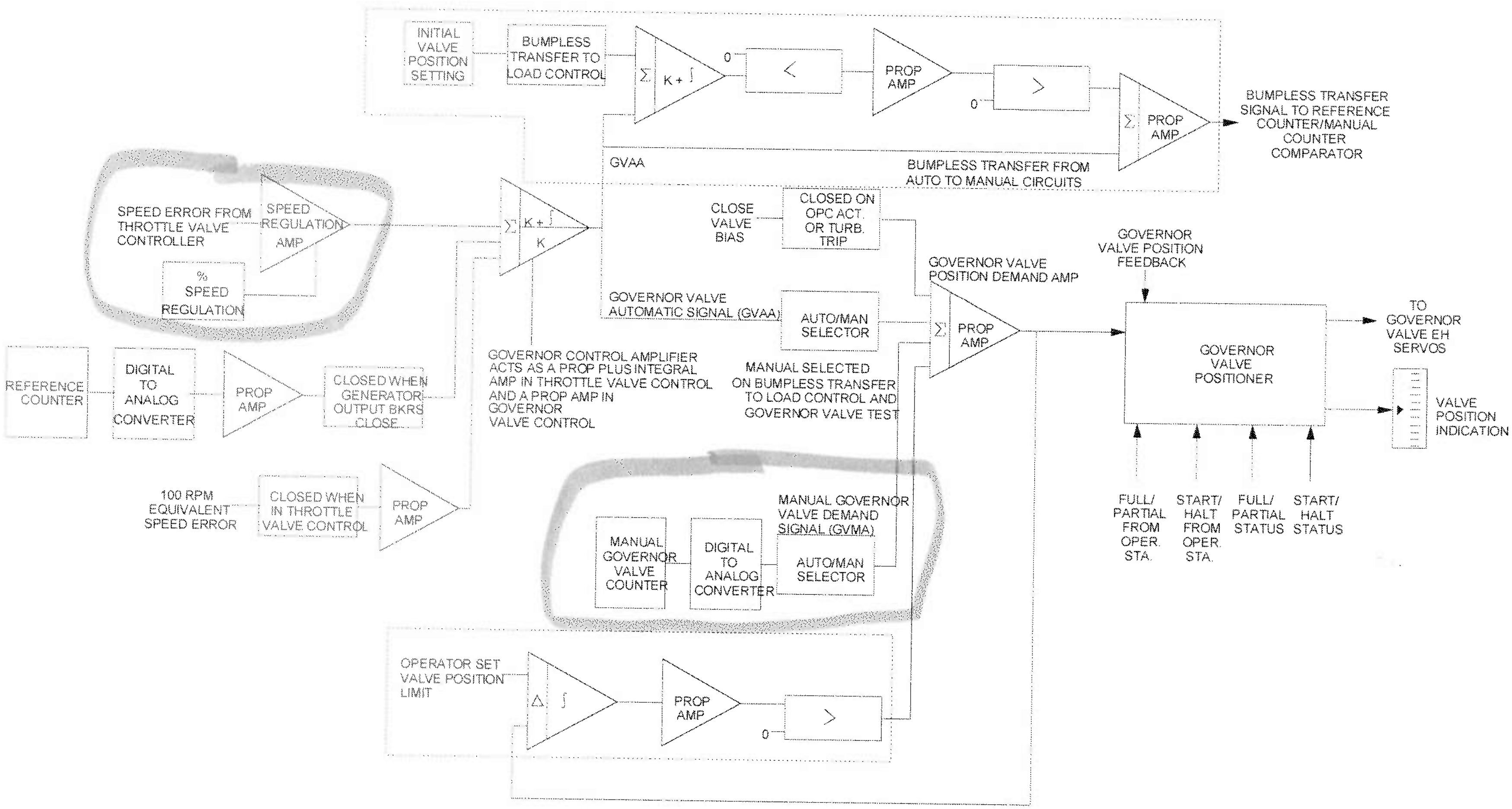


FIGURE 24.42: EHC GOVERNOR VALVE CONTROLLER

ANO UNIT 1 – 2014

TIER 1

GROUP 2

Questions 19 - 27

QID: 0897 **Rev:** 0 **Rev Date:** 9/10/14 **Source:** Modified **Originator:** Cork

TUOI: A1LP-RO-CRD **Objective:** 22 **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: Relationship of reactivity and reactor power to rod movement.

K/A Number: AK1.03 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1	**RO Imp:** 3.5	**RO Select:** Yes	**Difficulty:** 2
Group: 2	**SRO Imp:** 3.8	**SRO Select:** Yes	**Taxonomy:** K

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

In accordance with 1203.003, Control Rod Drive Malfunction Action, which of the following dropped or misaligned control rod scenarios would require a reactor shutdown instead of a control rod recovery attempt?

A. A dropped rod at 62% power

B. A dropped rod at <2% power

C. A rod misaligned by >6.5% for >1 hour

D. A rod misaligned for >24 hours

Answer:

B. A dropped rod at <2% power

Notes:

B is the correct answer since a dropped control rod at <2% power could cause the reactor to reach a subcritical state. Recovery of a dropped rod/rods from a subcritical condition can result in uncontrolled criticality and unanalyzed control rod configurations per 1203.003.
A is incorrect but plausible since a runback will occur if a rod drops when >60% power. A is incorrect because a single dropped rod when >2% power can be recovered per 1203.003.
C is incorrect since a misaligned rod can be recovered but is plausible since a power reduction is required at 2%/min to 60% of allowable thermal power.
D is incorrect but plausible since a fuel damaging event occurred at ANO with a misaligned rod. Procedural guidance is given to recover a rod misaligned >24 hours so this is incorrect.

This question was modified by adding the procedure reference. Distractor A was modified by changing power level. Distractor C was replaced with the current wording.

References:

1203.003, Control Rod Drive Malfunction Action

History:

Developed for 1998 RO/SRO Exam.
Modified QID 22 for 2014 Exam

SECTION 2
DROPPED ROD – REACTOR CRITICAL

INSTRUCTIONS

CAUTION

Recovery of dropped rod/rods from a subcritical condition can result in uncontrolled criticality and unanalyzed control rod configurations.

NOTE

Dropped rod actions also apply for control rods which did not latch and failed to withdraw resulting in asymmetric conditions during reactor startup.

1. IF dropped rod(s) exist
AND NI power is <2%,
THEN shutdown the reactor by inserting all regulating control rods in SEQ.

A. IF the reactor was shutdown by inserting regulating rods,
THEN place plant in Mode 3, >525°F per the applicable steps of Power Reduction and Plant Shutdown (1102.016).

2. IF more than one rod drops
AND NI power is ≥2%,
THEN trip the reactor and follow Emergency Operating Procedure series (1202.XXX).

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

NOTE

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) IF dropped safety rod
AND required to place CV-1248 in BLEED,
THEN verify CV-1250 NOT required AND closed.

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

QID: 0022 **Rev:** 0 **Rev Date:** 7/7/98 **Source:** Direct **Originator:** JCork

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

Section: 4.2 **Type:** Generic APEs

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: Relationship of reactivity and reactor power to rod movement.

K/A Number: AK1.03 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 2

Group: 2 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

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

Which of the following dropped or misaligned control rod scenarios would require a reactor shutdown instead of a control rod recovery attempt?

a. A fully dropped rod at 38% power.

b. A dropped rod at <2% power.

c. A partially dropped rod at 10% power.

d. A rod misaligned for >24 hours.

Answer:

b. A dropped rod at <2% power.

Notes:

(b) is the correct answer since a dropped control rod at <2% power could cause the reactor to reach a subcritical state. Recovery of a dropped rod/rods from a subcritical condition can result in uncontrolled criticality and unanalyzed control rod configurations.
(a) and (c) are incorrect because a single dropped rod when >2% power can be recovered safely. (d) is incorrect because procedural guidance is given to recover a rod misaligned >24 hours.

References:

1203.003 Rev. 19, page 3 & 4.

History:

Developed for 1998 RO/SRO Exam.

PARENT

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

QID: 0893 **Rev:** 0 **Rev Date:** 9/5/14 **Source:** New **Originator:** Possage

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

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

System Number: 005 **System Title:** Inoperable/Stuck Control Rod

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

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

Tier: 1 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** K

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

What is the reason for Rod Insertion Limits?

A. Ensures minimum required SDM is maintained

B. Ensures radial symmetry in the core power distribution

C. Ensures rod worth is approximately the same between the rod groups

D. Ensures RCS peak design pressure will not be exceeded during a rod ejection at low power

Answer:

A. Ensures minimum required SDM is maintained

Notes:

Answer A is correct and is consistent with the definition of shutdown margin in Tech Specs and the COLR. The COLR describes how Rod Insertion Limits preserve SDM..
Answer B is incorrect but plausible since radial symmetry is a component of core power distribution which a misalsigned control rod can lead to problems with quadrant power tilt but the QPT Tech Spec LCO adresses this issue.
Answer C is incorrect but plausible, sequential rod movement levelizes rod worth.
Answer D is incorrect as shown in the basis for answer A but plausible since a stuck rod does cause problems with rod worth and RCS peak pressure is a concern with a startup accident.

References:

Technical Specifications
COLR
STM 1-02, Control Rod Drive System

History:

New for 2014 Exam

1.1 Definition (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and c. There is no change in APSR position.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

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 actual shutdown margin to be ≥ 1 %Δk/k.

1.2.2 Control Modes

Two modes of control are allowed by the Control Rod Drive (CRD) system. Refer to figure 2.1 CRD system block diagram, page **Error! Bookmark not defined.**. Automatic mode allows the Integrated Control System (ICS) to move the group or groups selected by the sequence enabling circuits (Groups 5, 6, or 7) to control reactor power. The Manual Mode allows the operator to move any of the rods from the "Diamond" control panel.

Because the safety groups (1-4) are normally held in the full out position, they are powered from a static DC power supply. This power supply does not have the ability to move the rods and is referred to as the DC hold supply. Each of the other groups (5-8) has its own individual supply. When it becomes necessary to change the position of the safety groups, an auxiliary supply is used. The system is designed to prevent more than one group from being driven off the auxiliary power supply at the same time. Single or multiple combinations of rods within a group may be moved by the auxiliary supply but it is not designed to handle more than twelve CRDM's at a time. Clamping contactors, transfer relays and visual light indicators are used to aid the operator while making transfers to and from the auxiliary supply.

For startup, groups 1 - 4 are fully withdrawn, and then group 5 withdrawal is started. The regulating groups are normally withdrawn in sequence; as group 5 reaches a specified position, group 6 will start withdrawing, and then group 7 will commence withdrawing when group 6 has reached a predetermined position. The sequencing does not allow more than two numerically adjacent groups to move at any one time and then only during the first and last 20% of travel. This provides a more even reactivity change because adjacent control rod worth ($\Delta k/k$) is smaller at the extremes of their travel.

The safety groups are only controlled in manual, while the regulating groups may be controlled automatically or manually. The automatic mode is normal for power operations greater than 20%.

Group 8 may be moved by the operator manually at anytime regardless of the Diamond Panel's selection of auto or manual.

The CRD System is designed in accordance with the reactivity control limits established for both normal and emergency operating procedures. Sufficient control is available to produce an adequate shutdown margin. An available hot shutdown margin of 1.5% $\Delta k/k$ is maintained throughout core life assuming the highest worth CRA is stuck out of the core.

2.0 Detailed System Description

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

QID: 0895 **Rev:** 0 **Rev Date:** 9/5/14 **Source:** New **Originator:** Possage

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

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

System Number: 024 **System Title:** Emergency Boration

Description: Ability to determine and interpret the following as they apply to the Emergency Boration: When boron dilution is taking place.

K/A Number: AA2.06 **CFR Reference:** 43.5 / 45.13

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

Group: 2 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** An

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

Given:
- Reactor tripped 30 minutes ago due to loss of condenser vacuum
- Condenser vacuum is currently 18" Hg
- Group 6 Rod 4 did not fully insert
- Group 5 Rod 3 did not fully insert
- Emergency Boration in progress

During a brief Critical Parameters are given as follows:
- Pressurizer Level 220 inches
- Both S/G Levels 30 inches
- Both S/G Pressures 1020 psig
- SCM 70 °F
- CET 555 °F
- RCS Pressure 2155 psig
- Source Range 1000 cps

Which of the following is too high for the given conditions?

A. Pressurizer level

B. Subcooling Margin

C. S/G Pressure

D. Source Range

Answer:

D. Source Range

Notes:

D is correct, normally Source Range counts should be ≤50 cps 30 minutes after a trip so the flux level is too high even for an event that includes 2 stuck control rods which indicates that the boration is not being fully effective.
A is incorrect, although Pressurizer level is above normal post trip level of 100" but with Emergency Boration in progress the level is raised to 220".
B is incorrect, although this SCM seems high compared to the minimum value of 30°F, this is normal post trip SCM.
C is incorrect, normal post trip pressure is 995 psig so this seems high but with the loss of condenser vacuum the ADVs are controlling and their setpoint is 1020 psig.

References:

1202.001, Reactor Trip
1202.012, RT-12 Emergency Boration

History:

New for 2014 Exam

EMERGENCY BORATION

1. (Continued)

H. Start Batch Controller by depressing RUN key.

I. Adjust Batch Controller Flow CNTRL VLV (CV-1249) to 100% open as follows:

1) Depress VALVE SET.

2) Depress numbers: 1, 0, 0.

3) Depress ENTER.

4) Depress lower DISPLAY.

5) Depress RATE.

J. **IF** Batch Controller output rate < 5 gpm,
THEN perform the following:

1) Stop running Boric Acid pump(s):

- P39A
- P39B

2) Close Batch Controller Outlet (CV-1250).

3) Stop Batch Controller by depressing STOP key.

4) **GO TO step 2.**

K. Adjust Pressurizer Level Control Setpoint to 220".

L. Verify BWST T3 Outlet to OP HPI pump (CV-1407 or CV-1408) open.

M. **WHEN** PZR level is ≥ 100",
THEN establish maximum Letdown flow allowed by cooling capacity and component limitations.

(1. CONTINUED ON NEXT PAGE)

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. **Advise Shift Manager to implement Emergency Action Level Classification (1903.010).**	
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".**	
8. **Control RCS press within limits of Figure 3 (RT-14).**	

INSTRUCTIONS	CONTINGENCY ACTIONS
17. Check Turb BYP Valves operate to maintain SG press 950 to 1020 psig.	17. Operate TURB BYP Valves in HAND to maintain SG press 950 to 1020 psig.

A. **IF** TURB BYP Valves are failed closed, **THEN** perform the following:

1) **IF** Condenser Vacuum ≥ 23", **THEN** depress Low Vacuum Reset PBs on C02:

- HS-2850
- HS-2851

a) Verify TURB BYP Valves operate in HAND.

b) Place TURB BYP Valves in AUTO:

- TURB BYP Valves Loop A
- TURB BYP Valves Loop B

2) **IF** Condenser Vacuum is < 23"
OR
TURB BYP Valves fail to operate after Low Vacuum Reset PBs are depressed,
THEN perform the following:

a) Verify associated ATM Dump ISOL open:

SG A	SG B
CV-2676	CV-2619

b) Verify ATM Dump Control System controls SG press 1000 to 1040 psig.

SG A	SG B
CV-2668	CV-2618

(17. CONTINUED ON NEXT PAGE)

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

QID: 0423 **Rev:** 0 **Rev Date:** 4/25/2002 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-EOP01 **Objective:** 11 **Point Value:** 1

Section: 4.3 **Type:** Babcock & Wilcox EPEs/APEs

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

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

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

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

Group: 2 **SRO Imp:** 3.3 **SRO Select:** Yes **Taxonomy:** A

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

A power escalation in progress, current power is 50%

The following annunciators alarm and conditions are present:

- MSSV OPEN (K07-C5)
- MAIN STEAM PRESSURE HI/LO (K07-C6)
- PZR LEVEL HI (K09-D3)
- RCS PRESSURE HI/LO (K09-C2)
- GENERATOR L.O. RELAY TRIP (K04-A8)

What operator actions are required?

A. Reduce reactor power to within capacity of main turbine load.

B. Monitor runback to 40% load.

C. Trip the reactor and go to 1202.001.

D. Open Pressurizer Spray (CV-1008) in MAN.

Answer:

C. Trip the reactor and go to 1202.001.

Notes:

Answer "c" is correct since the reactor should have tripped due to indications of a turbine trip with reactor power > 43%.
Answer "a" is incorrect, although this action is in load rejection AOP, the presence of K04-A8 indicates a turbine trip should be present.
Answer "b" is incorrect, runbacks occur for other reasons but not in this instance.
Answer "d" is incorrect, although this action is in load rejection AOP, the presence of K04-A8 indicates a turbine trip should be present.

References:

1202.001, Reactor Trip

History:

New for 2002 RO/SRO exam.
Selected for 2014 Exam

ENTRY CONDITIONS

- An automatic reactor trip or DSS trip.
- Failure of RPS to trip the reactor upon reaching a limit listed below:
  - High power 104.9%
  - High power/pumps .... one pump per loop .... ≥ 55%
OR
0 pumps in one loop .... ≥ 0%
  - High power/imbalance/flow .... COLR Figure
  - High RCS temp .... ≥ 618°F (T-hot)
  - High RCS press .... ≥ 2355 psig
  - Low RCS press .... ≤ 1800 psig
  - Variable low RCS press .... COLR Figure
  - High RB press .... ≥ 18.7 psia
  - Turbine trip .... reactor power ≥ 43% **AND** Turbine is tripped
  - Both MFW pumps trip .... reactor power ≥ 9% **AND** both MFW pumps tripped
- Manual trip of the reactor is required due to reaching a limit listed below:
  - PZR level dropping < 100",
AND
no indication of recovery
  - PZR level > 290"
  - Any MSIV closure at power
  - Either SG level < 15" or > 95%,
AND
no indication of recovery
  - A system degradation that requires manual reactor trip based on operator judgment
  - Abnormal Operating Procedure requirement
- **IF** a system degradation occurs while shutdown, above DHR operation,
THEN perform applicable steps

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

QID: 0894 **Rev:** 0 **Rev Date:** 9/5/14 **Source:** New **Originator:** Possage

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

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

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: S/G tube failure

K/A Number: AA2.06 **CFR Reference:** 43.5 / 45.13

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

Group: 2 **SRO Imp:** 4.5 **SRO Select:** Yes **Taxonomy:** Ap

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

Following a reactor trip, MSLI on 'A' SG had to be actuated to correct an Overcooling

The following are observed 20 minutes later:
- TAVE 532 °F and stable
- Pressurizer Level 90 inches and rising at 1 inch/min
- 'B' TBVs are being controlled in Manual
- M/U Flow 58 gpm
- L/D Flow 43 gpm
- Seal Inj Flow 40 gpm
- Seal Bleedoff Flow 6 gpm total
- RB Leak Detector RE-7461 80 cpm
- (T-37A) ICW Surge Tank - 0.8 psid and stable
- (T-37B) ICW Surge Tank - 0.8 psid and stable
- L/D Temperature 95 °F

Where is the RCS Leak and what is the rate?

A. L/D Cooler Leak 37 gpm

B. Steam Generator Tube Leak 37 gpm

C. L/D Cooler Leak 61 gpm

D. Steam Generator Tube Leak 61 gpm

Answer:

B. Steam Generator Tube Leak 37 gpm

Notes:

B is the correct location and leak rate. (See filled out Exhibit 1 from 1203.039)
A is incorrect based on steady ICW Surge Tank levels.
C and D are incorrect based on leak rate, 61 gpm would be the calcuclated leak rate if the applicant erroneously applied the change in pressurizer level.

References:

1203.039, Excess RCS Leakage

History:

New for 2014 Exam

EXHIBIT 1

Page 1 of 2
Revised 2/20/14

Estimate of RCS Leakrate

> **NOTE**
> Pressurizer level changes can either be an addition or subtraction to the estimated leak rate.

1. **Perform the following to determine RCS leakrate:**

   A. Place Pressurizer Level Control Valve (CV-1235) in HAND and establish Makeup Flow as directed by CRS/SM.

   B. IF applicable,
   THEN record current cooldown rate for leak estimation: N/A

   C. Calculate Seal bleedoff flow for RCPs:

   _____ + _____ + _____ + _____ = 6
   F1270 F1271 F1272 F1273 Total

   D. Input parameters into table below and calculate total leakage:

Makeup Flow	F1238/C04	58 gpm	**Plus**
Seal Injection Flow	F1239/C04	40 gpm	**Plus**
HPI Flow	SPDS/C16 and C18	gpm	**Plus**
Letdown Flow	F1236/C04	43 gpm	**Minus**
Seal Bleedoff flow	Total from above	6 gpm	**Minus**
Makeup Flow Due to Cooldown	Exhibit 1, page 2	-0- gpm	**Minus**
Pressurizer Level change	X 12.4 gal/in	-12.4 gpm	**Minus (IF rising)- Plus (IF lowering)**
	TOTAL	36.6 gpm	

   E. IF desired by CRS/SM,
   THEN place CV-1235 in AUTO.

> **NOTE**
> When the BWST is aligned to the Makeup Tank, Makeup Tank Level changes should generally not be used for leak rate estimation.

2. **IF BWST T-3 Outlets are closed,**
   AND desired to confirm leakrate,
   THEN calculate flow into the RCS based on Makeup Tank level change (30.86 gal/in).

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

QID: 0543 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Direct **Originator:** NRC

TUOI: A1LP-RO-EOP05 **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APE

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: Actions contained in EOP for High Reactor Coolant Activity.

K/A Number: AK3.06 **CFR Reference:** 41.5, 41.10 / 45.6 / 45.13

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

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

Given:

- Unit 1 is in the EOP for Inadequate Core Cooling, 1202.005.
- Region 4 of Figure 4 has been entered.
- Fuel damage is suspected.

Which of the following will result in HIGHER radiation exposures to personnel in the Aux Building?

A. Throttling RB spray before initiating sump recirculation

B. Initiating long-term cooling with DHR pumps

C. Aligning Pressurizer AUX spray to LPI system prior to initiating sump recirculation

D. Isolating Letdown by closing letdown orifice block (CV-1222)

Answer:

B. Initiating long-term cooling with DHR pumps

Notes:

"B" is correct. "A", "C", and "D" reduce exposure levels.

References:

1202.005 Inadequate Core Cooling, change 004-00-0,
1203.019 High Activity in Reactor Coolant, change 011-01-0.

History:

Developed by NRC.
Used on 2004 SRO Exam.
Selected for 2014 Exam.

SHIFT TO RB SUMP SUCTION

WARNING

If core is significantly damaged, initiation of sump recirculation may cause high radiation in areas near HPI, LPI, and RB Spray system piping.

CAUTION

- Failure to throttle RB Spray before initiating sump recirc may result in inadequate pump suction press.
- Full flow from both trains of HPI, LPI, and RB Spray can remove 6' of water from BWST in 5 minutes.

NOTE

If ES has actuated, individual component signals may be overridden as necessary to perform this task.

1. **Verify both Low Pressure Injection (Decay Heat) Pumps running:**

   - P34A
   - P34B

   A. **IF either** Low Pressure Injection (Decay Heat) Pump is unavailable,
   THEN stop associated HPI pump:

P34A	P34B
P36A/B	P36C/B

   B. Verify LPI (Decay Heat) Room Coolers running:

P34A Room	P34B Room
VUC1A or B	VUC1C or D

   C. Verify both Low Pressure Injection (Decay Heat) Blocks fully open:

   - CV-1400
   - CV-1401

QID: 0898 **Rev:** 0 **Rev Date:** 9/10/14 **Source:** New **Originator:** Possage

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

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

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

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

K/A Number: EA1.1 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 2 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** An

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

Given:
- RCS pressure 1800 psig
- CET average 600 °F
- The B S/G has been isolated
- Due to valve control issues, the ATC is manually feeding EFW
- A S/G Pressure 925 psig
- A S/G level 60"

Per RT-5, which of the following is closest to the MINIMUM rate the ATC is required to feed the A S/G?

A. 380 gpm

B. 480 gpm

C. 580 gpm

D. 680 gpm

Answer:

C. 580 gpm

Notes:

C is correct per RT-5 for a loss of subcooling margin with only one SG available while in manual control of EFW, flow rate must be >/= 570 gpm.
A is incorrect but plausible as this is close to the 370 gpm value if two SGs are available but is incorrect for only one SG.
B and D are incorrect, they are the A and C values plus 100 gpm.

References:

1202.002, Loss of Subcooling Margin
1202.012, Repetitive Tasks, RT-5

History:

New for 2014 Exam

INSTRUCTIONS	CONTINGENCY ACTIONS

CAUTION

Tripping all RCPs > 2 minutes after loss of adequate SCM could cause reactor core to become uncovered.

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Check elapsed time since loss of adequate SCM **<u>AND</u>** **perform the following:**	
A. <u>IF</u> ≤ 2 minutes have elapsed, **<u>THEN</u>** trip all RCPs: • P32A • P32C • P32B • P32D	A. <u>IF</u> > 2 minutes have elapsed, **<u>THEN</u>** leave currently running RCPs on. 1) Perform rapid cooldown per **step 20**, while continuing with this procedure.
B. Initiate full HPI (RT-3).	B. <u>IF</u> HPI flow is < full flow from one HPI pump **<u>AND</u>** RV Head void is indicated, **<u>THEN</u>** perform rapid cooldown per **step 20**, while continuing with this procedure.
1) <u>IF</u> Makeup Tank level drops below 18", **<u>THEN</u>** close Makeup Tank Outlet (CV-1275).	
C. Verify proper EFW actuation and control (RT-5).	C. Perform the following: 1) <u>IF</u> AUX Feedwater Pump (P75) is available, **<u>THEN</u>** perform the following: a) Verify at least one Condensate Pump (P2A, P2B, or P2C) in service. b) Dispatch an operator to open Aux FW Pump RECIRC to E-11A Isolation (FW-1). c) Open Feedwater Pumps DISCH Crosstie (CV-2827).

(1. CONTINUED ON NEXT PAGE)

VERIFY PROPER EFW ACTUATION AND CONTROL

1. Verify EFW actuation indicated on C09:

Train A:

- Bus 1
- Bus 2

Train B:

- Bus 1
- Bus 2

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

2. Verify at least one EFW pump (P7A or P7B) running with flow to SG(s) through applicable EFW CNTRL valve(s).

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

3. 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 SG levels 370 to 410".

(3. CONTINUED ON NEXT PAGE)

VERIFY PROPER EFW ACTUATION AND CONTROL

3. (Continued)

1) **IF** both SGs are available,
THEN verify SG level rising and tracking EFIC setpoint until 370 to 410" is established.

   a) **IF** EFW flow is less than adequate,
   THEN control EFW to applicable SG in HAND to maintain ≥ 340 gpm to applicable SG until level is 370 to 410".

   b) **IF** EFW flow is excessive
   AND
   > 340 gpm to either SG,
   THEN throttle EFW to applicable SG in HAND to limit SG depressurization. Do **not** throttle below 340 gpm on either SG until SG level is 370 to 410".

2) **IF** only one SG is available,
THEN feed available SG in HAND at ≥ 570 gpm until SG level is 370 to 410".

3) **IF** EFW is being controlled in HAND
AND
SG press drops below 720 psig due to EFW flow induced overcooling,
THEN continue feeding at required minimum rate
AND 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

   b) Place applicable EFW CNTRL valves in VECTOR OVERRIDE:

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

   c) Place applicable EFW ISOL valves in MANUAL.

SG A		SG B
CV-2627	P7A	CV-2620
CV-2670	P7B	CV-2626

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

QID: 0182 **Rev:** 1 **Rev Date:** 8/5/02 **Source:** Direct **Originator:** J. Selva

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

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

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

Description: Knowledge of abnormal condition procedures.

K/A Number: 2.4.11 **CFR Reference:** 41.10 / 43.5 / 45.13

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

Group: 2 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** K

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

A small break LOCA is in progress.

In accordance with 1203.041, Small Break LOCA Cooldown, which one of the following actions is required to be performed prior to the BWST level reaching 6 feet?

A. Secure running Reactor Coolant Pumps.

B. Align Pressurizer AUX Spray to LPI system.

C. Secure running High Pressure Injection Pumps.

D. Align one LPI train to gravity flow from RCS hot leg to RB sump.

Answer:

B. Align Pressurizer AUX Spray to LPI system.

Notes:

B is correct per 1203.041. Pressurizer AUX Spray must be aligned prior to sump recirc to limit personnel exposure once the primary fluids are allowed to enter the Auxiliary Building.
A. & C. are incorrect but plausible since RCPs and HPI pumps are secured during a LOCA based on meeting LPI flow criteria BUT not BWST level..
D incorrect but plausible since this action is required for boron precipitation concerns during a LOCA but is based on time from LOCA and not BWST level.

References:

1203.041, Small Break LOCA Cooldown

History:

Developed for use in 98 exam.
Selected for use in 2002 SRO exam. KA E08 EK1.2
Selected for 2014 Exam

CAUTION

If RCS is solid, a 1°F change in RCS temp can result in a 100 psig RCS press change.

NOTE

- If adequate SCM exists, maximum allowable RCS cooldown rates are:
  - 100°F/hr when > 300°F
  - 50°F/hr when 300 to 170°F
  - 25°F/hr when < 170°F
- If SCM is less than adequate, no cooldown rate limits apply.
- If cooling down by feeding ruptured SGs, SGs might not pressurize and SG level might not be established. Cooldown is controlled by adjusting feed rate.

8. **IF available,**
   THEN steam SGs as necessary to expedite RCS cooldown within applicable limits.

9. **Before BWST level reaches 6', perform the following:**

   A. Evacuate all unnecessary personnel from Auxiliary Building in preparation for RB sump recirculation.

   B. IF RB Spray is operating,
      THEN verify RB Spray flow throttled to maintain 1050 to 1200 gpm per train.

   NOTE

   Aligning Pressurizer Aux Spray to LPI system before going on sump recirc reduces personnel exposure if the boron precipitation mitigation lineup is required later.

   C. Dispatch an operator to align Pressurizer Aux Spray to LPI system using Decay Heat Removal Operating (1104.004), "DH System Aux Spray Alignment Prior to RB Sump Recirc" section.

WARNING

If core is significantly damaged, initiation of sump recirculation can cause high radiation in areas near HPI, LPI, and RB Spray system piping.

10. **IF BWST level reaches 6',**
    THEN switch LPI suction to RB Sump per Repetitive Tasks (1202.012), step 15.

11. **WHEN SG press is < 720 psig AND controlled,**
    THEN bypass MSLI by momentarily placing SG Bypass toggle switch on each EFIC cabinet initiate module in BYPASS.

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

QID: 0124 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork

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

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

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

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

K/A Number: EA1.3 **CFR Reference:** 41.7 / 45.5 / 45.6

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

Group: 2 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** K

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

Given:
- Unit tripped from 100% power due a loss of off site power
- Both EDG's are powering vital busses.
- Natural circulation has been established for one hour with Tave at 545°F and Thot 565°F.

In accordance with 1203.013, Natural Circulation Cooldown, with natural circulation cooldown established you would expect to see:

A. core exit temperature tracking Thot

B. core delta T continuing to rise

C. demand for EFW drops to zero

D. TBVs opening periodically to maintain SG pressure

Answer:

A. core exit temperature tracking Thot

Notes:

If natural circulation is cooling the core, then core exit TC's should drop, therefore "A" is correct. Note: A was slightly revised to agree with AOP.
"B" is incorrect, core delta T increasing is a sign of a loss of natural circulation.
"C" and "D" are incorrect, as long as heat removal is continuing, secondary heat must be removed by steaming continuously, demand for EFW should be greater than zero.

References:

1203.013, Natural Circulation Cooldown

History:

Used question no. 55 from NRC developed RO exam 8/24/92 KA 2.4/2.4.21 (Tier 3)
Selected for 2014 Exam

SECTION 1 - Degraded Power

NOTE

Indication of primary to secondary heat transfer is as follows:

- T-cold tracking associated SG T-sat
- T-hot tracking CET temps
- T-hot/T-cold ΔT stable or dropping

6. Verify EFW actuated and EFW Pump (P7B) controlling SG level 300 to 340".

A. **IF** SCM is less than adequate, **THEN** verify SG levels 370 to 410".

B. **IF** DG1 is not available,
THEN energize A3 from AAC Gen or A4 to supply P7B using ES Electrical System Operation (1107.002).

C. Verify primary to secondary heat transfer in progress.

D. **IF** P7B is operating, **THEN** stop steam-driven EFW Pump (P7A) as follows:

1) Place EFW Pump Turbine K3 Steam Admission Valves (CV-2613 and CV-2663) in MANUAL **AND** close.

2) Verify P7A speed drops to ~0 rpm **AND** P7A discharge press drops to ~0 psig (C09).

3) Verify P7B maintains proper SG levels.

E. **IF** P7B is not operating **THEN** maintain P7A in service.

1) Maintain SG press ≥80 psig as long as P7A is the only source of feed.

2) Take action to restore P7B to operation prior to depressurizing SGs below 70 psig.

ANO UNIT 1 – 2014

TIER 2

GROUP 1

Questions 28 - 55

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

QID: 0899 **Rev:** 0 **Rev Date:** 9/10/14 **Source:** New **Originator:** Possage

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

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

System Number: 003 **System Title:** Reactor Coolant Pump System (RCPS)

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

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

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

Group: 1 **SRO Imp:** 3.5 **SRO Select:** Yes **Taxonomy:** Ap

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

Given:
- Plant Power 85%
- ICS in full automatic
- One RCP has tripped

In accordance with 1203.022, Reactor Coolant Pump Trip, when the plant stabilizes (without operator action) Main Feedwater flow be ________ for the RCS loop with two running RCPs and __________ for the RCS loop with one running RCP?

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

B. 4.125 X e6 lbm/hr
   2.06 X e6 lbm/hr

C. 4.4 X e6 lbm/hr
   2.2 X e6 lbm/hr

D. 5.5 X e6 lbm/hr
   2.75 X e6 lbm/hr

Answer:

D. 5.5 X e6 lbm/hr
   2.75 X e6 lbm/hr

Notes:

D is correct, plant will runback to ~75% and ICS will re-ratio FW to 2/3 flow for the power level for loop with both RCPs and 1/3 flow for loop with 1 RCP.
A is incorrect but plausible, this would be a re-ratio of FW flow to the correct loops but the total is 40% flow which is for a MFW Pump trip or an asymmetric rod runback but not an RCP runback.
B is incorrect but plausible, this would be a re-rataio of FW with half of 75% flow for the loop with two RCPs and half again for the loop with 1 RCP but the incorrect total FW flow.
C is incorrect but plausible, this would be the correct ratio of FW but to the wrong runback power level of 60%.

References:

1203.022, Reactor Coolant Pump Trip

History:

New for 2014 Exam

INSTRUCTIONS

1. **<u>IF</u> unit load is above limit for current RCP configuration, <u>THEN</u> verify ICS in track and runback in progress.**

2. **Verify main feedwater loop flow ratio responding to match RCS loop flow ratio.**

3. **<u>IF</u> reactor trips, <u>THEN</u> carry out Emergency Operating Procedure (1202.001).**

> **NOTE**
>
> Refer to COLR for calculated high ØΔØflow RPS trip setpoint.

4. **Verify ICS establishes and maintains proper steady state conditions:**

A. **<u>IF</u>** 3 RCPs in operation,
<u>THEN</u> unit load at ~675 MWe
(75% of 902 MWe).

> **CAUTION**
>
> Operation with only 1 RCP in each loop is permitted for 18 hours with the Rx Critical per TS 3.4.4 Action "A". Mode 3 is required to be attained within an additional 6 hours per TS 3.4.4 Action "B".

B. **<u>IF</u>** 1 RCP per loop in operation,
<u>THEN</u> unit load at ~405 MWe
(45% of 902 MWe).

> **NOTE**
>
> At 75% power, an expected feed flow ratio would be: $\frac{\text{1 RCP loop 2.7 MPPH}}{\text{2 RCP loop 5.55 MPPH}}$

C. Proper feed flow ratio with ΔT-cold near zero.

D. T-ave selected to loop with highest flow.

E. **<u>IF</u>** 3 RCPs in operation,
<u>THEN</u> maximum feedwater flow to a steam generator of 5.7×10^6 lbm/hr.

(continued)

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

QID: 0900 **Rev:** 0 **Rev Date:** 9/10/14 **Source:** New **Originator:** Possage

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

Section: 3.2 **Type:** RCS Inventory Conrol

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

Description: Knowledge of the effect of a loss or malfunction on the following CVCS components: Purpose of VCT divert valve.

K/A Number: K6.09 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** An

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

Given:
- Plant Power 100%
- CZ-9 Vacuum Degasifier Bypass, inadvertently left CLOSED following a tag out.

What impact would the above alignment have on the plant if the ATC were to place CV-1248, 3-Way valve, into the BLEED position?

A. Letdown flow would be isolated, no flow would be indicated on C04

B. Letdown flow would be aligned to the in-service T-12, Clean Waste Receiver Tank

C. Letdown flow would be through the Vacuum Degasifier to the Makeup Tank

D. Letdown flow would be diverted through a relief valve to the
Auxiliary Building Equipment Drain Tank

Answer:

D. Letdown flow would be diverted through a relief valve to the Auxiliary Building Equipment Drain Tank

Notes:

D is correct, this lineup would block letdown causing Letdown Relief to lift and divert flow to ABEDT.
A is incorrect, but plausible Letdown would be isolated but only for the split second it would take for the Letdown Relief to lift. Flow would still be indicated on C04.
B is incorrect but plausible this would be the result for a normal "Bleed" operation.
C is incorrect but plausible if the examinee came to the incorrect conclusion that closing the bypass would send flow through the Vacuum Degasifier.

References:

1103.004, Soluble Poison Concentration Control
STM 1-04, Primary Makeup and Purification
STM 1-33, Clean Liquid Radioactive Waste

History:

New for 2014 Exam

17.0 Bleeding Operations

NOTE

- The design of P-36 Suction Stop Check Isolations from BWST (BW-3, BW-2) typically allow hydraulic communication between the Makeup Tank and the DH/LPI Pumps (P-34A/B) and RB Spray Pumps (P-35A/B) ECCS suction pressure associated with the operating HPI train. When Red Train HPI is aligned for service, P-34A and P-35A idle suction pressure can trend with Makeup Tank pressure, and likewise, when Green train HPI is operating, P-34B and P-35B idle suction pressure can trend with Makeup Tank pressure.
- If it is desired to reduce idle ECCS suction pressure, then it is permissible to reduce Makeup Tank level and pressure by cycling Letdown 3-Way Valve (CV-1248), provided Makeup Tank limits are observed such as "Minimum Makeup Tank Water Level vs. Makeup Tank Pressure" Exhibit A of Makeup & Purification System Operation (1104.002).
Ref. CR-ANO-1-2012-0620

17.1 Vacuum degasifier either bypassed or in-service per Vacuum Degasifier Operations (1104.016).

17.2 Clean liquid waste system in operation per Clean Waste System Operation (1104.020), "Initial Startup" section.

17.3 Estimate the volume of bleed and ensure that adequate storage capability exists.

- Compare volume of bleed vs. available Clean Waste Receiver Tanks (T-12s) and any excess letdown due to heatup if applicable.
- IF Clean Waste Receiver Tanks (T-12s) cannot receive required volume,
THEN transfer applicable Clean Waste Receiver Tanks (T-12s) to Unit 2 per Clean Waste System Operation (1104.020).

17.4 Place 3-Way Valve (CV-1248) in BLEED position to begin reducing inventory.

17.5 WHEN desired volume has been removed,
THEN return Letdown 3-Way Valve (CV-1248) to LETDOWN position.

17.6 IF bleeding evolution was to reduce ECCS suction pressure,
THEN verify Station Log entry includes the following info:

- Initial and final ECCS suction pressure
- Initial and final Makeup Tank level
- Initial and final Makeup Tank pressure

18.0 Operability

18.1 The 24 hour time clock per TRM 3.5.1 Condition A is not required to be entered when the BAAT is aligned to alternate flow paths. In an alternate lineup, the BAAT retains the ability to borate the RCS to Cold Shutdown. If the BAAT maintains the ability to be manually realigned to the makeup tank, then time clock entry is not required.

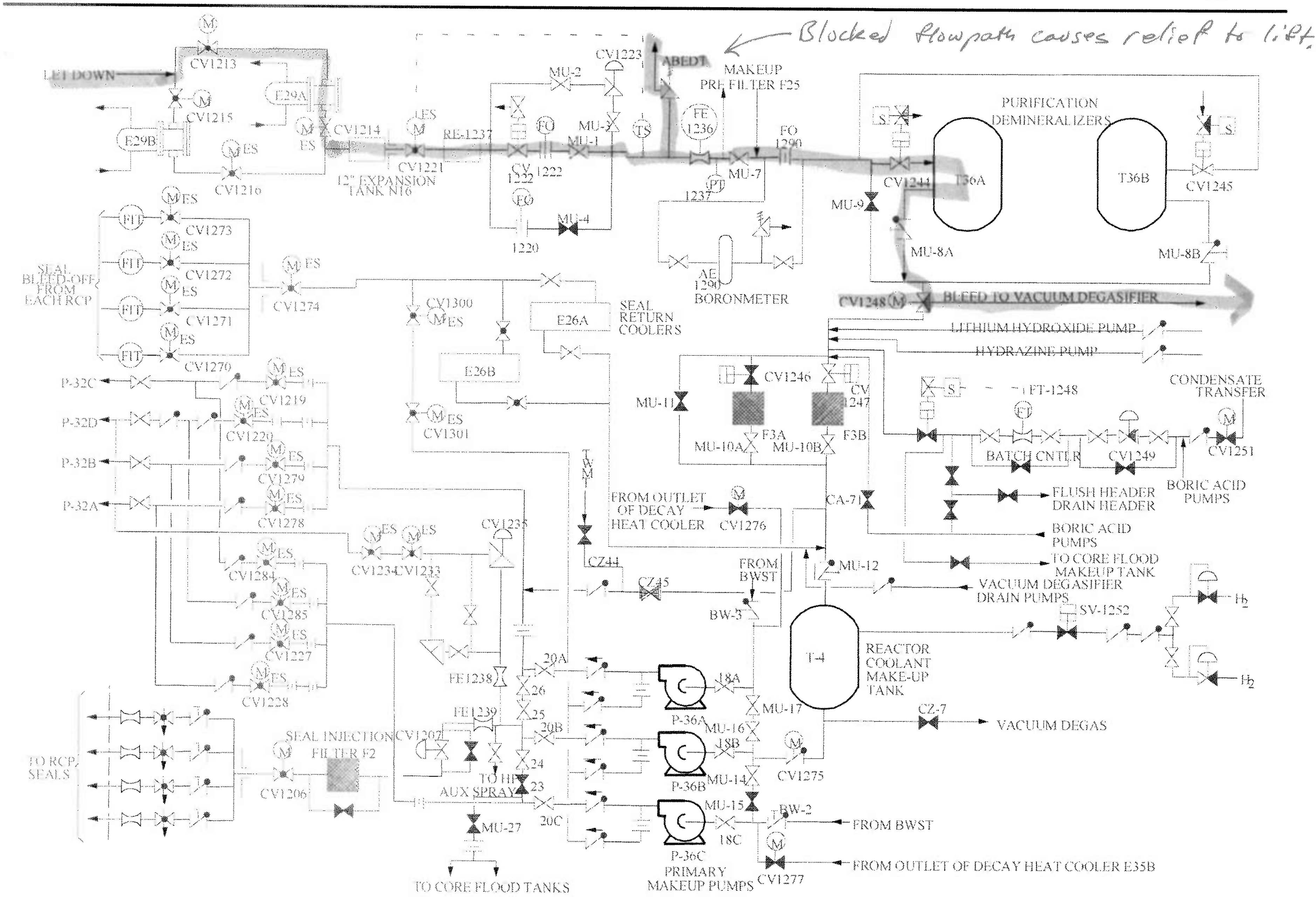
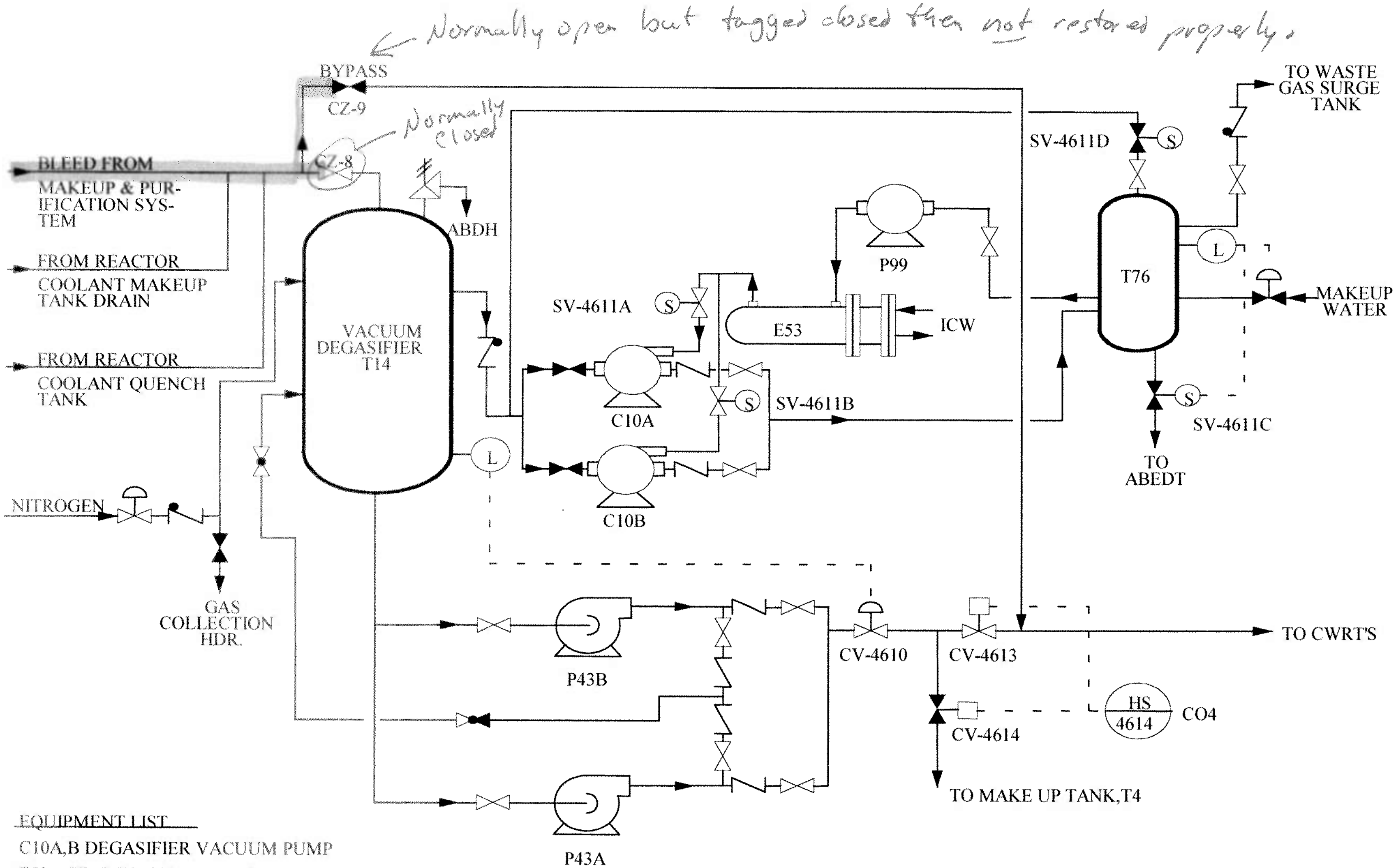


FIGURE 04.01: MU & P ONE LINE



EQUIPMENT LIST

C10A,B DEGASIFIER VACUUM PUMP
E53 SEAL WATER COOLER
P43A,B VACUUM DEGASIFIER DRAIN PUMPS
P99 SEAL WATER PUMP
T76 MOISTURE SEPARATOR

FIGURE 53.02: VACUUM DEGASIFIER

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

QID: 0902 **Rev:** 0 **Rev Date:** 9/5/14 **Source:** New **Originator:** Possage

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

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

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

Description: Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following: System protection logics, including high-pressure interlock ,reset controls, and valve interlocks.

K/A Number: K4.07 **CFR Reference:** 41.7

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

Group: 1 **SRO Imp:** 3.5 **SRO Select:** Yes **Taxonomy:** K

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

Which of the following correctly describes the CV-1050 Decay Heat Removal Suction Valve interlock?

CV-1050 closes at ____________ and receives its pressure input from ESAS Analog _____________ .

A. 290 psig
Channel 1 (PT-1020)

B. 320 psig
Channel 1 (PT-1020)

C. 290 psig
Channel 2 (PT-1041)

D. 320 psig
Channel 2 (PT-1041)

Answer:

B. 320 psig
Channel 1 (PT-1020)

Notes:

B contains the correct setpoint and pressure input for CV-1050.
A, C, and D are combinations of wrong setpoint for the other DH suction valve, or wrong setpoint, or incorrect pressure input, or both. 290 psig is the reset pressure for CV-1410 (i.e., pressure at which it can be reopened).

References:

1104.004, Decay Heat Removal Operating Procedure
STM 1-05, Decay Heat Removal System

History:

New for 2014 Exam

5.9 Two successive starts are allowed with motor initially at ambient temperature. With motor at rated temperature, one start is allowed. Thereafter, an interval of 5 minutes with motor running or motor stopped shall elapse before any additional start.

{4.3.2, 4.3.4}

5.10 Following any significant core damage, the effects on access to vital areas due to high radiation levels should be considered prior to placing the Decay Heat system into service.

{4.3.6}

5.11 To prevent Decay Heat pump damage due to vortex formation in DH suction piping and to provide adequate NPSH for the pump, maintain RCS level and total DH flow according to Attachment B.

{4.3.6}

5.12 Severe water hammer can damage DH system pipe if DH pump is started with high flow when RCS is drained.

5.13 Decay Heat system operation with RC pressure >150 psig can result in exceeding DH system design pressure if DH pump is dead-headed.

5.14 Decay heat flow through cooler should be established gradually if the Service Water and Decay Heat suction ΔT is >200°F.

5.15 Decay heat pump will be damaged if either Decay Heat Suction Valve (CV-1050 or CV-1410) closes while pump is operating.

5.15.1 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.

5.15.2 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.

5.16 If the Decay Heat system has been drained for any reason, the Decay Heat pump discharge piping to the HPI pumps' suction shall be refilled and vented to preclude vapor binding of the HPI Pumps (P-36A, P-36B, P-36C).

5.16.1 If the Decay Heat Pumps have been drained, the mechanical seals should be vented by Maintenance disassembling appropriate fittings on Cyclone Separators.

NOTE

- If Decay Heat Cooler E-35A/E-35B SW Outlet Isol valves (SW-22A and SW-22B) are positioned to the scribed "T" marking (~ 1/3 open) by going in the closed direction and the associated DHR Clr Service Water E-35A/E-35B Inlet valve (CV-3822 or CV-3821) is fully opened, then SW flowrates through the coolers will be >1600 gpm. CV-3822 and CV-3821 are not throttled for temperature control when RCS temperatures are >200°F.
- SW-22A/B must be opened past the scribed "T" position then closed down to the scribed "T" position to maintain SW and DH Loops operable.

5.17 Throttling Service Water flow <1600 gpm through the DH coolers when RCS temperatures are >200°F can result in unacceptably high Service Water piping temperatures.

2.2.3 RCS DHR Suction Isolation Valve CV-1050

(Refer to Figure 05.11)

The first suction isolation valve in the supply line is CV-1050. CV-1050 is designed to the same specifications of the RCS. Design pressure and temperature for CV-1050 is 2500 psig and 650°F. CV-1050 is located in the northwest quadrant of the Reactor Building at the 335' elevation. The valve is near the Reactor Coolant Quench Tank, and is mounted at floor level. CV-1050 is a 12" stainless steel gate valve manufactured by the Velan Valve Corp.

Indication and controls associated with CV-1050 are located on panel C-18. Additional information is provided in Table 5.1.

Interlocks associated with CV-1050 are based on RCS pressure and CFT "A" outlet isolation valve position. Interlocks are designed to protect the low pressure DHR piping from over pressurization. Interlock set points are high enough to insure DHR suction valves remain open during normal DHR operations. RCS pressure for CV-1050 auto closure and open permissive is provided by PT-1020 located in the ESAS analog channel 1 cabinet (C-88). RCS pressure can be read at ESAS channel 1 cabinet to determine pressure sensed by PT-1020. PT-1020 provides RCS pressure to a contact buffer that provides an open or close signal to the ACI logic for CV-1050. The contact buffer automatically resets when RCS pressure is less than setpoint of 290 psig. The contact buffer provides a white "open permissive" light above HS-1050. The open permissive light informs the operator that RCS pressure is less than setpoint for opening of CV-1050. ZS-2415A provides CFT isolation valve position for open permissive and auto closure of CV-1050.

The following conditions must be met for CV-1050 to be opened. RCS pressure must be less than 290 psig **and** the "A" CFT isolation valve must be 100% closed. If either of these conditions do not exist CV-1050 will not open.

The following conditions will provide an auto closure of CV-1050. CV-1050 will close when RCS pressure reaches 320 psig **or** "A' CFT isolation valve is removed from its closed seat.

Due to the importance of maintaining DHR pump suction, an annunciator alarm for CV-1050 auto closure and a bypass feature are provided. The annunciator alarm K09 B7 will alarm when an auto closure of CV-1050 occurs. Additional failures associated with CV-1050 will also cause this alarm to annunciate. Failures that will cause this alarm are, high closing torque, motor overload and CV-1050 not 100% open or closed. Refer to annunciator corrective actions 1203.012H (K09 B7) for additional information.

2.2.3.1 CV-1050 Auto Closure Bypass Feature

The bypass feature associated with CV-1050 was added by DCP-86-1129. This DCP added an additional handswitch (HS-1050A) to allow the operators to bypass the auto closure feature associated with CV-1050. This allows the auto closure feature of CV-1050 to be bypassed whenever directed by the SM or when the

QID: 0903 **Rev:** 0 **Rev Date:** 9/5/14 **Source:** New **Originator:** Possage

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:**	Yes	**Taxonomy:**	K

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

Which of the following provides motor power to CV-1407, BWST Outlet Valve?

A. B-51

B. B-55

C. B-56

D. B-57

Answer:

A. B-51

Notes:

A is the correct source of power to CV-1407.
B, C, D are plausible since they are also red train MCCs but are incorrect.

References:

1107.002, ES Electrical System Operation

History:

New for 2014 Exam.

ATTACHMENT C

BREAKER NUMBER	DESCRIPTION Ref. Drawing (E-15)	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INI-TIAL
5163	Quench Tank T42 Sample CV-1054 (E-246)	Closed			
5164	BWST T-3 Outlet CV-1407 (E-184)	Closed			
5171	RB Spray Block CV-2401 (E-219)	Note 1			
5172	SG-A ATM Dump Isol CV-2676 (E-442)	Closed			
5173	EFW P-7B Suction from CST CV-2800 (E-296)	Closed			
5174	Core Flood Tank T-2A Sample CV-2416 (E-239)	Closed			
5181	SW Loop I Supply to ICW Coolers CV-3820 (E-282)	Closed			
5182	Decay Ht CLR Service Water E-35A Inlet CV-3822 (E-283)	Closed			
5183	BWST Purification Upstream Recirc Isolation CV-1441 (E-184-6)	Closed			
5185	Spare	Open			
5191	Pressurizer Gas Sampling Iso. CV-1814 (E-249)	Closed			
5192	Pressurizer Water Sampling Iso. CV-1816 (E-249)	Closed			
5193	EFW P-7B Suction From SW CV-2803 (E-296)	Closed			
5194	EFW Serv Wtr Loop 1 Iso. CV-3850 (E-296)	Closed			
51101	Discharge Flume Rad Monitor RE-3618 (E-411)	Closed			
51102A	DG1 Soak Back Pump P-106A3 (E-108-3)	Closed			
51102B	VCH-4B Compressor (E-368-3)	Closed			
(NOTE 2) 51103	Spent Fuel Pool Cooling Pump P-40A (E-244)	Closed			

Note 1: This breaker will be closed by Plant Startup 1102.002.

Note 2: IF EC-31408 (temporary power to P-40A) is installed, THEN P-40A is powered from B2145.

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

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to manually operate and/or monitor in the control room: Recognition of leaking PORV/code safety.

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

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

Group: 1 **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** Ap

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

The following plant conditions exist:

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

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

A. Approximately 212 °F

B. Approximately 260 °F

C. Approximately 280 °F

D. Approximately 588 °F

Answer:

B. Approximately 260 °F

Notes:

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

References:

ASME Steam Tables

History:

Developed for 1998 RO Exam.
Selected for 2002 RO exam under 008 AK1.01.
Modified for 2005 RO exam but not used.
Selected for 2005 RO re-exam.
Selected for use in 2007 RO Exam.
Selected for 2014 Exam.

Location: C14

Device and Setpoint: RCS Relief Alarm Switch (VBS-1000)

RELIEF VALVE OPEN

Alarm: K09-A1

1.0 OPERATOR ACTIONS

1. Identify open relief by checking analog position indications and Hi-Alarm lights on panel C486-1.

2. IF desired,
THEN place Relief Valve Audio Monitor (XI-1000) keyswitch in SILENCE ALARM for the following reasons:

   - Testing
   - Drawing steam bubble
   - SM permission

3. Refer to Pressurizer Systems Failure (1203.015).

NOTE

If a relief valve is open, Quench Tank (T-42) temperature should go to saturation for its pressure.

4. Monitor Quench Tank pressure, level and temperature.

NOTE

Temperature elements downstream of each PSV indicate valve position. Temps are available on SPDS:

- PZR PSV-1002 Outlet Temp (T1027)
- PZR PSV-1001 Outlet Temp (T1026)
- ERV PSV-1000 Outlet Temp (T1025)

5. IF relief valve TEs read normal
AND Quench Tank level reads normal
AND Quench Tank temperature reads normal,
THEN use audio monitor to listen for flow noise through valve.

   A. IF alarm is due to monitor channel malfunction,
   THEN switch channels using Pressurizer Relief Valve Monitoring System Operation (1105.013).

6. IF XI-1000 keyswitch was placed in SILENCE ALARM,
THEN when alarm clears, place keyswitch in normal-vertical.

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

QID: 0905 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** Modified **Originator:** Possage

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:** Yes **Taxonomy:** C

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

Given:
- Plant is at 100% power.
- ICW pump P-33B is in service on Non-Nuclear ICW.
- ICW PUMP AUTO START, K12-A4 alarms

The Nuclear ICW pump has tripped, what would you observe on C09?

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 Nuclear loop, P-33C would auto-start.

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

Answer:

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

Notes:

D is correct P-33C is the Nuclear ICW pump, P-33A is the Non-Nuclear ICW pump, and P-33B can supply either. If P-33C trips, then
P-33B suction and discharge cross-connect valves will align to the Nuclear Loop and P-33A will auto start on the non-nuclear loop.
A is incorrect but plausible if examinee thought P-33A was standby for the -nuclear ICW pump.
B is incorrect but plausible, although P-33C will auto-start, P-33B is the swing pump and will re-align to the non-nuclear loop.
C is incorrect, P-33B will shift to loop with lowest pressure (nuclear) and the non-swing non-nuclear pump P-33A would auto-start. Plausible if examinee cannot remember which pump goes to each loop.

References:

1203.012K, Annunciator K12 Corrective Action

History:

Modified QID 627 for 2014 Exam

Location: C19

Device and Setpoint: see page 2 of 2.

ICW
PUMP
AUTO START

Alarm: K12-A4

1.0 OPERATOR ACTIONS

1. Check pump status on C09 to determine which ICW Pump (any of P-33A thru P-33C) auto started.

2. IF P-33A OR P-33C auto started on low P-33B discharge pressure,
THEN perform the following:

- Verify ICW Pumps Discharge Crossconnects (CV-2238 and CV-2239) closed.
- Verify ICW Pumps Suction Crossconnects (CV-2240 and CV-2241) closed.

A. Trip P-33B using HS on C09.

3. IF P-33A OR P-33B auto started on low P-33C discharge pressure,
THEN perform the following:

- Verify ICW Pumps Discharge Crossconnect (CV-2238) closed.
- Verify ICW Pumps Discharge Crossconnect (CV-2239) open.
- Verify ICW Pumps Suction Crossconnect (CV-2240) closed.
- Verify ICW Pumps Suction Crossconnect (CV-2241) open.

A. Trip P-33C using HS on C09.

4. IF P-33B OR P-33C auto started on low P-33A discharge pressure,
THEN perform the following:

- Verify ICW Pumps Discharge Crossconnect (CV-2238) open.
- Verify ICW Pumps Discharge Crossconnect (CV-2239) closed.
- Verify ICW Pumps Suction Crossconnect (CV-2240) open.
- Verify ICW Pumps Suction Crossconnect (CV-2241) closed.

A. Trip P-33A using HS on C09.

5. IF P-33B auto started,
THEN monitor ICW Surge Tank (T-37A and T-37B) levels.

A. IF leakage from a discharge crossconnect valve causes surge tank to overflow,
THEN verify both ICW loop suction valves open:

- ICW Pumps Suction Crossconnect (CV-2240)
- ICW Pumps Suction Crossconnect (CV-2241)

6. Initiate steps to determine and repair cause of low ICW pump discharge pressure.

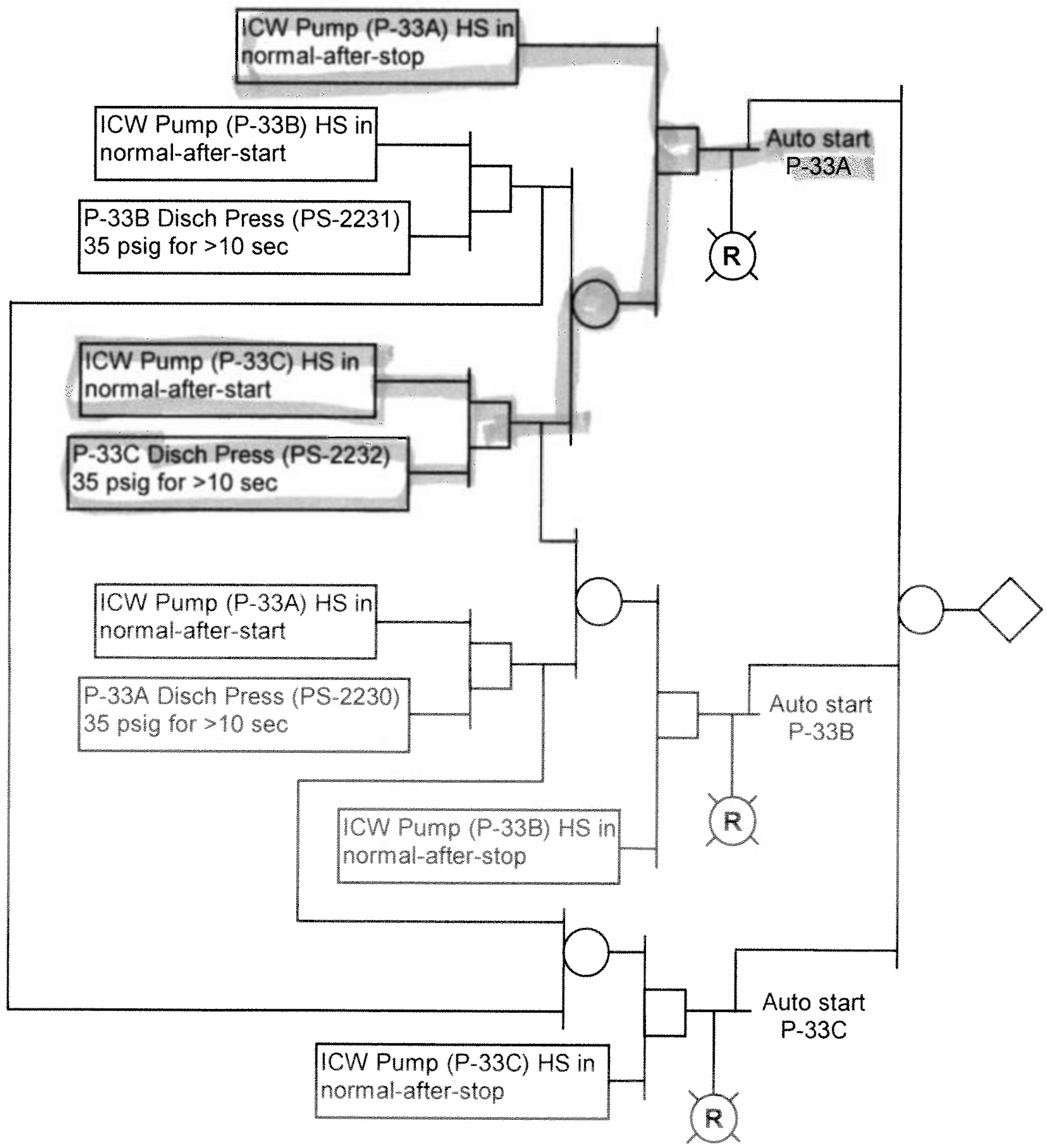
7. To clear alarm, place HS to normal-after-start for auto-started pump.

2.0 PROBABLE CAUSES

- Pump malfunction
- Electrical bus failure

3.0 REFERENCES

Schematic Diagram Annunciator K12 (E-462, sheets 1-3)



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

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

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

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

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

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

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

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

Question: **RO:** ☐ **SRO:** ☐

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

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

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

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

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

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

Answer:

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

Notes:

"C" is correct, P-33B will shift to loop with lowest pressure (non-nuclear) and the non-swing nuclear pump P-33C would auto-start.
"A" is incorrect, P-33A is the non-nuclear ICW pump.
"B" is incorrect, although P-33C will auto-start, P-33B is the swing pump and will re-align to the non-nuclear loop.
"D" is incorrect, P-33A is the non-nuclear ICW pump.

References:

STM 1-43, Rev.8

History:

New for 2005 RO re-exam.

PARENT

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

QID: 0906 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** New **Originator:** Possage
TUOI: A1LP-RO-RCS **Objective:** 6 **Point Value:** 1

Section: 3.3 **Type:** Reactor Pressure Control
System Number: 010 **System Title:** Pressurizer Pressure Control System (PZR PCS)
Description: Ability to monitor automatic operation of the PZR PCS, including: PZR pressure.

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

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	Yes	**Taxonomy:**	K

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

Uni1 is at 100% power.
The ATC observes RCS pressure is slowly dropping.

At what pressure would Pressurizer Heater Bank 4 automatically energize?

A. 2105 psig

B. 2120 psig

C. 2135 psig

D. 2140 psig

Answer:

B. 2120 psig

Notes:

B is correct.
A is incorrect but plausible, Bank 5 energizes at this pressure.
C is incorrect but plausible, Bank 3 energizes at this pressure.
D is incorrect but plausible, Bank 4 de-energizes at this pressure.

References:

1103.005, Pressurizer Operation

History:

New for 2014 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: 0907 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** Modified **Originator:** Possage

TUOI: A1LP-RO-EOP01 **Objective:** 11 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

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

Description: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A Number: 2.1.7 **CFR Reference:** 41.5 / 43.5 / 45.12 / 45.13

Tier: 2 **RO Imp:** 4.4 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** C

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

Given:

Plant is operating at 100% power.
A plant transient occurs.

Which of the following parameters would indicate a failure of the Reactor Protection System (RPS), and require the operator to manually trip the reactor?

A. Reactor Building pressure 19 psia

B. RCS pressure 1850 psig

C. RCS Thot 615 °F

D. NI power 103%

Answer:

A. Reactor Building pressure 19 psia

Notes:

A is correct, Must recognize that a manual reactor trip is required based on the RB pressure instrument. RPS should have tripped CRDs at 18.7 psia in the RB.
B is incorrect but plausible as this is near the low RCS pressure trip of 1800 psig.
C is incorrect but plausible as this is near the RCS Thot trip of 618°F.
D is incorrect but plausible as this is near the NI power trip of 104.9%

References:

1202.001, Reactor Trip

History:

Modified 659 for 2014 Exam

ENTRY CONDITIONS

- An automatic reactor trip or DSS trip.
- Failure of RPS to trip the reactor upon reaching a limit listed below:
  - High power 104.9%
  - High power/pumps .... one pump per loop .... ≥ 55%
    OR
    0 pumps in one loop .... ≥ 0%
  - High power/imbalance/flow .... COLR Figure
  - High RCS temp .... ≥ 618°F (T-hot)
  - High RCS press .... ≥ 2355 psig
  - Low RCS press .... ≤ 1800 psig
  - Variable low RCS press .... COLR Figure
  - High RB press .... ≥ 18.7 psia
  - Turbine trip .... reactor power ≥ 43% **AND** Turbine is tripped
  - Both MFW pumps trip .... reactor power ≥ 9% **AND** both MFW pumps tripped
- Manual trip of the reactor is required due to reaching a limit listed below:
  - PZR level dropping < 100",
    AND
    no indication of recovery
  - PZR level > 290"
  - Any MSIV closure at power
  - Either SG level < 15" or > 95%,
    AND
    no indication of recovery
  - A system degradation that requires manual reactor trip based on operator judgment
  - Abnormal Operating Procedure requirement
- **IF** a system degradation occurs while shutdown, above DHR operation,
  THEN perform applicable steps

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

QID: 0659 **Rev:** 0 **Rev Date:** 12/14/06 **Source:** Direct **Originator:** Possage
TUOI: A1LP-RO-EOP01 **Objective:** 11 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation
System Number: 012 **System Title:** Reactor Protection System
Description: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

K/A Number: 2.4.49 **CFR Reference:** 41.10 / 43.2 / 45.6

Tier:	2	**RO Imp:**	4.0	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	Ap

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

Given:

Plant is operating at 100% power.
A plant transient occurs.

Which of the following parameters would indicate a failure of the Reactor Protection System (RPS), and require the operator to manually trip the reactor?

A. Reactor Building pressure 3.5 psig.

B. Pressurizer level 95 inches and lowering.

C. RCS Cold Leg Temperature 540 °F.

D. Reactor Coolant System Pressure 1750 psig.

Answer:

D. Reactor Coolant System Pressure 1750 psig.

Notes:

"D" is correct. RPS low pressure trip setpoint is 1800 psig.
"A" is incorrect. RPS high RB pressure setpoint is 18.7 psia (4 psig)
"B" is incorrect. Although a manual Rx trip would be required of the operator at 100 inches with no idication of recovery, RPS does not have a setpoint associated with the pressurizer level, therefore RPS did not fail.
"C" is incorrect. All the conditions could be caused by a steam line break in the reactor building, and an RCS Tcold of 540 degrees requires entry into the Overcooling EOP.

References:

1202.001 Chg 028-05-0

History:

New for 2007 RO Exam.

PARENT

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

QID: 0908 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** New **Originator:** Possage

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

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

System Number: 013 **System Title:** Engineered Safety Features Acutation System (ESFAS

Description: Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: AFW System.

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

Tier:	2	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	K

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

Which of the following will cause a FULL actuation of EFW?

A. ES Channels 1 & 2

B. ES Channels 3 & 4

C. ES Channels 1 & 3

D. ES Channels 2 & 4

Answer:

B. ES Channels 3 & 4

Notes:

B is correct, ESAS Channel 3 will actuate Train A and Channel 4 will actuate Train B EFW.
A, C, & D are all incorrect but plausible combinations of ES Channels.

References:

STM 1-66, Emergency Feedwater Initiiation and Control

History:

New for 2014 Exam

signal from each OTSG is output through a buffer to the SPDS. Indication on C09, indicator and recorder, is provided for each OTSG from channels A and B.

2.3.4 RPS Inputs to EFIC

Refer to Figure 66.07 and figure 66.18.

The Reactor Protection System provides the following signals to the EFIC system:

- Nuclear Instrumentation (NI) power >10%
- Reactor Coolant Pump status
- Main Feedwater Pump trip status
- RPS Channel Bypass

The reactor coolant pump monitors send an RCP status signal to EFIC. EFIC Channel "A" receives four inputs from RPS Channel "A", EFIC Channel "B" from RPS Channel "B", and so on. These pump status signals are deenergize to trip within the RPS cabinets and are not bypassed by an RPS channel bypass. Any loss of power to the RPS cabinet will result in a loss of all RCP's signal to it's corresponding EFIC channel.The RCP status signals are input to the EFIC channel initiate module. Electrical isolation is provided for the RC Pump signals by opto-isolators.

The NI signal, which inputs into the initiate module, shows when reactor power is less than 10%. This signal is used to allow for shutdown bypass of the loss of all RCP's EFW actuation.

The RPS cabinet MFW pump anticipatory trip modules send a signal to the EFIC cabinet initiate module. Whenever RPS sees two main feedwater pumps tripped (except when bypassed in the RPS cabinet) a signal will be sent to the EFIC cabinets for EFW actuation. Each RPS cabinet will send a signal to its respective EFIC cabinet. A light on the alarm panel labeled MFP will indicate the status of this signal. A solid bright light indicates a normal condition. Light off indicates an abnormal "loss of both main feedwater pumps" EFW actuation signal. The RPS signal is automatically bypassed < 7% power. The EFIC MFP light will be off only when > 7% power with the trip signal present.

The RPS, through an auxiliary relay sends an RPS channel bypassed signal to its respective EFIC channel. This signal ensures that only the EFIC channel corresponding to the bypassed RPS channel can be bypassed.

The EFIC channel shutdown bypass light on the alarm panel will flash bright to dim whenever it's corresponding RPS channel is in channel bypass. Note, the EFIC channel is not in bypass with just the associated RPS channel bypassed, it merely indicates bypass.

2.3.5 ESAS Channel Inputs

Refer to Figures 66.07 and 66.19.

ESAS digital channel 3 inputs two signals to EFIC train "A" EFW trip module. ESAS digital channel 4 inputs two signals to the EFIC train "B" EFW trip module. These signals ensure an EFW action upon an actuation of ESAS channel 3 <u>or</u> channel 4. Channel 3 ESAS will actuate train A EFW only while channel 4 ESAS will

actuate train B EFW only. The status of these signals can be observed by looking at status lights on EFIC cabinets A and B alarm panels. There are two lights, labeled ESAS, on each alarm panel. These lights will be on solid in a normal state and will go out on an ESAS digital channel actuation. The two lights on EFIC cabinet A represent the two signals from ESAS digital channel 3 to the train "A" EFW trip module for trip bus 1 and 2. The two lights on EFIC cabinet "B" represents the two signals from ESAS digital channel 4 to the train "B" EFW trip module for trip bus 1 and 2.

ESAS auto/man pushbuttons are located in the ESAS Digital cabinets for channels 3 and 4. There are two sets of auto/man pushbuttons in each cabinet. One set is for each signal from the ESAS channels to their respective EFIC trip module. Two signals are required to each trip module in order to produce a full trip of EFW (both trip busses energized). The status of these signals is indicated on the alarm panel "EXT TRIPS" ESAS lights. Two lights are on cabinet "A" for the signals from ESAS ch. 3 to the train "A" EFW trip module. Two lights are on cabinet "B" for the signals from ESAS ch. 4 to the train "B" EFW trip module.

These lights are off when an actuation signal is sent to EFIC for EFW actuation on ESAS ch. 3 or 4 actuation. Once ESAS has actuated, these signals can be reset in two ways: (1) reset the ESAS digital channel, or (2) select "MAN" in the ESAS digital cabinet for both signals to each train. Once reset the ESAS lights on the Channel A and B alarm panels will be solid.

2.3.6 DROPS input to EFIC

The AMSAC (ATWS Mitigation System Actuation Circuitry) portion of DROPS (Diverse Reactor Overpressure Prevention System) monitors total main feedwater flow and reactor power. There are two channels of DROPS. Channel 1 inputs a signal to the channel A EFIC initiate module. Channel 2 inputs a signal to the channel D EFIC initiate module. When total FW flow <15% with RX power >45%, AMSAC will send initiate signals to EFIC resulting in a full actuation of both trains of EFW.

2.4 EFW Actuation

EFW can be actuated by any of eight separate conditions. Regardless of the actuating condition, the EFW trip module will perform the same function to start the EFW system. All actuating conditions except one, ESAS, are processsed through each channels "INITIATE MODULE". Refer to Figures 66.18, 66.19, and 66.20. When the initiate module determines EFW is required, the initiate module will send an EFW INITIATE signal to the train A and train B EFW trip modules.

2.4.1 Initiate Logic

Each initiate module has inputs from four bistables; two "Pressure Initiate" and two "Level Initiate" for both OTSGs. Each initiate module also has five inputs from RPS; four "RCP tripped" and one "both MFW pumps tripped" signal. Channels A and D initiate modules have inputs from AMSAC. One EFW actuation condition does not actuate through the INITIATE module. ESAS

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

QID: 0909 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** Modified **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:** Yes **Taxonomy:** K

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

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.

References:

STM 1-65, Engineered Safeguards Actuation System

History:

Modified 139 for 2014 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: 0139 **Rev:** 1 **Rev Date:** 04/07/94 **Source:** Direct **Originator:** K. Canitz

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

Section: 3.5 **Type:** Containment Integrity

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

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

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

Tier: 2 **RO Imp:** 3.9 **RO Select:** No **Difficulty:** 2

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

Question: RO: SRO:

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

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

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

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

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

Answer:

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

Notes:

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

References:

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

History:

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

PARENT

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

QID: 0454 **Rev:** 0 **Rev Date:** 5/6/2002 **Source:** Direct **Originator:** S.Pullin

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

Section: 3.5 **Type:** Plant Systems

System Number: 026 **System Title:** Containment Spray

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of spray pump.

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

Tier:	2	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	C

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

A large break LOCA is in progress.

- RCS pressure is ~ 25 psig.
- RB pressure is 45 psia and trending down.
- Shift to RB Sump Suction has just been completed.

Subsequently, annunciator K11-C7, "RB SPRAY P35B ES FAILURE" alarms.

Which of the following actions should be taken?

A. Close Decay Heat Supply to Makeup Pump Suction CV-1276.

B. Establish maximum flow through "A" RB Spray Pump.

C. Maintain "A" RB Spray flow at 1050 to 1200 gpm.

D. Throttle "B" LPI Pump flow to 2800 gpm.

Answer:

C. Maintain "A" RB Spray flow at 1050 to 1200 gpm.

Notes:

Answer "C" is the correct response, the indications given are that of "A" RB Spray pump suction vortexing on the RB sump.
Answer "A" is incorrect, although this action will reduce flow out of the RB sump, no procedural direction exists for this action.
Answer "B" is incorrect, although one pump has tripped and it seems sensible to maximize flow on the remaining pump, flow should not be maximized due to indications of vortexing.
Answer "D" is incorrect, throttling LPI flow to the minimum value could lessen vortexing, but no procedural guidance of this kind exists. At one time it was discussed to throttle both RB Spray and LPI flow after swap to RB Suction but only Spray flow is throttled. Also, this value is the flow where HPI can be stopped if both LPI pumps have greater than or equal to this flow rate.

References:

1202.010, ESAS

History:

Created for 2002 SRO exam.
Used on 2004 SRO Exam.
Selected for 2014 Exam.

CAUTION

Full flow from both trains of HPI, LPI, and RB Spray can reduce BWST level to 6' within 25 minutes of ESAS actuation.

18. Before BWST level reaches 6', perform the following:

A. Verify RB Sump Outlets open:

- CV-1414
- CV-1415

B. Evacuate all unnecessary personnel from Auxiliary Building in preparation for RB sump recirculation.

C. <u>IF</u> RB Spray has actuated,
<u>THEN</u> verify RB Spray flow throttled to maintain 1050 to 1200 gpm per train.

D. Verify both Low Pressure Injection (Decay Heat) Pumps running:

- P34A
- P34B

1) <u>IF</u> either Low Pressure Injection (Decay Heat) Pump is unavailable,
<u>THEN</u> stop associated HPI pump.

P34A	P34B
P36A/B	P36B/C

(18. CONTINUED ON NEXT PAGE)

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:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** C

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

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?

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

5.8 Cooldown rate from Mode 3 to Mode 5 shall be ≤50°F/hour to minimize stresses on bowed tie rods in SGs.

5.9 During cooldown minimize temperature transients and stresses on bowed tie rods in SGs as follows:

- Maintain SG level <348"
- Make SG level changes slowly
- Terminate SG level change if SG steam temp/downcomer ΔT approaches limit (15°F when RCS >250°F, 25°F when RCS 200-250°F). If exceeded write a CR to document. When temperature within limit, SG level change may resume.

5.10 SGs have been evaluated for the thermal cycles imposed by each refueling shutdown/cooldown at 50°F/hr cooldown rate.

5.11 During a non-refueling shutdown, exceeding a downcomer cooldown rate >15°F over a 30-minute period during a partial or complete RCS cooldown must be recorded and will require engineering evaluation from a thermal cycle standpoint.
(Ref. CR-ANO-1-2007-0959, CR-ANO-1-2011-1925)

5.12 If cooldown is greater than a change of more than 50°F pressurizer temperature then TR 3.4.3.2 requires verification that the pressurizer cooldown rate is not greater than 100°/hour.

5.13 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).

6.0 SETPOINTS

6.1 Refer to applicable system operating procedures for setpoints.

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

QID: 0911 **Rev:** 0 **Rev Date:** 9/11/14 **Source:** New **Originator:** Possage

TUOI: A1LP-ROEOP01 **Objective:** 10 **Point Value:** 1

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overfeeding event.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	Yes	**Taxonomy:**	C

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

Given:
- Reactor tripped from 100% due to a Turbine Trip
- Immediate Actions are complete
- ATC reports 'A' S/G level rising rapidly and approaching 410 inches

What is required by 1202.001, Reactor Trip, and why?

A. Trip 'A' MFWP to prevent flooding the aspirating ports

B. Trip Both MFWPs to prevent flooding the aspirating ports

C. Trip 'A' MFWP to prevent feedwater carryover into Main Steam Line

D. Trip Both MFWPs to prevent feedwater carryover into Main Steam Line

Answer:

D. Trip Both MFWPs to prevent feedwater carryover into Main Steam Line

Notes:

D is correct per 1202.001.
A is incorrect but plausible as this action is taken in the Overcooling EOP for overfeed events but if SG level is approaching the high limit both pumps are tripped to prevent putting water in the steam lines.
B is incorrect but plausible as this is the correct action but the wrong reason.
C is incorrect but plausible as this is the correct reason but both pumps should be tripped.

References:

1202.001, Reactor Trip

History:

New for 2014 Exam

INSTRUCTIONS	CONTINGENCY ACTIONS
10. Check OP HPI pump supplying normal Makeup and Seal Injection.	10. Perform the following: A. Isolate Letdown by closing either: • Letdown Coolers Outlet (RCS) (CV-1221) **OR** • Letdown Coolers Outlets (RCS): – CV-1214 – CV-1216 B. Restore normal Makeup and Seal Injection (RT-1).
11. Check both SG levels remain ≤ 410".	11. Perform the following: A. Trip both Main Feed Pumps: • A Main Feed Pump • B Main Feed Pump B. Place overfilled SG EFW Pump Turbine K3 Steam Supply Valve(s) in MANUAL **AND** close: SG A: CV-2667; SG B: CV-2617 C. Actuate EFW **AND** verify proper actuation and control (RT-5). 1) **IF** all MFW and EFW is lost, **THEN GO TO 1202.004, "OVERHEATING"** procedure. D. **IF** SG press is < 650 psig, **THEN** trip all Condensate pumps: • P2A • P2B • P2C

SG A	SG B
CV-2667	CV-2617

EOP Setpoint No.:	G.1	**Revision:**	3
Parameter:	STEAM GENERATOR LEVEL		
Setpoint Value:	410"		
Applicability:			
Associated System/Component:	STEAM GENERATOR		

Description:

Level above which with no indication of recovery, actions for SG overfill must be taken AND as criteria for Emergency Cooldown of RCS and SG Isolation during SGTR.

Key Assumptions:	NONE

Basis:

The intent of this setpoint is to protect the Steam Generator against overfilling and passing water into the steam lines. It also minimizes the possibility of passing water out the Turbine Bypass Valves or Atmospheric Dump Valves or MSSV's which could cause damage to them. The bottom of the steam outlet nozzles is located at approximately 423 inches above the lower tube sheet (Ref. 1). This limit allows for 13" margin to the bottom of the steam line at 423". If accident error were to exist such that the indicator is reading low, the level could then actually be above the steam line entrance elevation. However, reference 2 documents the fact that a full steam line of water has been analyzed and an incipient failure of the steam line is not expected to ensue. This argument has been used to establish the SG level band for establishing reflux boiling heat transfer (setpoint G.2).

This level is also used as a Tube Rupture Alternate Control Criteria (TRACC) for Emergency Cooldown to 500 F and for isolating a bad SG during a SG Tube Rupture (Reference 3, section III.E.3.4.3).

References:

1) Drawing M1D-202, "ANO-1 EOTSG Shell Outline Drawing," Revision 0.

2) Entergy memo, ANO-91-08158, "Effects of Carryover into Main Steam Lines," dated December 10, 1991.

3) B&W Document Identifier 74-1152414-08, "EOP Technical Basis Document," Vol. 3.

QID: 0912 **Rev:** 0 **Rev Date:** 9/12/14 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EFIC **Objective:** 32 **Point Value:** 1

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

System Number: 061 **System Title:** Auxiliary / Emergency Feedwater (AFW) System

Description: Knowledge of the physical connections and/or cause effect relationships between the AFW and the following systems: MFW System.

K/A Number: K1.02 **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:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	C

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

Which of the following is the LOWEST power level when an EFIC actuation on loss of both MFW Pumps will occur during a plant shutdown?

A. 8%

B. 9%

C. 10%

D. 11%

Answer:

A. 8%

Notes:

A is correct, the RPS trip is automatically bypass when Rx power is < 7%.
B and C are incorrect but plausible since Rx power < 10% is when the EFIC trip on loss of all RCPs is bypassed.
D is incorrect but plausible since the Loss of Main Feedwater Pumps Trip is required at greater than or equal to 10%

References:

STM 1-66, Emergency Feedwater Initiation and Control
T.S. Table 3.3.1-1

History:

New for 2014 Exam

signal from each OTSG is output through a buffer to the SPDS. Indication on C09, indicator and recorder, is provided for each OTSG from channels A and B.

2.3.4 RPS Inputs to EFIC

Refer to Figure 66.07 and figure 66.18.

The Reactor Protection System provides the following signals to the EFIC system:

- Nuclear Instrumentation (NI) power >10%
- Reactor Coolant Pump status
- Main Feedwater Pump trip status
- RPS Channel Bypass

The reactor coolant pump monitors send an RCP status signal to EFIC. EFIC Channel "A" receives four inputs from RPS Channel "A", EFIC Channel "B" from RPS Channel "B", and so on. These pump status signals are deenergize to trip within the RPS cabinets and are not bypassed by an RPS channel bypass. Any loss of power to the RPS cabinet will result in a loss of all RCP's signal to it's corresponding EFIC channel.The RCP status signals are input to the EFIC channel initiate module. Electrical isolation is provided for the RC Pump signals by opto-isolators.

The NI signal, which inputs into the initiate module, shows when reactor power is less than 10%. This signal is used to allow for shutdown bypass of the loss of all RCP's EFW actuation.

The RPS cabinet MFW pump anticipatory trip modules send a signal to the EFIC cabinet initiate module. Whenever RPS sees two main feedwater pumps tripped (except when bypassed in the RPS cabinet) a signal will be sent to the EFIC cabinets for EFW actuation. Each RPS cabinet will send a signal to its respective EFIC cabinet. A light on the alarm panel labeled MFP will indicate the status of this signal. A solid bright light indicates a normal condition. Light off indicates an abnormal "loss of both main feedwater pumps" EFW actuation signal. The RPS signal is automatically bypassed < 7% power. The EFIC MFP light will be off only when > 7% power with the trip signal present.

*

The RPS, through an auxiliary relay sends an RPS channel bypassed signal to its respective EFIC channel. This signal ensures that only the EFIC channel corresponding to the bypassed RPS channel can be bypassed.

The EFIC channel shutdown bypass light on the alarm panel will flash bright to dim whenever it's corresponding RPS channel is in channel bypass. Note, the EFIC channel is not in bypass with just the associated RPS channel bypassed, it merely indicates bypass.

2.3.5 ESAS Channel Inputs

Refer to Figures 66.07 and 66.19.

ESAS digital channel 3 inputs two signals to EFIC train "A" EFW trip module. ESAS digital channel 4 inputs two signals to the EFIC train "B" EFW trip module. These signals ensure an EFW action upon an actuation of ESAS channel 3 or channel 4. Channel 3 ESAS will actuate train A EFW only while channel 4 ESAS will

Table 3.3.1-1
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower –				
a. High Setpoint	1,2[(a)],3[(d)]	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5	≤ 104.9% RTP
b. Low Setpoint	2[(b)],3[(b)] 4[(b)],5[(b)]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 618°F
3. RCS High Pressure	1,2[(a)],3[(d)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 2355 psig
4. RCS Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 1800 psig
5. RCS Variable Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
6. Reactor Building High Pressure	1,2,3[(c)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 18.7 psia
7. Reactor Coolant Pump to Power	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 55% RTP with one pump operating in each loop.
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
9. Main Turbine Trip (Oil Pressure)	≥ 45% RTP	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 40.5 psig
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ 10% RTP	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 55.5 psig
11. Shutdown Bypass RCS High Pressure	2[(b)],3[(b)] 4[(b)],5[(b)]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 1720 psig

(a) When not in shutdown bypass operation.
(b) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.
(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.
(d) With any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

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

QID: 0913 **Rev:** 0 **Rev Date:** 9/12/14 **Source:** New **Originator:** Possage

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

Section: 3.6 **Type:** Electrical

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

Description: Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard).

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

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

Group: 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** C

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

While checking for Proper Electrical Response per RT-19, you find the Main Generator Output Breakers and Exciter Field Breaker CLOSED.

What is the correct action to take based on 125V DC Bus D01 status?

If D01 is _________ ,then open _________.

A. Energized
Both Main Generator Output Breakers ONLY

B. Energized
Both Main Generator Output Breakers AND Exciter Field Breaker

C. De-Energized
Both Main Generator Output Breakers ONLY

D. De-Energized
Both Main Generator Output Breakers AND Exciter Field Breaker

Answer:

B. Energized
Both Main Generator Output Breakers AND Exciter Field Breaker

Notes:

B is correct per RT-19.
A is incorrect but plausible since the output breakers are opened but the field breaker must be opened also.
C and D are incorrect combinations of A and B. If D01 is de-energized, then the procedural direction is to leave them closed.

References:

1202.012, Repetitive Tasks, RT-19

History:

New for 2014 Exam

CHECK PROPER ELECTRICAL RESPONSE

1. Check 125 V DC Bus D01 energized:

- **Turbine Trip Solenoid Power Available light lit**
- **Breaker position indications available on left side of C10**

A. **IF** 125 V DC Bus D01 is de-energized,
THEN inform CRS to perform "Loss of D01" section of Loss of 125 V DC (1203.036) in conjunction with Reactor Trip procedure, while continuing.

2. Check Main Generator and Exciter Field breakers open.

- **5114**
- **5118**
- **Exciter Field breaker**

A. **IF** Main Generator and Exciter Field breakers are closed,
THEN perform the following:

1) **IF** 125 V DC Bus D01 is energized,
THEN perform the following:

a) Inform the CRS.

b) Manually trip Main Generator breakers:

- 5114
- 5118

c) Manually trip Exciter Field breaker.

2) **IF** 125 V DC Bus D01 is de-energized,
THEN leave Main Generator and Exciter Field breakers closed.

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

QID: 0914 **Rev:** 0 **Rev Date:** 9/12/14 **Source:** New **Originator:** Possage

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

Section: 3.6 **Type:** Electrical

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

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

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

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

Group: 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** K

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

What is the power supply to DC Distribution Panel RA2?

A. D01 - 125 V DC Bus

B. D11 - Distribution Panel

C. D02 - 125 V DC Bus

D. D21 - Distribution Panel

Answer:

C. D02 - 125 V DC Bus

Notes:

C is the correct power supply for RA2.
A, B, and D are incorrect choices but are plausible since they are major DC busses or panels

References:

STM 1-32, Electrical Distribution

History:

New for 2014 Exam

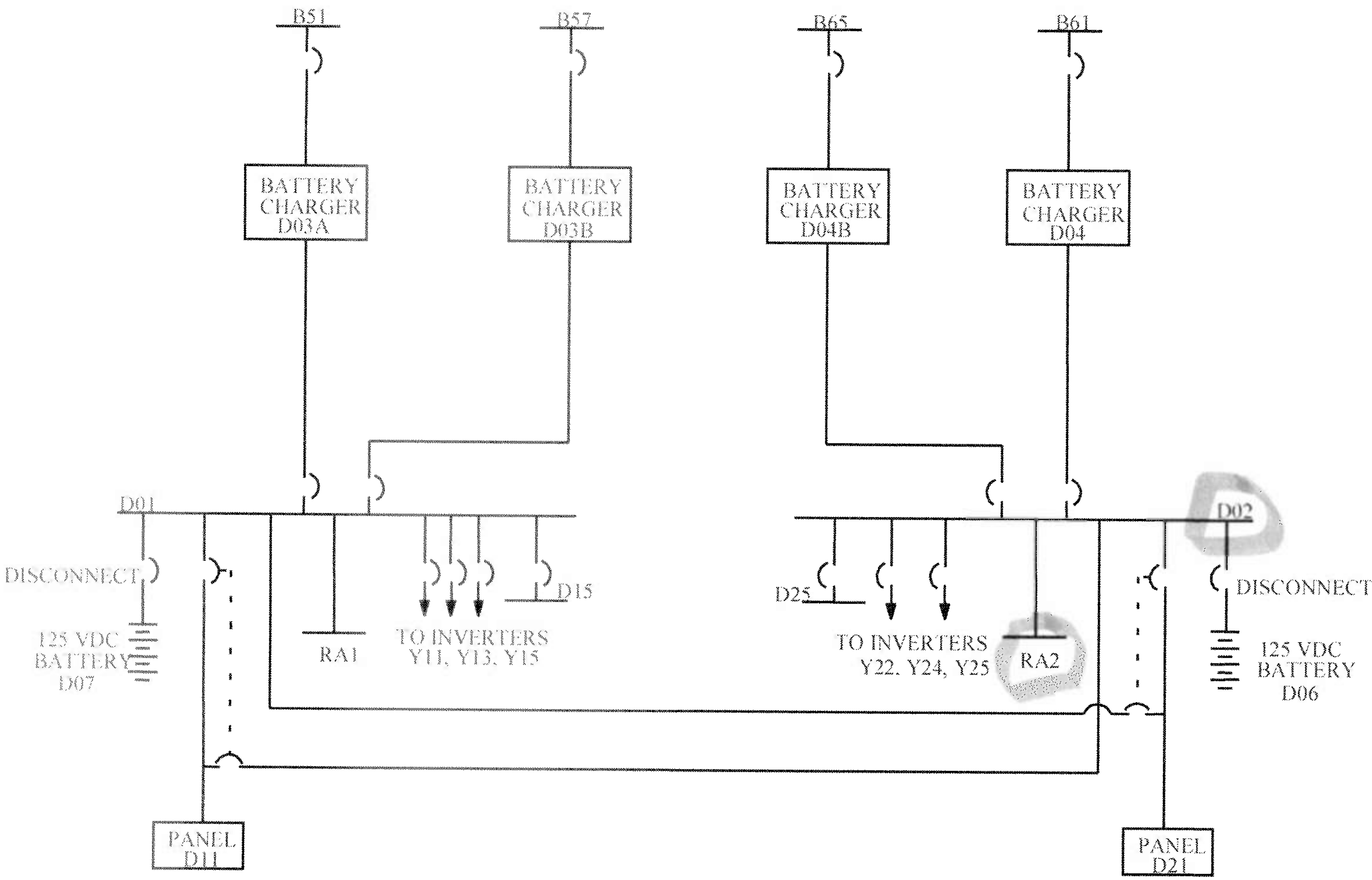


FIGURE 32.47: 125 VDC DISTRIBUTION

QID: 0915 **Rev:** 0 **Rev Date:** 9/12/14 **Source:** New **Originator:** Possage

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

Section: 3.6 **Type:** Electrical

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

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ED/G system controls including: Crankcase temperature and pressure.

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

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 2.9 **SRO Select:** Yes **Taxonomy:** C

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

The #1 EDG just tripped during a surveillance test.

If a positive crankcase pressure tripped the EDG, this would be indicated by ________ and Emergency Diesel Generator was tripped by ________.

A. High Cranckcase Pressure Critical Trouble Alarm
Energizing Emergency Trip Relay (K11)

B. High Cranckcase Pressure Trouble Alarm
De-Energizing Emergency Trip Relay (K11)

C, Low Lube Oil Pressure Critical Trouble Alarm
Energizing Emergency Trip Relay (K11)

D. Low Lube Oil Pressure Critical Trouble Alarm
De-Energizing Emergency Trip Relay (K11)

Answer:

C, Low Lube Oil Pressure Critical Trouble Alarm
Energizing Emergency Trip Relay (K11)

Notes:

C is correct, the Crankcase Trip Device drains the lube oil from the sensing line for the Low Lube Oil Pressure trip switiches, thus generating the associated alarm and the Emergency Trip Relay is energized to trip.
A and B are incorrect but plausible since it stands to reason that a high crankcase pressure would generate a corresponding alarm but no such alarm exists.
D is incorrect but plausible since this choice has the correct alarm but the K11 relay is energized, not de-energized, to trip the EDG.

References:

1104.036, Emergency Diesel Generator Operation

History:

New for 2014 Exam

18.0 Resetting Crankcase Pressure Trip Mechanism

CAUTION

- A crankcase pressure trip can indicate serious damage within engine and a potentially explosive condition.
- Except in the most serious emergency situations, trip should NOT be reset until cause of problem has been determined and corrected.
- To prevent internal explosions, do NOT open any engine inspection port until engine has cooled for at least 2 hours.

NOTE

The crankcase pressure trip device (PS-5283 for DG1, PS-5284 for DG2) is located on northeast end of diesel below fuel rack lever. A trip is indicated when round plunger in center of diaphragm protrudes. The trip device acts to dump lube oil in the sensing line for the low lube oil pressure trip switches, thereby causing the engine to trip on an indicated low lube oil pressure. There is no separate crankcase pressure trip alarm.

18.1 IF desired to reset DG1 crankcase pressure trip, THEN perform the following:

NOTE

Re-pressurization of crankcase pressure diaphragm can take several seconds.

18.1.1 Depress AND hold reset plunger on the crankcase pressure diaphragm until plunger remains latched in.

18.1.2 At DG1 Engine Control Panel (C107), depress RESET pushbutton (PB-5231A).

CAUTION

If an automatic start signal is present, the DG will restart immediately after resetting the DG lockout relay.

18.1.3 Verify all personnel are clear of DG1 room.

18.1.4 At DG1 Output Breaker (A-308), reset DG1 lockout relay (186-DG1) by turning lockout relay handswitch clockwise to vertical latched position.

18.1.5 WHEN emergency situation requiring DG1 is corrected, THEN crankcase pressure trip device shall be replaced or proven operational per vendor guidelines.

18.1.6 WHEN cause of crankcase pressure trip has been determined and corrected, THEN DG1 shall be proven operable per "DG1 Monthly Test" Supplement 1 of this procedure.

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

QID: 0917 **Rev:** 0 **Rev Date:** 9/12/14 **Source:** New **Originator:** Cork

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

Section: 3.7 **Type:** Instrumentation

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

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

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** An

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

Given:
- A release of Waste Gas Decay Tank T-18C is in progress.
- RADIATION MONITOR TROUBLE, K10-C1, alarms
- Gaseous Radwaste Monitor RI-4830 FAILURE ALARM light is on
- Automatic valve closures were verified to have occurred

Which of the following procedurally required actions shall be taken for the above conditions?

A. Reset RI-4830 and re-establish the release per the release permit.

B. A licensed operator, other than the individual who initially adjusted RI-4830, shall reset RI-4830 and re-establish the release per the release permit.

C. Perform an independent verification of the release lineup per the ODCM and re-establish the release per the release permit.

D. Reinitiate the release permit procedure prior to re-establishing the release.

Answer:

D. Reinitiate the release permit procedure prior to re-establishing the release.

Notes:

D is correct, two independent samples must be taken and analyzed so the current release paperwork is invalid.
A is incorrect but plausible as this is the action taken if an instantaneous spike of RI-4830 occurs.
B is incorrect but plausible as this is how the setpoint for RI-4830 is verified prior to the release.
C is incorrect but plausible since this is the action taken if RI-4830 is inoperable but the tank must also be independently sampled and anlyzed..

References:

1203.012I, Annunciator K10 Corrective Action
1104.022, Gaseous Radwaste System

History:

New for 2014 Exam

Location: C16

Device and Setpoint:
De-energization of or FAILURE ALARM on any radiation monitor in Radiation Monitoring System Panel (C25 Bays 1-3 and Bay 4 of C24). Monitors are listed on page 3.

RADIATION
MONITOR
TROUBLE

Alarm: K10-C1

1.0 OPERATOR ACTIONS

1. Observe monitors at C24 and C25 for FAILURE ALARM light(s) on or POWER ON light(s) off.

2. IF power is off to all monitors in a bay,
THEN check supply breaker closed:

- Rad Monitor Panel C24, Rad Monitor Panel C25, Bay 1 (RS1, bkr 8)
- Rad Monitor Panel C24, Rad Monitor Panel C25, Bay 2 (RS2, bkr 8)
- Rad Monitor Panel C25, Bay 3 (RS4, bkr 8)

A. IF breaker is tripped,
THEN reclose tripped breaker per "Reclosing Tripped Individual Load Supply Breakers" section of Electrical System Operations (1107.001).

3. IF either of the following monitors is inoperable (FAILURE ALARM or power loss):

- Spent Fuel Pool (RI-8009)
- Fuel Handling Area (RI-8017)

AND fuel handling in progress,
THEN stop fuel handling until radiation monitoring requirement is satisfied per Control of Unit 1 Refueling (1502.004) OR Control of Fuel and Control Rod Movement in the U-1 Spent Fuel Area (1502.010).
(TRM 3.9.1 and TRM 3.9.2)

4. IF Liquid Radwaste (RI-4642) is de-energized,
THEN verify CZ Disch to Flume Flow (CV-4642) is closed or auto closes.
(ODCM L2.1.1)

NOTE

The following alignment stops gaseous release and diverts flow to Waste Gas Surge Tank (T-17).

5. IF Gaseous Radwaste (RI-4830) is de-energized,
   THEN verify the following: (ODCM L2.2.1)
   - T-18s Discharge to Gaseous Radwaste Discharge Header Flow Control (CV-4820) closed
   - Gaseous Radwaste Discharge Isol (CV-4830) closed
   - ABVH Diversion to T-17 (CV-4806) open
6. IF RB Atmos Gaseous Monitor is inoperable,
   THEN refer to Reactor Building Ventilation (1104.033). (TS 3.4.15)
7. Initiate steps to survey areas for which radiation monitors are inoperable.
8. Initiate steps to have failed monitor(s) checked and repaired.
9. IF alarm was caused by FAILURE ALARM on monitors,
   THEN all monitors that are failed must be reset using ALARM RESET switch on front of monitor to clear K10-C1.

2.0 PROBABLE CAUSES

NOTE

- This annunciator has reflash capability. If the alarm window is lit solid due to one cause and another cause actuates, the alarm will go to fast flash with an audible alarm.
- FAILURE ALARM light on monitor indicates that the monitor has had no input from the detector for one minute; detector failure.

1. Any radiation monitor FAILURE ALARM in C25 or Bay 4 of C24
2. De-energization of any radiation monitor in C25 or Bay 4 of C24
3. Any radiation monitor in C25 or Bay 4 of C24 alarm lamp removed or burned out

ATTACHMENT C

1.6 Shift Manager/CRS Approval ______________________
SM/CRS

1.7 Submitted to Chemistry for analysis

by: ____________________ Date __________ Time ________

2.0 ANALYSIS (Chemistry)

2.1 Sample of Waste Gas Decay Tank ______ for gamma spectroscopy obtained.

by: ____________________

2.1.1 Record M&TE number __________

Cal Expiration Date __________

> **NOTE**
> If an independent sample and analysis is needed per step 2.4, independent sampling and analysis may be performed concurrently with the following steps.

2.2 Gamma spectroscopy performed by: ______________________

2.3 Gamma spectroscopy report reviewed by: ______________________

2.4 <u>IF</u> Gaseous Radwaste Process Monitor (RI-4830) is non-functional <u>OR</u> is unavailable (per steps 1.5 or 3.7.1), <u>THEN</u> perform the following: <u>OTHERWISE</u> mark 2.4.1 and 2.4.2 N/A. (ODCM App.1, L2.2.1.a)

2.4.1 Two independent samples shall be obtained and analyzed. Independent sample and analysis performed by:

______________________ Date __________

2.4.2 Two independent verifications of computer input data shall be performed. Independent verification performed by:

______________________ Date __________

2.5 Preliminary release report generated by: ______________________

2.6 Tank pressure at which release is to be terminated __________psig

by: ____________________.

2.7 SPING 2 setpoint value(s) adjusted per 1604.051 and Form 1604.051E

by: ____________________.

2.8 Preliminary report returned to operations

by: ____________________ Date __________ Time ________

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

QID: 0918 **Rev:** 0 **Rev Date:** 9/12/14 **Source:** New **Originator:** Possage

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

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

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

Description: Knowledge of the effect that a loss or malfunction of the SWS will have on the following: RHR components, controls, sensors, indicators, and alarms, including rad monitors.

K/A Number: K3.05 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** C

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

Given:
- 'A' DHR Train in service
- Chemistry reports 0.5% Failed Fuel

Which of the following would indicate a leak in DH Cooler (E-35A)?

A. Rising counts on SW Loop I Process Monitor (RI-3814)

B. Rising counts on SW Loop II Process Monitor (RI-3815)

C. Rising counts on Decay Heat Loop A Process Monitor (RI-3809)

D. Rising counts on Failed Fuel Process Monitor (RI-1237)

Answer:

C. Rising counts on Decay Heat Loop A Process Monitor (RI-3809)

Notes:

C is correct, RI-3809 monitors SW coming out fo the A DH cooler.
A is incorrect but plausible since this is on Loop I but this RM monitors SW from the RB coolers.
B is incorrect but plausible since this is on Loop II (wrong loop) but this RM monitors SW from the RB coolers.
D is incorrect but plausible as this monitor will be rising due to the failed fuel condition, but a rise in the failed fuel monitor would not indicate a leak in the A DH cooler.

References:

STM 1-05, Decay Heat Removal System
STM 1-42, Service and Auxiliary Cooling Water

History:

New for 2014 Exam

air operated control valve with no manual isolation valves provided. Operating the cooler outlet and bypass valves in conjunction with each other allows the operator to maintain a constant discharge flow of approximately 3000 gpm. While varying the amount of flow through and bypassing the cooler the operator can control the RCS cooldown rate.

The coolers also have vent tanks with sight glasses and a separate flow gauge. Water that collects in the vent tanks is drained to the Auxiliary Building Sump. Gasses vented through the flow gauge are piped to the Gaseous Radwaste header.

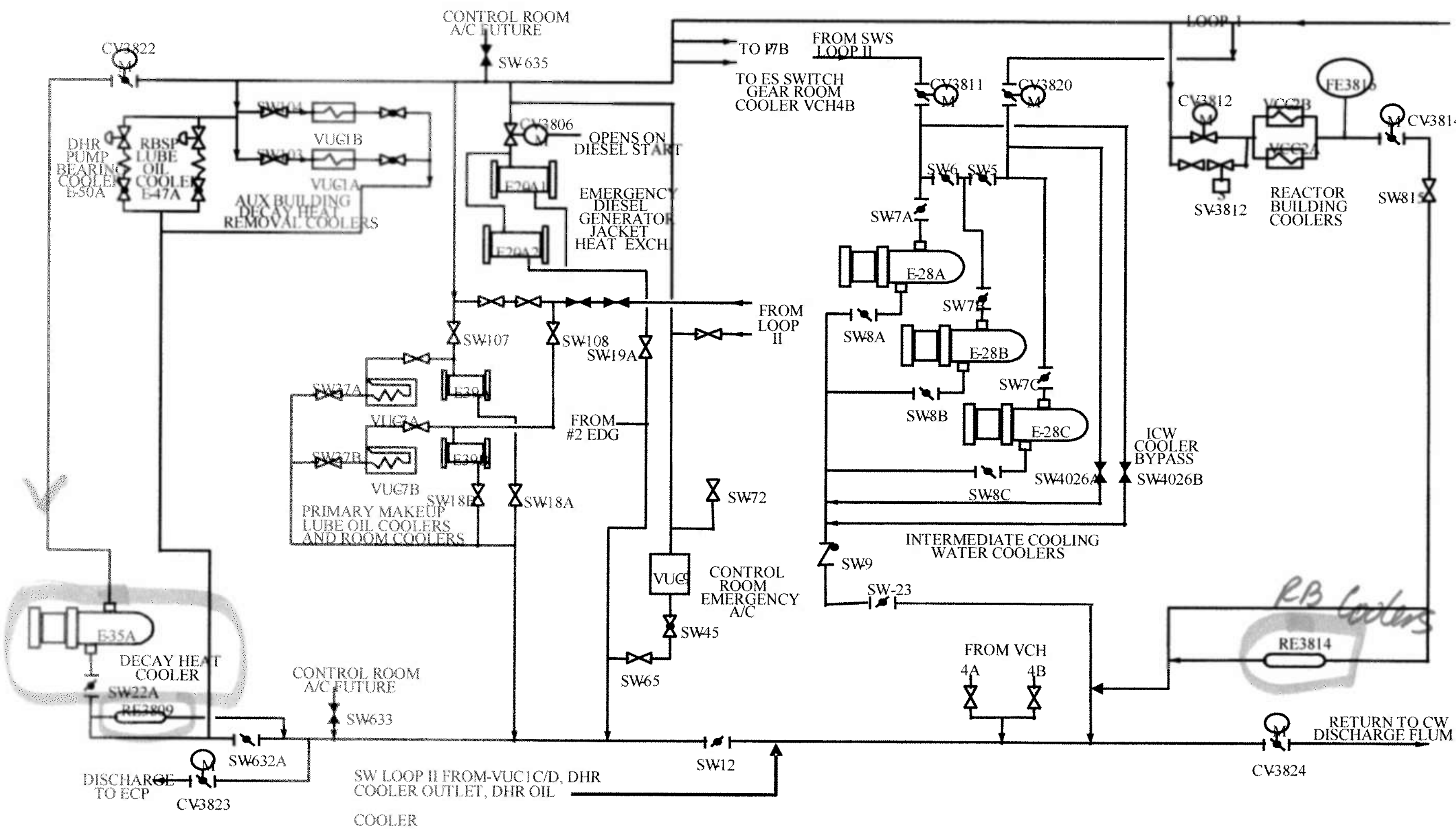
The DHR Cooler shell side (RCS / BWST) has a design pressure and temperature of 450 psig and 300°F. DHR/LPI supply line to each cooler is equipped with a pressure relief valve, which provides overpressure protection for the cooler and associated piping. The relief valves have a setpoint of 445 psig and relieve to the auxiliary building sump. The relief valves are designated as PSV-1407 (P-34A) and PSV-1406 (P-34B). The relief valves are located in their associated pumps decay heat vault. Due to the ability to cross-connect the decay heat pumps through the purification loop and auxiliary spray system piping, procedural guidance is provided during shutdown conditions.

The tube side (SW) has a design pressure and temperature of 120 psig and 300°F.

Service Water supply line to each cooler is equipped with a Motor Operated isolation valve (MOV). Each SW isolation valve receives an open signal when the associated pump breaker is closed. CV-3822 opens when P-34A is started and CV-3821 opens when P-34B is started. SW enters the cooler through a 12" pipe entering the top of the heat exchanger end-bell. SW flows through the tubes of the "U" tube heat exchanger exiting at the lower half of the inlet end-bell. SW flow through the cooler without regulating the flow would be ~3000 gpm. During operation when RCS temperature is greater than 200°F, SW flow through the cooler is controlled by throttling the SW discharge isolation valve. The service water discharge isolation valves are designated as SW-22A & SW-22B. SW flow through the cooler is maintained >1600 gpm by positioning SW-22A & SW-22B to a scribed "T" mark located on the valve.

The throttling of the service water to the DHR coolers provides sufficient backpressure to ensure other components cooled by the service water system have adequate flow during an ESAS actuation. SW-22A & SW-22B can not be removed from the scribed "T" mark except when RCS temperature is less than 200°F or during Surveillance testing of associated DH pump. SW-22A & SW-22B are category "E" controlled manual valves.

Service water out of the cooler is equipped with a process monitor used to detect a cooler leak. The process monitor associated with E-35A outlet is RE-3809 and RE-3810 for E-35B. When a cooler leak occurs the process monitor will alarm causing K10-B2



NOTE:
DRAWING DOES NOT SHOW INTERCONNECTIONS
FROM SW LOOP II TO COMMON RETURN
HEADER BETWEEN VALVES CV3823 & CV3824

FIGURE 42.01A: SERVICE WATER SYSTEM LOOP I

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

QID: 0714 **Rev:** 2 **Rev Date:** 4/10/2008 **Source:** Direct **Originator:** Exam Bank

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

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

System Number: 078 **System Title:** Instrument Air System (IAS)

Description: Knowledge of the affect that a loss or malfunction of the IAS will have on the following: systems having pneumatic valves and controls.

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

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	K

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

A Loss of Instrument Air has occurred.

What is the expected position of the Startup (CV-2623 and CV-2673) and Low Load Control Valves (CV-2622 and 2672)?

A. Startup valves fail closed,
Low Load valvess fail closed.

B. Startup valves fail closed,
Low Load valves fail as is.

C. Startup valves fail as is,
Low Load valves fail as is.

D. Startup valves fail as is,
Low Load valves fail closed.

Answer:

C. As Is

Notes:

C is correct.
Although A, B, and D are plausible since they are possible failure modes of air valves, these Feedwater control valves do not fail in these modes.

References:

1203.024, Loss of Instrument Air

History:

Selected for the 2008 RO Exam.
Selected for 2014 Exam

ATTACHMENT A

CRITICAL AIR OPERATED COMPONENTS

Startup Valves (CV-2623 and CV-2673)

Startup valves fail AS IS on a loss of Instrument Air. At low power levels, this can result in a loss of feedwater control to the SGs. At high power levels, this will have no effect on feedwater since Main Feedwater Block valves are open.

Low Load Valves (CV-2622 and CV-2672)

Low load valves fail AS IS on a loss of Instrument Air. Manual reset is required at the valve to reset. At low power levels, this can result in a loss of feedwater control to the SGs. At high power levels, closure of these valves will have a minor effect on feedwater since Main Feedwater Block valves are open.

TURB BYP Valves (CV-6687, CV-6688, CV-6689, and CV-6690)

Turbine bypass valves fail CLOSED on a loss of Instrument Air (≤ 55 psig). Each valve is equipped with two small air reservoirs to maintain the valve closed for a short duration. If the respective steam line is pressurized, the line pressure will force open the turbine bypass valve after the air reservoirs are depleted. If the reactor is tripped, overcooling could result from open turbine bypass valves. At high power levels, a reactivity excursion could result from open turbine bypass valves.

MSIVs (CV-2691 and CV-2692)

Main steam isolation valves are provided with air reservoirs to minimize the effect of a loss of Instrument Air. They are MSIV CV-2692/ADV CV-2618 Air Accum Tank (T-93A) and MSIV CV-2691/ADV CV-2668 Air Accum Tank (T-93B). MSIVs will fail CLOSED on a loss of Instrument Air as their reservoirs lose pressure. Testing has shown that MSIVs close as pressure drops as follows:

	Begin Closing	Fully Closed
MSIV(CV-2691)	67 psig	37 psig
MSIV(CV-2692)	66 psig	34 psig

ATM Dump CNTRL valves (CV-2618 and CV-2668)

ATM Dump CNTRL valves fail CLOSED on a loss of Instrument Air (≤ 42 psig). Each valve is equipped with two small air reservoirs to maintain the valve closed. Air for these valves is supplied from MSIV CV-2692/ADV CV-2618 Air Accum Tank (T-93A) and MSIV CV-2691/ADV CV-2668 Air Accum Tank (T-93B).

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

QID: 0919 **Rev:** 0 **Rev Date:** 9/13/14 **Source:** New **Originator:** Possage

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

K/A Number: 2.4.34 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 2 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** C

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

1203.002, Alternate Shutdown AOP in progress.

While RO #1 is manually opening CV-1408, BWST OUTLET he notices significant flow noise and vibration indicating ~1000 gpm flow.

What should RO #1 do based on the above conditions?

A. Close CV-1408, piping failure is indicated.

B. Close CV-1408, go to Decay Heat Vaults and Close RB Sump Outlet Valves.

C. Throttle CV-1408, reduce the flow rate to prevent waterhammer as the piping fills.

D. No actions necessary, flow is expected due to HPI Pump running.

Answer:

B. Close CV-1408, go to Decay Heat Vaults and Close RB Sump Outlet Valves.

Notes:

B is correct per 1203.002 Section 1C.
A is incorrect but plausible as it contains part of the correct response but not the sump valves.
C is incorrect but plausible as these are the actions for filling a voided line.
D is incorrect but plausible as flow noise could be expected if an HPI pump were running with suction aligned from the MU Tank. When CV-1408 was opened the head of the BWST would overcome the pressure in the MUT.

References:

1203.002, Alternate Shutdown

History:

New for 2014 Exam

3.4 At ERV Isolation Valve CV-1000 breaker (B6142), place Local/Remote handswitch in LOCAL.

3.4.1 IF B6142 is energized,
THEN perform the following:

A. Verify CV-1000 closed.

B. WHEN CV-1000 is closed,
THEN open B6142.

C. Notify SM that CV-1000 is in LOCAL, CV-1000 is closed and B6142 is open.

CAUTION

Opening BWST Outlet (CV-1408) may drain BWST to the RB sump.

3.5 Perform the following to open BWST Outlet (CV-1408):

3.5.1 Begin to open CV-1408 while monitoring for significant flow through valve indicated by throttling noise and vibration relative to ~1000 gpm.

3.5.2 IF significant flow is detected,
THEN perform the following:

A. Close CV-1408.

B. Go immediately to Decay Heat vaults
AND verify the following valves closed:

- RB Sump - Line A Outlet (CV-1405)
- RB Sump - Line B Outlet (CV-1406)

3.5.3 WHEN there is no significant flow,
THEN open CV-1408.

3.6 In Aux Bldg stairwell, verify Makeup Tank Outlet valve (CV-1275) closed.

3.7 In UNPPR, open the following Green train valves:

- "B" HPI Block Valve (CV-1227)
- "A" HPI Block Valve (CV-1228)

3.8 In UNPPR, close RCP Seal Injection Block (CV-1206).

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

QID: 0326 **Rev:** 0 **Rev Date:** 9-8-99 **Source:** Direct **Originator:** Stanley

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

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

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

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

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

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

Group: 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** C

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

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

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

a. Rise due to loss of flow to the seal cooler.

b. Rise due to bleedoff in excess of seal cooler capacity.

c. Remain the same due to seal bleedoff cooling flow.

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

Answer:

b. Increase due to bleedoff in excess of seal cooler capacity.

Notes:

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 180°F. Therefore "b" is correct. "a" is incorrect because ICW supplies the seal cooler. "c" is incorrect because there is no seal bleedoff cooling. "d" is incorrect because the recirc impeller provides flow of RCS through the RCP seal cooler and back to the seal.

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.

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. IF any of the following conditions exist,
THEN 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. IF another stage shows sign of failure,
THEN 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: 1022 **Rev:** 0 **Rev Date:** 9/23/14 **Source:** Modified **Originator:** Possage

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

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

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

Description: Ability to monitor automatic operation of the CVCS, including: Water and boron inventory.

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

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

Group: 1 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** AP

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

Given:

Plant at 100%
Letdown flow 80 gpm indicated on FI-1236
Letdown pressure 50 psig on PI-1237

Subsequently, CV-1244 and CV-1245 Letdown DI Inlet Isolation valves lose power.

With no operator action what would be the expected automatic response of the pressurizer level control system ?

PI-1237 would read ______ psig and Makeup Tank Level will ________ .

A. 150 ; rise

B. 150 ; drop

C. 50 ; rise

D. 50 ; drop

Answer:

B. 150 ; drop

Notes:

B is correct, letdown DI Inlet Isolation Valves fail closed on a loss of power which would isolate letdown, letdown pressure would rise to the automatic action of the letdown relief opening (150 psig) , causing Makeup Tank Level to drop.
A, C, and D are variations of these possible combinations.

References:

STM 1-04, Makeup and Purification System

History:

Modified QID 797 for 2014 Exam

To cope with these impurities, the purification system incorporates 1 prefilter, 2 DI's and 2 post filters. The demineralizers remove ionic impurities and have some filtering capability for suspended materials. Each demineralizer contains a bed of mixed cation and anion resins. One unit is normally operating while the second unit is in standby.

The F-3A & B filters are used to keep resin fines and any particulate material that may pass through the DI's from entering the remainder of the purification system and the RCS.

2.13.1 Purification Demineralizer Inlet Valves, CV-1244 and CV-1245

These air operated gate valves are used to place either of the purification demineralizers in service. CV-1244 is for Demineralizer T-36A and CV-1245 is for Demineralizer T-36B. Both valves are operated from panel C-04 and have solenoid actuated, air operated, single acting cylinder operators. One valve is normally open, the other normally closed. These valves will fail closed on loss of air or power.

2.13.2 Demineralizer Bypass Valve, MU-9

This valve is used to bypass the purification demineralizers. It's use is directed during recovery from high temperature conditions to prevent LD DI resin depletion. During high temperature conditions in the letdown line, this valve should be opened prior to opening the letdown isolation valve (CV-1221). MU-9 is a manual valve.

2.13.3 Purification Demineralizers (DI's), T-36A/B

The HOH mixed bed demineralizers (DI's) are used to remove reactor coolant impurities from the letdown stream. Since the reactor coolant may be contaminated with dissolved fission and corrosion products, ion exchange resins are used to clean the reactor coolant. The resins remove radioactive impurities and reduce the radiation levels that might otherwise be present in the RCS piping.

Normally, the operating demineralizer is saturated with boron at a concentration equal to RCS boron concentration. The standby demineralizer may be unsaturated. This allows use of the standby demineralizer to remove boron late in core life to keep the reactor operating. A positive reactivity addition hazard may occur if the wrong DI is placed in service during power operation. The mixed bed HOH resin will remove the boron from the water passing through it. This will continue until the HOH resin comes up to an equiliabrium concentration of boron that equals RCS concentration. The RCS water passing into the demineralizer also may have an excess concentration of lithium-7, in the form of Lithium Hydroxide (LiOH). LiOH is used for pH control, thus corrosion control of the reactor coolant system. The HOH resin will also remove the Lithium from the water passing through it. This will continue until the HOH resin comes up to an equilibrium concentration of lithium that equal the RCS concentration.

Maximum and minimum flows through one demineralizer are 123 gpm (to prevent resin compacting) and 25 gpm (to avoid channeling) respectively. Table 4.2 contains design data for the purification demineralizers.

The standby DI may be used to remove fission products when the need arises. Operating experience has demonstrated that mixed

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

QID: 0797 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** Direct **Originator:** S. Pullin

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

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

System Number: 011 **System Title:**

Description:

K/A Number: A3.03 **CFR Reference:** 41.7 /45.5

Tier: 2 **RO Imp:** **RO Select:** No **Difficulty:** 3

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

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

Given:

Plant at 100%
Letdown flow 80 gpm indicated on FI-1236
Letdown pressure 50 psig on PI-1237

CV-1244 and CV-1245 Letdown DI Inlet Isolation valves lose power.

With no operator action what would be the expected automatic response of the pressurizer level control system ?

A. PI-1237 would read 50 psig and Pressurizer level control valve CV-1235 position would close.

B. PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would open.

C. PI-1237 would read 50 psig and Pressurizer level control valve CV-1235 position would open.

D. PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would close.

Answer:

B. PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would open.

Notes:

B is correct, due to letdown DI Inlet Isolation Valves fail closed on a loss of power. Which would isolate letdown, letdown pressure would rise to the letdown relief setpoint of 150 psig, causing a LOCA. Pressurizer level would go down causing CV-1235 to open.
A,C, and D are variations of these possible combinations.

References:

STM 1-04, Makeup and Purification System

History:

New for 2010 RO/SRO exam. KA 011 A3.03

PARENT

QID: 0920 **Rev:** 0 **Rev Date:** 9/13/14 **Source:** New **Originator:** Possage

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

Section: 3.7 **Type:** Instrumentation

System Number: 012 **System Title:** Reactor Protection System (RPS)

Description: Ability to manually operate and/or monitor in the control room: Channel blocks and bypasses.

K/A Number: A4.03 **CFR Reference:** 41.7 / 45.5 to 45.8

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

Group: 1 **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** C

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

Given:
- Unit 1 is at 100% power.
- I&C is performing a surveillance on the "A" RPS channel.
- NI input to ICS is selected to "C" RPS channel.

What would occur if someone inserted the key and placed "C" RPS channel in bypass?

A. Reactor trip

B. "A" RPS channel would automatically be removed from bypass

C. Two RPS channels would be in bypass

D. No effect, interlock would prevent this action

Answer:

D. No effect, interlock would prevent this action

Notes:

D is correct, an interlock prevents two channels from being in bypass.
A is incorrect but plausible if the examinee believes this would trip the C channel and I&C has tripped the A channel.
B is incorrect but plausible since EFIC works in this fashion.
C is incorrect but plausible if the examinee does not recall the interlock.

References:

STM 1-63, Reactor Protection System

History:

New for 2014 Exam

front panel of each reactor trip module. Four are on each reactor trip module (one for each logic relay in that module) with a "SIMULATE TRIP" label below them. Provided for testing, each switch will de-energize one of the four logic relays in that module. If two switches are taken to test, or if a channel trip signal is present from another channel when one switch is taken to test, the coincidence logic will be satisfied and the breaker associated with that channel will open.

The reactor trip modules also contain the coincidence logic for the reactor protection system. This logic is in the form of eight contacts in two networks of four contacts each. Each contact logic network consists of two sets of two contacts in series, with the two sets being parallel. The provision of two networks assures that the two out of four coincidence logic is met.

A key operated switch is included on the front plate layout of the reactor trip module. The function of this switch is to provide manual bypass of the channel trip. Placing of the switch into the bypass position energizes the manual bypass relay, provided that none of the other reactor protection system channels are in channel bypass. Contacts operated by the manual bypass relay close to supply -15 volt DC to the channel trip relay, bypassing all other contacts which, when opened, would de-energize it. Other contacts are also operated by the manual bypass relay to prevent actuation of the bypass function in any other reactor protection system channel. The bypass function allows a channel to be placed in test without initiating a trip signal to the coincidence logic. With a channel bypassed, the reactor protection system is placed into a two out of three coincidence logic, meaning that two of the three remaining channels must trip for a reactor trip to occur. This bypass condition is also referred to as maintenance bypass.

Eight indicating lamps are provided on the reactor trip module. Each lamp has two states, bright or dim. In the normal condition logic relay energized the lights will be dim. When in a trip condition (logic relays de-energized) the indicating lights will be bright. Four lamps provide status of each channel logic relay. The lamp labeled "Subsystem Trip", located just above the channel reset toggle switch will be dim if a flow-path for current through the channel contact string exists. If contacts open and break the current flow path, a relay repositions a contact, resulting in a bright "Subsystem Trip" lamp. This lamp will be bright if the contact string current flow path is broken even if the channel bypass switch is in bypass. (The "Subsystem Trip" lamp is not an indication of the channel trip relay status). The lamp labeled "Test Trip" goes bright if the module-test /interlock relay is de-energized by placing of a test module in other than the operate position or by removal of a critical module from the reactor protection system. The "Test Trip" lamp will indicate bright under these circumstances even when the channel is in maintenance bypass. The "Manual Bypass" lamp will go to bright when the channel bypass key-switch is placed in the bypass condition as long as none of the other RPS channels are in bypass.

The seven lights covered to this point all have white lens caps. The Reactor Trip lamp is red and differs in one other aspect, the lamp does not indicate the operation of a relay within the reactor trip module. The lamp, "when bright", indicates the satisfactory

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

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

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

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

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

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

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

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** 52 **SRO:** 52

Given:
- Reactor tripped due to loss of all offsite power.
- RCS T cold is 535°F and dropping.
- RCS pressure 1800 psig and dropping.
- OTSG pressures are ~910 psig and dropping.
- "A" OTSG level is 210" and rising.
- "B" OTSG level is 195" and rising.
- "A" EFW flow is 370 gpm.
- "B" EFW flow is 350 gpm.

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

A. Maintain >590 gpm to each SG in HAND.

B. Throttle EFW to prevent overcooling.

C. Select Reflux Boiling setpoint.

D. Actuate MSLI on both OTSGs.

Answer:

B. Throttle EFW to prevent overcooling.

Notes:

"B" is correct since overcooling is evident by RCS pressure and T cold dropping. SCM is adequate and both SGs have >340 gpm flow so EFW should be throttled per RT-5.
"A" is incorrect, this is EFW flow value is higher than the current flow rates and is above the value to be used if SCM was inadequate and only one SG is available.
"C" is incorrect, although no RCPs are running, SCM is adequate, therefore the REFLUX BOILING setpoint would be inappropriate.
"D" is incorrect, although overcooling is evident, MSLI is too drastic a measure as SGs are much greater than 600 psig, EFW should be throttled first.

References:

1202.012, Repetitive Tasks, RT-5

History:

Created for 2002 RO/SRO exam.
Selected for 2005 RO re-exam.
Selected for 2011 RO Exam. KA 061 K5.01
Selected for 2014 Rxam

VERIFY PROPER EFW ACTUATION AND CONTROL

4. IF SCM is adequate,
THEN perform the following:

CAUTION
Excessive EFW flow can result in loss of SCM due to RCS shrinkage.

NOTE

- Table 2 contains examples of less than adequate/excessive EFW flow.
- Expect CETs to rise until natural circ conditions are established. If EFW flow control is in HAND, additional flow may not be necessary to prevent rising CETs until natural circ conditions are established.

A. Verify EFW CNTRL valves operate to establish and maintain applicable SG level band per Table 1.

1) IF EFW flow is less than adequate
OR
EFW flow is excessive,
THEN control EFW to applicable SG in HAND as necessary to ensure the following:

- Maintain sufficient EFW flow to prevent rise in CET temp.
- Maintain continuous EFW flow until applicable level band is reached.
- Maintain sufficient EFW flow to ensure SG level is either stable **OR** rising until applicable level band is reached.

5. IF all 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

6. Monitor EMERGENCY FEEDWATER and EFIC alarms on K12.

VERIFY PROPER EFW ACTUATION AND CONTROL

Table 1		
EFIC Automatic Level Control Setpoints		
Condition	**Level Band**	**Automatic Fill Rate**
Any RCP running	20 to 40”	No fill rate limit
All RCPs off and Natural Circ selected	300 to 340”	2 to 8”/min
All RCPs off and Reflux Boiling selected	370 to 410”	2 to 8”/min

Table 2
Examples of Less Than Adequate EFW Flow Indications
• SG level < 20” and no EFW flow indicated • All RCPs off and SG level not tracking EFIC calculated setpoint • All RCPs off and EFIC level setpoint not trending toward applicable level band
Examples of Excessive EFW Flow Indications
• SG press drops ≥ 100 psig due to EFW flow induced overcooling • SCM approaching minimum adequate due to EFW flow induced overcooling • EFW CNTRL valve open with associated SG level > applicable setpoint level band

END

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

QID: 0568 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** Direct **Originator:** S.Pullin

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

Section: 3.6 **Type:** Electrical

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

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:**	2
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	K

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

Given:

- #1 EDG has one Air Start Compressor and it's associated Air Receiver Tanks tagged out.
- The remaining Air Start Compressor on #1 EDG trips while running.
- The Air Receiver Tanks' pressure is 176 psig.

What is the MAXIMUM number of start attempts assured with the above #1 EDG conditions?

A. One

B. Three

C. Five

D. Seven

Answer:

C. Five

Notes:

"C" is correct per the STM, the others are odd numbered incorrect values.

References:

STM 1-31, Emergency Diesel Generator

History:

New for 2005 RO exam. KA 064 - A3.04
Selected for 2014 Exam

The alumina desiccant has a high affinity for moisture. The moisture attracted to the absorbent is held in pores in the desiccant. The air dryers reduce the dew point to -30°F.

Reactivating the idle tower is accomplished by allowing air to bleed through the idle tower. The air to reactivate the idle tower flows from the in service tower through a filter and flow restricting orifice into the idle tower outlet. The idle tower purge valve is open allowing the air to flow in a reverse direction through the tower.

To shift air dryer tower alignment, the idle tower purge valve closes. A solenoid valve opens to pressurize the tower. After a short time delay, the inlet valves of the towers shift. The reactivated air dryer tower is in service and the exhausted air dryer tower is out of service. The purge valve on the exhausted tower then opens to reactivate the exhausted air dryer tower.

The air dryers are automatically aligned by timer motor. The time cycle starts when the air compressor starts. The timer places an air dryer tower in service for five minutes and reactivates an air dryer tower for five minutes. The timer turns a set of cams which energize solenoid valves. The solenoid valves align or vent air to operate the air dryer inlet and purge valves. When the air start compressor stops, the timer motor is de-energized. When the air start compressor is again started, the air dryer alignment will be the same as when the air start compressor was stopped.

An inlet filter to the air dryers removes particulates and any oil mists which may degrade desiccant operation. An outlet filter removes any desiccant dust which may be carried with the air discharged from the dryer.

3.4.4 Air Start Receiver Tanks

The air start receiver tank capacity is 85.45 cubic feet. Air storage capacity is sufficient for five starts of the EDG from each pair of receiver tanks. Pressure switches monitor the receiver tank pressure. The switches provide a low pressure alarm and start/stop of the air start compressors.

3.4.5 Air Start Solenoid Valve

(Refer to figure 35

The air start solenoid valve provides the interface with the EDG control circuits. When an EDG start signal is received, the solenoid valve is energized. The resulting magnetic field pulls the valve plunger down. The vent port is closed and air is aligned to the starter motor bendix drive piston. When the engine is running the air start solenoid valve is de-energized. Spring force will close the air supply and vent the air from the starter motor bendix drive piston.

If DC control power is lost, a manual override is provided. The override pin is depressed to override and open the air start solenoid valve.

3.4.6 Air Start Valve

(Refer to figure 36)

The air start valve aligns air to the starter motors. Air is aligned to the air start valve actuating piston to open the valve. The air is

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

QID: 0535 **Rev:** 1 **Rev Date:** 10/13/05 **Source:** Direct **Originator:** J.Cork

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

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

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

Description: Knowledge of the physical connections and / or cause-effect relationships between the IAS and the following systems: Service Air

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

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.8	**SRO Select:**	Yes	**Taxonomy:**	K

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

Instrument Air pressure has dropped to 50 psig.

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

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

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

B. Unit 1 to Unit 2 Instrument Air cross-connect automatically opens.

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

D. Close Letdwon Cooler Outlet to isolate letdown

Answer:

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

Notes:

"B" is incorrect, this valve is closed when either unit IA pressure reaches 60 psig.
"A" is correct, this automatically occurs when pressure drops to 50 psig.
"C" is incorrect, this would not be done until pressure was less than 35 psig.
"D" is incorrect, this would not be done until pressure was less than 35 psig

References:

1104.025, Service Air System

History:

Developed for 1998 RO exam (similar to QID 102)
Modified question for 98 RO Re-exam
Modified for 2005 RO re-exam.
Selected for 2010 RO/SRO exam.
Selected for 2014 RO Exam.

1.0 PURPOSE

Provide instructions for startup and normal operation of the Service Air System.

2.0 SCOPE

Instructions are provided for aligning the Unit 1 SA system by crossconnecting Unit 1 and Unit 2 SA systems.

3.0 DESCRIPTION

The Service Air System provides a continuous supply of air for the plant. The system has outlets for pneumatic tool operation and is connected to various tanks for blowout and cleanup. Service air is also supplied to each RB spray header for testing the RB spray nozzles.

The Unit 1 and Unit 2 SA systems must be crossconnected unless a temporary compressor is installed for Unit 1. System pressure is maintained by the Unit 2 Sullair Yard Compressor (2C-43) located east of the Unit 2 turbine building. The Sullair compressor has enough capacity (650 scfm @ 125 psig) to supply the load requirements of both Units.

The major component of this system is the Service Air Receiver (T-63).

The Pallet Air Compressor (C-3C) is abandoned in place and not aligned to the system.

The Service Air System is connected to the Instrument Air System (IA) through the Instrument Air X-Over (SV-5400). The crossover valve automatically opens on low IA pressure, immediately providing the IA system with service air.

Air is pumped from the Unit 2 SA system to the air receiver. From the receiver, the air flows into a distribution header for use in the various parts of the plant.

4.0 REFERENCES

4.1 REFERENCES USED IN PROCEDURE PREPARATION

4.1.1 Instrument & Service Air (P&ID M-218, Sheet 2)

4.2 REFERENCES USED IN CONJUNCTION WITH THIS PROCEDURE

4.2.1 ICW System Operating Procedure (1104.028).

4.3 NRC COMMITMENTS

None

5.0 LIMITS AND PRECAUTIONS

5.1 Keep air intake filters on operating compressors free of:

Loose paper	Rags
Sheet plastic	Etc.

5.2 Do not operate compressor with dirty or fouled inlet air filter.

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

QID: 0921 **Rev:** 0 **Rev Date:** 9/13/14 **Source:** Modified **Originator:** Cork

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent SIS actuation.

K/A Number: A2.13 **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:** Yes **Taxonomy:** AN

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

iven

- Plant at 100% power
- P-2B Condensate Pump OOS
- Inadvertent actuation of ES Channel #1
- S/U #1 OOS for maintenance LCO 3.8.1.A 72 hour Time Clock in effect

What would be the impact to the plant due to this malfunction AND which of the following procedural actions would have the HIGHEST priority to mitigate the effects?

A. #1 Emergency Diesel Generator would start, reset the tripped channel and secure EDG

B. Red Train High Pressure Injection would occur, override and throttle HPI

C. Loss of power to A-1 bus, perform immediate actions for reactor trip.

D. All Seal Return isolates, realign RCP seal bleed off.

Answer:

C. Loss of power to A-1 bus, perform immediate actions for reactor trip.

Notes:

C is correct, the Unit Aux supply breaker to A-1 would open on ES Channel #1 actuation and would result in a reactor trip due to a loss of all Condensate pumps resulting in a losss of Main Feedwater.
A is incorrect, although the EDG would start with a reactor trip the EOP would have priority over securing the EDG
B is incorrect, although HPI would occur the ESAS EOP would not be utilized to secure HPI for an inadvertant actuation.
D is incorrect, seal return would be realigned to the Quench Tank rather than isolate.

Modified QID 812 for RO level by removing procedure references and just using operator actions.

References:

STM 1-32, Electrical Distribution
STM 1-20, Condensate System
1202.001, Reactor Trip

History:

Modified QID 812 for 2014 Exam.

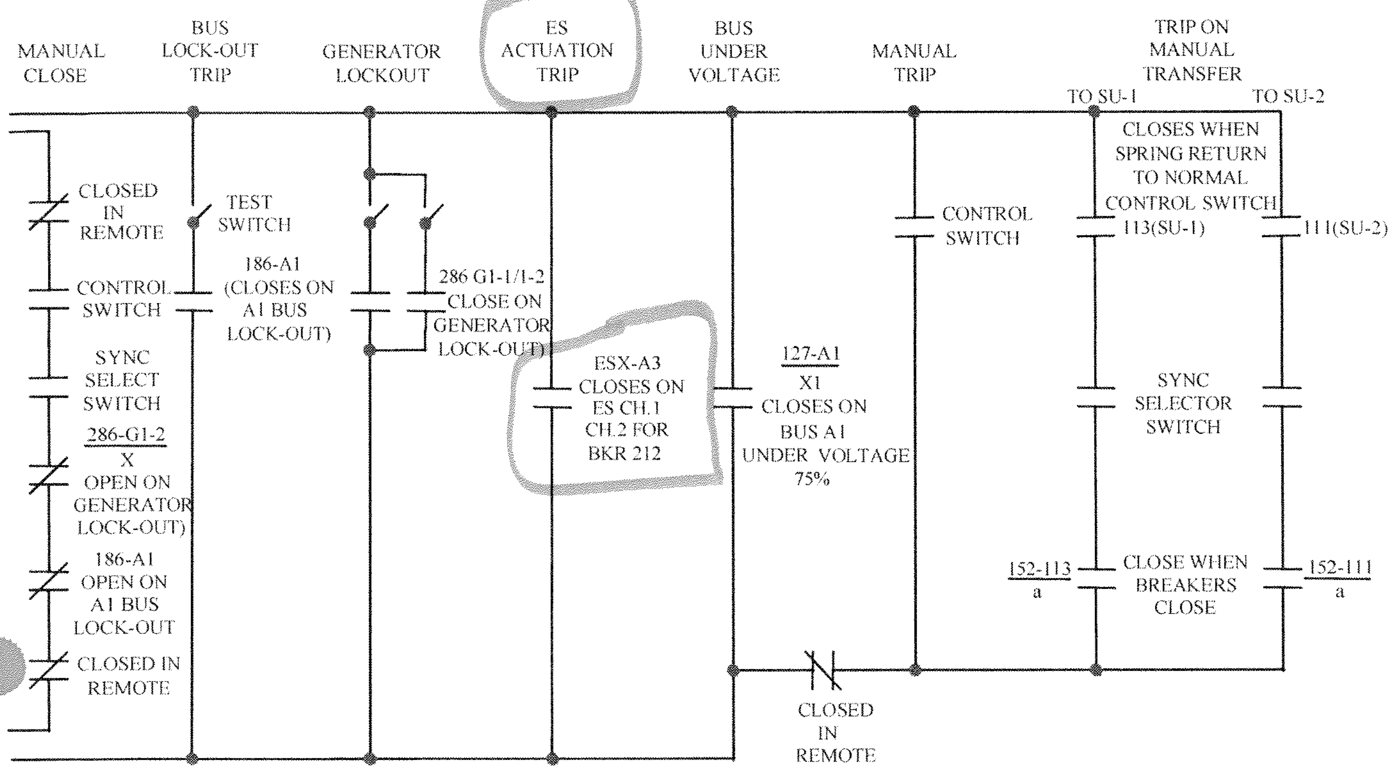


FIGURE 32.62: UNIT AUXILIARY TRANSFORMER FEEDER BRK 112(212)

1166 ft TDH. Pump run-out is 10,000 gpm at 1000 ft TDH. Shutoff head for the condensate pumps is 650 psig. The pump shaft seal utilizes a conventional stuffing box with six rings of packing. Seal cooling water is supplied from the condensate pump common discharge header through a pressure control valve.

During normal full power operation, two condensate pumps will supply 100% of the required condensate flow with the remaining pump in an auto-start configuration.

The Condensate pump motors are Westinghouse 3000 HP, 4160 volt, and 3 phase induction motors vertically mounted on the pump. Motor rotates at 1200 rpm drawing approximately 370 amps. The motor bearings are oil lubricated, water cooled by an internal bearing oil cooler (E-51A, B & C). Cooling water to the bearing oil cooler is supplied from the Auxiliary Cooling Water System (ACW). Each condensate motor is provided with instrumentation for monitoring motor bearing and winding temperatures.

The Condensate pumps are powered from non-vital 4160-volt buses A-1 and A-2. Condensate pump power supplies are listed below.

* P2A - Switchgear A-105
* P2B - Switchgear A-205
* P2C - Switchgear A-106

2.2.1 Seal Cooling Water Supply

(Refer to Figure 20.04)

Gland seal water for the condensate pumps is taken from the condensate pump discharge header through a one-inch line that supplies water to all three pumps. Each seal supply line is equipped with a pressure control valve (PCV) with a bypass valve, relief valve and local pressure gauge. The PCV's regulate the gland seal water pressure to 10 psig. To ensure the PCV's operate at their mid range to control seal supply pressure, EC-1785 was installed to provide a continous flow demand from the seal supply line back to the pump suction. Gland seal water relief valves are set to lift when supply pressure reaches 75 psig. Seal water from the three-condensate pumps combine to form a common return line to the condenser.

Refer to Table below for associated condensate pump PCV, PSV and local pressure gage.

	PCV	PSV	PI
P-2A	PCV-2901	PSV-2901	PI-2901
P-2B	PCV-2902	PSV-2902	PI-2902
P-2C	PCV-2903	PSV-2903	PI-2903

2.2.2 Condensate Pump Instrumentation

(Refer to Figure 20.04)

Each condensate pump is provided with instrumentation to monitor motor bearing and winding temperatures during pump operation. Pump discharge pressure is used for indication, an auto-start signal to standby pump and provides signal to ICS for plant

ENTRY CONDITIONS

- An automatic reactor trip or DSS trip.
- Failure of RPS to trip the reactor upon reaching a limit listed below:
  - High power 104.9%
  - High power/pumps .... one pump per loop .... ≥ 55%
    OR
    0 pumps in one loop .... ≥ 0%
  - High power/imbalance/flow .... COLR Figure
  - High RCS temp .... ≥ 618°F (T-hot)
  - High RCS press .... ≥ 2355 psig
  - Low RCS press .... ≤ 1800 psig
  - Variable low RCS press .... COLR Figure
  - High RB press .... ≥ 18.7 psia
  - Turbine trip .... reactor power ≥ 43% **AND** Turbine is tripped
  - Both MFW pumps trip .... reactor power ≥ 9% **AND** both MFW pumps tripped
- Manual trip of the reactor is required due to reaching a limit listed below:
  - PZR level dropping < 100",
    AND
    no indication of recovery
  - PZR level > 290"
  - Any MSIV closure at power
  - Either SG level < 15" or > 95%,
    AND
    no indication of recovery
  - A system degradation that requires manual reactor trip based on operator judgment
  - Abnormal Operating Procedure requirement
- **IF** a system degradation occurs while shutdown, above DHR operation,
  THEN perform applicable steps

INSTRUCTIONS	CONTINGENCY ACTIONS

CAUTION

ES-actuated components overridden in other than ES position will prevent fulfillment of the associated ES function if actual trip signal is present.

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Check the following: • **RCS pressure remains > 1590 psig** • **RB pressure remains < 18.7 psia**	1. **GO TO Reactor Trip (1202.001)**.
2. Check multiple ES digital channels have actuated.	2. <u>IF</u> a single channel has actuated, **<u>THEN</u> GO TO attachment for channel tripped.**
3. Verify reactor tripped. A. Perform Reactor Trip (1202.001) concurrently with this procedure.	
4. Check ES channel 5 or 6 has <u>not</u> actuated.	4. Perform the following: A. Trip all RCPs: • P32A • P32C • P32B • P32D B. Dispatch an operator to perform Service Water And Auxiliary Cooling System (1104.029) Exhibit B, "Restoring SW to ICW following ES Actuation", while continuing. C. Establish ICW to RB as follows: 1) Verify the following pumps running: • Two ICW Pumps • One RCP Seal Cooling pump (P114A or P114B) • One CRD Cooling pump (P79A or P79B

(4. CONTINUED ON NEXT PAGE)

ATTACHMENT 1

Recovery From Inadvertent ES Digital Channel 1 Actuation

CAUTION
ES-actuated components overridden in other than ES position will prevent fulfillment of the associated ES function if actual trip signal is present.

1. Verify reactor tripped
<u>AND</u> perform Reactor Trip (1202.001) in conjunction with this procedure.

2. Override and open HPI RECIRC Block (CV-1301).

3. <u>IF</u> HPI Pump (P-36A) is the ES HPI Pump,
<u>THEN</u> perform the following:

A. On C18, depress MAN ES pushbutton for HPI Pump (P-36A).

B. Start Aux Lube Oil pump (P-64A).

C. Stop HPI Pump (P-36A).

NOTE
Aux Lube Oil Pump (P-64A) will automatically stop 20 seconds after starting on an ES signal.

D. Stop Aux Lube Oil pump (P-64A).

E. <u>IF</u> desired,
<u>THEN</u> depress CH 1 MAN ES pushbutton and close BWST T3 Outlet (CV-1407).

4. Perform the following to restore Service Water and ACW:

A. On C18, depress MAN ES pushbutton for P-4A to P-4B Crosstie (CV-3646) and P-4B to P-4C Crosstie (CV-3640)

B. On C18, place HS-3646 and HS-3640 in open.

C. On C16, open ACW Isolation Valve (CV-3643).

D. Verify the following valves open:

- P-4B to P-4C Crosstie (CV-3640)
- P-4A to P-4B Crosstie (CV-3646)

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

QID: 0812 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based ability on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertant ESFAS actuation.

K/A Number: A2.06 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

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

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

Given

- Plant at 100% power
- P-2B Condensate Pump OOS
- Inadvertent actuation of ES Channel #1
- S/U #1 OOS for maintenance LCO 3.8.1.A 72 hour Time Clock in effect

What would be the impact to the plant due to this malfunction and what procedure would be used to mitigate the effects?

A. #1 Emergency Diesel Generator would start and use OP-1105.003, Engineered Safeguards Actuation System to reset the tripped channel.

B. Red Train High Pressure Injection would occur and use 1202.010, ESAS EOP to override HPI

C. Loss of power to A-1 bus and use 1202.001, Reactor Trip EOP

D. All Seal Return isolates and use OP1203.031, Reactor Coolant Pump and Motor Emergencies to realign seal bleed off.

Answer:

C. Loss of power to A-1 bus and use 1202.001, Reactor Trip EOP

Notes:

C is correct, the Unit Aux supply breaker to A-1 would open on ES Channel #1 actuation and would result in a reactor trip due to a loss of all Condensate pumps resulting in a losss of Main Feedwater.
A is incorrect, although the EDG would start with a reactor trip the EOP would have priority over securing the EDG
B is incorrect, although HPI would occur the ESAS EOP would not be utilized to secure HPI for an inadvertant actuation.
D is incorrect, seal return would be realigned to the Quench Tank rather than isolate.

References:

STM 1-32 Rev 33
OP-1107.001 Change 073

History:

New selected for 2010 SRO exam

PARENT

ANO UNIT 1 – 2014

TIER 2

GROUP 2

Questions 56 - 65

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

QID: 0717 **Rev:** 1 **Rev Date:** 4/14/2008 **Source:** Direct **Originator:** Exam Bank

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

Section: 3.1 **Type:** Reactivity Control

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

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:** 41.5 / 43.5 / 45.12

Tier: 2 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** C

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

Given:

Reactor Power is 100%.
The Group 7 Rods average position is 92% withdrawn.

An operator moves the group 6 rod 4 Relative Position Indication (RPI) to 0%.
Group 6 has 8 rods.

What is the expected result?

A. Asymmetric rod runback.

B. Asymmetric rod alarm.

C. Out inhibit.

D. Sequence inhibit.

Answer:

D. Sequence inhibit.

Notes:

A is incorrect. Asymmetric rod runback requires a control rod in limit which is from the Absolute Position Indication system.
B is incorrect Asymmetric rod alarm is based on Absolute Position Indication and is not affected by the Relative Position error.
C is incorrect. Out inhibit is based on Absolute Position Indication and Startup Rate and is not affected by the Relative Position error.
D is correct. Sequence inhibit of the CRD system utilized input from the RPI system. The group 6 average (87.5%) < 95% with group 7 > 25% causes a Sequence inhibit.

References:

STM 1-02, Control Rod Drive System

History:

Direct from Exam Bank: OpsUnit1 QuestionID: ANO-OPS1-3153
Selected for the 2008 RO Exam
Selected for 2014 Exam

2.8.1.2 Asymmetric Rods Lamp

Asymmetric Rods Lamp (amber): When on, indicates that one or more rods within a group are more than 6.5% out of alignment with the group average position. More on Asymmetric Rods later.

2.8.1.3 Out Inhibit Lamp

Out Inhibit Lamp (amber): Indicates that control rods will not respond to any out command. Refer to figure 2.42. The following conditions will result in an "Out Inhibit":

* High startup rate signal from the RPS of 2 DPM in source range and 3 DPM in intermediate range. The high startup rate signals are bypassed in the RPS when greater than 10% power.
* If the Diamond is in automatic and greater than 40% reactor power, a loss of any safety group out limit, (refer to Safety Rods Out relay logic figure 2.43) or a 6.5% asymmetric rod fault. If either of these conditions occur they will "seal in" and the "Fault Reset" switch must be used to reset the circuit.

2.8.1.4 Sequence Inhibit Lamp

Sequence Inhibit Lamp (Amber): This lamp, when lighted, indicates excessive overlap between regulating groups (> 25%).

The signal for sequence inhibit (Sometimes referred to as sequence fault) is developed by one of two sequence monitor circuits. Refer to figure 2.44. Input to the sequence monitor circuits is group average from relative position indication (RPI). RPI is utilized in order to provide the monitoring without incurring a sequence fault in the event an asymmetric rod condition alters the group average.

The same relay which causes the sequence inhibit will de-energize the sequence light in the "SEQ/SEQ OR" pushbutton.

Sequence inhibit will reject the Diamond Panel to manual

2.8.1.5 Auto Inhibit Lamp

Auto Inhibit Lamp (amber) indicates that the Diamond cannot be placed in automatic because:

* The safety groups are not at the out limit.
* The neutron error signal (demand versus actual) exceeds ± 1.00%.
* ICS power not available.

If the Diamond panel is in automatic, a loss of ICS Power will bring in this alarm and reject the Diamond to manual (refer to figure 2.45).

2.8.1.6 APSR Overlap Fault Lamp

APSR Overlap Fault Lamp (amber) Indicates that the group 6 lower poison section is less than three inches from group 8 upper poison section. This alarm is used for indication only.

2.8.2 Diamond Panel Indication Lights

2.8.2.1 Out Limit Lamps

Out Limit Lamps Groups 1 - 8 (red): Indicates that at least one rod out of its respective group is at the out limit. This will stop rod withdrawal for all rods within that group. On group 7, out limit

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

QID: 0132 **Rev:** 0 **Rev Date:** 3/8/96 **Source:** Direct **Originator:** J. Cork

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

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

System Number: 002 **System Title:** Reactor Coolant System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: PZR and makeup tank level.

K/A Number: A1.02 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	Ap

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

The makeup tank has decreased from 81 inches to 73 inches. Assuming that the level decrease was due to a loss of RCS inventory, how much has been lost?

a. 220 to 229 gallons

b. 230 to 239 gallons

c. 240 to 249 gallons

d. 250 to 259 gallons

Answer:

c. 240 to 249 gallons

Notes:

T-4 contains 30.8 gallons per inch
81 - 73 = 8 inches
8 in. x 30.8 gal/in = 246.4 gallons
Since 30.8 is an oddball value to remember, if candidate uses anywhere from 30 to 31.1 gal/in, he should arrive at answer [c].
All other choices are incremental bands close to the correct answer.

References:

1104.002, Makeup & Purifcation System Operation

History:

Taken from Exam Bank QID # 1709 - answer bands modified slightly
Used in 98 RO Re-exam
Selected for 2014 Exam.

Table 2.9.1 - Makeup Tank Level Data					
Start Time:		Stop Time:		ΔT= Min.	
Variable	Final Level	Initial Level	Change Level	Conversion Factor	Level Change
Makeup Tank Level	( ) -	( )		30.86	GAL

K. Determine leakage past Makeup Tank Outlet (CV-1275) as follows:

$$\frac{\text{Level Change (} \quad \text{gallons)}}{\Delta T \text{ (} \quad \text{minutes)}} = \text{CV-1275 Leak Rate (\_\_\_\_\_\_ gpm)}$$

L. Enter CV-1275 leak rate in section 3.0.

2.9.2 IF testing CV-1275 when aligned to P-34B, THEN perform the following:

A. IF closed,
THEN unlock and open P-36 B&C Makeup Pump Suction Cross-Over (MU-14).

B. Verify P-36 B&C Makeup Pump Suction Cross-Over (MU-15) open.

C. IF available,
THEN select Diagnostic Instrumentation display on SPDS for HPI pumps.

D. Open Decay Heat Supply to Makeup Pump Suction (CV-1277) AND pressurize MU pump suction header.

NOTE

Due to piping arrangement, Makeup Pump suction pressure may be adjusted using CV-1400, CV-1429 and CV-1432.

E. Adjust Makeup Pump (P-36B or P-36C) suction pressure to obtain 30 to 60 psig by throttling CV-1400, CV-1429 and CV-1432.

- Verify Makeup Pump suction pressure 30 to 60 psig.

QID: 0922 **Rev:** 0 **Rev Date:** 9/13/14 **Source:** New **Originator:** Cork

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

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

System Number: 011 **System Title:** Pressurizer Level Control System (PZR LCS)

Description: Knowledge of bus power supplies to the following: PZR heaters.

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

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

Group: 2 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 58 **SRO:** 58

Which of the following list of pressurizer heaters are ALL vital powered?

A. Banks 1, 2, and 3

B. Banks 1, 2, 4 and 5

C. Banks 1, 2, and RUB 13

D. Banks 1, 2, RUB 14 and 15

Answer:

D. Banks 1, 2, RUB 14 and 15

Notes:

T.S. 3.4.9 requires >126 kW of vital powered heaters to be operable, in order to satisfy the requirement all of the vital powered heater "groups" must be operable. Group 5 is made up of RUB 13, 14 and 15 but only RUB 14 and 15 are vital powered. Banks 1 and 2 are also vital powered.

D is correct and is the only choice with all of the ES powered PZR heaters.
A, B, C are incorrect but plausible since each contain some of the ES powered heaters.

References:

1103.005, Pressurizer Operation
1107.022, ES Electrical System Operation
Techncial Specifications 3.9.4

History:

New for 2014 Exam

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 320 inches; and

b. A minimum of 126 kW of Engineered Safeguards (ES) bus powered pressurizer heaters OPERABLE.

--NOTE--
OPERABILITY requirements on pressurizer heaters do not apply in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 262°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limits.	A.1 Restore level to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with RCS temperature ≤ 262°F.	6 hours 24 hours
C. Capacity of ES bus powered pressurizer heaters less than limit.	C.1 Restore pressurizer heater capacity.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

NOTE

- In order to satisfy the requirements of TS 3.4.9, Bank 1 Proportional heaters, Bank 2 Proportional heaters, and Group 5 vital powered heaters shall be operable. This ensures that ≥126 KW (nominal) is available in the event of a loss of offsite power concurrent with a single failure of one EDG.
- Group 5 vital powered heaters shall be capable of manual transfer via B55/56. In the event B55/56 can not be manually transferred, then one train of Pressurizer Heaters is considered inoperable.
- If Group 5 vital powered heaters are declared inoperable, then both trains of Pressurizer Heaters are considered inoperable. Per Licensing, TS 3.0.3 is NOT applicable. Entry into TS 3.4.9 Condition C is required for inoperability of both trains of Pressurizer Heaters.
- Unit 1 Emergency-Powered Pressurizer Heater Checkout (1307.009) is used to determine operability of vital powered Pressurizer Heaters.

15.3 Bank 1 Proportional heaters, Bank 2 Proportional heaters, and Group 5 vital powered heaters.

15.3.1 TS 3.4.9 requires 2 trains of Pressurizer Heaters with 126 KW (nominal) capacity in Modes 1, 2, 3 and 4 with RCS temperature > 262°F to be operable and capable of being powered from its associated Engineered Safeguards bus.

A. TS 3.4.9 is not satisfied if EITHER of the following heater combinations are determined to be <126 KW:

- Bank 1 Proportional heaters AND Group 5 vital powered heaters (capable of manual transfer)
- Bank 2 Proportional heaters AND Group 5 vital powered heaters (capable of manual transfer)

B. Failure to satisfy TS 3.4.9 requires the following:

1. Enter 72 hour time clock per Condition C.
2. Initiate a Condition Report.

15.4 Failure of Valve Stroke to Meet Acceptable Normal Range.

Failure of a Valve stroke to meet its Acceptable Normal Range requires the following actions per ASME Code.

Immediately retest or declare valve inoperable. If valve is retested and its stroke time remains outside its ANR, an engineering evaluation shall be performed within 96 hours to verify component operability. If valve is retested and its stroke time is within its ANR, the cause of the initial deviation shall be analyzed and the results documented in section 4.0 of the test supplement.

ATTACHMENT D

BREAKER NUMBER	DESCRIPTION Ref. Drawing (E-16)	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INI-TIAL
5643B	Emergency Power Supply for SPDS AC Unit 2VUC-30/2VE6	Closed			
5644A	Aux Bldg Lighting Panel 1PC (E-431)	Closed			
5644B	Lighting Transformer XL-7 (E-431)	Note 5 Closed			
5645	Spare	Open			
5646	HPI Pump Rm Cooler VUC-7B (E-368)	Closed			
5651	Decay Heat Suction from RCS CV-1404 (E-182)	Note 6 Closed			
5652	Makeup Tank Outlet CV-1275 (E-222)	Closed			
5653	Serv Water to ACW Isolation CV-3643 (E-278)	Closed			
5654	Non-Nuclear ICW CRD Inlet CV-2235 (E-221)	Closed			
5656A	Pressurizer Heater RUB 15 (E-202) Series fed from B5663	Note 7 Closed			
5656B	Pressurizer Heater RUB 14 (E-202) Series fed from B5663	Note 7 Closed			
5661	Core Flood Tank T-2A Outlet CV-2415 (E-238)	Locked Open			
5663	PZR Heater Group 5 RUB-14 & 15 (feeds breakers 5656A & 5656B) (E-203)	Closed			
5666	Control Rm Emer Stby Filter Fan 2VSF-9 (E-359-3)	Open			
	Space Heater Breaker HTR 1	Open			
	Space Heater Breaker HTR 2	Open			

Note 5: Breaker may be opened upon Reactor Building exit per "Securing Reactor Building Lighting for Power Operations", Attachment 2 of 1107.005.

Note 6: Locked open Category E by Plant Startup 1102.002.

Note 7: Breakers B5656A and B5656B are hard wired from B5663 and are NOT de-energized when racked out from bus

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

QID: 0058 **Rev:** 0 **Rev Date:** 7/8/98 **Source:** Direct **Originator:** JCork
TUOI: A1LP-RO-NI **Objective:** 3 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation
System Number: 015 **System Title:** Nuclear Instrumentation System
Description: Knowledge of the effect of a loss or malfunction of the following will have on the NIS: sensors, detectors, and indicators.
K/A Number: K6.01 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.2	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** 59 **SRO:** 59

A startup is in progress.
The reactor is critical.
The CBOR is commencing power escalation to <2% reactor power.

The following indications are observed:

NI-3 1 x 10-8 amps
NI-4 8 x 10-9 amps
NI-5 0.8%
NI-6 1.1%
NI-7 1.3%
NI-8 1.2%

What conclusion should you deduce from the above indications?

A. Power Range channel 5 requires calibration.

B. The Intermediate Range channels are overcompensated.

C. The POAH has not yet been reached.

D. The Intermediate Range channels are undercompensated.

Answer:

B. The Intermediate Range channels are overcompensated.

Notes:

Answer "B" is correct, overcompensation will cause the IR channels to read low. The indications given show the PR channels starting to come on scale which is indicative of the POAH being reached. The POAH is roughly 4 x 10-8 amps, both IR channels are less than this, indicating a case for overcompensation.
Answer "A" is incorrect, although NI-5 is reading less than the other PR's, this is far too low in the scale to jump to the conclusion that it requires calibration.
Answer "C" is incorrect, the POAH has obviously been reached as the PR channels are coming on scale.
Answer "D" is incorrect, undercompensation would cause the IR channels to read high, not low.

References:

1102.002, Plant Startup
STM1-67, Nuclear Instrumentation

History:

Developed for the 1998 RO/SRO Exam.
Used in 1999 exam. Was KA A3.03

Selected for use on 2007 RO Exam
Selected for 2014 Exam.

chambers are mounted in a housing assembly which electrically insulates the detector case from ground and provides a means for handling the detector. Each assembly is seal welded, evacuated, baked dry, and filled with an inert nitrogen atmosphere to prevent moisture from entering.

2.4.2 Detector Construction and Operation

The detectors are Westinghouse Compensated ion chambers, WL-23635, and are electrically compensated for Gamma radiation. The detector is of the double can type with a boron lining on the inside of the outer can. Both cans are filled with nitrogen gas. The incoming neutrons interact with the boron lining of the outer can causing the same reaction seen in the BF3 Source Range detectors. The outer can is sensitive to both neutrons and gamma because of the Boron lining. As can be seen on Figure 67.22, the outer wall has a +600 volt potential and the inner wall is grounded. In the outer can, ionization causes current to flow from the outer wall to the power supply and through ground to the inner wall.

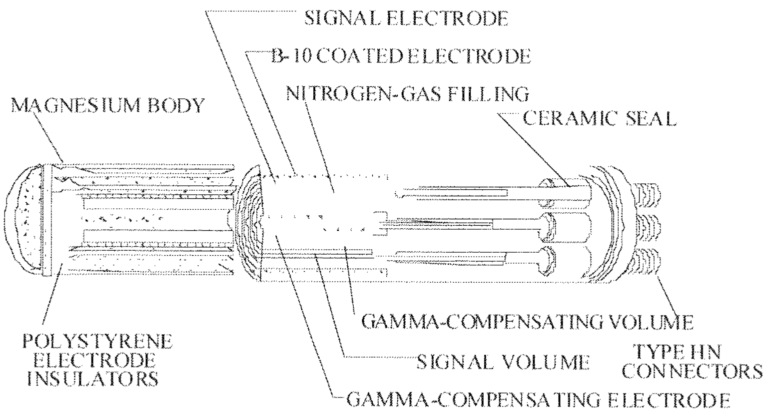
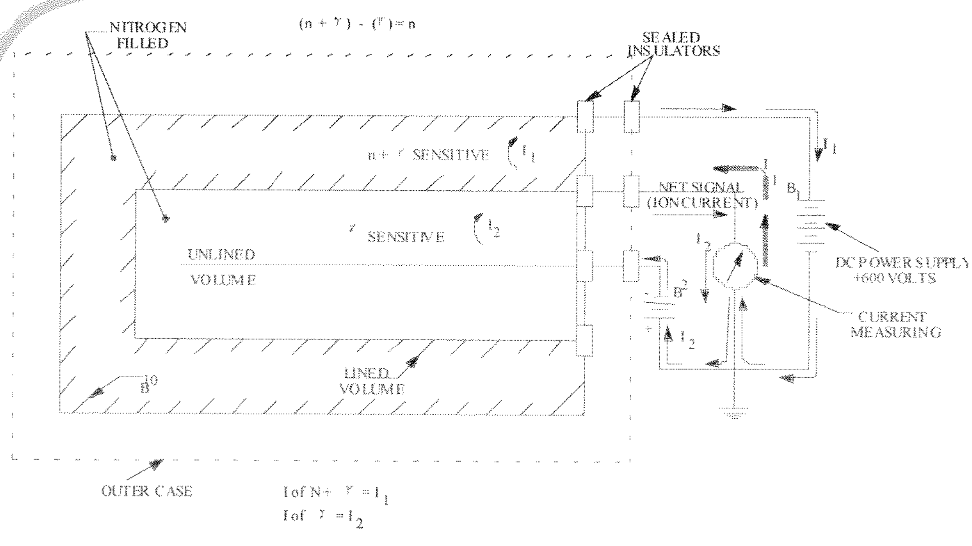


FIGURE 67.21: COMPENSATED ION CHAMBER

The inner can is not Boron lined and is sensitive to gamma only. The center electrode has a potential of about - 20 VDC applied to it. The outer wall is grounded. Ionization within the inner can causes current from gamma to flow in the opposite direction to the current from the outer can. Therefore, current from the gamma field is subtracted from the current produced by both neutrons and gamma. The net ion current displayed is the signal generated by the neutron flux. In order to obtain a true neutron signal, the two gamma signals must be balanced exactly to make them cancel. A problem may develop because the gamma flux is not always evenly distributed. One volume may see more gamma interactions than the other. A method used to deal with this problem is electrical compensation.



NOTE: BOTH NEUTRON AND GAMMA CURRENT FLOWS DOWNWARD THRU THE METER BUT ONLY GAMMA CURRENT FLOWS UPWARD THRU THE METER THEREBY CANCELLING OUT THE GAMMA CURRENT.

FIGURE 67.22: COMPENSATED ION CHAMBER OPERATION

By applying a variable

negative voltage, it is possible to "electrically" adjust the size of the inner can. This gamma sensitive volume operates in a partial recombination (region I) region. (Figure 67.06) Therefore, by adjusting the compensating voltage, only a part of the total ionization is collected. This method effectively increases or decreases the gamma current from the inner volume to achieve proper compensation.

FIGURE 67.23: TYPICAL GAMMA COMPENSATION CURVE

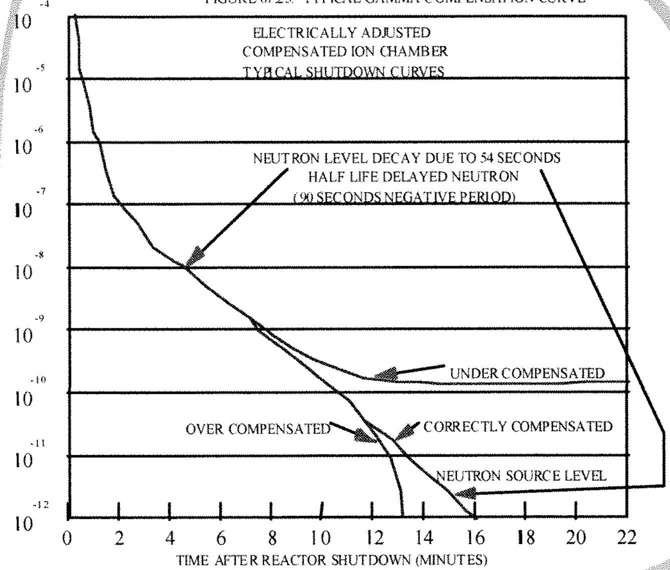


Figure 67.23 illustrates that under compensation or overcompensation will lead to variation in the indicated neutron flux level. If the CIC is overcompensated, the indicated neutron level will be lower than actual. This is because the gamma sensitivity of the inner volume is too high. If the CIC is under compensated, the neutron level will appear higher than actual. Generally, our Intermediate Range detectors are maintained about 5% under compensated. This makes them more responsive at the low end. It also makes our readings more conservative.

2.4.3 Detector Power Supplies

Refer to Figure 67.24

The Intermediate Range detectors are unique. They are the only detectors that require two power supplies. The detector and auxiliary power supplies. One for detector chamber excitation and one for compensation voltage. The power supplies are functionally identical to the Source Range power supplies. In accordance with revision 24 to OP 1015.03A, Auxiliary Power Supply voltage is normal at 23 volts. Detector Power supply must be between 550 and 650 volts to meet Tech Spec operability criteria.

2.4.4 Logarithmic Amplifier

Refer to Figure 67.24

The logarithmic amplifier converts the compensated ion chamber output current to an output of 0 to +10 Volts DC that is proportional to the logarithm of the input signal current. The range of signal is 1 X 10^{-11} to 1 X 10^{-3} amperes DC. Standard buffer amplifiers on the output furnish isolation. The module front plate contains a meter calibrated in terms of the input signal, DC amps. The meter and other channel instrumentation for NI-3 is located in RPS channel C (C43) and NI-4 is located in channel D RPS (C44).

The outputs are also sent to the SUR Rod Withdrawal interlock, plant computer, SPDS and to remote indicators (see table 67.1) The frontplate (figure 67.25) also contains adjustments for amplifier balance and calibration and test jacks used for on line testing.

2.4.5 Rate of Change Amplifier

The rate of change amplifier feeds a Control Room panel indicator (.5 to 5 DPM), a SUR signal to the plant computer, and a

17.6 As reactor power is being raised, following actions shall be accomplished continuously.

NOTE

Point of adding heat (POAH) as observed during Cycle 15 is ~4.0 x 10^{-8} amps.

- Limit steady state SUR to ≤0.5 dpm.
- Maintain RC pressure between 2080 and 2230 psig.
- Maintain Makeup Tank (T-4) level per Exhibit A, of Makeup & Purification System Operation (1104.002) by diverting letdown to radwaste system (Vacuum Degasifier T-14, if available).
- Maintain pressurizer level within limits specified in Attachment C of this procedure.
- Maintain T-ave vs. reactor power level within limits of Attachment D of this procedure.
- Verify power range channels come on scale when intermediate range channels read ~2 x 10^{-6} amps.
- Limit reactor power escalation according to Power Operation (1102.004), Attachment L, "Reactor Maneuvering Limits."

CAUTION

Power maneuvering requires close attention to prevent exceeding Control Rod Index Limits.

NOTE

Above 2% power, use Heat Balance Power for assessing rod index limits.

17.7 CBOR compare rod index with a copy of the Regulating Rod Insertion Limits curves of the COLR at ~15 minutes intervals to verify limits are not exceeded.

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

QID: 0169 **Rev:** 0 **Rev Date:** 11/19/98 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-NNI **Objective:** 19 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumention System (NNIS)

Description: Ability to monitor automatic operation of the NNIS, including: Relationship between meter readings and actual parameter value.

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

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 4

Group: 2 **SRO Imp:** 2.9 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** 60 **SRO:** 60

Given:

- Plant is at 100% power.
- PZR level transmitter LT-1001 selected via HS-1002 on C04.
- PZR temperature element TE-1001A selected via HS-1000 on C04.

The PZR temperature indicator, TI-1000, on C04 drops suddenly to 50°F (bottom of scale).

Without operator action, what will be the effect on the PZR Level Control System?

a. PZR Level Control Valve, CV-1235, will open to establish a higher steady-state PZR level.

b. PZR Level Control Valve, CV-1235, will maintain the same steady-state PZR level.

c. PZR Level Control Valve, CV-1235, will close to establish a lower steady-state PZR level.

d. PZR Level Control Valve, CV-1235, will fail open to continuously raise PZR level.

Answer:

a. PZR Level Control Valve, CV-1235, will open to establish a higher steady-state PZR level.

Notes:

[a] is correct. A loss of temperature compensation will result which will appear as a low PZR level. This is the same reason which makes [b] & [c] incorrect.
(d) is incorrect. The loss of temperature compensation does not produce an indication that is similar to a high off scale indication.

References:

STM 1-69, Non-Nuclear Instrumentation System

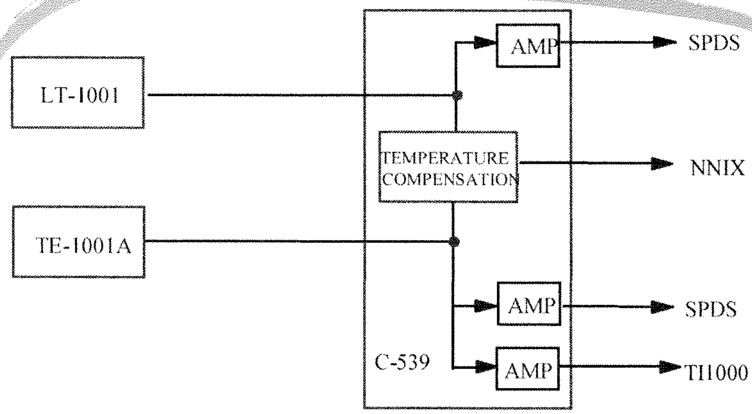
History:

Developed for 98 exam.
Used in 2001 Exam.
Selected for use in 2002 SRO exam. KA 011 K4.03
Selected for 2014 Exam

displayed on FR-1032 that is located on C13. The display range is 0 to 160 E^6 lbm/hr. The total flow inputs into the RCS low flow alarm.

3.3.12 Pressurizer Temperature and Level

Pressurizer level and temperature detectors input into instrument cabinets C539 and C540. LT-1001 and TE-1001A input into C539



and LT-1002 and TE-1002A input into C540. Buffered temperature and pressure outputs are sent to the SPDS computer. Pressurizer temperature is indicated on TI-1000, which is located on C04. Selector switch HS-1000 allows selection of either TE-1001A or TE-1002A for indication. The range of indication is 50 to 750 °F. Pressurizer temperature also supplies density compensation signals for pressurizer level. LT-1001/LT-1002 are temperature compensated by TE-1001A/TE-1002A respectively. The compensated pressurizer level outputs are sent to the NNIX system.

Within the NNIX system the compensated pressurizer level signals are processed for indication and control of the make-up valve. The summing amplifiers provide input buffering. The output of the summing amplifiers supplied input to the plant computer via a buffer amp and remote shutdown indicators LI-1001/LI-1002. LI-1002 is located at R-33 (South Electrical Equipment Room), and LI-1001 is located at R08 (outside makeup tank room).

The output is also supplied to the pressurizer level controller via the pressurizer level selector switch HS-1002. HS-1002 operates control relays, which selects the desired pressurizer level signal. LT-1001 is normally selected and supplies LI-1000 (located on the dasey panel), the pressurizer level controller, and the HI HI/LO LO pressurizer level alarms. The level alarms are actuated at 275" and 55" respectively. The LO LO level alarm also interlocks the pressurizer heaters off.

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

QID: 0312 **Rev:** 1 **Rev Date:** 11/16/00 **Source:** Direct **Originator:** J Cork

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

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

System Number: 033 **System Title:** Spent Fuel Pool Cooling System

Description: Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling system will have on the following: Area ventilation systems.

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

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.1	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 61 **SRO:** 61

The WCO reports the Spent Fuel Pool level is +1.5 ft.

What problem could this level pose for Spent Fuel Pool operations or fuel handling in the SFP?

A. SFP minimum water temperature limit will be exceeded.

B. SFP ventilation ducts will be flooded.

C. Area dose rates will rise.

D. SFP must be sampled within 5 hours.

Answer:

B. SFP ventilation ducts will be flooded.

Notes:

Answer [B] is correct since normal level is 0 ft with a maximum allowable level of +1.0 ft which prevents water carryover into the ventilation ducts.
Answer [A] is incorrect because this answer is associated with SF cooling capacity which is largely unaffected by pool level.
Answer [C] is incorrect since this problem is associated with a low water level.
Answer [D] is incorrect but plausible since the time for sampling is correct but level is greater than maximum allowed.

References:

STM 1-7, Spent Fuel Cooling System

History:

Developed for 1999 exam.
Modified for 2001 RO/SRO Exam.
Used on 2004 RO/SRO Exam.
Selected for 2014 Exam.

* Spent Fuel Pool Demineralizer
* Borated Water Recirculation Pump
* Cask Loading Pit and Fuel Tilt Pit Submersible Pumps

The SF storage pool is filled with borated water, an economical, effective, and transparent radiation shield as well as a reliable cooling medium for removal of decay heat. Racks are installed in the pool to ensure proper spacing of the assemblies.

The SF pool circulating pumps circulate the pool water through the SF coolers to remove the decay heat. The SF coolers are cooled by nuclear ICW. Water in the pool is also circulated through filters and a demineralizer to maintain water clarity. After refueling, the purification loop can also be used to purify water in the BWST.

During refueling, fuel access to the storage pool from the Reactor Building is provided by the fuel transfer canal and transfer tube.

2.0 Detailed System Description

2.1 Spent Fuel Storage Pool

The Spent Fuel Storage Pool, located in the fuel handling area of the Auxiliary Building, serves as a storage facility for spent fuel assemblies from the Reactor Core.

Two smaller pools are located adjacent to the Spent Fuel Pool. One is the Fuel Tilt Pit, located on the south end, in which the fuel transfer mechanism is located. The other is a loading area for Spent Fuel Shipping Casks, located on the north end. Both are connected to the main pool but may be isolated by watertight gates and pumped down for dry handling of the shipping cask, or maintenance on the fuel transfer mechanism. The cask area is also used at times for fuel assembly inspection equipment.

2.1.1 SF Pool Design Features

The Pool is a reinforced concrete pool lined with welded stainless steel and fitted with tell-tale drains to indicate any leak in the liner. The tell-tale drains are connected to the Auxiliary Building Equipment Drain Tank (T-11) through valves on the 354′ elevation south of the elevator. The stainless steel provides leak-tightness and ease of decontamination.

The pool is filled with borated water to a normal level of -3.5 feet (below floor level). This corresponds to an indicated level of 0 ft. The water level is normally maintained between the high and low level alarms (+0.5 ft. and -0.5 ft.) and below 0.0 ft. during refueling. The maximum allowable pool water level is an indicated 1 ft. which prevents water from entering the ventilation exhaust ducts around the pool. Minimum allowable pool water level is an indicated -1.5 ft. to ensure sufficient shielding of the fuel assemblies and to reduce the possibility of uncovering the fuel assemblies.

Tech Specs require > 2000 ppm boron in the pool at all times to remain within the limit of the Spent Fuel Pool accident and criticality analysis. To prevent approaching the Tech Spec limit a minimum concentration of 2100 ppm boron is maintained in the Spent Fuel

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

QID: 0662 **Rev:** 1 **Rev Date:** 9/23/14 **Source:** Direct **Originator:** Possage

TUOI: A1LP-RO-EOP04 **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:** Yes **Taxonomy:** Ap

Question: **RO:** 62 **SRO:** 62

Given:
Recovery from an Overheating condition is in progress.
Auxiliary Feedwater Pump, P-75 is the only available source of water.
"A" S/G level is 18 inches and stable.
"B" S/G level is 21 inches and lowering.
Subcooling Margin is adequate.

Which of the following indicate the proper action to take and why?

A. Neither S/G can be fed due to unanalyzed stresses of feeding a dry S/G with Aux Feedwater.

B. "A" S/G can not be fed until primary to secondary heat transfer is established. "B" S/G can be fed while monitoring tube to shell delta T for unanalyzed stresses of feeding a dry S/G with Aux Feedwater.

C. Both S/Gs can be fed while monitoring tube to shell delta T until primary to secondary heat transfer is established.

D. Both S/Gs can be fed, tube to shell delta T is not a concern until primary to secondary heat transfer is established, then maintain tube to shell delta T within limits.

Answer:

C. Both S/Gs can be fed while monitoring tube to shell delta T until primary to secondary heat transfer is established.

Notes:

"C" is correct. ANO has been analyzed for feeding dry steam generators with Aux Feedwater. As long as SCM is adequate tube to shell delta T limits apply as a throttling criteria of the feedwater both prior to and following the establishment of primary to secondary heat transfer.
"A" is incorrect. Stresses have been analyzed and "B" S/G would not be considered dry.
"B" is incorrect. Stresses have been analyzed and "B" S/G would not be considered dry.
"D" is incorrect. Tube to shell delta T is a concern as long as SCM is adequate.

References:

1202.012, Repetitive Tasks, RT-16
1202.004, Overheating

History:

New for 2007 RO Exam, K/A 054 AK1.02
Selected for 2011 RO Exam. KA 054 AK3.04
Selected for 2014 Exam

INSTRUCTIONS	CONTINGENCY ACTIONS
12. <u>WHEN</u> CET temps begin to drop, <u>THEN</u> perform the following:	
A. <u>IF</u> SCM is adequate, <u>THEN</u> maintain RCS cooldown rate ≤ 100°F/hr by throttling HPI and Letdown flow.	A. Maintain full HPI flow.
13. <u>IF</u> EFW restoration is imminent, <u>THEN</u> GO TO step 14.	13. Perform one of the following: A. <u>IF</u> Main or AUX Feedwater Pump becomes available, <u>THEN</u> refill available SG(s) using RT-16. 1) **GO TO step 15.** B. <u>IF</u> Main and AUX Feedwater Pumps are <u>not</u> available, <u>THEN</u> **GO TO 1202.011, "HPI COOLDOWN"** procedure.
14. <u>IF</u> EFW becomes available, <u>THEN</u> refill available SG(s) using RT-16.	14. **RETURN TO step 13.**

CAUTION

- With RCS solid, 1°F temp change can cause 100 psig press change.
- A large reduction in out-flow without a corresponding reduction in in-flow will result in RCS press rise.

INSTRUCTIONS	CONTINGENCY ACTIONS
15. Check adequate primary to secondary heat transfer established <u>AND</u> perform the following:	15. **GO TO 1202.011, "HPI COOLDOWN"** procedure.
A. Check Letdown in service.	A. <u>IF</u> conditions permit: • fuel damage does <u>not</u> exist • RCS to ICW leak is <u>not</u> suspected <u>THEN</u> restore Letdown (RT-13).

(15. CONTINUED ON NEXT PAGE)

FEEDING INTACT SG

2. (Continued)

F. **<u>IF</u>** AUX Feedwater Pump (P75) is available,
<u>THEN</u> perform the following:

1) Dispatch an operator to verify AUX FW Pump RECIRC to E-11A Isolation (FW-1) open.

2) Verify Aux Feedwater Pump (P75) running.

3) **GO TO step 2.H.**

G. **<u>IF</u>** MFW pump is available,
<u>THEN</u> verify MFW pump running.

1) Place RFR Override handswitch in OVERRIDE.

H. **<u>IF</u>** SCM is **<u>not</u>** adequate,
<u>THEN</u> establish **<u>AND</u>** maintain SG levels 370 to 410" within 25 minutes of SCM loss using Startup valve H/A stations in HAND.

1) **<u>IF</u>** SCM becomes adequate prior to establishing 370 to 410",
<u>THEN</u> GO TO step 2.I.

2) **<u>IF</u>** any good SG press drops below 720 psig,
<u>THEN</u> 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)

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)

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

QID: 0662 **Rev:** 0 **Rev Date:** 12/15/06 **Source:** Direct **Originator:** Possage

TUOI: A1LP-RO-EOP04 **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:**	Yes	**Taxonomy:**	Ap

Question: **RO:** 62 **SRO:** 62

Given:
Recovery from an Overheating condition is in progress.
Auxiliary Feedwater Pump, P-75 is the only available source of water.
"A" S/G level is 18 inches and stable.
"B" S/G level is 21 inches and lowering.
Subcooling Margin is adequate.

Which of the following indicate the proper action to take and why?

A. Neither S/G can be fed due to unanalyzed stresses of feeding a dry S/G with Aux Feedwater.

B. "A" S/G can not be fed until primary to secondary heat transfer is established. "B" S/G can be fed while monitoring tube to shell delta T due to unanalyzed stresses of feeding a dry S/G with Aux Feedwater.

C. Both S/G can be fed while monitoring tube to shell delta T until primary to secondary heat transfer is established.

D. Both S/G can be fed, tube to shell delta T is not a concern until primary to secondary heat transfer is established, then maintain tube to shell delta T within limits.

Answer:

C. Both S/G can be fed while monitoring tube to shell delta T until primary to secondary heat transfer is established.

Notes:

"C" is correct. ANO has been analyzed for feeding dry steam generators with Aux Feedwater. As long as SCM is adequate tube to shell delta T limits apply as a throttling criteria of the feedwater both prior to and following the establishment of primary to secondary heat transfer.
"A" is incorrect. Stresses have been analyzed and "B" S/G would not be considered dry.
"B" is incorrect. Stresses have been analyzed and "B" S/G would not be considered dry.
"D" is incorrect. Tube to shell delta T is a concern as long as SCM is adequate.

References:

1202.012, Repetitive Tasks, RT-16
1202.004, Overheating

History:

New for 2007 RO Exam, K/A 054 AK1.02
Selected for 2011 RO Exam. KA 054 AK3.04
Selected for 2014 Exam

ORIGINAL

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

QID: 0923 **Rev:** 0 **Rev Date:** 9/14/14 **Source:** Modified **Originator:** Cork

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

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

System Number: 041 **System Title:** Steam Dump System (SDS) / Turbine Bypass Control

Description: Knowledge of SDS design feature(s) and/or interlock(s) which provide for the following: Turbine trip.

K/A Number: K4.18 **CFR Reference:** 41.7

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

Group: 2 **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** 63 **SRO:** 63

Given:
- Reactor Power reduced to 25% due to lowering condenser vacuum
- Condenser vacuum is 25" Hg
- Main Turbine was manually tripped 30 minutes ago
- Instrument Air pressure is 75 psig and slowly dropping.

How is SG pressure being controlled for these conditions?

A. TBVs at ~895 psig

B. TBVs at ~995 psig

C. ADVs at ~1020 psig

D. MSSVs at 1050 - 1100 psig

Answer:

A. TBVs at ~895 psig

Notes:

A is correct. The Main Turbine is required to be manually tripped if MWe is <270 and vacuum is <26.5"Hg. The normal setpoint of 895 psig will be used for auto control of SG pressure.
B is incorrect but plausible since the TBVs have a bias of 100 psig applied to their setpoint on a Reactor trip but the Reactor will not auto trip if the Main Turbine trips below 43% power.
C is incorrect but plausible if the examinee cannot recall when the TBVs are interlocked closed on low condenser vacuum (<23").
D is incorrect but plausible if the examinee cannot recall what Instrument Air pressure will cause these valves to use their accumulator air or when the ADVs will run out of accumulator air.

References:

STM 1-15, Main Steam

History:

Modified regular exambank QID ANO-OPS1-6658 for 2014 Exam

TBV's have a sophisticated air supply system. IA supply is supplied to the valve positioner and provides supply air for valve operation. In addition to supplying motive air for operation it also maintains the two volume tanks pressurized to maintain the TBV closed on loss of instrument air.

The positioner is supplied with regulated air at 65 psig. With this supply of air, the positioner aligns to either the open volume booster or close volume booster through the trip valve. The instrument air supply through the volume booster then operates the valve. Operation of the TBV volume booster is identical to the ones used in the ADV's. Refer back to page 14 for explanation of volume booster operation.

The comparator is a spring-loaded valve that monitors air pressure continuously. If instrument air header pressure drops to 55 psig, the comparator's spring forces the internal valve down allowing the trip valve to realign the air accumulators to the turbine bypass valves closed volume booster thus maintaining the valve closed. Once air pressure in the two volume tanks is depleted, for whatever reason, system pressure can open the turbine bypass valve.

2.2.6.1 Turbine Bypass Valve Operations--Automatic

During normal operation (plant operating at 100% power), when in automatic, the valves are controlled by the steam header pressure controller ICC-0013. Header Pressure Controller, ICC-0013, is located on C03. The Header Pressure Controller analog meter provides Turbine Header pressure indication with a range of 600 - 1200 psig. Turbine header pressure is also indicated on chart recorder PR-2634 (on C03 vertical section, above ICC-0013). The setpoint dial on the Header Pressure Controller reads out in 0 to 100% of 600 to 1200 psig. The normal setpoint for this controller is 49% which is equal to ~895 psig. If the turbine is tripped or on Throttle Valve control, the Turbine Bypass Valves control their associated OTSG pressure.

As stated, the TBV's control turbine header pressure normally, however, during normal operation (plant operating at 100% power), the main turbine is also controlling turbine header pressure via the same controller. In order to prevent the turbine and TBV's from competing against each other, the TBV's have a 50-psig bias added to their setpoint provided the following conditions are met:

Turbine Header Pressure is within 10 psig of its setpoint.

All of the Turbine Bypass Valves are closed or if all of the valves are not closed, and the unit's load demand is >15%.

Reactor is not tripped.

Turbine is not tripped.

During a Reactor trip, another bias is applied to these valves. In this case, the bias added to the dial setpoint is 100 psig. Additional information on Turbine Bypass Valve automatic operation can be found in the Integrated Control System STM (STM 1-64).

 ANO-OPS1-6658

A large Instrument Air line has ruptured at 100% power.The reactor was tripped 10 minutes ago.Instrument Air header pressure is 0 psig.Condenser Vacuum is 20" Hg and falling.How is OTSG pressure being controlled?

A. TBVs via accumulator tanks controlling at ~ 895 psig.
B. TBVs via accumulator tanks controlling at ~ 995 psig.
C. ADVs via MSIV accumulator tanks controlling at ~ 1020 psig.
D. Floating on MSSVs at 1050 - 1100 psig.

Answer: C

Question Comments: Knowledge of events that will cause the Main Steam Isolation Valves to close is necessary to allow control of the plant.;

Image Reference: None

QuestionID: ANO-OPS1-6658

Objectives:

1. CourseID: A1LP-RO-STEAM Objective: 3
2. CourseID: A1LP-RO-STEAM Objective: 6

KA References:

1. 065 AA2.08

References:

1. 1-15
2. 1203.024

Training Programs:

Categories:

1. CAT-9

Systems:

Task References:

Cognitive Level: 2: Comprehension or Analysis

Point Value: 1.0

Exam Bank: OpsUnit1

Review Status: Reviewed

Comments:

PARENT

QID: 0929 **Rev:** 0 **Rev Date:** 9/16/14 **Source:** New **Originator:** Cork

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

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-Core Temperature Monitor System (ITM)

Description: Knowledge of the operational implications of the following concepts as the apply to the ITM system: Saturation and subcooling of water.

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

Tier: 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** 64 **SRO:** 64

Given:
- RCS pressure 1760 psig and dropping
- Unit 1 Reactor tripped
- ICCMDS CETs indicate 618 °F
- ICCMDS display for SCM shows -1°F and is flashing
- ICCMDS timer is displaying "1:46"

Which of the following procedural actions are required for these conditions?

A. Trip one RCP in each loop and perform rapid RCS cooldown

B. Trip ALL RCPs

C. Go to 1202.004, Overheating

D. Go to 1202.005, Inadequate Core Cooling

Answer:

B. Trip ALL RCPs

Notes:

B is correct, the operational implication of loss of subcooling margin is to trip all RCPs if less than two minutes have elapsed.
A is incorrect but plausible, this is the action taken previously if SCM lost and greater than two minutes had elapsed.
C is incorrect but plausible, CETs 618°F is an entry condition for overheating but RCPs should be tripped first and loss of subcooling margin takes priority.
D is incorrect but plausible, if the SCM is negative and flashing, this is an indication of superheated conditions but the CETs must be moving away from the saturation line for entry into ICC.

References:

120.022, Loss of Subcooling Margin

History:

New for 2014 Exam

INSTRUCTIONS	CONTINGENCY ACTIONS

CAUTION

Tripping all RCPs > 2 minutes after loss of adequate SCM could cause reactor core to become uncovered.

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Check elapsed time since loss of adequate SCM <u>AND</u> perform the following:	
A. <u>**IF**</u> ≤ 2 minutes have elapsed, <u>**THEN**</u> trip all RCPs: • P32A • P32C • P32B • P32D	A. <u>**IF**</u> > 2 minutes have elapsed, <u>**THEN**</u> leave currently running RCPs on. 1) Perform rapid cooldown per **step 20**, while continuing with this procedure.
B. Initiate full HPI (RT-3).	B. <u>**IF**</u> HPI flow is < full flow from one HPI pump <u>**AND**</u> RV Head void is indicated, <u>**THEN**</u> perform rapid cooldown per **step 20**, while continuing with this procedure.
1) <u>**IF**</u> Makeup Tank level drops below 18", <u>**THEN**</u> close Makeup Tank Outlet (CV-1275).	
C. Verify proper EFW actuation and control (RT-5).	C. Perform the following: 1) <u>**IF**</u> AUX Feedwater Pump (P75) is available, <u>**THEN**</u> perform the following: a) Verify at least one Condensate Pump (P2A, P2B, or P2C) in service. b) Dispatch an operator to open Aux FW Pump RECIRC to E-11A Isolation (FW-1). c) Open Feedwater Pumps DISCH Crosstie (CV-2827).

(1. CONTINUED ON NEXT PAGE)

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

QID: 0924 **Rev:** 0 **Rev Date:** 9/14/14 **Source:** New **Originator:** Cork

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

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

System Number: 086 **System Title:** Fire Protection System (FPS)

Description: Knowledge of the physical connections and/or cause effect relationships between the Fire Protection System and the following systems: Raw service water.

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

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

Group: 2 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** 65 **SRO:** 65

Given:
- In preparation for a surveillance test an operator is transferring A SW bay from the lake to the ECP.
- During the transfer, SG-5 fails to open.
- The operator tries to open SG-1 but it will not open.
- The operator also tries to open SG-3 but it will not open.

What is the impact on the Fire Protection System?

A. P-6A, Electric Fire Pump, is non-functional

B. P-6B, Diesel Fire Pump, is non-functional

C. P-11, Jockey Fire Pump is non-functional

D. Temporary Fire Pump can not be operated

Answer:

B. P-6B, Diesel Fire Pump, is non-functional

Notes:

B is correct since P-6B takes suction on the A SW bay.
A is incorrect but plausible since the A pump logically should be in the A bay, but it isn't.
C and D are incorrect and are the other available choices for fire pumps.

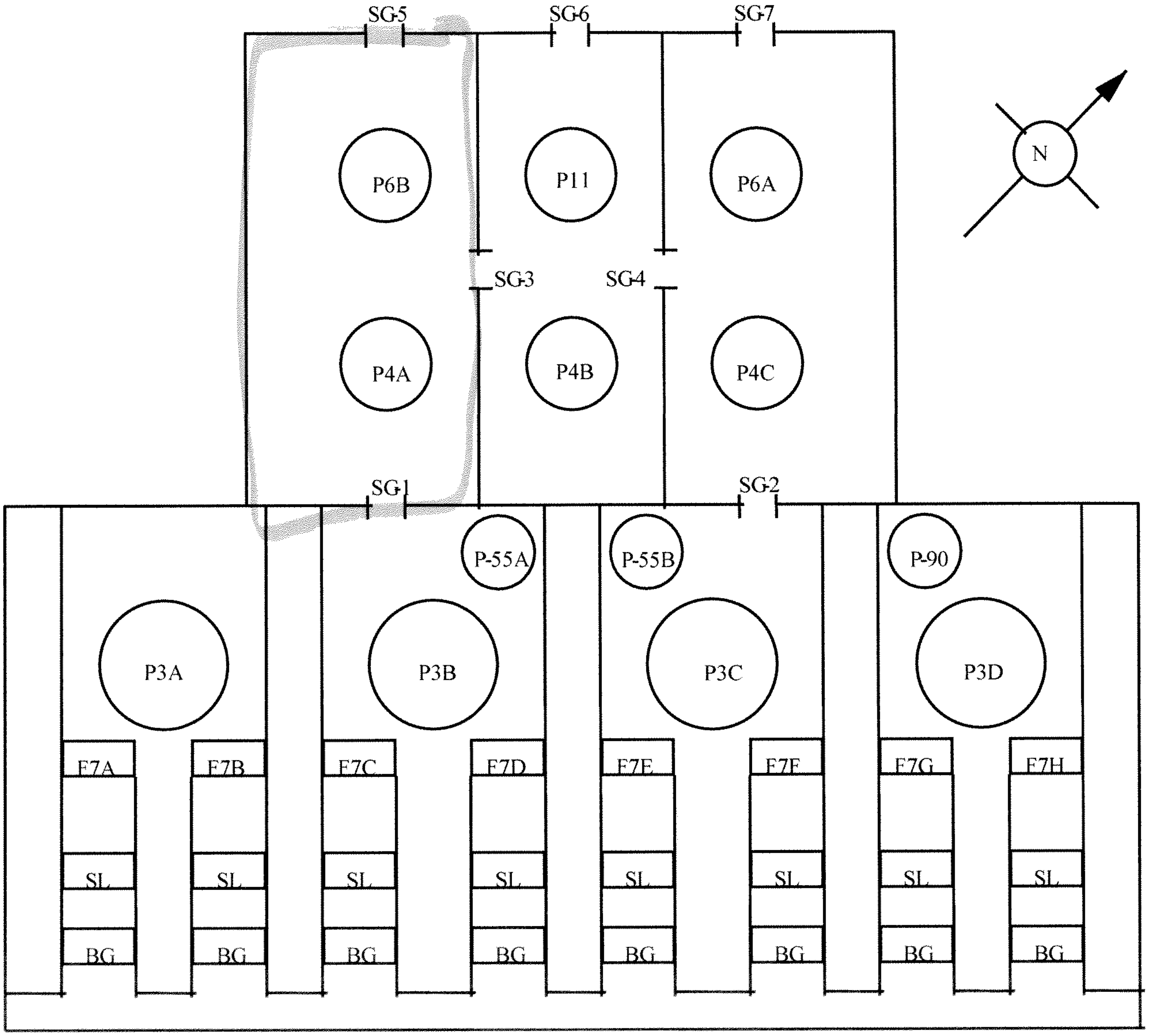
References:

STM 1-42, Service & Auxiliary Cooling Water

History:

New for 2014 Exam

FIGURE 42.03: RELATIVE POSITION OF MAJOR COMPONENTS



P6A, P6B	- FIRE WATER PUMPS
P4A, P4B, P4C	- SERVICE WATER PUMPS
P3A, P3B, P3C, P3D	- CIRCULATING WATER PUMPS
P55A, P55B	- SCREEN WASH PUMPS
P90	- SODIUM BROMIDE/HYPERCHLORIDE PUMP
F7A thru F7H	- TRAVELING SCREENS
SL's	- STOP LOGS
BG's	- BAR GRATES
P11	- JOCKEY FIRE PUMP

ANO UNIT 1 – 2014

TIER 3

Questions 66 - 75

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

QID: 0389 **Rev:** 3 **Rev Date:** 12/7/00 **Source:** Direct **Originator:** S.Pullin

TUOI: ASLP-RO-PRCON **Objective:** 8 **Point Value:** 1

Section: 2 **Type:** Generic

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to verify the controlled procedure copy.

K/A Number: 2.1.21 **CFR Reference:** 41.10 / 45.10 / 45.13

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

Group: **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 66 **SRO:** 66

Given:

-A job is in progress that will last for several weeks.
-The procedure has been verified at the start of the job.
-A pre-job brief has been completed for all participants.

How often should the procedure for this job be verified current and what source is used?

A. Once every 24 hours,
eB change number

B. Once every 24 hours,
the work order reference

C. Prior to each use,
eB change number

D. Prior to each use,
the work order reference

Answer:

C. Prior to each use,
eB change number

Notes:

Answer [c] is correct IAW EN-AD-102, Rev 3 all other choices are familiar frequencies of tasks.
Answer C is still correct but this guidance has been moved to EN-HU-106 Rev 3., Page 6 of 24

References:

EN-HU-106, Procedure and Work Instruction Use and Adherence

History:

Modified regular exambank QID# 6054 for use in 2001 RO Exam.
Selected for use in 2002 RO exam.
Selected for the 2008 RO Exam.
Selected for 2014 Exam

Q-66

4.0 RESPONSIBILITIES

[1] **Managers** are responsible for ensuring that personnel in their organizations adhere to the requirements provided in this Procedure.

[2] **Supervisors** are responsible for providing direction to personnel in their organization on the use of, and any clarifications to, this Procedure. Specific responsibilities include the following:

(a) Resolution or correction of identified problems is expected before the Procedure or Work Instruction is commenced.

(b) Determine which, if any, Procedure or Work Instruction steps are not applicable for the evolution being performed and mark as Not Applicable (N/A), initial, and date.

(c) Clarify Procedure or Work Instruction steps not understood by the user.

(d) Reinforce expectations to Procedure or Work Instruction Users to stop if the Procedure or Work Instruction cannot be performed as written and contact supervision.

(e) Reinforce expectations for Procedure or Work Instruction compliance

[3] **Procedure or Work Instruction Users** are responsible for:

(a) Verifying the correct revision of the Procedure before performing any activity identified in the Procedure.

(b) Complying with the requirements identified in this Procedure. [P-20286]

(c) Understanding Procedure or Work Instruction requirements prior to starting any activity identified in the Procedure or Work Instruction. [P-177727]

(d) Identifying any potential problem with the Procedure or Work Instruction prior to starting any activity identified in the Procedure or Work Instruction.

(e) Resolving any potential problem with the Procedure or Work Instruction prior to starting any activity identified in the Procedure or Work Instruction.

(f) Using placekeeping techniques in accordance with this procedure (Sec 5.6.3).

(g) Knowing the placekeeping method to be used prior to performance.

(h) Stopping a Procedure or Work Instruction activity if a deficiency is identified and activities cannot proceed per the Procedure or Work Instruction.

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

QID: 0245 **Rev:** 1 **Rev Date:** 9-1-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ASLP-RO-OPSPR **Objective:** 4j **Point Value:** 1

Section: 2 **Type:** Generic

System Number: 2.1 **System Title:** Conduct Of Operations

Description: Knowledge of conduct of operations requirements.

K/A Number: 2.1.1 **CFR Reference:** 41.10 / 45.13

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

Group: G **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 67 **SRO:** 67

The feedwater/condensate system startup is in progress.
A main feedwater isolation valve had been closed by operation of the manual handwheel to isolate the system.

Prior to declaring this valve operable, what action must be taken?

A. The valve must be fully opened using the local handwheel.

B. Electricians must check the torque switch adjustment.

C. The measured torque value required to remove the valve from its seat is verified below the limit.

D. The valve must be stroked electrically to confirm proper clutch engagement.

Answer:

D. The valve must be stroked electrically to confirm proper clutch engagement.

Notes:

a. is incorrect because the valve only needs to be cracked from the closed seat using the local handwheel.
b. is incorrect because the torque switch only needs to be adjusted when suspected to be out of adjustment, manual operation of the valve will not affect the torque switch setting.
c. is incorrect because the torque values are only required when seating a MOV and operability is desired.

References:

1015.001, Conduct of Operations

History:

Used in 1999 exam.
Direct from ExamBank, QID# 3090
Selected for use in 2002 RO exam.
Selected for use on 2007 RO Exam.
Selected for the 2008 RO Exam KA - 2.1.29
Selected for 2014 Exam

11.5 IF any of the following conditions are applicable:

- Seismic restraint to be removed
- Valve operator to be removed.
- Valve to be removed.
- Piping to be disconnected from MOV.
- Other condition that could make MOV inoperable due to seismic concerns.

THEN perform EITHER of the following:

- Enter applicable Tech Spec action for affected system(s).
- Verify Engineering Calculation justifies Piping Code Qualification.

11.6 WHEN desired to return MOV to service following manual seating, de-clutching, locked, sealed, or otherwise secured,
THEN perform the following:

11.6.1 IF installed,
THEN remove danger tag or locking device from handwheel.

11.6.2 IF MOV manually seated,
THEN manually free MOV from its seat.

11.6.3 IF installed,
THEN clear danger tag from MOV breaker.

11.6.4 IF locking device removed,
THEN verify independent verification performed.

NOTE

- Stroke testing does not require the MOV to travel its full cycle.
- Electrical MOV movement ensures proper engagement of the clutch mechanism.

11.6.5 IF MOV manually seated or de-clutched,
THEN confirm proper clutch mechanism engagement as follows:

A. IF full stroke of MOV NOT desirable,
THEN manually position valve until intermediate indication observed.

B. Electrically stroke MOV to desired position and verify proper engagement of clutch mechanism by observing indicating light changes (if available) and local observation of stem movement. Smooth valve movement should be observed.

11.6.6 IF installed,
THEN clear caution tag(s).

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

QID: 0143 **Rev:** 3 **Rev Date:** 11/3/05 **Source:** Direct **Originator:** D.Walls

TUOI: ASLP-AO-DUTYS **Objective:** 14 **Point Value:** 1

Section: 2 **Type:** Generic K&As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc..

K/A Number: 2.1.29 **CFR Reference:** 41.10 / 45.1 / 45.12

Tier: 3 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

Group: G **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 68 **SRO:** 68

Given:

- The plant is shut down for Refueling.
- A Core Flood system valve alignment is in progress inside Controlled Access.
- The primary sample room has become a high radiation area due to hydrogen peroxide cleanup.
- The first check was made on SS-81, Core Flood Tanks Sample Isolation, but the Shift Manager decided to waive the second check to reduce exposure to high radiation.

Which one of the following statements most accurately describes why the Shift Manager's decision is acceptable or unacceptable?

A. Acceptable, independent verifications for manual valves can be waived for valve alignments inside High Radiation Areas.

B. Unacceptable, independent verification should not be waived if remote valve position indication is provided.

C. Acceptable, independent verification can be waived at any time with the Shift Manager's approval.

D. Unacceptable, independent verifications cannot be waived for valve alignments without the approval of the Manager of Plant Operations.

Answer:

A. Acceptable, independent verifications for manual valves can be waived for valve alignments inside High Radiation Areas.

Notes:

"A" is correct, this meets the guidance of 1015.001.
"B" is untrue, independent verification of a manual valve in a high rad area should be waived, but only if no other means are available to verify it's position.
"C" is untrue, the Shift Manager can only waive verification in specific situations.
"D" is untrue, it lists the wrong approval authority.

References:

1015.001, Conduct of Operations

History:

Taken from Exam Bank QID # 3273

Used in 98 RO Re-exam
Modified for use in 2001 RO Exam.
Selected for 2005 RO re-exam.
Selected for 2014 RO Exam

14.3.7 WHEN operating manual valves (including reach rod operated valves with no valve position indicator pin or dial indicator),
THEN proper position should be verified using at least two of the following methods:

- Handwheel movement stops
- Visual observation of valve stem position
- Local position indication
- Remote position indication
- Verification of system parameters (flow, pressure, etc.)

14.3.8 IF manual valve located in high radiation area
OR area where safety concern exists,
THEN independent verification can be waived by CRS/SM
AND shall be documented with reason, time, and date.

14.3.9 Extra care should be used to determine valve position prior to operating Knocker Valves. Knocker valves can be easily damaged by driving them into open or closed seats due to force applied by knocker.

14.3.10 When pipe caps removed for tagging or other evolutions, FME bags (available from both the store room and tool room), plastic bags or other storage containers can be used for storage of pipe caps.

- FME bags should not be used in Containment/Reactor Buildings (sump clogging concerns) or in Aux Buildings (radwaste concerns).
- Do not attach bag or other storage container with multiple caps to piping due to possible seismic issues.
- IF multiple caps stored together,
  THEN they should be labeled and staged appropriately.

14.3.11 IF necessary to climb on equipment to access components,
THEN refer to Operations Expectations and Standards (COPD-001).

14.3.12 Large manual valves can be very difficult to operate with excessive differential pressure and may NOT stroke full open or closed (e.g. condensate pump discharge, MFP suction, etc.). Bypass valves should be opened if available to equalize pressures prior to operating these valves.

14.4 Guidance concerning Mechanical agitation of any plant component.

- Mechanical agitation of any plant component should only be performed under guidance of an approved work order, procedure, or approved trouble shooting plan. Condition report shall be written for any safety related check valve that is leaking by.

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

QID: 0928 **Rev:** 0 **Rev Date:** 9/16/14 **Source:** New **Originator:** Possage

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

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

K/A Number: 2.2.22 **CFR Reference:** 41.5 / 43.2 / 45.2

Tier:	3	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 69 **SRO:** 69

Which of the following lists ALL of the MODES of applicability for Technical Specification 2.1.2, RCS Pressure Safety Limit?

A. 1 & 2 ONLY

B. 1, 2 & 3 ONLY

C. 1, 2, 3 & 4 ONLY

D. 1, 2, 3, 4 & 5

Answer:

D. 1, 2, 3, 4 & 5

Notes:

Reactor core safety limits are MODES 1 &2 which adds credibility to (A), and the SL Violations discusses if RCS Pressure SL is violated in MODES 3, 4, and 5 then take actions to restore within 5 minutes which adds credibility to additional combinations used in (B) and (C) .

D is the only correct answer per TS 2.1.2.
A, B, C are partial lists but contain the word ONLY and thus are incorrect.

References:

Technical Specifications 2.1.2

History:

New for 2014 Exam

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5 x 10^{-3} x (Burnup, MWD/MTU)°F) for TACO2 applications, ≤ 4642 - (5.8 x 10^{-3} x (Burnup, MWD/MTU)°F) for TACO 3 applications, and ≤ 4901 °F, decreasing linearly by 13.7 °F per 10,000 MWD/MTU of burnup for COPERNIC applications.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation, 1.18 for the BWC correlation, and 1.132 for the BHTP correlation.

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the Core Operating Limits Report, so that the safety limits are met.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits AND be in MODE 3 within 1 hour.

2.2.3 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits AND be in MODE 3 within 1 hour.

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

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

QID: 0161 **Rev:** 1 **Rev Date:** 4/24/2002 **Source:** Direct **Originator:** J. Cork

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

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

System Number: 2.2 **System Title:** Equipment Control

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

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

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

Group: G **SRO Imp:** 4.5 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 70 **SRO:** 70

Given:

- Power escalation is in progress following a shutdown.
- Reactor power is 35%.
- Rod 6 of Group 7 drops.

Which of the following actions should be taken?

A. Insert all regulating rods in sequential mode.

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

C. Verify plant stabilizes at 320 MWe after ICS runback.

D. Verify SDM within COLR limit within one hour.

Answer:

D. Verify SDM within COLR limit within one hour.

Notes:

[a] would only be performed if power was <2%.
[b] would not be done because only one rod dropped.
[c] power is <360 MWe so there wouldn't be any runback, the value given would require a power increase.
[d] is the correct answer per TS.

References:

1203.003, Control Rod Drive Malfunction Action

History:

Developed for use in 98 RO Re-exam.
Used in 2001 RO/SRO Exam.
Selected for 2002 RO/SRO exam. Revised to agree with ITS.
Selected for 2010 RO/SRO exam KA2.4.11
Selected for 2014 Exam.

SECTION 2
DROPPED ROD – REACTOR CRITICAL

NOTE

- Technical Specifications defines an inoperable rod as follows:
  - Safety Rod that is NOT fully withdrawn within one hour, except during performance of rod exercise surveillance (TS 3.1.5). If the Safety Rod is declared inoperable in TS 3.1.5, then TS 3.1.4 must also be entered.
  - Inability to move control rod (SR 3.1.4.2) or APSR (TS 3.1.6).
  - Rod can not be located with API, RPI or limit lights (TS 3.1.7). Not meeting TS 3.1.7 results in not meeting either TS 3.1.4 or 3.1.6.
- The misaligned (>6.5%) rod's position is NOT to be used in the calculation of the rod group average position.

4. IF rod is declared inoperable
OR is misaligned >6.5%,
THEN perform the following:

NOTE

If the inoperable control rod is fully inserted, then it is not necessary to consider it inoperable for the purposes of shutdown margin calculations because it has inserted its negative reactivity. A control rod is considered to be inoperable if it is not free to insert into the core within the required insertion time, or does not have at least one position indicator channel operable, i.e., cannot be located. (Ref. TS 3.1.4 Bases)

- Within 1 hour **AND** once every 12 hours thereafter, verify 1.5% available shutdown margin per Reactivity Balance Calculation (1103.015) OR initiate boration to restore SDM to be within COLR limit within 1 hour.

A. IF control rod is NOT fully inserted,
OR the control rod can NOT be located,
THEN use worksheet 4 and use the inoperable rod option (does NOT apply to APSRs).

B. IF rod is fully inserted,
THEN use worksheet 4 and do NOT use the inoperable rod option.

QID: 0925 **Rev:** 0 **Rev Date:** 9/15/14 **Source:** Modified **Originator:** Cork

TUOI: ASLP-RO-RADP **Objective:** 15 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of 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:** Yes **Taxonomy:** K

Question: **RO:** 71 **SRO:** 71

What is the federal occupational exposure limit to the LDE (Lens Dose Equivalent) in accordance with 10CFR20?

A. 0.1 rems/calendar year

B. 5.0 rems/calendar year

C. 15.0 rems/calendar year

D. 50.0 rems/calendar year

Answer:

C. 15.0 rems/calendar year

Notes:

C is the correct answer per 10CFR20 and EN-RP-201.
A is incorrect but plausible, this is the TEDE limit for the general public.
B is incoreect but plausible, this is the TEDE limit.
D is incorrect but plausible, this is the SDE limit.

References:

EN-RP-201, Dosimetry Administration

History:

Modified QID 121 for 2014 Exam

5.2 INDIVIDUAL MONITORING CLASSIFICATIONS

[1] Monitored – An individual likely to receive occupational dose, which requires monitoring per 10CFR20.1502.

(a) Unescorted – Any occupationally monitored individual who has successfully completed and maintained site specific Radiation Worker Training (RWT), and Plant Access Training (PAT) along with any site specific training.

(b) Escorted – An individual who has a need to access an RCA and is required to be monitored per 10CFR20.1502, but whose qualification status requires escorted access to the RCA.

[2] Unmonitored – Any occupationally exposed individuals not requiring monitoring per 10CFR20.1502.

5.3 LIMITS AND GUIDELINES

[1] Annual Regulatory Limits

- TEDE = 5 rem
- LDE = 15 rem
- SDE, WB = 50 rem
- SDE, ME = 50 rem
- TODE = 50 rem
- Declared Pregnant Woman (DPW) TEDE = 50 mrem/month, 500 mrem/gestation period.
- Minors = 10% of any Regulatory Limit
- Unmonitored Individuals = 10% of any Regulatory Limit
- Members of the Public TEDE = 100 mrem/year

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

QID: 0121 **Rev:** 1 **Rev Date:** 12/16/06 **Source:** Direct **Originator:** Possage

TUOI: ASLP-RO-RADP **Objective:** 15 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of 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:**	K

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

What is the federal occupational exposure limit to the whole body TEDE (Total Effective Dose Equivalent) in accordance with 10CFR20?

A. 0.1 rems/calendar year

B. 5.0 rems/calendar year

C. 15.0 rems/calendar year

D. 50.0 rems/calendar year

Answer:

B. 5.0 rems/calendar year

Notes:

"B" is the correct answer.
"A", "C", and "D" are incorrect values for general public TEDE, LDE, and SDE.

References:

EN-RP-201, Rev. 3

History:

New question developed for 2001 RO/SRO NRC Exam.
Selected for use in 2002 RO/SRO exam.
Selected for 2005 RO re-exam. Question was on skin (SDE) and the answer was "C" 50.0 rems/calendar year.
Modified and used on 2007 RO Exam
Selected for 2011 RO Exam.

PARENT

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

QID: 0234 **Rev:** 0 **Rev Date:** 12/3/98 **Source:** Direct **Originator:** J. Cork

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

Section: 2.0 **Type:** Generic K/As

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

Description: Knowledge of the specific bases for EOPs.

K/A Number: 2.4.18 **CFR Reference:** 41.10 / 43.1 / 45.13

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

Group: **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 74 **SRO:** 74

During a SGTR, which of the following actions is performed specifically to reduce plant personnel exposure?

A. Maintaining RCS pressure low within limits of Fig. 3.

B. Steaming bad SG to maintain tube-to-shell DT <150°F.

C. Aligning HPI to provide PZR Aux Spray.

D. Removing all but C & D condensate polishers from service.

Answer:

D. Removing all but C & D condensate polishers from service.

Notes:

A is incorrect. This is performed to curtail primary system losses.
B is incorrect. This is done to alleviate tube stresses and to prevent failing more tubes.
C is incorrect. Aligning aux spray during SGTR is done for pressure control without RCPs and to reduce primary system losses.
D is correct. This task is performed to clean up the secondary while using centrally located polishers to maintain doses ALARA to the train bay and polisher panel.

References:

1202.006, Tube Rupture
1203.014, Control of Secondary System Contamination

History:

Developed for use in 98 RO Re-exam
Selected for use in 2002 RO/SRO exam.
Selected for 2005 RO re-exam. KA 2.3.10
Selected for 2014 Exam

Location: C16

Device and Setpoint:

Any process monitor in Radiation Monitoring System Panel (C25 Bays 1 thru 3)	HIGH ALARM or loss of power
OR	
Turb Bldg Drn Rad Monitor (RI-5641)	HIGH ALARM or loss of power

PROC MONITOR
RADIATION
HI

Alarm: K10-B2

Monitors are listed in step 3.

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 12.

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.

   E. IF alarm was caused by instantaneous spiking, THEN reset alarm by performing the following:

      1) IF Gaseous Radwaste (RI-4830), THEN GO TO step 14.

      2) IF any other alarm, THEN select "ALARM RESET" on the appropriate drawer and exit this procedure.

NOTE

- HIGH alarm condition on RI-7461 is indicated by a red lamp and a flashing "H" on monitor display.
- Alarm setpoint is adjustable at RI-7461 on C25 and is set per Radiation Monitoring System Check and Test (1305.001), Supplement 5.
- Alarm setpoint for RI-7461 can be read by repeatedly pressing the MODE key until HighAlm value is displayed.

12. IF RB Atmos Gaseous Monitor (RI-7461) radiation is high, THEN perform the following:

    A. Compare counts to High Alarm setpoint by depressing MODE key until HighAlm value is displayed.

    B. IF alarm is caused by instantaneous spiking, THEN depress RESET button to clear alarm AND exit this procedure.

    C. IF alarm is due to rise in activity and confirmation is warranted, THEN perform the following:

    - IF reactor building is open for access, THEN notify RP of condition and to sample RB air.
    - Notify Chemistry to obtain grab sample of RB air.

    D. IF alarm is confirmed, THEN perform the following:

    - IF RCS is > 200°F, THEN perform RCS Leak Detection (1103.013).
    - IF reactor building is open for access, THEN perform the following:
        - Assess need to perform localized evacuation of Reactor Building per Evacuation procedure (1903.030).
        - Notify RP of degrading RB radiological conditions.

    E. Monitor RDACs.

    - IF rising trend is observed, THEN notify Chemistry to perform Offsite Dose Projections (1904.002).
        - Notify SM to review Emergency Action Level Classification (1903.010).

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

QID: 1025 **Rev:** 0 **Rev Date:** 9/28/2014 **Source:** New **Originator:** Possage

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

K/A Number: 2.3.15 **CFR Reference:** 41.12 / 43.4 / 45.9

Tier:	3	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.1	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 73 **SRO:** 73

The Control Room Emergency Ventilation System can be actuated by a high radiation condition monitored by fixed process radiation detectors in the duct work.

What type of detectors are 2RITS-8001A/B?

A. Proportional Detector

B. Geiger - Mueller Detector

C. Ion Chamber Detector

D. Scintillation Detector

Answer:

D. Scintillation Detector

Notes:

D is the correct answer the Control Room Emergency Ventilation system uses scintillation detectors to monitor the ventilation ductwork.
The incorrect choices are different types of detectors on the Gas Amplification curve and are familiar terms at ANO.

A is incorrect because none of the radiation monitors use a proportional type detector.
B is incorrect because while many of the radiation monitors use Geiger-Mueller type detectors, the Control Room Emergency Ventilation system does not.
C is incorrect because none of the radiation monitors use an ion chamber type detector.

References:

STM 1-62

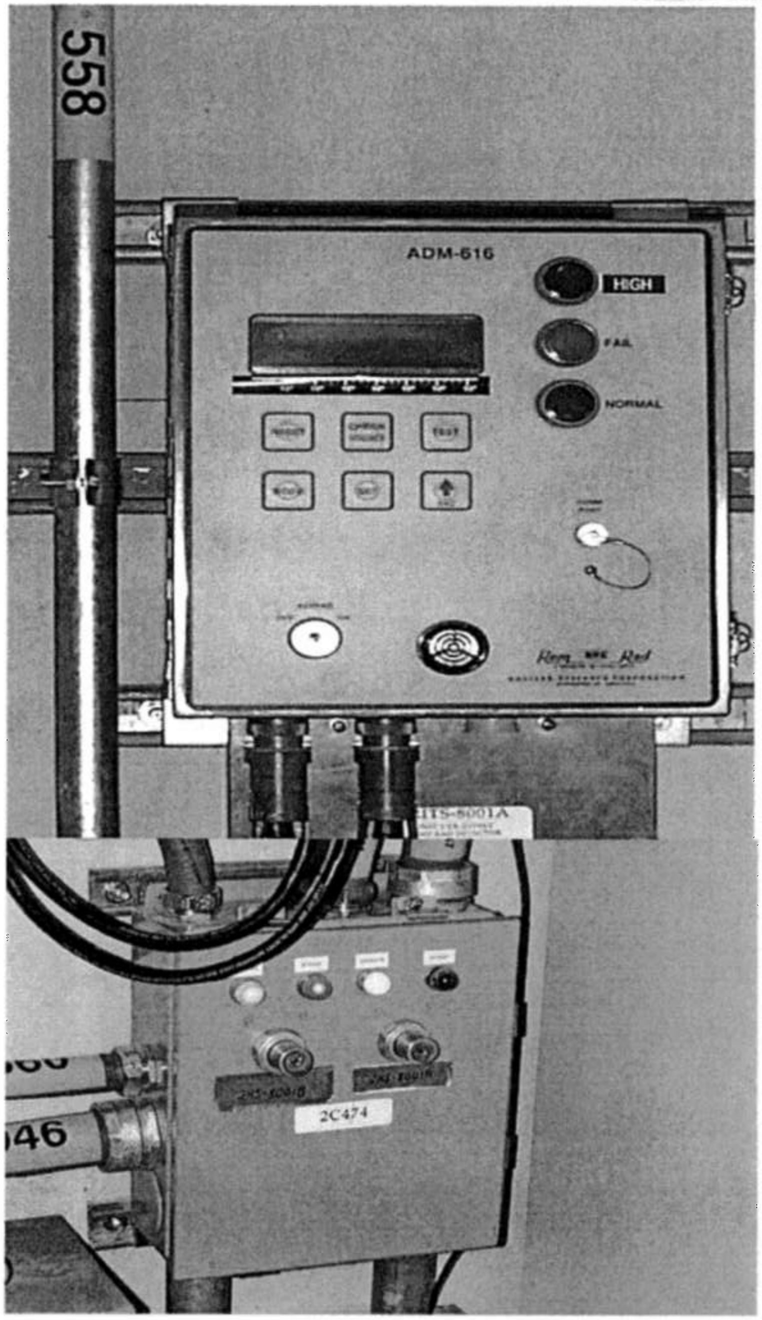
History:

New for 2014 RO exam

documents control room isolation that occurred due to RI-8001 being de-energized. LER 91-006 documents a control room isolation occurring after the warning alarm setpoint was reset too low and a signal voltage spike caused an inadvertent actuation. The necessity to report an inadvertent control room isolation has been relaxed by a change to 10CFR 50.73 (Refer to TREDS for reportability guidance).

2.1.2.1 Control Room Ventilation Radiation Detectors

Radiation monitors 2RE-8001A and 2RE-8001B are mounted in the ductwork for the Unit 1 supply air in the Unit 1 PMS computer room. Radiation detectors 2RE-8750-1A and 2RE-8750-1B are mounted in the supply duct for the Unit-2 control room.



All four of these radiation monitors have scintillation type detectors with associated pre-amplifiers (2RY-8001A and 2RY-8001B) and rate meters (2RITS-8001A and 2RITS-8001B). The rate meters are wall mounted on the Unit 1 PMS computer room West wall. The rate meters automatically select the correct range for display.

Mounted below the rate meters is 2C474 that contains two key operated switches. The BYPASS position permits maintenance or repair of any component on one detector string while the other detector string performs the design radiation monitoring and protection functions.

The Unit 1 control room normal supply duct radiation monitors also output to the Unit 1 PMS computer as PID R8001A and R8001B.

Actuation of RI-8001, 2RITS-8001A, or 2RITS-8001B closes the isolation dampers for both control rooms, stops supply fans VSF-8A/B, and starts VSF-9. Circuit failure of 2RITS-8001A or 2RITS-8001B also results in the associated protective actuations. These monitors annunciate in the Unit 1 control room to warn operators of this event. The contacts for the Unit 1 detectors are normally open, as shown in Figure 12.05, with the associated protective relay (RIAX-8001) de-energized.

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

QID: 0234 **Rev:** 0 **Rev Date:** 12/3/98 **Source:** Direct **Originator:** J. Cork

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

Section: 2.0 **Type:** Generic K/As

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

Description: Knowledge of the specific bases for EOPs.

K/A Number: 2.4.18 **CFR Reference:** 41.10 / 43.1 / 45.13

Tier:	3	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 74 **SRO:** 74

During a SGTR, which of the following actions is performed specifically to reduce plant personnel exposure?

A. Maintaining RCS pressure low within limits of Fig. 3.

B. Steaming bad SG to maintain tube-to-shell DT <150°F.

C. Aligning HPI to provide PZR Aux Spray.

D. Removing all but C & D condensate polishers from service.

Answer:

D. Removing all but C & D condensate polishers from service.

Notes:

A is incorrect. This is performed to curtail primary system losses.
B is incorrect. This is done to alleviate tube stresses and to prevent failing more tubes.
C is incorrect. Aligning aux spray during SGTR is done for pressure control without RCPs and to reduce primary system losses.
D is correct. This task is performed to clean up the secondary while using centrally located polishers to maintain doses ALARA to the train bay and polisher panel.

References:

1202.006, Tube Rupture
1203.014, Control of Secondary System Contamination

History:

Developed for use in 98 RO Re-exam
Selected for use in 2002 RO/SRO exam.
Selected for 2005 RO re-exam. KA 2.3.10
Selected for 2014 Exam

INSTRUCTIONS | CONTINGENCY ACTIONS

5. IF Reactor power is > 20%,
THEN begin controlled plant shutdown at ≥ 5% per minute.

6. Determine bad SG using one or more of the following:

- OTSG N-16 Gross Detectors:

SG A	SG B
RI-2691	RI-2692

- SGTR display on SPDS

- Plant Monitoring System Alarms

- Steam Line High Range Radiation Monitors:

SG A	SG B
RI-2682	RI-2681

- Local steam line radiation survey

- Nuclear Chemistry sample

- At low FW flow rates:

  * Higher than expected SG level
  * Lower than expected FW flow rate
  * Lower than expected MFW pump speed

7. Verify Control of Secondary System Contamination (1203.014) being performed in conjunction with this procedure.

8. WHEN bad SG is known,
THEN place bad SG EFW Pump Turbine K3 Steam Supply valve in MANUAL AND close:

SG A	SG B
CV-2667	CV-2617

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.

QID: 0927 **Rev:** 0 **Rev Date:** 9/15/14 **Source:** New **Originator:** Cork

TUOI: ASLP-FP-FBLDR **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

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

Description: Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.

K/A Number: 2.4.26 **CFR Reference:** 41.10 / 43.5 / 45.12

Tier:	3	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 75 **SRO:** 75

The Fire Brigade responded to a fire in the Startup Boiler building.
The fire was extinguished by partially expending a local fire extinguisher.

The ________ will report the fire was extinguished, the fire extinguisher would be _________, and Unit 2 Mechanical Maintenance would be notified.

A. Unit 1 WCO
hung in its assigned location

B. Unit 2 WCO
hung in its assigned location

C. Unit 1 WCO
placed on its side

D. Unit 2 WCO
placed on its side

Answer:

C. Unit 1 WCO
placed on its side

Notes:

C is correct, the Unit 1 WCO will be the Fire Brigade leader and a fire extinguisher which has been used is placed on its side near its assigned location.
A is incorrect, this has the correct person but the wrong action for the exrtinguisher.
B is incorrect, this has both the wrong person and the wrong action for the exrtinguisher.
D is incorrect, this has the correct action for the fire but the wrong person.

References:

1015.007, Fire Brigade Organization and Responsibilities

History:

New for 2014 Exam

NOTE

The use of an operator to perform the role of Fire Brigade Support Member vice a Security Officer is permissible so long as the operator shall be independent of any collateral or control room duties they may need to perform as a result of the fire. Based on Alternate Shutdown, Remote Shutdown, and Fire Brigade requirements, Fire Brigade Support Member supplied by Operations shall be an extra person not standing watch.

6.0 INSTRUCTIONS

6.1 Assignment of Fire Brigade Personnel

6.1.1 The Unit 1 Fire Brigade consists of the following:

A. Unit 1 Fire Brigade Leader

B. Three Fire Brigade Members from Unit 2

C. Fire Brigade Support Member from Security Force or other Fire Brigade qualified individual designated by either unit Shift Manager.

6.1.2 The Unit 2 Fire Brigade consists of the following:

A. Unit 2 Fire Brigade Leader

B. Three Fire Brigade Members from Unit 1

C. Fire Brigade Support Member from Security Force or other Fire Brigade qualified individual designated by either unit Shift Manager.

NOTE

The ANO Fire Brigade can only respond to fires within the Protected Area. All other fires on ANO property (outside the Protected Area) require notification of the London Fire Department instead of the ANO Fire Brigade (CR-ANO-C-2011-0098).

6.2 The fire is reported to the Control Room of the affected unit.

6.2.1 The SM/CRS of the affected unit dispatches the Fire Brigade to the scene of the fire. The SM/CRS of the unaffected Unit will dispatch the Fire Brigade for zones identified in 1203.049/2203.049, Fires In Areas Affecting Safe Shutdown.

6.2.2 The Fire Brigade Leader of the affected unit responds and assumes command of the fire fighting activities.

6.2.3 The Fire Brigade Members from the unaffected unit respond along with the Security Fire Brigade Support Member. This comprises the initial fire fighting force.

8.4 Fire Extinguishers

8.4.1 Return all fire extinguishers to their assigned location.

8.4.2 IF extinguisher is discharged,
THEN place the extinguisher on its side.

8.4.3 Coordinate with Unit 2 Mechanical Maintenance for replacement of expended extinguishers.

8.5 IF automatic Fire Water valve, Turbine Generator CO_2 or Control Room Halon has tripped, THEN reset per the appropriate sections of Fire Protection Systems (1104.032) and Unit 2 Fire Protection Systems Operations (2104.032).

8.6 IF Fire Water pump (P-6A or P-6B) started,
THEN perform "Securing Fire Water Pumps (P-6A and P-6B)" section of Fire Protection Systems (1104.032).

8.7 IF Fire Water valves have been repositioned,
THEN perform the following per the appropriate Attachments and Supplements of Fire Protection Systems (1104.032) and Unit 2 Fire Protection Systems Operations (2104.032).

8.7.1 Verify valves properly aligned.

8.7.2 As applicable, attach tamper seals or verify tamper switches engaged.

8.8 Reset and clear, alarm and trouble circuits as needed:

8.8.1 For Unit 1, per "Trouble Action and Circuit Restoration", Attachment A of Fire Protection System Annunciator Corrective Action (1203.009).

8.8.2 For Unit 2, per Fire Protection System Annunciator Corrective Action (2203.009).

ANO UNIT 1 – 2014

SRO EXAM

Questions 76 - 100

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

QID: 0871 **Rev:** 0 **Rev Date:** 9/2/14 **Source:** New **Originator:** Possage

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

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

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

Description: Ability to determine or interpret the following as they apply to a small break LOCA: RCS parameters.

K/A Number: EA2.15 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	3.3	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.4	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 76

Unit 1 is at 100% power when the following conditions are noted:

- M/U Tank Level 85" and trending down at 1"/min
- M/U Flow 41 gpm and stable
- Pressurizer Level 220" and stable
- K08-A7 "RCP SEAL INJ FLOW LO" is in alarm
- Seal Injection Flows:
  A - 15 gpm
  B - 3 gpm
  C - 3 gpm
  D - 3 gpm

What procedure should be used to mitigate the given conditions?

A. 1203.031, Reactor Coolant Pump and Motor Emergency, Section 1 Seal Degradation

B. 1203.039, Excess RCS Leakage

C. 1203.031, Reactor Coolant Pump and Motor Emergency, Section 2 Seal Failure

D. 1203.012G, ACA - RCP Seal Injection Flow Low

Answer:

B. 1203.039, Excess RCS Leakage

Notes:

Answer B is correct, these are indications of a seal cooler failure in the A RCP. Seal injection flow is at the maximum of the indication for seal injection flow and the leak is causing the other RCPs seal flows to be lower. 1203.039 step 8 address an RCP seal cooler leak.
Answers A and C are incorrect but plausible since these two sections within the RCP and Motor Emergency AOP deal with seal problems but not a seal cooler leak.
Answer D is incorrect but plausible if the examinee believes this ACA should be entered for RCPs B, C, D seal injection flows being low outside of the normal band of 8-10 gpm.

References:

1203.039, Excess RCS Leakage
1103.006, Reactor Coolant Pump Operation

History:

New for 2014 Exam

8. IF RCP Seal Cooler RCS to ICW leak is indicated
OR
RCS leak into Letdown Cooler can not be isolated,
THEN perform the following:

NOTE
Minimum seal injection flow for each RCP is 2.5 gpm.

A. IF seal injection is available,
THEN verify ≥ 2.5 gpm seal injection flow per RCP.

B. IF seal injection is not available
AND
RCP(s) are operating,
THEN perform the following:

1) IF the Reactor is critical,
THEN trip the reactor
AND perform Reactor Trip (1202.001), while continuing with this procedure.

2) Actuate EFW AND verify proper actuation and control (RT-5).

3) Trip running RCPs.

4) Place RCP Seal Bleedoff (Alternate Path to Quench Tank) controls in CLOSE:

- SV-1270
- SV-1272
- SV-1271
- SV-1273

5) Isolate RCP Seal Bleedoff (Normal) by closing either:

- CV-1274

OR

- CV-1270
- CV-1272
- CV-1271
- CV-1273

(8. CONTINUED ON NEXT PAGE)

5.33 Operation with P-32C secured will result in a higher pressure to be sensed in the pressurizer due to low spray flow. The spray valve will cycle at a lower indicated pressure (Cycle 25 startup ~35 psig lower) than with P-32C in service. CR-ANO-1-2013-2241

6.0 SETPOINTS

The following conditions must be satisfied to start an RCP from the control room.

6.1 Rx power <22%.

6.2 RCP seal injection flow >3 gpm.
If <3 gpm, alarms RCP SEAL INJ FLOW LO (K08-A7).

RCP P-32A Seal Injection Flow (FS-1280)

RCP P-32B Seal Injection Flow (FS-1281)

RCP P-32C Seal Injection Flow (FS-1282)

RCP P-32D Seal Injection Flow (FS-1283)

6.3 RCP motor cooling flow >250 gpm (non-nuclear ICW).
If <250 gpm, alarms RCP MOTOR COOLING FLOW LO (K08-E6).

P-32A MTR Air LO CLR ICW RTN Flow (PDIS-2260)

P-32B MTR Air LO CLR ICW RTN Flow (PDIS-2261)

P-32C MTR Air LO CLR ICW RTN Flow (PDIS-2262)

P-32D MTR Air LO CLR ICW RTN Flow (PDIS-2263)

6.4 RCP seal cooling flow >30 gpm (nuclear ICW).
If <30 gpm, alarms RCP SEAL COOLING FLOW LO (K08-E7).

P-32A Seal CLR ICW RTN Flow (PDIS-2250)

P-32B Seal CLR ICW RTN Flow (PDIS-2251)

P-32C Seal CLR ICW RTN Flow (PDIS-2252)

P-32D Seal CLR ICW RTN Flow (PDIS-2253)

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

QID: 0872 **Rev:** 0 **Rev Date:** 9/2/14 **Source:** New **Originator:** Possage

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

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

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

Description: Ability to determine operability and/or availability of safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 41.7 / 43.5 / 45.12

Tier:	1	**RO Imp:**	3.6	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 77

****REFERENCES PROVIDED********

Given:
- RCS Temperature is 280 F
- 'A' Decay Heat Removal Pump is in service
- SPDS indicates P-34A flow 3000 gpm
- FIS-1401 (C18) has failed low (DHR Loop A Flow)

The CRS should declare FIS-1401 INOPERABLE and ______________________________.

A. Write a Work Request

B. Perform a Safety Function Determination

C. Enter T.S. 3.5.3 Condition A, ECCS-Shutdown action statement

D. Enter T.S. 3.3.15 Condition A, PAM Instrumentation action statement

Answer:

A. Write a Work Request

Notes:

Include TS 3.3.15, 3.5.3, and last 2 pages of operability section from 1104.004 in students' handout.

A is correct per TS 3.3.15 bases which states that SPDS is an acceptable means of flow indication.
B is incorrect but plausible since FIS -1401 should be declared inoperable and supports the DHR System, T.S. 3.0.6 would allow continued operation if a Safety Function Determineation revealed that the safety function was satisfied.
C is incorrect but plausible since TS 3.5.3 could be exercised if the train was declared inoperable due to no flow indication.
D is incorrect but plausible since FIS -1401 should be declared inoperable but TS 3.3.15 does not have to be entered due to SPDS indication.

References:

Technical Specification 3.3.15 and bases
1104.004, Decay Heat Removal Operating Procedure

History:

New for 2014 exam

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.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the availability of information so that the control room operating staff can:

- Perform the diagnosis specified in the abnormal and emergency operating procedures. These variables include preplanned actions for the primary success path of DBAs (e.g., loss of coolant accident (LOCA);
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, which are required for safety systems to accomplish their safety functions;
- Determine whether systems important to safety are performing their intended functions;
- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and estimate the magnitude of any impending threat.

SAR Section 7.3.4 (Ref. 4) documents the results of the Regulatory Guide 1.97 analysis process which identified Type A and Category I non-Type A variables.

In MODE 1, PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 2 and 3, Category I, non-Type A, instrumentation must be retained in Technical Specifications because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category I, non-Type A variables are important for reducing public risk, and satisfy Criterion 4 of 10 CFR 50.36 (Ref. 5).

LCO

LCO 3.3.15 requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the unit and to bring the unit to, and maintain it in, a safe condition following that accident. Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. When a channel includes more than one qualified control room indication, such as both an indicator and a recorder, or an indicator and Safety Parameter Display System (SPDS) readout, etc., only one indication is required for channel OPERABILITY.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 Two LPI trains shall be OPERABLE.

--NOTE--
An LPI train may be considered OPERABLE during alignment and when aligned for decay heat removal, if capable of being manually realigned to the LPI mode of operation.

APPLICABILITY: MODE 3 with Reactor Coolant System (RCS) temperature ≤ 350°F,
MODE 4.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to ECCS DHR loops.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LPI train inoperable.	A.1 Restore LPI train to OPERABLE status.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 --------------NOTE--------------- Only required if one DHR train is OPERABLE. ------------------------------------ Be in MODE 5.	24 hours
C. Two LPI trains inoperable.	C.1 Initiate action to restore one LPI train to OPERABLE status. AND C.2 --------------NOTE--------------- Only required if one DHR train is OPERABLE. ------------------------------------ Be in MODE 5.	Immediately 24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.3.1	-------------------------------NOTE------------------------------- An LPI train may be considered OPERABLE during alignment and operation for DHR, if capable of being manually realigned to the LPI mode of operation. ---	
	For all equipment required to be OPERABLE, the following SRs are applicable: SR 3.5.2.1, SR 3.5.2.4, SR 3.5.2.2, SR 3.5.2.5. SR 3.5.2.3,	In accordance with applicable SRs

23.11 Instruments Removed from Service

As a general rule, flow measurements derived from differential pressure across a restriction are inaccurate below 10% of the flow span. Under zero flow conditions, readings between 0 and 5% of indicated flow span are to be expected and do not necessarily represent a need for instrument calibration. Under zero flow conditions, if the indicated flow is above 10% of the flow span it will be required to be calibrated, but will not be considered inoperable.

When a channel includes more than one qualified control room indication, such as both an indicator and a recorder, or an indicator and Safety Parameter Display System readout, etc., only one indication is required for channel operability (TS 3.3.15 Bases).

23.12 Failed LPI Flow Instruments

Any qualified indication can serve to meet the requirements of TS 3.3.15-1.19 Condition "A". LPI flow indicators FIS-1401 and FIS-1402, SPDS and FIRS-1500 meet the requirements of TS 3.3.15-1.19. An inoperable LPI flow instrument (i.e. transmitter and/or associated instrument loop) requires the following actions to be performed:

23.12.1 Declare associated train of LPI inoperable and applicable TS LCO (3.5.2 or 3.5.3) not met based on Support SSC inoperability.

23.12.2 Enter TS 3.3.15 Condition A.

23.12.3 Perform one of the following:

A. Enter one of the applicable Tech Specs:

- If RCS >350°F, then enter TS 3.5.2 Condition A.
- If RCS ≤350°F, then enter TS 3.5.3 Condition A.

B. IF desired to enter TS 3.0.6,
THEN perform the following:

NOTE
Compliance with the Conditions and Required Action of TS 3.5.2 or TS 3.5.3 may be delayed until it is determined that LCO 3.0.6 cannot be applied.

1. IF opposite train and required support equipment are operable,
THEN perform 1015.045, Unit 1 Safety Function Determination Program (refer to Att. 2 of the procedure).

a. IF opposite train is inoperable,
THEN enter applicable TS 3.5.2 or TS 3.5.3.

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

QID: 0873 **Rev:** 0 **Rev Date:** 9/2/14 **Source:** New **Originator:** Possage

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

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

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

Description: Knowledge of 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:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 78

Given:
- ESAS actuation has occurred due to a steam line break inside the Reactor Building.

Based on the above condition, which procedure specifically directs tripping Reactor Coolant Pumps?

A. 1202.010, ESAS

B. 1202.003, Overcooling

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

D. 1203.031, Reactor Coolant Pump and Motor Emergency

Answer:

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

Notes:

Answer C is correct, only this procedure contains the direction to trip RCPs on ESAS actuation of channels 5 or 6.
Answer A is incorrect, although this procedure is entered upon ESAS actuation in most instances, it simply directs performance of RT-10.
Answer B is incorrect, although a steam line break will cause an overcooling, only RT-10 contains the direction to trip RCPs.
Answer D is incorrect, although an isolation of cooling to RCPs would normally cause entry into this AOP, this procedure would not be used with these conditions.

References:

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

History:

New for 2014 Exam

VERIFY PROPER ESAS ACTUATION

3. **IF any RCP is running,**
   THEN perform the following:

   A. IF ES Channel 5 or 6 has actuated,
      THEN perform the following:

      1) IF SCM is adequate,
         THEN trip all running RCPs due to loss of ICW:

         - P32A
         - P32C
         - P32B
         - P32D

      2) IF SCM is **not** adequate,
         THEN 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 perform the following:

            (1) Leave currently running RCPs on.

            (2) IF RCS press > 150 psig,
                THEN notify CRS to **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure.

            (3) Restore RCP services per RT-8 while continuing.

   B. IF neither ES channel 5 nor 6 has actuated,
      THEN dispatch an operator to perform Service Water And Auxiliary Cooling System (1104.029) Exhibit B, "Restoring SW to ICW Following ES Actuation" while continuing.

      1) WHEN ICW Cooler SW Outlets and Bypasses are aligned per 1104.029, Exhibit B,
         THEN override AND open one Service Water to ICW Coolers Supply
         (CV-3811 or CV-3820).

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

QID: 1026 **Rev:** 0 **Rev Date:** 9/30/14 **Source:** New **Originator:** Cork

TUOI: A1LPR-RO-EOP08 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to determine or interpret the following as they apply to a Station Blackout: RCS core cooling through natural circulation cooling to S/G cooling.

K/A Number: EA2.02 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 4.4 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 79

Given:
- Both units have tripped due to a loss of offsite power
- SU1 voltage 15.3 KV
- SU2 voltage 60.1 KV
- K01-A2 "EDG1 TRIP" in alarm
- K02-B7 "A4 L.O. RELAY TRIP" in alarm
- CETs 600 °F
- RCS pressure 1850 psig
- RVLMS Level 1 and 2 indicate "Dry"

Based on the above conditions, which of the following procedure actions are required to be performed?

A. Go to 1202.002, Loss of Subcooling Margin

B. Perform rapid cooldown per 1202.008, Blackout

C. Perform RT-4, Initate HPI Cooling

D. Dispatch operator to perform Att. 2, Recovery from Blackout Breaker Alignment and UV Relay Defeat, of 1202.008, Blackout.

Answer:

B. Perform rapid cooldown per 1202.008, Blackout

Notes:

B is correct, with inadequate SCM and head voids indicated, then a rapid plant cooldown is required per floating step of 1202.008.
A is incorrect but plausible since a Loss of Subcooling Margin is indicated but no power exists so 1202.008 should be in use to restore power.
C is incorrect but plausible since step 8 in RT-4 provides for continuing the RT without HPI pumps but no steps in 1202.008 direct performance of this RT.
D is incorrect but plausible since Att. 2 will be performed during a Blackout with degraded voltage indicate on SU transformers but not until after the buses are energized following performance of Att. 1 (which has a similar sounding title).

References:

1202.008, Blackout

History:

New for 2014 SRO Exam

Floating Steps

RCS Inventory/Press

- **IF** SCM is **not** adequate
  AND
  RV Head void is indicated,
  THEN perform rapid cooldown per step **54**, while continuing.

Electrical

- **IF** an EDG or AAC Gen becomes available before offsite power is restored,
  THEN GO TO step 3.

- **IF** offsite power becomes available,
  THEN verify steps **1** through **7** have been performed
  AND
  GO TO step 8.

Secondary

- **IF** EFW is lost,
  THEN perform step **2** Contingency.

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

QID: 0874 **Rev:** 0 **Rev Date:** 9/24/14 **Source:** New **Originator:** Cork

TUOI: A1LP-RO-EOP07 **Objective:** 10 **Point Value:** 1

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

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

Description: Ability to interpret control room inidcations 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:** 41.5 / 43.5 / 45.12

Tier: 1 **RO Imp:** 4.2 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 80

Given:
- Tornado touched down in the switchyard 30 minutes ago
- Emergency Diesel Generators are supplying their respective busses,
- Only P-4A Service Water Pump is running

Which Emergency Operating Procedure and action is required for the given conditions?

A. Per 1202.007, Degraded Power, Open P-4A to P-4B Crossties (CV-3644 & CV-3646)

B. Per 1203.030, Loss of Service Water, Open P-4A to P-4B Crossties (CV-3644 & CV-3646)

C. Per 1203.030, Loss of Service Water, Verify ACW Isolation (CV-3643) and BOTH SW to ICW Coolers Supply Valves closed

D. Per 1202.007, Degraded Power, Verify ACW Isolation (CV-3643) and BOTH SW to ICW Coolers Supply Valves closed

Answer:

D. Per 1202.007, Degraded Power, Verify ACW Isolation (CV-3643) and BOTH SW to ICW Coolers Supply Valves closed

Notes:

D is the correct contingency action for both EDGs running and only one SW pump running in 1202.007.
A is incorrect but a plausible action to take, however, the Crossties are normally open and need to be maintained open. 1203.030 will isolate the SW loops if only one SW pump is running since it doesn't consider a loss of offsite power condition.
B is incorrect but plausible, with only one service water pump running crosstie of the loops would supply service water to the Green train EDG but the listed procedure is not correct for the given condition.
C is incorrect but plausible, the actions are correct but the listed procedure is not correct for the given condition.

References:

1202.007, Degraded Power

History:

New for 2014 Exam

INSTRUCTIONS	CONTINGENCY ACTIONS
3. Verify SERV WTR to DG1 and DG2 CLRs open to operating EDGs: • **CV-3806** • **CV-3807**	
4. Verify a Service Water pump running on each operating DG, after 15-second time delay (P4A, P4B, P4C).	**4.** <u>IF</u> both EDGs are operating **<u>AND</u>** only one Service Water pump can be started, **<u>THEN</u>** perform the following: A. Verify ACW Isolation (CV-3643) closed. B. Verify both Service Water to ICW Coolers Supply valves closed: • CV-3811 • CV-3820
5. Actuate MSLI for both SGs **<u>AND</u>** **verify proper actuation and control of EFW and MSLI (RT-6):** A. Operate ATM Dump CNTRL valves in HAND to minimize cycling and conserve Instrument Air. B. <u>IF</u> Instrument Air to ATM Dump CNTRL valves is lost, **<u>THEN</u>** perform the following: 1) Establish SG press control using ATM Dump ISOL valves in MANUAL. 2) Dispatch an operator with a radio to place ATM Dump CNTRL valves on hand jack and fully open. (Refer to Alternate Shutdown (1203.002), Exhibit A). 3) Maintain SG press 1000 to 1040 psig using ATM Dump ISOL valves.	5. <u>IF</u> all EFW is lost, **<u>THEN</u> GO TO step 55.**

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

QID: 0756 **Rev:** 0 **Rev Date:** 11/10/2008 **Source:** Direct **Originator:** Don Slusher

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

Section: 2 **Type:** Generic APE

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

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

K/A Number: 2.2.22 **CFR Reference:** 41.5/43.2/45.2

Tier:	1	**RO Imp:**	4.0	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 81

*** REFERENCE PROVIDED ***

Given:
- RCS Temperature 450 degrees
- RCS Pressure 1600 psig

- During a battery charger swap, D03A and D03B chargers are damaged causing a loss of red train DC power.

Assuming no Operator actions, which two of the following Technical Specifications are most limiting?

A. 3.8.4, DC Sources Operating, and 3.8.7 Inverters Operating.

B. 3.8.5 DC Sources Shutdown, and 3.8.8 Inverters Shutdown.

C. 3.8.5 DC Sources Shutdown, and 3.8.10 Distribution Systems Shutdown.

D. 3.8.4, DC Sources Operating and 3.8.9 Distribution Systems Operating.

Answer:

D. 3.8.4, DC Sources Operating and 3.8.9 Distribution Systems Operating.

Notes:

A is incorrect with a 8 and 12 hour specs.
B is incorrect as the first condition does not apply.
C is incorrect as the first condition does not apply.
D is correct with two 8 hour specs.

References:

Unit 1 Technical Specifications

Must provide TS 3.8.4, 3.8.5, 3.8.7, 3.8.9, and 3.8.10 to students

History:

New for the 2009 Retake SRO Exam
Selected for 2014 SRO Exam

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Both DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	8 hours
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 124.7 V on float charge.	7 days
SR 3.8.4.2	Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Two AC, DC, and 120 VAC electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystem(s) inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One or more 120 VAC electrical power distribution subsystem(s) (RS1, RS2, RS3, RS4) inoperable.	B.1 Restore 120 VAC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
C. One or more DC electrical power distribution subsystem(s) inoperable.	C.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours
	D.2 Be in MODE 5.	36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	7 days

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

QID: 0875 **Rev:** 0 **Rev Date:** 9/2/14 **Source:** New **Originator:** Possage

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

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

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

Description: Knowledge of the specific bases for EOPs.

K/A Number: 2.4.18 **CFR Reference:** 41.10 / 43.1 / 45.13

Tier:	1	**RO Imp:**	3.3	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 82

Given
- Reactor Power 100%

In accordance with Technical Specification bases which RPS Trip Setpoint is designed to protect against a rod withdrawl accident?

A. Rx Power/Imbalance/Flow Trip

B. High Power Trip

C. High Temperature Trip

D. High Pressure Trip

Answer:

B. High Power Trip

Notes:

The RPS bases and associated setpoints are derived from the anticipated accidents, one of which is the rod withdrawl accident. The RPS setpoint for High Power Trip is described in the T.S. Bases section to mitigate the effects of the rod withdrawl accident. The EOP for Reactor Trip is designed to rely on RPS to perform its function to mitigate accidents.

Answer B is correct in accordance with T.S. bases for 3.3.1.
Answer A is incorrect but plausible since this does protect against DNB but does not protect against rapid reactivity transients.
Answer C is incorrect but plausible since this does protect against DNB but this trip is a backup to the low pressure and variable low pressure trips.
Answer D is incorrect but plausible since it does provide protection for slow reactivity transients such as a continuous rod withdrawal but does not provide protection against fast reactivity transients like a rod ejection.

References:

Technical Specifications, Bases for 3.3.1

History:

New for 2014 Exam

APPLICABLE SAFETY ANALYSES (continued)

considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1.

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel bypass. Bypass effectively places the unit in a two-out-of-three logic configuration that can still initiate a reactor trip, even with a single failure within the system.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Trip setpoints are specified in the setpoint calculations or calibration procedures. The setpoints are selected such that the setpoint measured by CHANNEL FUNCTIONAL TESTS is not expected to exceed the Allowable Value if the bistable is performing as required.

For most RPS Functions, the Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific figures for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the consequences of unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the specified deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

   a. Nuclear Overpower - High Setpoint

      The Nuclear Overpower - High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core fast neutron leakage flux.

      The Nuclear Overpower - High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to a maximum value of the design overpower.

APPLICABLE SAFETY ANALYSES (continued)

Thus, the Nuclear Overpower - High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE, also provide protection. The role of the Nuclear Overpower - High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower - High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident, the rod ejection accident, and the steam line break accident. By providing a trip during these events, the Nuclear Overpower - High Setpoint trip protects the unit from excessive power levels and also serves to reduce reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

The specified Allowable Value is selected to initiate a trip at or before reactor power exceeds the highest point at which the RCS Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trips are analyzed to provide protection against DNB and fuel centerline melt. The Allowable Value does not account for harsh environment induced errors, because the trip will actuate prior to degraded environmental conditions being reached.

b. Nuclear Overpower - Low Setpoint

While in shutdown bypass, the Nuclear Overpower - Low Setpoint is instated with a trip setpoint of ≤ 5% RTP. The low power setpoint, in conjunction with the Shutdown Bypass RCS High Pressure setpoint, protect the unit from excessive power conditions when other RPS trips are bypassed.

The Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. Reactor Outlet High Temperature

The Reactor Outlet High Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor outlet temperature approaches the conditions necessary for DNB. Portions of each Reactor

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

QID: 0876 **Rev:** 0 **Rev Date:** 9/2/14 **Source:** New **Originator:** Possage

TUOI: A1LP-RO-EOP06 **Objective:** 14 **Point Value:** 1

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

System Number: 037 **System Title:** Steam Generator (S/G) Tube Leak

Description: Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: S/G tube failure.

K/A Number: AA2.06 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	4.3	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.5	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 83

Given:

- Reactor was tripped due to an 'A' SG Tube Leak
- 'A' SG level is 390 inches and rising at a rate of 6"/min
- 'B' SG is 30 inches and stable

What procedural required action should be taken based on the given conditions?

A. Per 1202.001, Reactor Trip, trip both MFW Pumps

B. Per 1202.006, Tube Rupture, trip both MFW Pumps

C. Per 1202.006, Tube Rupture, establish RCS cooldown at ≤ 100°F/hr

D. Per 1202.006, Tube Rupture, establish RCS cooldown at > 100°F/hr but ≤ 240°F/hr

Answer:

D. Per 1202.006, Tube Rupture, establish RCS cooldown at > 100°F/hr but ≤ 240°F/hr

Notes:

Answer D is correct per 1202.006 Floating Step page and step 26.
Answer A is incorrect but plausible since water in steam lines could damage MFW pumps and this action does exist in 1202.001 but the reactor trip EOP is not used in a tube rupture situation.
Answer B is incorrect, the MFW pumps are tripped in a later step but only after the AFW pump is placed in service and not due to high affected SG level. The action in this case is to close the affected MSIV.
Answer C is incorrect but plausible as this is the normal cooldown rate in Step 27 of 1202.006, i.e., if the SG level was NOT approaching 410" due to a tube rupture.

References:

1202.006, Tube Rupture

History:

New for 2014 exam.

Floating Steps

RCS Temp

- **<u>WHEN</u>** RCS T-hot is < 500°F,
  <u>THEN</u> 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

- **<u>IF</u>** RCS T-hot is < 490°F **<u>AND</u>** any of the following occur:

  - Bad SG Level approaches 410".
  - BWST Level reaches 23'.
  - Projected activity at the site boundary reaches Alert criteria.

  <u>THEN</u> perform **step 45.**

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

SG A	SG B
P32D	P32B

ESAS

- **<u>IF</u>** ESAS actuates,
  <u>THEN</u> perform **step 34.**

Secondary

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

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

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

SF Pool Cooling

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

INSTRUCTIONS | **CONTINGENCY ACTIONS**

26. IF bad SG level is approaching 410" due to leakage
OR
dose rate ≥ Alert criteria is projected at Site boundary,
THEN establish emergency cooldown rate of ≤ 240°F/hr (≤ 4°F/min) to 500°F T-hot as follows:

A. For good SG, place TURB BYP Valves in HAND
AND
adjust to maintain cooldown rate ≤ 240°F/hr.

Contingency: A. IF TURB BYP Valves are **not** available, THEN operate ATM Dump Control System for good SG in HAND to maintain cooldown rate ≤ 240°F/hr.

SG A		SG B
CV-2676	ATM DUMP ISOL	CV-2619
CV-2668	ATM DUMP CNTRL	CV-2618

1) IF both SGs are bad, THEN steam both SGs.

B. IF RCS press drops below 1700 psig
AND SCM is adequate
AND RCS press is controlled,
THEN bypass ESAS.

C. IF only one SG is bad,
THEN steam **bad** SG only as necessary to maintain Exhibit 1 limits.

Contingency: C. IF both SGs are bad, THEN steam both SGs as necessary to maintain Exhibit 1 limits.

INSTRUCTIONS	CONTINGENCY ACTIONS

27. IF emergency cooldown rate is not required
OR
RCS T-hot is ≤ 500°F,
THEN establish RCS cooldown rate of ≤ 100°F/hr as follows:

A. For good SG, place TURB BYP Valves in HAND
AND
adjust to maintain cooldown rate ≤ 100°F/hr.

Contingency: A. IF TURB BYP Valves are not available, THEN operate ATM Dump Control System for good SG in HAND to maintain cooldown rate ≤ 100°F/hr.

SG A		SG B
CV-2676	ATM DUMP ISOL	CV-2619
CV-2668	ATM DUMP CNTRL	CV-2618

1) IF both SGs are bad, THEN steam both SGs.

B. IF RCS press drops below 1700 psig
AND SCM is adequate
AND RCS press is controlled,
THEN bypass ESAS.

C. IF only one SG is bad,
THEN steam **bad** SG only as necessary to maintain Exhibit 1 limits.

Contingency: C. IF both SGs are bad, THEN steam both SGs as necessary to maintain Exhibit 1 limits.

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

QID: 0188 **Rev:** 1 **Rev Date:** 9/10/14 **Source:** Direct **Originator:** Cork

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

Section: 4.2 **Type:** Generic AOP

System Number: 036 **System Title:** Fuel Handling Incidents

Description: Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: ARM system indications.

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

Tier: 1 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 84

Given:

- Core reload in progress in the Reactor Building.
- A fuel assembly is in the Main Bridge mast and in transit to the reactor core
- "RADIATION MONITOR TROUBLE" K10-C1 alarms
- CBOT reports Reactor Building Fuel Handling Area (RI-8017) area radiation monitor has a Failure alarm in solid.

What procedurally reqiured actions should be taken by the refueling team?

A. Return assembly to the Spent Fuel Pool and then secure refueling activities until repairs are made to RI-8017.

B. Continue refueling activities and notify Radiation Protection to perform area surveys of the Main Bridge every 15 minutes.

C. Continue refueling activities as long as two subcritical core neutron flux monitors are available.

D. Stop fuel movement into the reactor core until RI-8017 is operable or a suitable portable survey instrument is obtained.

Answer:

D. Stop fuel movement into the reactor core until RI-8017 is operable or a suitable portable survey instrument is obtained.

Notes:

D is the correct answer.per 1203.012I and the TRM.
A is incorrect. In the event of a severe weather condition this action might be taken, but for RI-8017 failure the fuel movement should stop until the rad monitor is restored or a suitable instrument is in place.
B is incorrect. Refueling activities can continue in some circumstances whiile repairs are being made but they should be stopped and periodic surveys does not satisfy the TRM.
C is incorrect. Refueling activities can continue in some circumstances whiile repairs are being made and the two neutron flux monitors are necessary to perform refueling activities but RI-8017 must also be operable or a suitable replacement must be in place.

References:

Technical Requirments Manual 3.9.1
1203.012I, Annunciator K10 Corrective Action

History:

Developed for use in 98 RO Re-exam KA 036 AK2.02

Revised and selected for 2014 SRO Exam.

Location: C16

Device and Setpoint:
De-energization of or FAILURE ALARM on any radiation monitor in Radiation Monitoring System Panel (C25 Bays 1-3 and Bay 4 of C24). Monitors are listed on page 3.

RADIATION
MONITOR
TROUBLE

Alarm: K10-C1

1.0 OPERATOR ACTIONS

1. Observe monitors at C24 and C25 for FAILURE ALARM light(s) on or POWER ON light(s) off.

2. IF power is off to all monitors in a bay,
THEN check supply breaker closed:

- Rad Monitor Panel C24, Rad Monitor Panel C25, Bay 1 (RS1, bkr 8)
- Rad Monitor Panel C24, Rad Monitor Panel C25, Bay 2 (RS2, bkr 8)
- Rad Monitor Panel C25, Bay 3 (RS4, bkr 8)

A. IF breaker is tripped,
THEN reclose tripped breaker per "Reclosing Tripped Individual Load Supply Breakers" section of Electrical System Operations (1107.001).

3. IF either of the following monitors is inoperable (FAILURE ALARM or power loss):

- Spent Fuel Pool (RI-8009)
- Fuel Handling Area (RI-8017)

AND fuel handling in progress,
THEN stop fuel handling until radiation monitoring requirement is satisfied per Control of Unit 1 Refueling (1502.004) OR Control of Fuel and Control Rod Movement in the U-1 Spent Fuel Area (1502.010).
(TRM 3.9.1 and TRM 3.9.2)

4. IF Liquid Radwaste (RI-4642) is de-energized,
THEN verify CZ Disch to Flume Flow (CV-4642) is closed or auto closes.
(ODCM L2.1.1)

TRM 3.9 REFUELING OPERATIONS

TRM 3.9.1 Fuel Handling - Reactor Building

TRO 3.9.1 Radiation levels shall be monitored by RE-8017.

APPLICABILITY: During movement of fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RE-8017 inoperable.	A.1 Monitor area with portable survey instrument of appropriate range and sensitivity.	Immediately
B. Required Action and associated Completion Time not met.	B.1 Cease movement of fuel into reactor core	Immediately
	<u>AND</u>	
	B.2 Cease activities that might increase the reactivity of the core.	Immediately

TEST REQUIREMENTS

TEST	FREQUENCY
None.	

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

QID: 0188 **Rev:** 0 **Rev Date:** 11/18/98 **Source:** Direct **Originator:** L. Kilby

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

Section: 4.2 **Type:** Generic AOP

System Number: 036 **System Title:** Fuel Handling Incidents

Description: Knowledge of the interrelations between the Fuel Handling Incidents and the following: Radiation monitoring equipment (portable and installed).

K/A Number: AK2.02 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** C

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

Given:

- Fuel handling is in progress in the Reactor Building.
- A fuel assembly is in the Main Bridge mast.
- Reactor Building Fuel Handling Area (RI-8017) area radiation monitor has a Failure alarm in solid.

What actions should be taken by the refueling team?

a. Place the assembly in the nearest core location and then secure refueling activities until repairs are made to RI-8017.

b. Continue refueling activities and notify Health Physics to perform area surveys of the Main Bridge every 15 minutes.

c. Continue refueling activities as long as two subcritical core neutron flux monitors are available.

d. Secure fuel handling activities until RI-8017 is operable or a suitable portable survey instrument is obtained.

Answer:

d. Secure fuel handling activities until RI-8017 is operable or a suitable portable survey instrument is obtained.

Notes:

(a.) is incorrect. In the event of a severe weather condition this action would be taken, but for RI-8017 failure the fuel movement should stop until it is restored or a suitable instrument is in place.
(b.) is incorrect. Refueling activities should be stopped and periodic surveys does not satisfy tech specs.
(c.) is incorrect. These are necessary to perform refueling activities but RI-8017 must also be operable or a suitable replacement must be in place.
(d.) is the correct answer.

References:

TRM 3.9.1

History:

Developed for use in 98 RO Re-exam
Selected for 2005 RO exam but not used.

ORIGINAL

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

QID: 1027 **Rev:** 0 **Rev Date:** 10/1/14 **Source:** Modified **Originator:** Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 067 **System Title:** Plant Fire On Site

Description: Ability to determine and interpret the following as they apply to a plant fire on site: systems that may be affected by the fire.

K/A Number: AA2.17 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	3.5	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.3	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 85

A fire in the Cable Spreading Room has occurred.

Which procedure is required for this occurrence AND which monitoring system has indications that will NOT be affected by this fire?

A. 1203.002, Alternate Shutdown
Dasey Panel

B. 1203.029, Remote Shutdown
Dasey Panel

C. 1203.002, Alternate Shutdown
Safety Parameter Display System

D. 1203.029, Remote Shutdown
Safety Parameter Display System

Answer:

C. 1203.002, Alternate Shutdown
Safety Parameter Display System

Notes:

C is correct, 1203.002 should be used for a fire that forces control room evacuation and SPDS contains displays with instruments routed outside of the cable spreading room.
A is incorrect, this is the right procedure but the Dasey Panel has indications for use in a remote shutdown situation.
B is incorrect, this is the wrong procedure and the Dasey Panel has indications for use in a remote shutdown situation.
D is incorrect, this is the wrong procedure but the right source of non-affected indications.

References:

1203.002, Alternate Shutdown, Att. 9 step 2.5

History:

Modified QID 0478 for 2014 SRO Exam

The vital 480 volt buses (B5 and B6) will be de-energized to prevent spurious actuations. The vital 480 volt load centers will later be re-energized after certain breakers have been opened to prevent undesirable spurious actuations. These buses are required to support extended diesel generator operations, HPI and to power the battery chargers.

2.3 Service Water

One service water pump per bus (A3, A4) is verified operating, then breaker control power is de-energized to prevent spurious shutdown. The valves and sluice gates required to pressurize both loops are de-energized in their pre-fire alignment to prevent spurious misalignment. Subsequently, the positions of these valves and sluice gates will be verified locally. Service water pressure is verified locally to assure adequate component cooling. Service water is required to support sustained diesel generator, HPI pump and possibly EFW system operations.

2.4 High Pressure Injection

Initially, the "B" train (Green-powered) of HPI will be manually aligned with the BWST suction valve open and two HPI valves full open. HPI capability will be controlled by the CRS locally starting/stopping P-64C(B) and closing/opening the "C" (or "B" if its MOD to A4 is closed) HPI pump breaker.

As time permits, the redundant train of HPI will be aligned and verified for backup purposes. If necessary, the remaining HPI valves may be opened.

2.5 Instrumentation

Instrument strings totally independent of the Control Room and Cable Spread Room have been provided via the SPDS Alternate Shutdown display (A/S-G). The power source and cable routing of these instrument strings will assure their reliability for Control Room/Cable Spread Room fires. All other instrumentation should be considered unreliable. See Attachment 10 for available instrumentation and SPDS points.

These instrument strings provide the following parameters via the SPDS Alternate Shutdown display (A/S-G):

A) SG-A and SG-B levels
B) SG-A and SG-B pressures
C) SG-A and SG-B Loop T-cold temperatures (wide range)
D) SG-A and SG-B Loop T-hot temperatures (wide range)

E) Pressurizer level (compensated)
F) Neutron flux (source range)
G) RCS pressure (wide range)
H) EFW CST (T-41B) level

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

QID: 0478 **Rev:** 0 **Rev Date:** 10/6/2003 **Source:** Direct **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 067 **System Title:** Plant Fire On Site

Description: Ability to determine and interpret the following as they apply to a plant fire on site: systems that may be affected by the fire.

K/A Number: AA2.17 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 2.5

Group: 2 **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** C

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

A fire in the Cable Spreading Room has occurred.
Alternate Shutdown, 1203.002, is in progress.

Which monitoring system has indications that will NOT be affected by this fire?

A. Plant Monitoring System

B. Plant Data Server

C. Safety Parameter Display System

D. Dasey Panel

Answer:

C. Safety Parameter Display System

Notes:

Only answer "c" has instrumentation that will not be afffected by a fire in the cable spreading room, all of the other choices contain cables that are routed through this area.

References:

1203.002, Alternate Shutdown, Att. 9 step 2.5

History:

New
Used on 2004 SRO Exam
Determined in 2014 to be non-SRO since it does not meet any of the 55.43 criteria

PARENT

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

QID: 0607 **Rev:** 1 **Rev Date:** 8/4/2005 **Source:** Direct **Originator:** J.Cork

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

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

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

Description: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A Number: 2.1.23 **CFR Reference:** 41.10 / 43.5 / 45.2 / 45.6

Tier: 2 **RO Imp:** 4.3 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 86

Given:

- P-36A is the in-service Makeup Pump.
- Pressurizer level has dropped from 220" to 218" in 10 minutes.

- P-36A suction pressure is 40 psig and going down slowly.
- Makeup Tank level is 78" and trending down slowly.

- Seal Injection flow is oscillating from 38 to 43 gpm.
- MU-34D HPI temperature TE-1069A is reading 255°F.
- Aux. Building sump level is going up.

Considering the above conditions, which procedure will direct the Makeup Pump to be secured?

A. 1203.039, Excess RCS Leakage

B. 1203.026, Loss of Reactor Coolant Makeup,
Section 1 - Loss of HPI Pump

C. 1203.026, Loss of Reactor Coolant Makeup,
Section 2 - Large Makeup and Purification System Leak

D. 1203.032, HPI Line Temperature High

Answer:

C. 1203.026, Loss of Reactor Coolant Makeup, Section 2 - Large Makeup and Purification System Leak

Notes:

Answer "C" is correct, indications are for a leak on the discharge of the Makeup Pump, Section 2 of 1203.026 contains actions to secure the pump.
Answer "A" is incorrect, this procedure is entered due to leakage indicated but direction to secure the pump is in 1203.026.
Answer "B" is incorrect, this is the correct procedure but this section secures the pump on a loss of suction which is not indicated by the conditions given.
Answer "D" is incorrect, although an HPI line temperature indicator is >200°F, there are no actions in this procedure for securing the pump.

References:

1203.026, Loss of Reactor Coolant Makeup

History:

New for 2005 SRO exam. System 006 KA A2.03
Selected for the 2008 SRO Exam.
Selected for 2011 SRO Exam
Selected for 2014 SRO Exam.

SECTION 2 -- LARGE MAKEUP AND PURIFICATION SYSTEM LEAK

INSTRUCTIONS

NOTE

Indications of loss of HPI suction are:

- Erratic flow
- Erratic discharge pressure
- Control valves stable

1. **IF HPI pump has lost suction, THEN stop the HPI pump.**

2. **IF ANY of the following:**

   - Leakage > normal makeup capacity (50 GPM)
   - CRS/SM determine it is desired
   - HPI pump is stopped

   THEN 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)

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

QID: 0466 **Rev:** 0 **Rev Date:** 7/19/2002 **Source:** Direct **Originator:** S.Pullin

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

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

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

Description: Knowledge of abnormal condition procedures.

K/A Number: 2.4.11 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 2 **RO Imp:** 4.0 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 87

Given:

- Plant is at 100%
- CBOR reports Makeup Tank level is unexpectedly trending down
- Annunciator K10-B2 "PROC MONITOR RADIATION HI" is in alarm
- Annunciator K08-C7 "RCP BLEEDOFF TEMP HI" subsequently alarms
- SE reports the process monitor is for Nuclear ICW

Which of the following contains the specific guidance for mitigation of this event?

A. 1203.012I, Annunciator K10 Corrective Action, K10-B2 "PROC MONITOR RADIATION HI"

B. 1203.039, Excess RCS Leakage

C. 1203.012G, Annunciator K08 Corrective Action, K08-C7 "RCP BLEEDOFF TEMP HI"

D. 1203.031, Reactor Coolant Pump and Motor Emergency

Answer:

D. 1203.039, Excess RCS Leakage

Notes:

Answer (b) is the only answer with the actions to mitigate an Inter System LOCA.
All of the remaining choices are related but do not contain mitigating actions.

References:

1203.039, Excess RCS Leakage

History:

Created for 2002 SRO exam. KA 008 A2.04
Selected for 2014 SRO 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**.

NOTE

Pressurizer Sample Cooler is known to have a small leak (CR-ANO-1-2012-0774). If a PZR sample is aligned and there is rising Nuclear Loop ICW activity, the leak may have deteriorated.

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

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: 0350 **Rev:** 1 **Rev Date:** 9/18/14 **Source:** Direct **Originator:** S.PULLIN

TUOI: ANO-1-LP-RO-ESAS **Objective:** 6 **Point Value:** 1

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of AC control power.

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

Tier:	2	**RO Imp:**	3.7	**RO Select:**	No	**Difficulty:**	3.5
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 88

Given: the plant is operating at 100% power.

ESAS Analog 2 RC pressure transmitter fails LOW due to loss of instrument power.

What action below will allow compliance with Technical Specifications and allow continued plant operation at 100% power?

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

B. Continued power operation is not allowed, plant shutdown is required.

C. Immediately trip one of the two remaining operable channels.

D. Test ES components associated with Analog Channel 2 within 24 hours.

Answer:

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

Notes:

A- is correct per Tech Spec 3.3.5. The impact of a loss of power is that the transmitter fails low which results in an analog channel trip since the channel is already in the required state for the TS required action only administrative controls are necessary.
B- is incorrect, with only one inoperable channel, plant shutdown is not required.
C- is incorrect, this action would result in ES actuation.
D- is incorrect, testing of ES components is not required.

55.43(2)

References:

Tech. Spec 3.3.5

History:

Developed for 2001 SRO exam
Revised for 2014 SRO Exam

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 --------------NOTE-------------- Only required for Reactor Building Pressure High setpoint and High High setpoint. ------------------------------------	
	Be in MODE 5.	36 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

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

QID: 0350 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Direct **Originator:** S.PULLIN

TUOI: ANO-1-LP-RO-ESAS **Objective:** 6 **Point Value:** 1

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based ability on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument bus.

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

Tier: 2 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 3.5

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

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

Given, the plant is operating at 100% power.

ESAS Analog 2 RC pressure transmitter fails LOW due to loss of instrument power.

What operator action will allow continued plant operation at 100% power?

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

b. Continued power operation is not allowed, plant shutdown is required.

c. Immediately trip one of the two remaining operable channels.

d. Test ES components associated with Analog Channel 2 within 24 hours.

Answer:

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

Notes:

"a" is correct per Tech Spec table 3.5.1-1, Note 6.
"b" is incorrect, with only one inoperable channel, plant shutdown is not required.
"c" is incorrect, this action would result in ES actuation.
"d" is incorrect, testing of ES components is not required.

References:

Tech. Spec 3.5.1-1 table note 6

History:

Developed for 2001 SRO exam

ORIGINAL

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

QID: 0932 **Rev:** 0 **Rev Date:** 9/18/14 **Source:** New **Originator:** Cork/Possage

TUOI: A1LP-RO-EFIC **Objective:** 43 **Point Value:** 1

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

System Number: 061 **System Title:** Auxiliary / Emergency Feedwater (AFW) System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: pump failure or improper operation.

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

Tier:	2	**RO Imp:**	3.4	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** **SRO:** 89

Given:
- Unit 1 is at 100% power
- It is a GREEN train maintenance week

Two MOV (Motor Operated Valve) PMs are on the schedule for the Turbine Driven EFW Pump P-7A.

Which of the following GREEN train maintenance activities can you APPROVE for work and which action will maintain operability of P-7A?

A. PM on Steam Admission Valve CV-2613,
Open D-2512 and lock CV-2613 in the OPEN position

B. PM on Steam Admission Valve CV-2613,
Open D-2512 and lock CV-2613 in the CLOSED position

C. PM on Steam Supply Valve CV-2617,
Open B-6241 and lock CV-2617 in the OPEN position

D. PM on Steam Supply Valve CV-2617,
Open B-6241 and lock CV-2617 in the CLOSED position

Answer:

C. PM on Steam Supply Valve CV-2617,
Open B-6241 and lock CV-2617 in the OPEN position

Notes:

The applicant has to determine the impact of each of the proposed activities and realize that maintenance on CV-2613 would render P-7A inoperable and therefore would be an improper activity to allow. Then they must determine how to control the position of CV-2617 which is prescribed in the operability section of 1106.006.

C is correct per section 18.0, Operability, of 1106.006.
A and B are plausible but incorrect, CV-2613 is green train DC powered and is required for operability of P-7A.
D is plausible but incorrect, while locking CV-2617 in the closed position sounds plausible for SGTR concerns, it is not the action to take for steam line break concerns.

References:

1106.006, Emergency Feedwater Pump Operation

History:

New for 2014 SRO Exam

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.

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.

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

QID: 0641 **Rev:** 0 **Rev Date:** 10/19/06 **Source:** Direct **Originator:** Cork/Possage

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

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

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

Description: Knowledge of limiting conditions for operation and safety limits.

K/A Number: 2.2.22 **CFR Reference:** 41.5 / 43.2 / 45.2

Tier: 2 **RO Imp:** 4.0 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 90

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

Given:

- Plant is at 100% ambient temperature ~55°F
- SW Pump P-4C is out-of-service for scheduled motor replacement
- SW Pump P-4B is in service on Loop II

The Outside AO notifies the Control Room that the gantry crane operator has pulled the center roof plug for the Intake Structure.

Which of the following actions would you recommend to the Shift Manager? (Reference Provided)

A. Restore Loop II SW to operable status within 72 hours.

B. Unit must be placed in Mode 3 within 6 hours.

C. Unit must be placed in Mode 3 within 7 hours per Tech Spec 3.0.3.

D. No Tech Spec action required as long as ambient temperature remains <70°F.

Answer:

A. Restore Loop II SW to operable status within 72 hours.

Notes:

Answer "A" is correct. With "C" pump oos and with the "B" (center) roof plug pulled, 1104.029, section 12 states that the "B" pump should be declared inoperable. TS 3.7.7.A is therefore in effect.
Answer "B" is incorrect, this would apply only after the 72 hours has expired.
Answer "C" is incorrect, 3.0.3 would only be applicable if both trains were inoperable.
Answer "D" is incorrect, although roof plugs are mentioned in the ventilation operability section, the ambient temperature does not affect the inoperability due to the position of the roof plugs.

References:

1104.029, Service Water and Auxiliary Cooling Water, Section 36.0 Operability
Technical Specification 3.7.7

NOTE: TS 3.7.7 reference must be in the SRO exam handout

History:

New for 2007 SRO exam
Selected for 2014 SRO Exam

3.7 PLANT SYSTEMS

3.7.7 Service Water System (SWS)

LCO 3.7.7 Two SWS loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS loop inoperable.	A.1 --------------NOTES------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by SWS. 2. Enter Applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal made inoperable by SWS. -------------------------------------- Restore SWS loop to OPERABLE status.	 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

36.0 Operability

36.1 Discussion -- This section aids in determining system operability so that consistency will be maintained. This is NOT a listing of all requirements necessary for system operability.

36.2 Testing Inoperable SW Pump On Operable SW Loop

If an inoperable SW pump is tested on an operable SW loop in Modes 1-4, then the SW loop must be declared inoperable and step 36.8 of this procedure performed. CR-ANO-1-2012-1456

36.3 Running Uncoupled SW Pump Motor

If an uncoupled SW pump motor is run on the same bus as an operating SW pump in Modes 1-4, then the SW loop must be declared inoperable and step 36.8 of this procedure performed. CR-ANO-1-2012-1456

36.4 Intake Structure Roof Plugs

Removal of an intake structure roof plug to access the SW pump motor renders the associated SW pump inoperable while the plug is removed.

36.5 Intake Structure Ventilation

36.5.1 Degraded Power Conditions

During degraded power conditions, normal intake structure ventilation is de-energized. Adequate SW pump motor cooling is dependent on natural circulation air flow from the lower level of the intake structure out through the upper level roof plugs. For this reason, the roof plugs must remain in the elevated position at all times. Door #172, on north end of intake structure, is normally open during summer months to provide more ventilation and may be opened at any time if a high temperature condition occurs. Having Door #172 open is not required for operability.

36.5.2 Post Accident Cooling Design Basis

If outside ambient temperature is ≤80°F during normal operations with no more than 2 SW pumps operating continuously, hatches may be installed AND operation may proceed without time limit AND without additional operator actions. If a DBA accident or a fire occurs, OR if fans VEF-25 AND VEF-32 become inoperable, AND outside ambient temperature remains ≤80°F AND no more than 2 SW pumps are operating continuously, no additional operator action is required.

If outside ambient temperature increases to >80°F during normal operations, with no more than 2 SW pumps operating continuously, OR following a DBA or fire event, OR if fans VEF-25 and VEF-32 become inoperable, then door hatches (HTC-70 and HTC-71) must be removed and locked before outside ambient temperature exceeds 85°F in order for the SW pumps to remain operable. This results in the following limits based on outside ambient temperature:

- The maximum temperature limit is 80°F.
- The operability limit is 85°F.

Continuous concurrent operation of three SW pumps with no forced ventilation and door hatches (HTC-70, HTC-71) installed is beyond the scope of and not permitted by evaluation for CR-ANO-C-2004-1848 Operability Version 2. Transient operation of three SW pumps, such as during pump swap, is acceptable.

36.6 SW Pump Strainer Differential Pressure

A SW Pump strainer ΔP of 10 psid leaves a minimum margin of operability for SW flow to various components. If SW Pump strainer ΔP exceeds 10 psid, System Engineering must be contacted to evaluate SW system/component operability.

36.7 SW Pump Packing Gland Leakoff

Packing gland leakoff is an important feature for cooling and lubricating the pump shaft. If the pump is run for extended periods without packing gland leakoff, shaft damage can result. Therefore, if a pump is found running without packing gland leakoff, a condition report shall be initiated documenting the event so that pump operability will be assessed. Furthermore, a WR/WO shall be initiated to inspect the pump shaft at the earliest opportunity.

36.8 Inoperable SW Loop

If a Service Water Loop is declared inoperable, then verify the LCOs Conditions and Completion Times entered for the following:

- Tech Spec 3.7.7 Condition A 72 hours (One SW Loop inoperable)
- Tech Spec 3.8.1 Condition B [See Tech Spec] (One DG inoperable)
  - Complete 1107.001, Sup 10, Verification of Two Offsite Circuit Power Sources (within 1 hour).
  - Verify Unit 2 has taken the appropriate actions as identified in Attachment B (Table 3) of Control Room Emergency Air Conditioning and Ventilation (2104.007).
- Enter TS 3.0.6 and apply to the following Tech Specs LCOs (entry into TS LCO Condition and Completion Time NOT required per TS 3.0.6):
  - Tech Spec 3.5.2 (One ECCS Train inoperable)
  - Tech Spec 3.6.5 (One RB Spray Train and RB Clng Train inop)
  - Tech Spec 3.6.6 (Spray Additive Sys inop)
  - Tech Spec 3.7.5 (One EFW Train inoperable)
  - Tech Spec 3.8.4 (One DC power subsystem inoperable)
  - Tech Spec 3.8.7 (One and Two Inverters inoperable)
  - Tech Spec 3.8.9 (One AC power subsystem, one 120 VAC electrical power distribution and one DC power subsystem inoperable)
- Make station log entry stating similar to:

  "Entered TS 3.0.6 and declared LCO 3.5.2, 3.6.5, 3.6.6, 3.7.5, 3.8.4, 3.8.7 and 3.8.9 NOT met."
- Without delay, perform Unit 1 Safety Function Determination Program (1015.045), Attachment 2.
- Enter TRO 3.5.1 Condition A (Makeup and Chem Addition System)
- <u>IF</u> predicted to be in TRO 3.5.1 Condition A for 24 hours,
  <u>THEN</u> also perform TRO 3.5.1 Condition B.

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

QID: 0744 **Rev:** 2 **Rev Date:** 6/3/2008 **Source:** Direct **Originator:** David Thompson

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

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation System (NNIS)

Description: Ability to (a) predict the impacts of the following malfunction or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector Failure.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier: 2 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 91

Given:

Plant power 100%
TE-1014 "A" Loop Narrow Range T-hot INSTANTLY fails low

What procedure should be used for this condition?

A.1203.001, "ICS Abnormal Operation"

B. 1203.012F, "SASS Mismatch" ACA

C. 1202.001, "Reactor Trip"

D. 1105.006, "Reactor Coolant System NNI"

Answer:

B. 1203.012F, "SASS Mismatch" ACA

Notes:

Applicant has to know that Narow Range T-hot is a SASS protected instrument so that a failure would cause a SASS Mismatch alarm, then based on the sudden failure realize that the only required corrective actions are contained in the ACA.

A is incorrect. The fast failure transfer of SASS precludes abnormal ICS operation.
B is correct and contains corrective actions for this event.
C is incorrect The fast failure transfer of SASS precludes impact. A different SASS failure would result in Reactor Trip.
D is incorrect. Although this is an NNI system, only recovery actions exist for this instrument.

References:

1203.012F, Annunciator K07 Corrective Action
STM 1-69, Non-Nuclear Instrumentation System

History:

New for the 2008 SRO Exam.
Selected for the 2009 SRO Exam Retake
Selected for 2014 SRO Exam

Location: C13

Device and Setpoint:

SASS
MISMATCH

Alarm: K07-B4

1.0 OPERATOR ACTIONS

1. Determine the cause of the alarm as follows:

NOTE

SASS ENABLE indicating light is not lit when:

- Channel mismatch or signal transfer occur.
- SASS is not in AUTO.
- Loss of AC or DC power to SASS.

A. Observe SASS indicating lights on C03, C04, and C13 and determine if a transfer from the preferred source (X) has occurred.

1. IF SASS MISMATCH due to controlling RCS Pressure Selector, THEN perform the following:

a. IF desired AND an automatic transfer has NOT occurred, THEN perform the following to select opposite transmitter:

1) Check that other indications of RCS pressure (e.g. ICCMDS, WR RCS pressure) indicate similar to signal to be selected.

2) IF RCS pressure signal transfer will NOT cause a pressure transient,
THEN select desired signal.

B. Observe the non-selected input on the plant computer for mismatch indication.

C. IF desired,
THEN monitor SASS modules in C47-2 for mismatch indications.

2. IF a transfer has occurred,
THEN place the selector switch in the position of the automatically selected input signal.

and 2 must be designated by special indication in the control room. This indication will provide qualified reliable instrumentation to base control room decisions during an accident situation.

Table 69.3 contains the Reg. Guide 1.97 instruments with their classifications for this system. Under each listing is the basis for the instrument and the indication range covered.

3.0 DETAILED SYSTEM DESCRIPTION

3.1 Smart Automatic Signal Selection System

Industry events have shown that the majority of plant transients caused by the Integrated Control System were caused by input signal failures. The Smart Automatic Signal Selection System auctions the ICS input signals, which reduces transients caused by input failures. In addition to installing the Smart Automatic Signal Selection System, DCP-87-1041 modifies the ICS and NNI systems to make them less vulnerable to power failures.

The Smart Automatic Signal Selection System (SASS) transfers to a redundant input when a rapid signal failure is detected. Each of the following parameters has signal selection by SASS.

- RCS pressure
- RCS T_c
- RCS T_h.
- RCS flow
- OTSG pressure
- Header pressure
- Main Feedwater Flow
- Main Feedwater Temperature
- OTSG S/U level
- OTSG Operate Level
- Reactor Power

The SASS system is located in the NNIX cabinets. The SASS system is composed of two rows of modules. Each row of modules consists of:

- A mismatch alarm bypass module
- Three SASS modules
- One power supply module

Each SASS module can monitor up to eight signals (two signals for each parameter). Normally, all signals are selected to the NNIX channel. If a failure of the NNIX signal occurs, the SASS system will cause the NNIY signal to be selected.

The exception to this is Tc, which can be selected from the average to the NNIY or to the NNIX signal.

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

QID: 0855 **Rev:** 0 **Rev Date:** 7/31/2011 **Source:** Direct **Originator:** Cork

TUOI: A1LP-RO-EOP **Objective:** 2 **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 (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam valve stuck open.

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

Tier: 2 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 92

Following a reactor trip, the following conditions exist:

- Both SG pressures are LOWERING
- A SG pressure 971 psig
- B SG pressure 940 psig
- K07-C5 "MSSV OPEN" is in alarm
- SPDS indicates that the open MSSV is on the "B" SG.

Which of the following procedural actions would be used in response to the above conditions?

A. Actuate EFW and MSLI for "B" SG, verify proper actuation and control using RT-6, per 1202.001, Reactor Trip.

B. Control Turbine Bypass valves (TBVs) to quickly reduce SG pressures to reseat the MSSV per 1202.003, Overcooling.

C. Actuate EFW and MSLI for "B" SG, verify proper actuation and control using RT-6, per 1202.003, Overcooling.

D. Control Turbine Bypass valves (TBVs) to quickly reduce SG pressures to reseat the MSSV per 1202.001, Reactor Trip.

Answer:

D. Control Turbine Bypass valves (TBVs) to quickly reduce SG pressures to reseat the MSSV per 1202.001, Reactor Trip.

Notes:

"D" is correct, if both SG pressures are >900 psig, then the Reactor Trip EOP would be in use and this action performed as a contingency to step 15.
"A" is incorrect, this action would be proper if a steam leak were downstream of the MSIVs, and is the incorrect procedure.
"B" is incorrect, this is the correct action but Overcooling would not be transitioned to per step 15 of 1202.001.
"C" is incorrect, this is an incorrect action and incorrect procedure.

References:

1202.001, Reactor Trip

History:

New question created for 2011 SRO Exam. KA E10 EA2.2
Selected for 2014 SRO Exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
15. Check MSSV OPEN (K07-C5) alarm clear.	15. **<u>IF</u>** stuck open MSSV has been validated, **<u>THEN</u>** perform the following: A. Attempt to reseat MSSV by quickly reducing associated SG press using TURB BYP or ATM Dump Control System in HAND until: • MSSV OPEN (K07-C5) alarm clears **<u>OR</u>** • 700 psig SG press is reached 1) Operate TURB BYP or ATM Dump Control System valves in AUTO or HAND as necessary to restore SG press control. 2) **<u>IF</u>** SG fails to repressurize ≥ 900 psig, **<u>THEN</u> GO TO 1202.003, "OVERCOOLING"** procedure. B. Initiate corrective actions to gag MSSV, while continuing with this procedure.

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

QID: 0600 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** J.Cork

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

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

System Number: 034 **System Title:** Fuel Handling Equipment

Description: Knowledge of refueling administrative requirements.

K/A Number: 2.1.40 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	2	**RO Imp:**	2.8	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 93

Given:

- Plant is in a Refueling outage.
- Core re-load is in progress.
- Approximately 90% of the core is in the Reactor vessel.

The Main Fuel Handling Bridge has a once-burned fuel assembly and is in the process of indexing over the specified core location when NI-502 fails to 0.1 cps.

What action should be taken?

A. No action necessary because with NI-501 operating, Tech Spec NI requirements for operablility are met.

B. Contact the Main Fuel Bridge operator and place the assembly in a core location without any adjacent fuel assemblies.

C. Halt operations on the Main Fuel Bridge. Core geometry cannot be changed unless two neutron flux monitors are operable.

D. Verify boron concentration in the Refueling Canal is greater than 2326 ppm and then continue fuel load.

Answer:

C. Halt operations on the Main Fuel Bridge. Core geometry cannot be changed unless two neutron flux monitors are operable.

Notes:

Answer "C" is correct per 1502.004, 5.3, and T.S. 3.9.2
Answer "A" is incorrect, although only one is required in Mode 6, two NI's are required during core alterations.
Answer "B" is incorrect, this is still a core alteration.
Answer "D" is incorrect, this is simply a requirement for refueling.

References:

1502.004, Control of Unit 1 Refueling
T.S. 3.9.2

History:

Direct from regular exam bank QID#3178
Selected for 2005 SRO exam.
Selected for 2010 SRO exam
Selected for 2014 SRO Exam.

5.0 LIMITS AND PRECAUTIONS

5.1 During movement of any fuel assemblies within the reactor building, radiation levels shall either be monitored by Fuel Handling Equip Rad Monitor (RE-8017) or applicable TRM 3.9.1 Condition has been entered and the Required Action to place a portable survey instrument of appropriate range and sensitivity in-service have been performed. (TRM 3.9.1).

5.2 During movement of any fuel assemblies within the auxiliary building, radiation levels shall either be monitored by Spent Fuel Pool Rad Monitor (RE-8009) or applicable TRM 3.9.2 Condition has been entered and the Required Action to place a portable survey instrument of appropriate range and sensitivity in-service have been performed. (TRM 3.9.2).

5.3 One source range neutron flux monitor shall be operable in Mode 6. Two source range neutron flux monitors shall be operable during core alterations (TS 3.9.2).

5.4 One decay heat removal loop shall be operable and in operation in Mode 6 with water level ≥23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.4 for contingencies and exceptions.

5.5 Two decay heat removal loops shall be operable, and one loop shall be in operation in Mode 6 with the water level <23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.5 for contingencies and exceptions.

{4.3.1}

NOTE

The Refueling Boron Concentration specified by Reactivity Balance Calculation (1103.015) Worksheet 5 or 6 provides a shutdown margin of 5% as required by NRC commitment P 205. This concentration also satisfies TS 3.9.1.

5.6 Boron concentration of the RCS and Fuel Transfer Canal shall be maintained within the limits specified in the COLR when the Fuel Transfer Canal is connected to the RCS (TS 3.9.1).

5.7 Direct communications between the control room and the refueling personnel in the reactor building shall exist during movement of irradiated fuel assemblies in the reactor building (TRM 3.9.4).

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2

a. One source range neutron flux monitor shall be OPERABLE, and

b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable during CORE ALTERATIONS.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately Immediately
B. No OPERABLE source range neutron flux monitor.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status. AND B.2 Perform SR 3.9.1.1.	Immediately Once per 12 hours

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:**	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 0310.

What is the LATEST time at which a replacement CRS must be in the Control Room BEFORE Technical Specifications are violated?

A. 0400

B. 0500

C. 0600

D. 0700

Answer:

B. 0500

Notes:

Answer [B] is the correct answer since the maximum time the shift can be below the minimum complement is two hours.
Answers [A], [C], [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
10CFR50.54(m)

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: 0407 **Rev:** 1 **Rev Date:** 9/3/14 **Source:** Modified **Originator:** J.Cork

TUOI: ASLP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of individual licensed operator responsibilities related 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:** No **Difficulty:** 2

Group: **SRO Imp:** 3.8 **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 0310.

What is the LATEST time at which a replacement CRS must be in the Control Room BEFORE Technical Specifications are violated?

A. 0400

B. 0500

C. 0600

D. 0700

Answer:

B. 0500

Notes:

Answer [A] is the correct answer since the maximum time the shift can be below the minimum complement is two hours.
Answers [B], [C], [D] are one hour increments around the correct answer.

References:

Technical Specifications 5.2.2 c
10CFR50.54(m)

History:

New created for 2001 SRO Exam.
Selected for use in 2002 SRO exam.
Used on 2004 SRO Exam.
Selected for 2011 SRO Exam. KA 2.1.4

PARENT

QID: 0409 **Rev:** 2 **Rev Date:** 12/4/06 **Source:** Direct **Originator:** JCork
TUOI: ASLP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities
System Number: 2.2 **System Title:** Equipment Control
Description: Knowledge of the process for making design or operating changes to the facility.

K/A Number: 2.2.5 **CFR Reference:** 41.10 / 43.3 / 45.13
Tier: 3 **RO Imp:** 2.2 **RO Select:** No **Difficulty:** 3
Group: **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 95

Which of the following changes would require a 10 CFR 50.59 Evaluation per EN-LI-101, 10 CFR 50.59 Evaluations, rather than only a PAD review per EN-LI-100, Process Applicability Determination?

A. A change to the table of contents for 1203.017, Moderator Dilution.

B. A change in the title of Shift Superintendent to Shift Manager.

C. A change to correct a HPI valve number in 1104.002, Makeup & Purification Procedure.

D. A change to the acceptance criteria for the LPI pumps' surveillance.

Answer:

D. A change to the acceptance criteria for the LPI pumps' surveillance.

Notes:

Answer [D] is correct, any time the method used to evaluate a safety system function is changed, a 50.59 review is required.
Answers [A], [B], [C] will have PAD reviews but do not require 50.59 reviews.

References:

EN-LI-101, 10 CFR 50.59 Evaluations

History:

Created for 2001 SRO Exam.
Selected for use in 2005 SRO exam.
Modified for use in 2007 SRO Exam.
Selected for 2014 SRO Exam

[6] Change – A modification or addition to, or removal from the facility or procedures that affects:

(a) A design function, or

(b) A method of performing or controlling the design function, or

(c) A method of evaluation that demonstrates the intended design functions will be accomplished.

An activity involving a system, structure, or component (SSC) not explicitly described in the UFSAR that affects the function of an SSC that is explicitly described in the UFSAR is also considered a change.

[7] Consequences of an Accident or Malfunction of Equipment Important to Safety – The radiological consequences (dose) that may result from an accident or equipment malfunction. Additionally, onsite dose consequences that restrict access to vital areas or otherwise impede actions to mitigate the consequences of accidents may require a license amendment.

[8] Departure from a Method of Evaluation described in the UFSAR used in establishing the Design Bases or in the Safety Analyses –

(a) Changing any element of the method described in the UFSAR unless the results of the analysis are conservative or essentially the same; or

(b) Changing from a method described in the UFSAR to another method unless that method has been approved by NRC for the intended application and all conditions specified in the associated NRC Safety Evaluation are met for the site adopting the new method.

Results are "essentially the same" if they are within the margin of error for the type of analysis being performed.

[9] Described in the UFSAR – SSCs, procedures, tests, descriptions, analyses, drawings, etc. that are described explicitly or implicitly in the UFSAR.

[10] Design Bases – As defined in 10 CFR 50.2, that information which identifies the specific functions to be performed by an SSC and the specific values or range of values chosen for controlling parameters as reference bounds for design. These values may be:

(a) Restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals; or

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

QID: 0879 **Rev:** 0 **Rev Date:** 6/3/14 **Source:** Direct **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:** 41.10 / 43.5 / 45.13

Tier:	3	**RO Imp:**	2.6	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 96

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

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)

[3] Emergency Maintenance – The correction of a condition or deficiency that:

(a) Constitutes an immediate and direct threat to the health and safety of the public.

(b) Requires immediate attention to prevent deterioration of plant conditions to a possible unsafe or unstable level, *which would then* constitute an immediate and direct threat to the health and safety of the public.

(c) Poses a significant industrial hazard that must be corrected immediately to prevent or mitigate actual serious injury or death.

(d) The Shift Manager can authorize the immediate start of repair efforts, in parallel with initiation and planning of a Priority 1 Work Request/Work Order. The Work Order should be completed as soon as practical to document the repairs.

[4] Emergent Work – Any work added after schedule freeze. This classification will not include FIN work.

[5] Expedited Work Order – Work determined to be "Emergency Maintenance" where the need exists, as determined by the Shift Manager, to commence work in the field prior to 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 safety. The Shift Manager is responsible for assessing the risk of performing work under an expedited work order.

[6] Fire Impairment - The degradation of a fire protection system or feature that adversely affects the ability of the system or feature to perform its intended function.

[7] Job Type – The code that identifies at the TASK level the type of work to be performed. Refer to EN-WM-105. The following definitions characterize the equipment degradation aspects of the identified issue. Attachment 9.7 provides examples of Corrective, Deficient and Other work order classifications.

(a) Corrective Maintenance (CM) – Represents a level of degradation of plant equipment that has failed or is significantly deficient such that failure is imminent (within its operating cycle/preventive maintenance interval) and it no longer conforms to or cannot perform its design function. Plant equipment should be considered failed or significantly deficient if the deficiency is similar to any of the following examples:

- Is removed from service because of actual or incipient failure

- Significant component degradation that affects system operability –The SSC may be determined operable by engineering assessment, but the degradation is significant and requires immediate corrective action. This normally includes any deficiency that requires a basis for continued operation as defined in NRC Generic Letter 91-18, and should be

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

QID: 0931 **Rev:** 0 **Rev Date:** 9/12/14 **Source:** Modified **Originator:** Cork

TUOI: A1LP-SRO-RAD **Objective:** 4 **Point Value:** 1

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

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of 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:**	4
Group:		**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 97

A worker arrives on site with 1.3 Rem accumulative dose for the calendar year.
The worker's NRC form 4 is on file.
The worker's expected exposure will be 1.6 Rem for his assigned job.

In accordance with Entergy administrative procedures, which of the SPECIFIC authorizations listed below is required to extend the worker's TEDE exposure limit?

A. The worker's Supervisor and Radiation Protection Manager

B. The worker's Supervisor, Radiation Protection Manager. and Plant General Manager.

C. Radiation Protection Manager, Plant General Manager and Site Vice President.

D. This exposure limit can not be authorized per Entergy Admin Exposure Limits.

Answer:

A. The worker's Supervisor and Radiation Protection Manager

Notes:

Answer [a] is correct IAW EN-RP-201 for doses >2 R but <3 R. Since the worker has 1.3 Rem and is expected to receive 1.6 Rem, his total exposure is 2.9 Rem.
Answer [b] is incorrect, this is the authorization required for doses >3 R but <4 R.
Answer [c] is incorrect, this is the authorization required for doses >4 R but <4..5 R.
Answer [d] is incorrect, this limit can be authorized.

Question was modified by making worker's accumulative dose prior to arrival on site to be 1.3 Rem and re-arranging the choices thereby changing the correct answer from "C" to "A".

References:

EN-RP-201, Dosimetry Administration

History:

Modified QID 391 for 2014 SRO Exam.

5.4, continued

[3] Extend a Radiation Workers' administrative TEDE ADG to the guidelines described in the following table, after obtaining the indicated approvals.

NOTE

Responsible individuals may be designated to authorize dose extensions.

Exposure Guideline	Requirements	Authorizations (Note)
Greater than 2000 mrem and less than or equal to 3000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends RP Manager approves
Greater than 3000 mrem and less than or equal to 4000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends Radiation Protection Manager approves Plant General Manager approves
Greater than 4000 mrem and less than 4500 mrem per year for Radiation Workers. Greater than 400 mrem but less than or equal to 450 mrem /gestation period	No undocumented quarters in the current year	Radiation Protection Manager approves Plant General Manager approves Site Vice President approves
Greater than 1000 mrem and less than or equal to 2000 mrem for individuals whose lifetime exposure greater than or equal to 1000 mrem * n where n = age	No undocumented quarters in the current year	Individual's Supervisor recommends Radiation Protection Manager approves

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

QID: 0391 **Rev:** 0 **Rev Date:** 5/26/11 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-SRO-RAD **Objective:** 4 **Point Value:** 1

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

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of 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:** 3.5

Group: **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** C

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

A worker arrives on site with 2.8 Rem accumulative dose for the calendar year.
The worker's NRC form 4 is on file.
The worker's expected exposure will be 1.6 Rem for his assigned job.

Whose authorization is required to extend the worker's TEDE exposure limit?

a. The worker's Supervisor, Radiation Protection Manager, and Plant General Manager.

b. The worker's Supervisor and Radiation Protection Manager.

c. Radiation Protection Manager, Plant General Manager and Site Vice President.

d. This exposure limit can not be authorized per Entergy Admin Exposure Limits.

Answer:

c. Radiation Protection Manager, Plant General Manager and Site Vice President.

Notes:

Answer [c] is correct IAW EN-RP-201 for doses >4 R but <4.5 R.
Answer [a] is incorrect, this is the authorization required for doses >3 R but <4 R.
Answer [b] is incorrect, this is the authorization required for doses >2 R but <3 R.
Answer [d] is incorrect, this limit can be authorized, it is the authorization required for doses >4.5 R.

References:

EN-RP-201, Rev. 3

History:

New question created for 2011 SRO Exam.

PARENT

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

QID: 0816 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** Direct **Originator:** S Pullin

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

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

K/A Number: 2.3.14 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier:	3	**RO Imp:**	3.4	**RO Select:**	No	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 98

During a fuel handling accident Krypton-85 is the major source of gaseous activity released from a damaged Fuel assembly that has decayed for >190 days.

Which portion of the body will receive the highest dose after a fuel handling accident?

A. Skin dose from Beta

B. Whole body dose from Gamma

C. Extremities dose from Beta

D. Internal Organ dose from Gamma

Answer:

A. Skin dose from Beta

Notes:

A is correct, skin dose rates from K-85 are 100 times higher than the whole body , gamma dose rates.
B, C, and D are all incorrect.

This question is SRO applicable due to relation to 10CFR55.43(4).

References:

1203.042, Refueling Abnormal Operation

History:

New selected for 2010 SRO exam
Selected for 2014 SRO Exam

SECTION 1 -- FUEL HANDLING ACCIDENT

INSTRUCTIONS

1. **IF damage to a spent fuel assembly is suspected, THEN perform the following:**

WARNING

Krypton-85, a beta emitter, is the major source of gaseous activity released from a damaged spent fuel assembly that has decayed >190 days. Skin dose rates from Kr-85 are 100 times higher than the whole body, gamma dose rate. Instruments not sensitive to beta, such as self-reading dosimeters and survey meters with their beta windows closed, will read less than the actual values.

A. Notify RP personnel to proceed to the area and a beta hazard associated with a damaged spent fuel assembly could exist.

B. Inspect the spent fuel assembly with available means to determine if damage has occurred.

C. **IF** even slight spent fuel assembly damage is detected, **THEN** GO TO step 2.

D. **IF** fuel assembly is NOT damaged, **THEN** proceed as directed by Operations Manager.

2. **IF spent fuel assembly damage is confirmed, THEN perform the following:**

A. **IF** accident has occurred outside of the Reactor Building, **THEN** perform the following while continuing with this section:

1) Make the following announcement over the plant PA system:

"Attention all personnel; attention all personnel. Due to a fuel handling accident in the (*location*) all personnel shall evacuate the (*state the area to be evacuated*). All other plant personnel continue normal activities until instructed otherwise."

2) Activate evacuation alarm for appropriate area.

3) Repeat announcement and alarm.

(continued)

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

QID: 1023 **Rev:** 0 **Rev Date:** 9/24/14 **Source:** New **Originator:** Cork

TUOI: ASCNT-EP-A0011 **Objective:** 3 **Point Value:** 1

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

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

Description: Knowledge of SRO responsibilities in emergency plan implementation.

K/A Number: 2.4.40 **CFR Reference:** 41.10 / 43.5 / 45.11

Tier:	3	**RO Imp:**	2.7	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	4.5	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 99

Given:
- An Alert has been declared on Unit 1 30 minutes ago.
- All notifications have been made for the Alert.

The U-1 Shift Manager tells the CRS he will be leaving the Control Room to meet with the Operations Manager.

Who can the U-1 Shift Manager turn over Emergency Direction and Control to before leaving the Control Room?

A. Unit 2 Shift Manager

B. Unit 1 Control Room Supervisor

C. EOF Emergency Director

D. TSC Director

Answer:

C. EOF Emergency Director

Notes:

C is correct per 1903.064
A is incorrect but sounds plausible since the Unit 2 Shift Manager is also Emergency Director qualified but he cannot assume those duties on Unit 1 without first being relieved as Shift Manager.
B is incorrect but plausible since the CRS is SRO licensed and can be responsible for oversight but this position is not qualified as ED.
D is incorrect but plausible since this position previously could take ED&C but can no longer do so.

References:

1903.064, Emergency Response Facility - Control Room

History:

New for 2014 SRO Exam

6.3 TURNOVER

6.3.1 Shift Manager/Emergency Director (ED)

A. The Shift Manager of the affected unit shall have responsibility and authority for Emergency Direction and Control of the incident response until relieved by the EOF Emergency Director (ED).

B. The Shift Manager SHALL NOT delegate the responsibility for making offsite Protective Action Recommendations (PARs) or for making decisions to notify offsite authorities while responsible for Emergency Direction and Control.

C. The Shift Manager must turn over responsibilities to a qualified individual before leaving the Control Room when he has responsibility for Emergency Direction and Control.

D. The responsibility for Emergency Direction and Control must be transferred from the Shift Manager to the Emergency Director (ED) within 60 - 90 minutes of an Alert, or higher, emergency class.

E. The Emergency Director (ED) shall notify the Shift Manager when he is prepared to assume the responsibility and authority for Emergency Direction and Control of the incident.

F. The Shift Manager shall promptly turn over responsibility and authority for the overall response as requested by the Emergency Director (ED).

G. The Shift Manager shall announce the turnover to the Initial Response Staff (IRS) personnel and report this turnover to the Emergency Plant Manager located in the TSC.

H. It is the responsibility of the Shift Manager to ensure that the Command and Control Status Board in the Control Room is updated as turnover occurs in the ERO.

6.3.2 Control Room Staff

A. Emergency Response personnel in the Control Room who must leave their assigned location temporarily must inform their immediate superior of their absence, destination, and estimated time of return (with the exception of the Shift Manager as outlined in Section 6.3.1.C).

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

QID: 0880 **Rev:** 0 **Rev Date:** 9/3/14 **Source:** New **Originator:** Possage

TUOI: ASCBT-EP-A0081 **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

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

Description: Knowledge of procedures relating to a security event (non-safeguards information).

K/A Number: 2.4.28 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 3 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 100

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

You receive a report from security that there is an on-going gun fight with multiple intruders near the intake structure.

What is the appropriate classification for this event?

A. HU-1, Unusual Event

B. HA-1, Alert

C. HS-1, Site Area Emergency

D. HG-1, General Emergency

Answer:

C. HS-1, Site Area Emergency

Notes:

Provide 1903.010, Emergency Action Level Classification - HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY - Security

Answer C contains the correct classification per 1903.010.
Answers A, B, D contain the other available classifications for a security event making them plausible but incorrect.

References:

1903.010, Emergency Action Level Classification

History:

New question for 2014 SRO Exam.

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Security			
HG1 1 2 3 4 5 6 D HOSTILE ACTION resulting in loss of physical control of the facility **Emergency Action Level(s):** 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.	**HS1** 1 2 3 4 5 6 D HOSTILE ACTION within the PROTECTED AREA **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANO Security Shift Supervision.	**HA1** 1 2 3 4 5 6 D HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANO Security Shift Supervision. **OR** 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.	**HU1** 1 2 3 4 5 6 D Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant **Emergency Action Level(s):** 1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANO Security Shift Supervision. **OR** 2. A credible site specific security threat notification. **OR** 3. A validated notification from NRC providing information of an aircraft threat.

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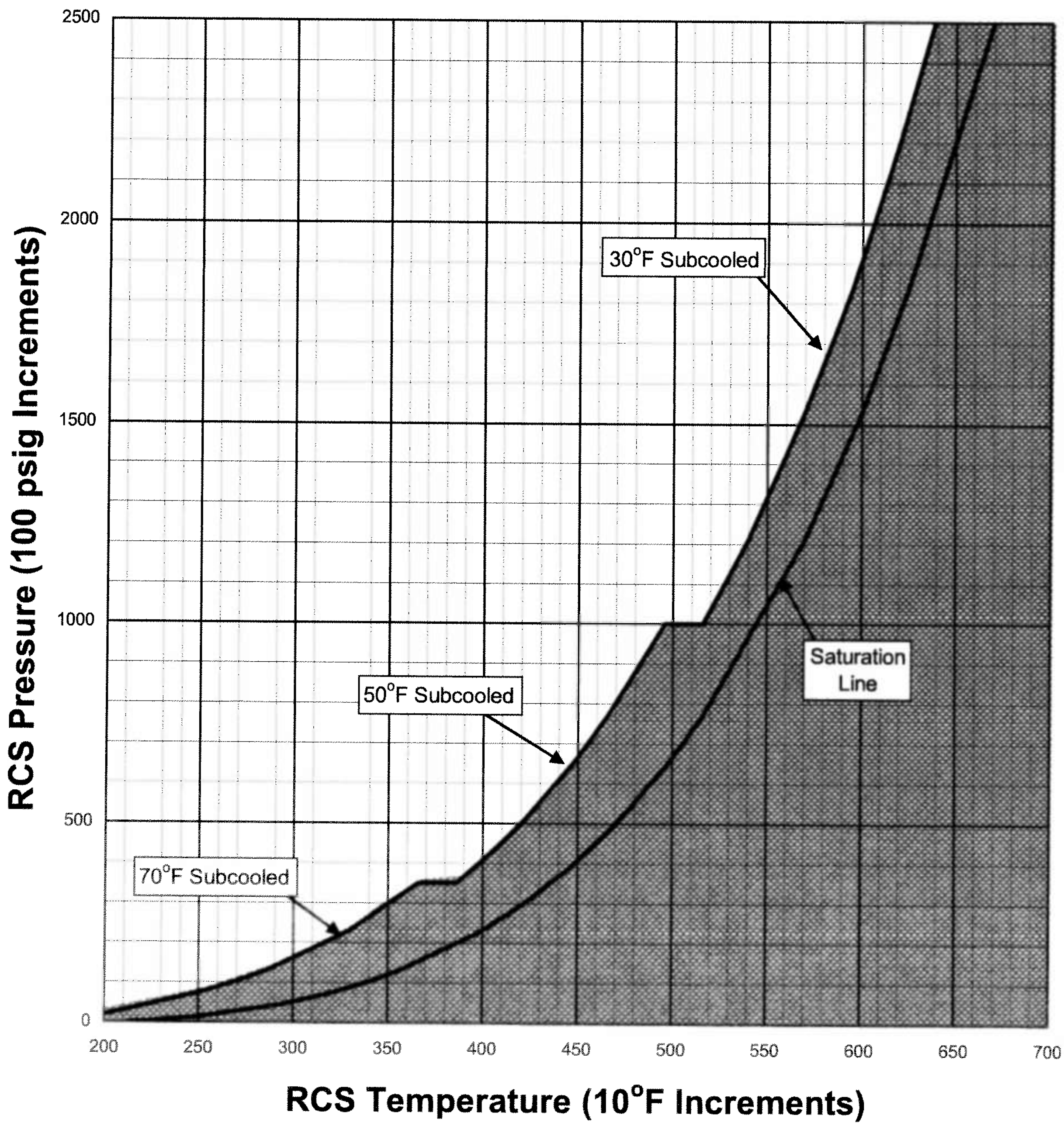
FIGURE 1
Saturation and Adequate SCM



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

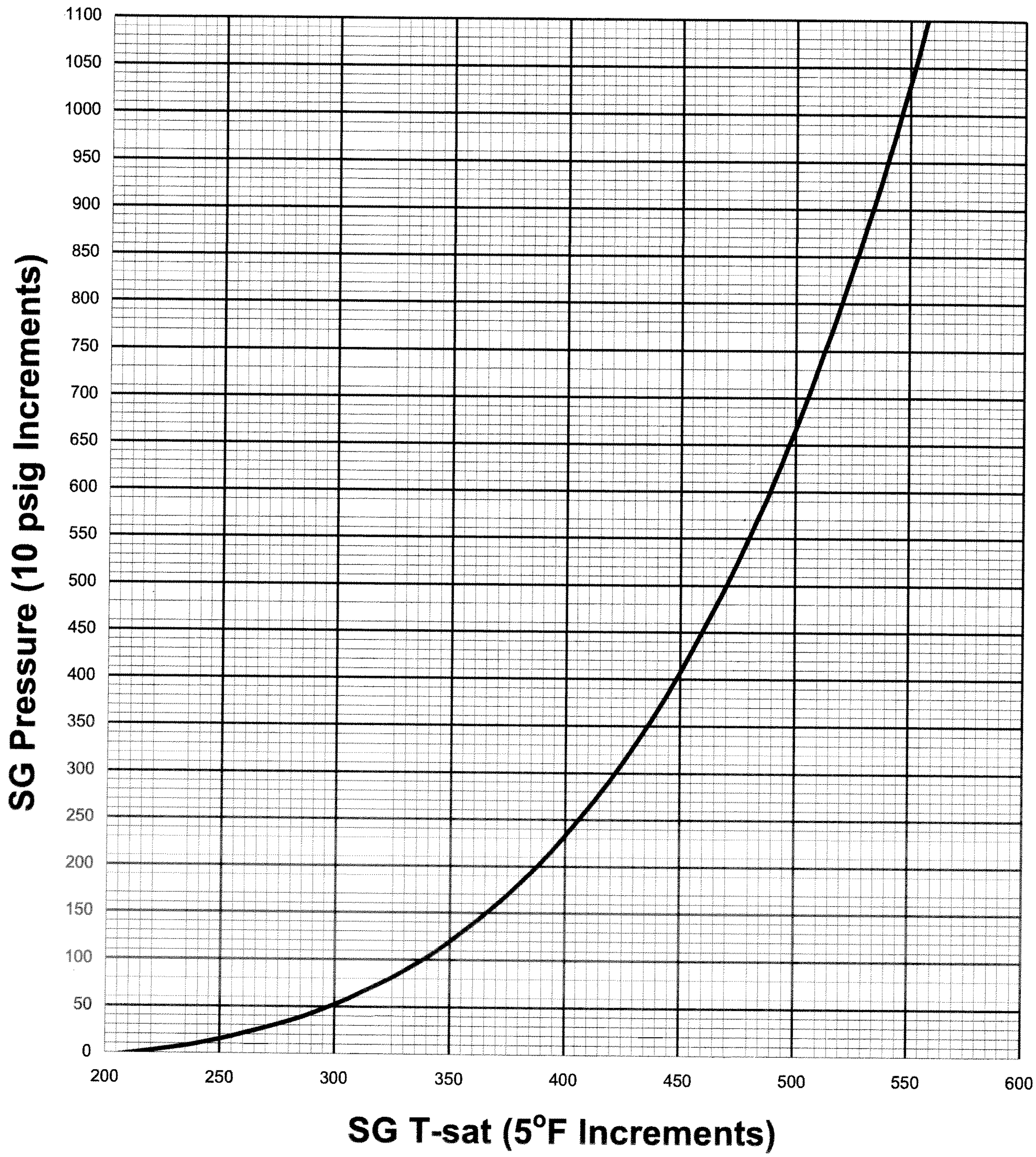


FIGURE 3
RCS Pressure vs Temperature Limits

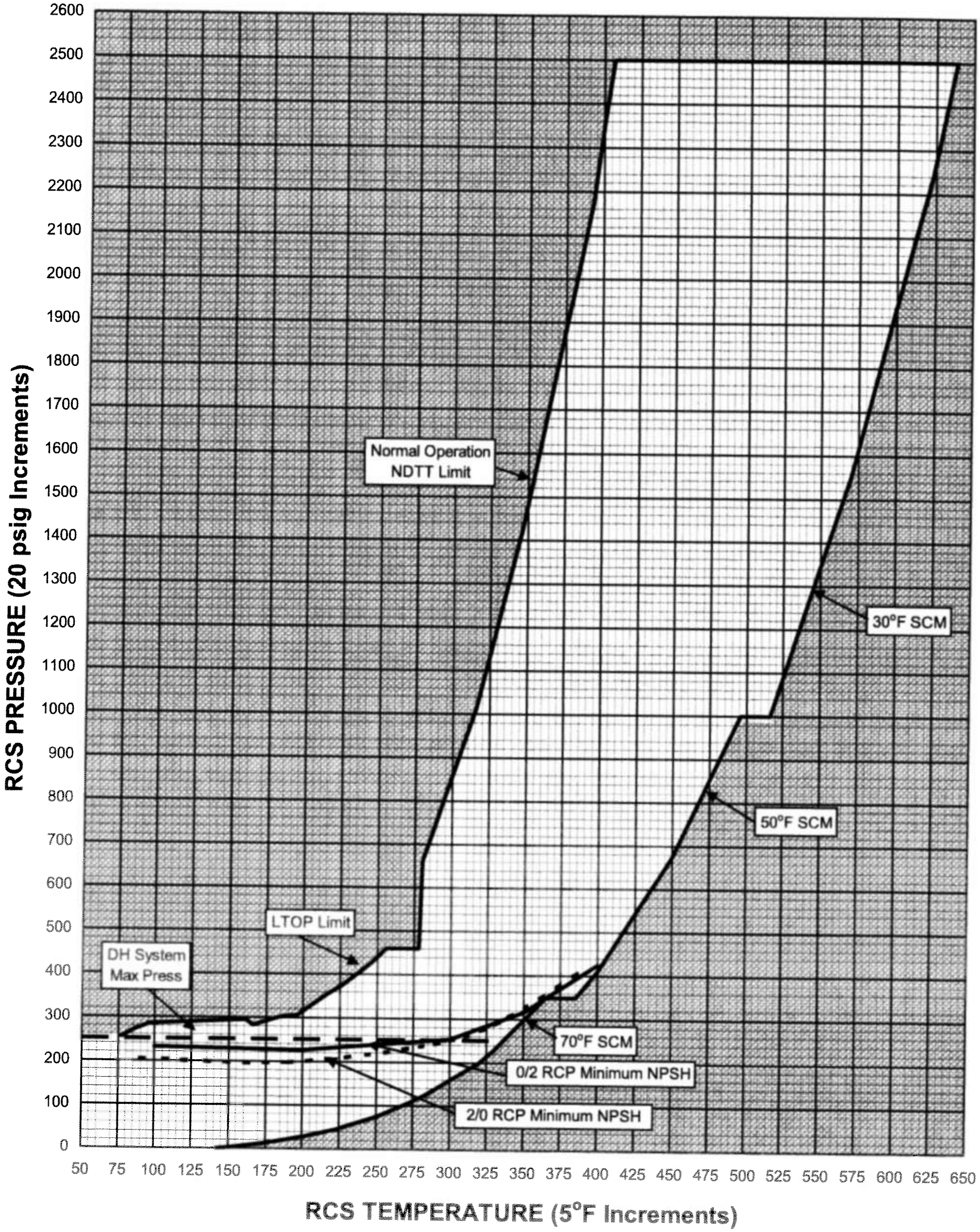


FIGURE 4
Core Exit Thermocouple for Inadequate Core Cooling

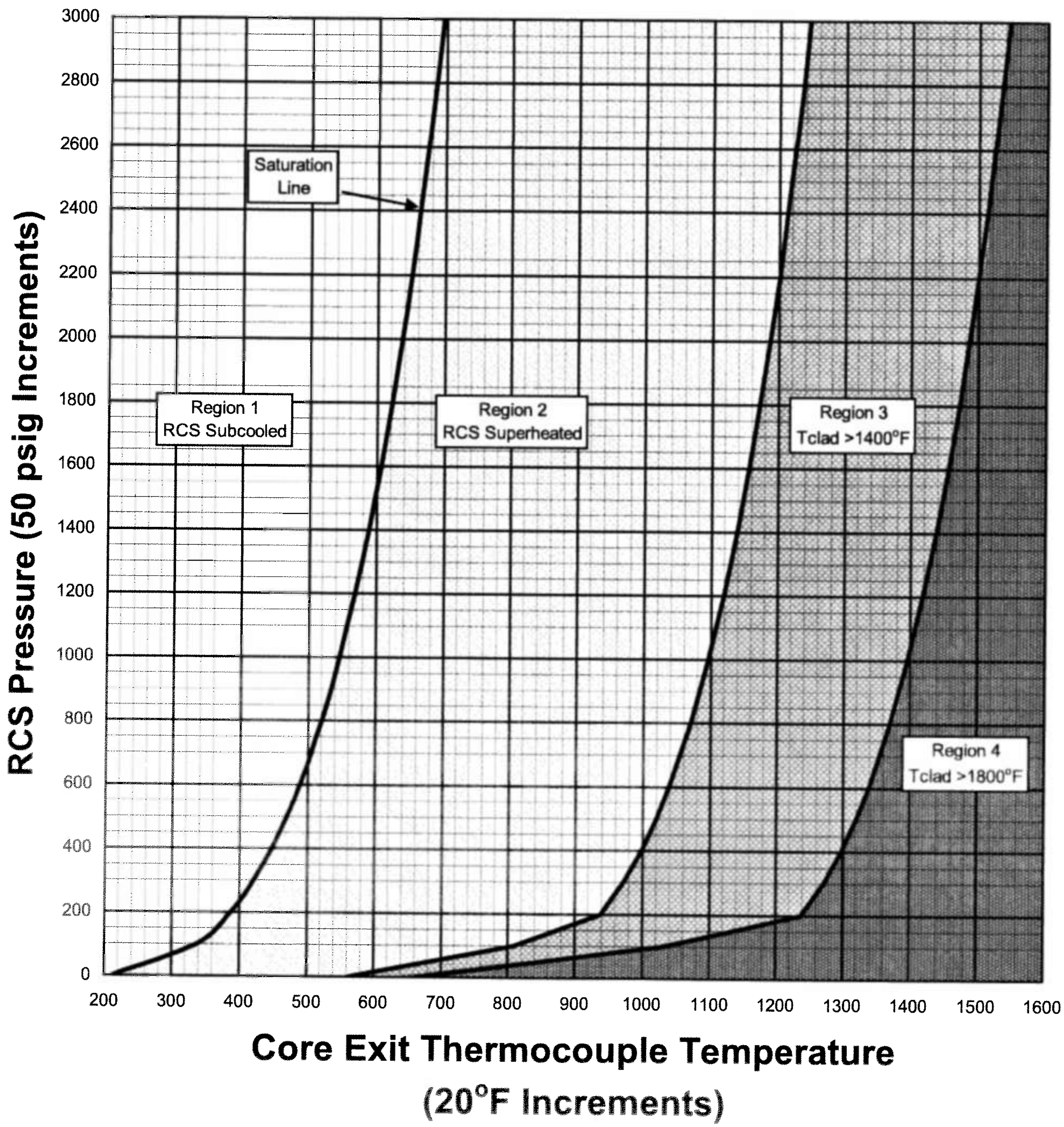


FIGURE 5

SG Pressure to Establish 40° to 60°F Primary to Secondary ΔT

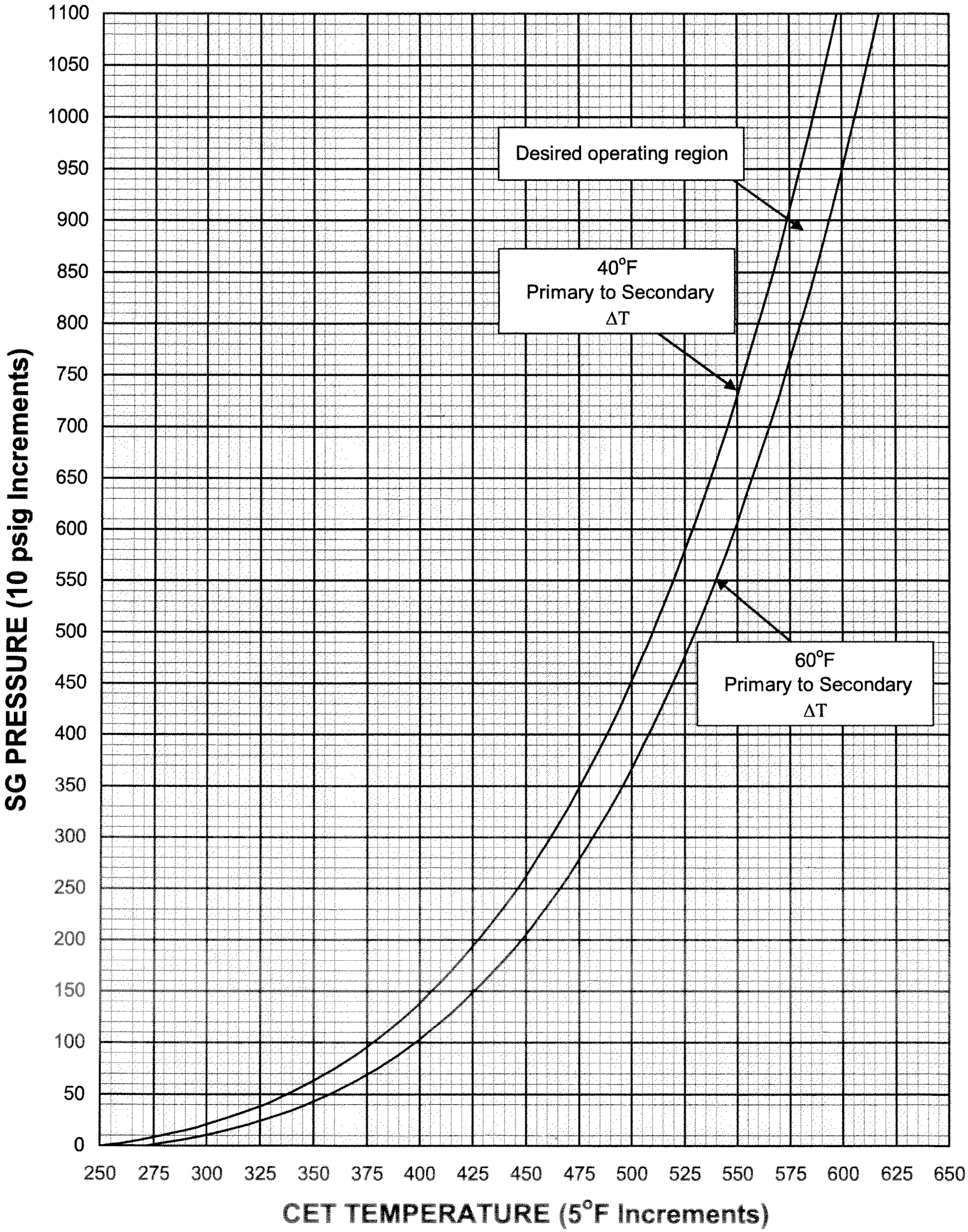
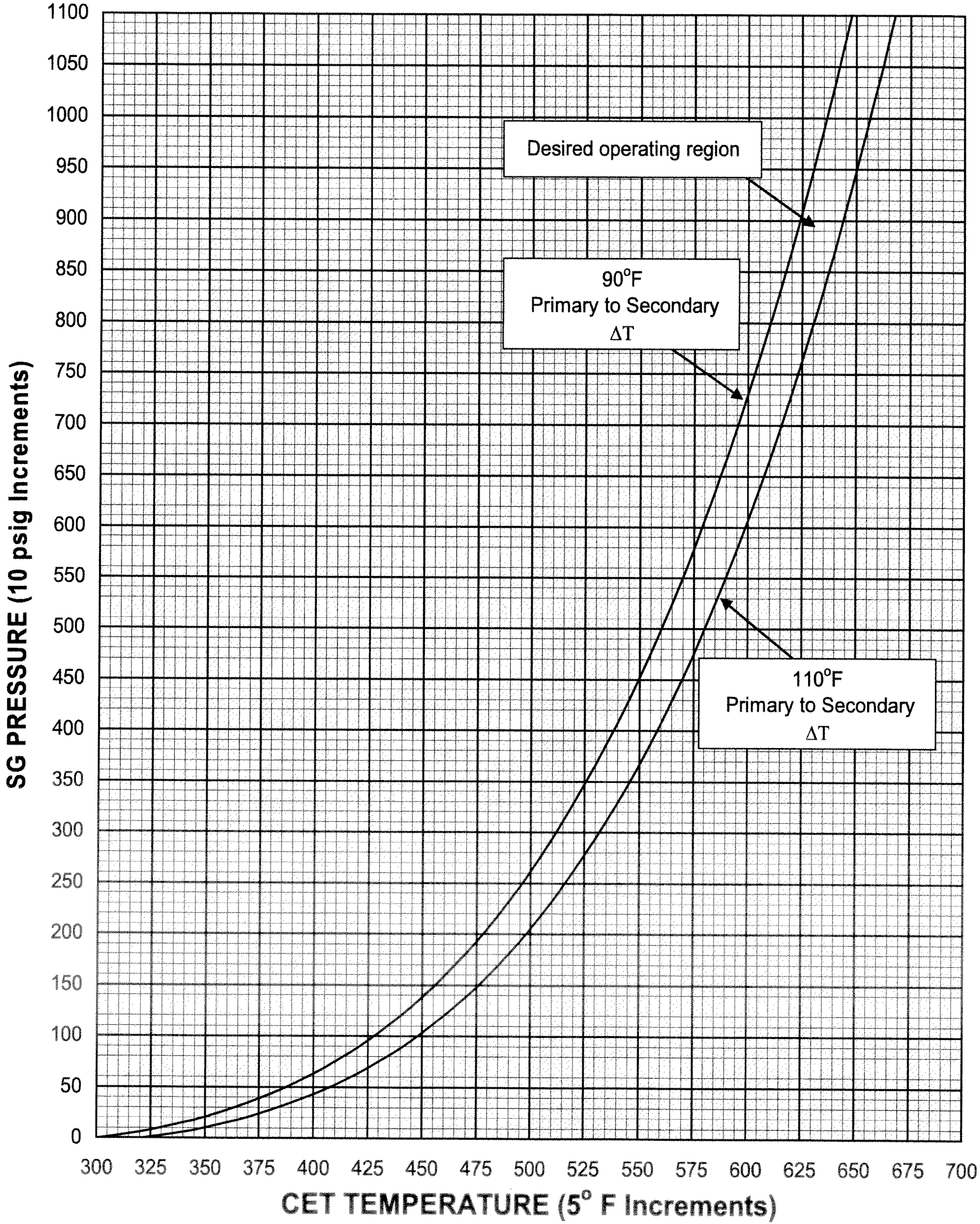


FIGURE 6

SG Pressure to Establish 90° to 110°F Primary to Secondary ΔT



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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. <u>AND</u> 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.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 Two LPI trains shall be OPERABLE.

-----NOTE-----
An LPI train may be considered OPERABLE during alignment and when aligned for decay heat removal, if capable of being manually realigned to the LPI mode of operation.

APPLICABILITY: MODE 3 with Reactor Coolant System (RCS) temperature ≤ 350°F,
MODE 4.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to ECCS DHR loops.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LPI train inoperable.	A.1 Restore LPI train to OPERABLE status.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 -----NOTE----- Only required if one DHR train is OPERABLE. ----- Be in MODE 5.	24 hours
C. Two LPI trains inoperable.	C.1 Initiate action to restore one LPI train to OPERABLE status. <u>AND</u> C.2 -----NOTE----- Only required if one DHR train is OPERABLE. ----- Be in MODE 5.	Immediately 24 hours

3.7 PLANT SYSTEMS

3.7.7 Service Water System (SWS)

LCO 3.7.7 Two SWS loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS loop inoperable.	A.1 --------------NOTES------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by SWS. 2. Enter Applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal made inoperable by SWS. -------------------------------------- Restore SWS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Both DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	8 hours
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 124.7 V on float charge.	7 days
SR 3.8.4.2	Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	18 months

SURVEILLANCE		FREQUENCY
SR 3.8.4.3	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 The DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.1.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.1.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.1.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately
	AND	

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1.5 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by Condition A.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	------------------------------NOTE------------------------------ The following SRs are not required to be performed: SR 3.8.4.2 and SR 3.8.4.3. --- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, and SR 3.8.4.3.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 The following inverters shall be OPERABLE.

a. Two Red Train inverters (Y11 and Y13, Y11 and Y15, or Y13 and Y15), and

b. Two Green Train inverters (Y22 and Y24, Y22 and Y25, or Y24 and Y25),

---NOTE---
One of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b may be disconnected from its associated DC bus for ≤ 2 hours to perform load transfer to or from the swing inverter, provided:

a. The associated 120 VAC bus is energized from its alternate AC source; and

b. The other three 120 VAC buses are energized from their associated OPERABLE inverters.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	A.1 ---------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any of the 120 VAC buses RS1, RS2, RS3, or RS4 de-energized. ------------------------------------- Restore inverter to OPERABLE status.	24 hours <u>AND</u> 96 hours from discovery of failure to meet LCO

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 Inverters shall be OPERABLE to support the onsite Class 1E AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately
	AND	

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.5	Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by AC vital bus inverter(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage and alignments to required 120 VAC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Two AC, DC, and 120 VAC electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystem(s) inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One or more 120 VAC electrical power distribution subsystem(s) (RS1, RS2, RS3, RS4) inoperable.	B.1 Restore 120 VAC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
C. One or more DC electrical power distribution subsystem(s) inoperable.	C.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours
	D.2 Be in MODE 5.	36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE by the following specifications:

LCO 3.3.9, "Source Range Neutron Flux,"
LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System,"
LCO 3.7.9, "Control Room Emergency Ventilation System (CREVS),"
LCO 3.7.10, "Control Room Emergency Air Conditioning System (CREACS),"
LCO 3.9.2, "Nuclear Instrumentation," for one monitor,
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or 120 VAC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable. OR A.2.1 Suspend CORE ALTERATIONS. AND	Immediately Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.4 Initiate actions to restore required AC, DC, and 120 VAC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2.5 Declare associated required decay heat removal subsystem(s) inoperable.	Immediately
	<u>AND</u>	
	A.2.6 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by Electrical Power Distribution System.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	7 days

23.11 Instruments Removed from Service

As a general rule, flow measurements derived from differential pressure across a restriction are inaccurate below 10% of the flow span. Under zero flow conditions, readings between 0 and 5% of indicated flow span are to be expected and do not necessarily represent a need for instrument calibration. Under zero flow conditions, if the indicated flow is above 10% of the flow span it will be required to be calibrated, but will not be considered inoperable.

When a channel includes more than one qualified control room indication, such as both an indicator and a recorder, or an indicator and Safety Parameter Display System readout, etc., only one indication is required for channel operability (TS 3.3.15 Bases).

23.12 Failed LPI Flow Instruments

Any qualified indication can serve to meet the requirements of TS 3.3.15-1.19 Condition "A". LPI flow indicators FIS-1401 and FIS-1402, SPDS and FIRS-1500 meet the requirements of TS 3.3.15-1.19. An inoperable LPI flow instrument (i.e. transmitter and/or associated instrument loop) requires the following actions to be performed:

23.12.1 Declare associated train of LPI inoperable and applicable TS LCO (3.5.2 or 3.5.3) not met based on Support SSC inoperability.

23.12.2 Enter TS 3.3.15 Condition A.

23.12.3 Perform one of the following:

A. Enter one of the applicable Tech Specs:

- If RCS >350°F, then enter TS 3.5.2 Condition A.
- If RCS ≤350°F, then enter TS 3.5.3 Condition A.

B. IF desired to enter TS 3.0.6, THEN perform the following:

NOTE

Compliance with the Conditions and Required Action of TS 3.5.2 or TS 3.5.3 may be delayed until it is determined that LCO 3.0.6 cannot be applied.

1. IF opposite train and required support equipment are operable, THEN perform 1015.045, Unit 1 Safety Function Determination Program (refer to Att. 2 of the procedure).

   a. IF opposite train is inoperable, THEN enter applicable TS 3.5.2 or TS 3.5.3.

2. IF SM/CRS determines TS 3.0.6 is applicable, THEN enter TS 3.0.6 AND immediately perform a Safety Function Determination as directed by 1015.045.

    a. IF entry into TS 3.0.6 is NOT approved for use per 1015.045, THEN enter applicable Conditions and Required Actions of TS 3.5.2 or TS 3.5.3.

23.13 Removing LPI Flow Instruments from Service

RB Spray, HPI and LPI flow instrumentation is used in the Emergency Operating Procedures in part, to verify system flow is proper prior to shifting suction source from the BWST to the RB Sump. Typically, when a flow transmitter is removed from service, the system becomes inoperable, however the pump and injection flowpath could still function and operate upon an actuation signal. In this case, should it be necessary to shift suction to the RB Sump, the operator cannot verify proper flow, hence the pump might need to be secured to ensure adequate NPSH to the other operating ECCS pumps. The Shift Manager should be consulted before securing the ECCS pump as it might be the only operating pump which has no flow indication.

23.13.1 Prior to removing an LPI flow transmitter from service, install a caution tag on the handswitch for associated flow injection valve which states:

"The flow transmitter associated with this flowpath is OOS. If required to shift suction source from the BWST to the RB Sump, consult with Shift Manager to determine need to secure pump."

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY – Security			
HG1 1 2 3 4 5 6 D HOSTILE ACTION resulting in loss of physical control of the facility **Emergency Action Level(s):** 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.	HS1 1 2 3 4 5 6 D HOSTILE ACTION within the PROTECTED AREA **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANO Security Shift Supervision.	HA1 1 2 3 4 5 6 D HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat **Emergency Action Level(s):** 1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANO Security Shift Supervision. **OR** 2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.	HU1 1 2 3 4 5 6 D Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant **Emergency Action Level(s):** 1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANO Security Shift Supervision. **OR** 2. A credible site specific security threat notification. **OR** 3. A validated notification from NRC providing information of an aircraft threat.