# ANO 2011
# RO Questions
# Tier 1 Group1

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

**QID:** 0023 **Rev:** 1 **Rev Date:** 3/16/05 **Source:** Direct **Originator:** S.Pullin
**TUOI:** A1LP-RO-EOP01 **Objective:** 13 **Point Value:** 1

**Section:** 4.1 **Type:** Generic EPEs
**System Number:** 007 **System Title:** Reactor Trip
**Description:** Ability to operate and monitor the following as they apply to a reactor trip: S/G pressure.

**K/A Number:** EA1.10 **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:** 1 **SRO:** 1

The following conditions exist immediately after a reactor trip:

- Group 2, Rod 4 failed to fully insert into the core
- RCS pressure is at 1750 psig and trending down
- Pressurizer level is at 50 inches and trending down
- A OTSG pressure is at 880 psig and trending down
- B OTSG pressure is at 885 psig and trending down
- CETs are 560°F and stable
- Turbine Trip Solenoid Power Available light is OFF

Which of the following contains the required operator response as well as the reason for the response?

A. Manually actuate MSLI for affected SG(s) and EFW due to overcooling.

B. Commence emergency boration per RT-12 for the stuck rod.

C. Trip all Reactor Coolant Pumps due to loss of subcooling margin.

D. Initiate High Pressure Injection per RT-2 due to low pressurizer level and low RCS pressure.

**Answer:**

A. Manually actuate MSLI for affected SG(s) and EFW due to overcooling.

**Notes:**

(a) is correct since OTSG pressure <900 psig requires actuation of MSLI for the affected SG and actuation of EFW.
(b) is incorrect since a single stuck rod does not require emergency boration.
"c" is incorrect because subcooling margin is adequate.
(d) is incorrect since pressurizer level is >30 inches and RCS pressure is >1700 psig.

**References:**

1202.001, Chg.031

**History:**

Developed for 1998 RO/SRO Exam.
Modified for 2005 RO Exam, but not used.
Selected for use in 2007 RO Exam, status remains as modified. KA changed from 007 EK3.01.
Selected for 2011 RO Exam.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| 2. **Manually trip Turbine.**<br><br>A. Verify Turbine throttle and governor valves closed. | A. Perform the following:<br><br>1) **IF** 125 V DC Bus D01 is de-energized as indicated by **both** of the following, **THEN** perform Loss of 125V DC (1203.036) "Loss Of Bus D01" section in conjunction with this procedure.<br><br>• Turbine Trip Solenoid Power Available light off.<br>• Breaker position indications on left side of C10 off.<br><br>2) **IF** SG press is < 900 psig, **THEN** perform the following:<br><br>a) Actuate MSLI for affected SG(s) **AND** actuate EFW **AND** verify proper actuation and control (RT 6).<br><br>b) Advise Shift Manager to implement Emergency Action Level Classification (1903.010).<br><br>c) **GO TO 1202.003, "OVERCOOLING"** procedure. |

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

**QID:** 0506 **Rev:** 2 **Rev Date:** 12/16/2003 **Source:** Repeat **Originator:** NRC

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

**Section:** 4.1 **Type:** Generic EPE

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

**Description:** Knowledge of the interrelationships between the small break LOCA and the following: S/G's.

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

**Tier:** 1 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 4

**Group:** 1 **SRO Imp:** 3.3 **SRO Select:** Yes **Taxonomy:** Ap

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

Given:

- A Small Break LOCA has occurred.
- ATC has tripped the reactor.
- CBOT has tripped the turbine.
- Both OTSG Pressure's are at 895 psig
- SCM is 25°F.
- All RCPs are OFF.

The desired final OTSG levels will be maintained with EFW using ___?___ Level instrument(s) and obtained with a fill rate of ___?___.

A. EFIC Low Range,
2" to 8"/min in Manual or 340gpm/SG in Auto

B. EFIC High Range
2" to 8"/min in Auto or 340gpm/SG in Manual

C. EFIC Low Range
2" to 8"/min in Auto or 340gpm/SG in Manual

D. EFIC High Range
2" to 8"/min in Manual or 340gpm/SG in Auto

**Answer:**

B. EFIC High Range
2" to 8"/min in Auto or 340gpm/SG in Manual

**Notes:**

"B" is correct, EFIC High Range, 2" to 8"/min in Auto or 340gpm/SG in manual until Steam Generator level is 370 to 410 inches.. EFIC Low Range does not cover the 370-410" level required for this EOP/RT. EOP/RT specifies the level and rate and the other three distracters are incorrect combinations of other instruments and manual versus auto modes of control.

**References:**

1202.012 "Repetitive Tasks", RT 5 Change 009

**History:**

Developed by NRC.
Used on 2004 RO/SRO Exam
Used on the 2008 RO Exam
Selected for 2011 RO Exam, repeat from last two exams

## 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 |

## 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.

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 $\Delta T$ stable or dropping

6. **Monitor EMERGENCY FEEDWATER and EFIC alarms on K-12.**

## 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:** 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:** 3 **SRO:** 3

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.

The RC will have to flow between the core and the SGs by natural circulation. This requires a higher SG level than does forced-RC flow. The SG level must be high enough to create a SG heat sink thermal center sufficiently above the core thermal center to induce adequate natural circulation of the RC. Although obtaining the necessary SG levels is the preferred method for establishing natural circulation, adequate natural circulation can occur due to EFW flow only. For certain in-plant conditions (e.g., low decay heat level), it may not be necessary to increase SG levels as long as EFW flow is providing adequate heat transfer.

3.3 SG Loss of Subcooling Margin

If the RC SCM is lost, the SG level MUST be controlled at or above the [loss of subcooling margin setpoint].

This SG level is even higher than the natural circulation setpoint. When the RC SCM is lost, the RC has a potential of being saturated. Therefore, the SG water level is raised to the loss of SCM setpoint. This setpoint has been determined for a saturated RCS with steam in the hot leg pipes, for the case when core cooling is assisted by boiler condenser cooling. The core heats the surrounding water creating steam which flows through the hot leg pipes to the SGs where it is condensed. The resulting pool of water in the SG tubes must be higher than the elevation of the RCP internal spill-over so that the cold leg water will flow to the RV. For this to happen, the SG condensing surface has to be higher than the RCP internal spill-over. Also the condensing surface of the SG tubes must be adequate, combined with HPI cooling, to remove all the heat being generated by the core. The elevation of the EFW nozzles is high enough to provide the required condensing surface. The level setpoint is set high enough such that one SG can provide the required condensing surface during periods of no EFW flow.

3.4 If ICC conditions exist with an indicated fuel clad temperature greater than 1400°F the SG levels SHOULD be raised to the [ICC level setpoint]. (Only applicable to Davis-Besse)

The SG water level should be raised to the maximum level possible without causing water to enter the steam lines or losing SG level measurement or causing SG overfill protection system actuation; i.e., SFRCS trip on high SG level. This will provide the greatest SG condensing surface.

4.0 FEEDWATER CONTROL TO STEAM GENERATORS THAT CAN HOLD PRESSURE

This section applies only to SGs that can hold pressure. Special considerations for SG(s) that cannot hold pressure are given in Section 5.0.

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

**QID:** 0825 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** New **Originator:** S. Pullin

**TUOI:** A1LP-RO-ARCP **Objective:** 19 **Point Value:** 1

**Section:** 4.2 **Type:** Generic APEs

**System Number:** 015 **System Title:** Reactor Coolant Pump Malfunctions

**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:** | 3 |
| **Group:** | 1 | **SRO Imp:** | 2.8 | **SRO Select:** | Yes | **Taxonomy:** | C |

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

Given:
-Plant is at 65% power
-P-32A, P-32C, and P-32D RCPs are running
-K08-C7 "RCP BLEEDOFF TEMP HIGH" alarms
-P-32A Seal Bleedoff temperature is 210°F

Which of the following operator actions are required?

A. Reduce Reactor power to 50% and secure RCP P-32A.

B. Trip RCP P-32A and verify proper ICS response.

C. Close P-32A Seal Bleedoff Isolation CV-1273 and Seal Bleedoff Isolation to Quench Tank SV-1273.

D. Trip the Reactor and then trip RCP P-32A.

**Answer:**

D. Trip the Reactor and then trip RCP P-32A.

**Notes:**

Answer "D" is correct, this meets criteria for seal failure and requires tripping pump, this leaves no pumps in the A loop so the Rx must be tripped per 1203.031.
Answer "A" is incorrect, this would be done only if at least one pump was remaining in each loop per 1203.031.
Answer "B" is incorrect, this would be done if tripping the RCP would NOT cause an automatic Rx trip per 1203.031.
Andwer "C" is incorrect, this would be done with a simultaneous loss of seal injection with a loss of seal cooling.

**References:**

1203.031, Chg. 019

**History:**

New for 2011 RO Exam.

SECTION 2
SEAL FAILURE

**INSTRUCTIONS**

1. **IF tripping the affected RCP(s) will result in an automatic reactor trip,**
   **THEN perform the following:**

   A. Trip reactor.

   B. Trip affected RCP(s).

   C. While continuing with follow-up actions, refer to Emergency Operating Procedure (1202.XXX).

2. **IF tripping the affected RCP(s) will NOT cause an automatic reactor trip,**
   **THEN perform the following:**

   A. Trip affected RCP(s).

   B. Verify proper ICS response.

   C. IF only 1 RCP in operation per loop,
      THEN enter Tech Spec 3.4.4 Condition A (18-hour time clock).

3. **IF HPI is required to maintain RCS inventory,**
   **THEN trip reactor**
   **AND refer to Emergency Operating Procedure (1202.XXX).**

(continued)

ATTACHMENT A



RCP PARAMETERS

**NOTE**

- RCP Seal chart recorders on C13 show actual cavity pressures.
- Seal $\Delta P$ is available on RCP mimic displays on PDS/PMS.
- RCP seal stage $\Delta P$ can be determined as follows:
  - 1st stage $\Delta P$ = system pressure - lower seal cavity press.
  - 2nd stage $\Delta P$ = lower seal cavity pressure - upper seal cavity press.
  - 3rd stage $\Delta P$ = upper seal cavity pressure - RB atmospheric press.
- Third stage seal leakage by design is 0 to 0.08 gpm. Third stage leakage in excess of design will affect upper seal cavity pressure and seal bleed off flow.

**Seal Degradation — any of the following requires securing the affected RCP per Section 1:**

- RCP seal cavity pressure oscillations exceed 800 psi peak-to-peak
- $\Delta P$ across any stage exceeds 2/3 of system pressure on a running RCP
  **OR** exceeds 80% of system pressure on an idle RCP.
- $\geq$2.5 gpm total seal outflow, including seal bleedoff (excluding shaft sleeve leakage),
  **AND** a loss of seal injection
- Seal bleed off temp >40°F above 1st stage seal temp
- RCP seal bleed off or seal stage temp reaches 180°F,
  **AND** no interruption of seal injection OR ICW flow.

**Seal Failure — any of the following requires tripping the affected RCP per Section 2:**

- $\geq$10 gpm rise in RCS leak
  **AND** a change in seal cavity pressure behavior.
- RCP seal bleed off or seal stage temp reaches 200°F
  **AND** no change in seal injection **OR** ICW flow.
- $\Delta P$ across a single stage equal to RCS press, with seal bleed off flow established.

ATTACHMENT A



RCP PARAMETERS

**Loss of Cooling Water to RCP Motors or Motor/Bearing Trouble**

1. **IF Motor Bearing Temperature >190°F (167°F for P-32B)**
   **AND continues to rise,**
   **THEN SECURE the affected RCP per section 4 and/or section 5 of this procedure.**

2. **ANY of the following are criteria to SECURE the RCP per section 5 of this procedure:**

   - P32B, P32C or P32D **PUMP SHAFT** vibration; more than one channel ≥25 mils, after startup stabilization
   - P32A **PUMP SHAFT** vibration; more than one channel ≥28 mils, after startup stabilization

3. **ANY of the following are criteria to TRIP the affected RCP per section 4 and/or section 5 of this procedure:**

   - Motor current exceeds 800 amps
   - Winding temperature exceeds 300°F
   - Bearing temperature exceeds 225°F (176°F for P32B)
   - P-32B, C or D **MOTOR** vibration; more than one channel >20 mils after startup stabilization
   - P-32A **MOTOR** vibration; more than one channel >0.8 in/sec after startup stabilization
   - ANY RC PUMP SHAFT vibration ≥29 mils after startup stabilization

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

**QID:** 0549 **Rev:** 0 **Rev Date:** 3/17/2005 **Source:** Direct **Originator:** J.Cork

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

**Section:** 4.2 **Type:** Generic APE

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

**Description:** Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Pump Makeup: Isolating Letdown.

**K/A Number:** AK3.04 **CFR Reference:** 41.45, 41.10 / 45.6 / 45.13

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

**Group:** 1 **SRO Imp:** 3.4 **SRO Select:** Yes **Taxonomy:** C

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

Given the following:

- Plant is at 100% power.
- HPI pump discharge pressure is oscillating from 1500 to 2500 psig.
- Makeup flow rate is oscillating from 0 to 70 gpm.
- Seal Injection total flow is oscillating from 30 to 60 gpm.
- Pressurizer level is 215 inches and dropping.
- Letdown flow is 80 gpm and stable.
- Makeup tank level 50 inches and dropping.

Which of the following actions, and reasons for the actions, should be perfomed in response to these indications?

A. Trip HPI pump and isolate Letdown by closing Letdown Isolation, CV-1221, due to indications of degraded suction.

B. Take manual control of RC Pumps Total Injection Flow, CV-1207, and maintain 30-40 gpm to prevent RCP seal damage.

C. Take manual control of Pressurizer Level Control, CV-1235, and stabilize Pressurizer level due to automatic valve control malfunction.

D. Trip HPI pump, trip reactor, and go to EOP 1202.001, Reactor Trip, due to loss of seal injection at power.

**Answer:**

A. Trip HPI pump and isolate Letdown by closing Letdown Isolation, CV-1221, due to indications of loss of suction.

**Notes:**

"A" is the correct response due to indications of loss of suction to HPI pump per section 2 of 1203.026.
"B" and "C" are incorrect because the correct action is to take manual control and close valves after pump tripped.
"D" is incorrect since the reactor does not have to be tripped for these conditions.

**References:**

1203.026, Chg. 011

**History:**

New for 2005 RO exam, but not used.
Selected for 2007 RO Exam.
Selected for 2011 RO 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. **Isolate letdown by performing one of the following:**

   - Close Letdown Coolers Outlet (CV-1221)
   - Close both of the following on C18:
     - Letdown Coolers Outlet (RCS) (CV-1214)
     - Letdown Coolers Outlet (RCS) (CV-1216)

> **NOTE**
>
> - With HPI pump off, ICW cooling of RCP seals should provide adequate time to isolate and correct leaks, providing no pre-condition exists, such as excessive RCP shaft sleeve leakage. HPI can provide necessary makeup for normal operations or plant shutdown.
> - Reactor Coolant Pump and Motor Emergency (1203.031), Attachment A can be used as an aid to assess seal parameters.

3. **IF HPI pump is stopped,**
   **THEN verify RC pump seals are being cooled by ICW.**

   A. IF ICW to RCP seals is **NOT** available,
   THEN perform Reactor Coolant Pump and Motor Emergency (1203.031), "Simultaneous Loss of Seal Injection and Seal Cooling Flow" section.

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

**QID:** 0164 **Rev:** 1 **Rev Date:** 05/03/91 **Source:** Repeat **Originator:** J. Cork

**TUOI:** ANO-1-LP-RO-DHR **Objective:** 23 **Point Value:** 1

**Section:** 4.2 **Type:** Generic APE

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

**Description:** Knowledge of the operational implications of the following concepts as they apply to a Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation.

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

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

**Group:** 1 **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** A

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

An outage is in progress with the following conditions:

- The RCS is drained to 371.5 feet as indicated by RCS hot leg level.
- Decay heat removal flow becomes erratic.
- Indicated decay heat removal flow is ~2500 gpm.

Which action should be taken first to prevent a Loss of Decay Heat?

A. Reduce decay heat removal flow until flow has stablized.

B. Start the Standby Decay Heat pump

C. Raise RCS level.

D. Raise decay heat removal flow.

**Answer:**

A. Reduce decay heat removal flow until flow has stablized.

**Notes:**

(A) is correct. With erratic flow, actions should be taken to stabilize the flow by throttling flow back.
(B) is incorrect. Due to vortexing starting the standby pump would only damage it also.
(C) is incorrect. Although this may be necessary in the long term, the immediate response to the condition is to reduce DH flow.
(D) is incorrect. This will make the condition worse instead of better.

**References:**

1203.028, Loss of Decay Heat Removal, Chg. 023

**History:**

Taken from Exam Bank QID # 3070
Used in 98 RO Re-exam
Used on 2002 RO/SRO exam.
Used on 2004 RO/SRO exam.
Selected for the 2008 RO exam.
Selected for 2011 RO Exam.

SECTION 4 – LOSS OF DH FLOW DUE TO VORTEXING

**INSTRUCTIONS**

1. **IF DH flow is lost,**
   **THEN stop the running DH pump.**

2. **IF DH flow is erratic,**
   **THEN attempt to stabilize flow by throttling one or more of the following:**

| | P-34A | P-34B |
|---|---|---|
| LPI Block | CV-1401 | CV-1400 |
| Decay Heat Cooler Outlet | CV-1428 | CV-1429 |
| Decay Heat Cooler Bypass | CV-1433 | CV-1432 |

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

4. **IF loss of inventory is indicated by rising level or annunciator alarm in any of the following,**
   - RB sump
   - Aux building sump
   - Aux Building Equipment Drain Tank (T-11)
   - Dirty Waste Drain Tank (T-20A/B)
   - Train A or Train B Decay Heat Room Flood (K09-C4, D4)
   - Process Monitor Radiation High (K10-B2), Decay Heat Loop A/B (RI-3809, RI-3810)

   **THEN GO TO "Loss of Inventory or DH Removal System Leak <20 gpm" section of this procedure:**

**NOTE**

- Minimum Height of Water to Avoid Vortex Formation vs. Decay Heat Flow is provided by Decay Heat Removal Operating Procedure(1104.004), Attachment B.
- SPDS Safety System Diagnostic Instrumentation display may be helpful in monitoring DH pump (P-34A or P-34B).

5. **IF throttling per step 2. stabilized DH flow,**
   **THEN exit this procedure.**
   **OTHERWISE, continue with this section.**

6. **IF running,**
   **THEN stop the affected DH pump (P-34A or P-34B).**

(continued)

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

**QID:** 0095 **Rev:** 2 **Rev Date:** 04/23/2002 **Source:** Direct **Originator:** JCork

**TUOI:** ANO-1-LP-RO-EOP10 **Objective:** 6 **Point Value:** 1

**Section:** 4.2 **Type:** Generic AOPs

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

**Description:** Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS.

**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:** | 3 |
| **Group:** | 1 | **SRO Imp:** | 3.9 | **SRO Select:** | Yes | **Taxonomy:** | C |

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

Given:

- RCS pressure has dropped to approximately 1500 psig.
- RB pressure has risen to 5 psig.
- CETs are approximately 500 degrees F.

Which of the following best describes the effects on the ICW system or the components it cools?

a. All RCPs must be secured due to loss of motor cooling.

b. All RCPs must be secured due to loss of subcooling margin.

c. ICW Booster pumps are protected by opening of bypass valve.

d. ICW pumps must be secured due to isolation of SW to ICW coolers.

**Answer:**

a. All RCPs must be secured due to loss of motor cooling.

**Notes:**

"A" is correct, RCP's are secured due to loss of motor cooling due to ESAS actuation channels 5&6 on RB pressure.
"B" is incorrect since SCM is approximately 100 degrees F.
"C" is incorrect, ICW Booster pumps should be secured during isolation of suction and discharge.
"D" is incorrect, SW will be isolated to coolers but this doesn't require securing ICW pumps.

**References:**

1202.012, Repetitive Tasks, RT-10, Chg. 009

**History:**

Developed for 1998 SRO exam
Revised after 9/98 exam analysis review.
Used in 98 RO Re-exam
Modified for use in 2002 RO/SRO exam.
Selected for 2011 RO Exam.

## VERIFY PROPER ESAS ACTUATION

**NOTE**

Obtain Shift Manager/CRS permission prior to overriding ES.

1. **Verify BWST T3 Outlets open:**
   - CV-1407
   - CV-1408

   A. **IF** BWST T3 Outlet (CV-1407 or CV-1408) fails to open,
   **THEN** override **AND** stop associated HPI, LPI, and RB Spray pumps until failed valve is opened.

2. **Verify SERV WTR to DG1 and DG2 CLRs open:**
   - CV-3806
   - CV-3807

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
      - P32B
      - P32C
      - P32D

   2) **IF** SCM is **not** adequate,
   **THEN** check elapsed time since loss of adequate SCM
   **AND** perform the following:

      a) **IF** $\leq$ 2 minutes have elapsed,
      **THEN** trip all RCPs:
         - P32A
         - P32B
         - P32C
         - P32D

(3. CONTINUED ON NEXT PAGE)

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

**QID:** 0824 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** New **Originator:** S. Pullin

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

**Section:** 4.2 **Type:** Generic APEs

**System Number:** 027 **System Title:** PZR Pressure Control System (PZR PCS)Malfunctions

**Description:** Ability to determine and interpret the following as they apply to the Pressurizer pressure control malfunctions: Tech-Spec limits for RCS pressure.

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

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

**Group:** 1 **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** K

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

Given:
-Plant at 100% power.
-PZR Spray valve, CV-1008, fails to 30% open.
-RCS pressure 2120 psig and slowly dropping.

When would the Technical Specification for RCS Pressure, Temperature, and Flow DNB Surveillance limits required to be entered?

A. 2135 psig

B. 2120 psig

C. 2105 psig

D. 2080 psig

**Answer:**

D. 2080 psig

**Notes:**

"D" is the correct answer per the Cycle 23 COLR for 4 pump operation.
"A", "B", "C" are the setpoints for PZR heater operation.

**References:**

Tech Specifications 3.4.1, Cycle 23 COLR, p. 29

**History:**

New for 2011 RO Exam.

**RCS Pressure, Temperature, and Flow**
**DNB Surveillance Limits**

(Limit is referred to by Technical Specification 3.4.1)

| | Four-Pump Operation | Three-Pump Operation | Two-Pump Operation |
|---|---|---|---|
| Minimum RCS Hot Leg Pressure (psig) Note 1 | 2082.2 | 2081.2 Note 4<br>2120.4 Note 5 | 2118.1 |
| Maximum RCS Hot Leg Temperature (°F) Note 2 | 602.6 | 602.9 | 603.15 |
| Minimum RCS Total Flow ($Mlb_m$/hr) Note 3 | 143.36 Note 6<br>138.01 Note 9 | 106.46 Note 7<br>102.45 Note 9 | 70.64 Note 8<br>67.96 Note 9 |

Note 1 -- Using individual indications P1021, P1023, P1038 and P1039 (or equivalent) from the plant computer.

Note 2 -- Using individual indications T1011NR, T1014NR, T1039NR, T1042NR, T1012, T1013, T1040 and T1041 or averages TOUTA, XTOUTA, TOUTB, XTOUTB, TOUT, XTOUT from the plant computer.

Note 3 -- Using indication WRCFT (or equivalent) from the plant computer, and can be linearly interpolated between these values provided the $T_{ave}$ versus Power level curve is followed.

Note 4 -- Applies to the RCS loop with two RCPs operating.

Note 5 -- Applies to the RCS loop with one RCP operating.

Note 6 -- For $T_{cold}$ = 556.57 °F.

Note 7 -- For $T_{cold}$ = 556.3 °F.

Note 8 -- For $T_{cold}$ = 556.1 °F.

Note 9 -- For $T_{cold}$ = 580 °F.

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

**QID:** 0345 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** D. Slusher

**TUOI:** ANO-1-LP-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 determine or interpret the following as they apply to a SGTR: Magnitude of rupture.

**K/A Number:** EA2.13 **CFR Reference:** 43.5 / 45.13

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 3.1 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 3.7 | **SRO Select:** | Yes | **Taxonomy:** | Ap |

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

Given:

- "A OTSG N-16 Trouble" is in alarm.
- Letdown flow is 70 gpm.
- Makeup flow is 90 gpm.
- Seal injection flow is 32 gpm.
- Seal Bleedoff Flow is 1.5 gpm per RCP
- Pressurizer level is constant.

What is the approximate primary to secondary leak rate?

a. ~6 gpm

b. ~12 gpm

c. ~46 gpm

d. ~52 gpm

**Answer:**

c. ~46 gpm

**Notes:**

Makeup flow + seal injection flow = 122 gpm
Letdown flow + seal bleedoff flow = 76 gpm
122 - 76 = 46 gpm, "c" is correct, the other answers are probable math errors

**References:**

1104.002, Chg. 070

**History:**

Developed for 1999 exam.
Selected for 2011 RO Exam.

3.0 DESCRIPTION

3.1 The makeup and purification system performs various functions in direct support of the reactor coolant system. The MU system is used to:

- Control RCS coolant inventory.
- Purify and recirculate reactor coolant.
- Maintain proper boron concentration of RCS.
- Add chemicals for RCS chemistry control:
  - Lithium Hydroxide (LiOH)
  - Hydrazine (N2H4)
  - Hydrogen (H2)
- Degasify the RCS.
- Supply the following:
  - Seal injection water for reactor coolant pumps
  - Borated makeup water to core flood tanks
  - HPI to RCS upon ESAS actuation

As stated in the scope, many of these functions are covered in other operating procedures.

3.2 To perform the functions listed above, the makeup and purification system uses a combination of the following components with related valves and instrumentation.

- Letdown Coolers (E-29A, E-29B)
- RCP Seal Return Coolers (E-26A, E-26B)
- Purification Demineralizers (T-36A, T-36B)
- Makeup Filters (F-3A, F-3B)
- Seal Injection Filter (F-2)
- Makeup Prefilter (F-25)
- N-16 expansion tank
- Makeup Tank (T-4)

**NOTE**

Throughout this procedure the terms makeup pumps and HPI pumps are used interchangeably.

- Makeup Pumps (P-36A, P-36B, P-36C)

3.3 RCS letdown comes from the cold leg of the B reactor coolant loop and goes through the letdown coolers. The coolers lower the letdown temperature so that (1) the demineralizer resin will not be damaged by high letdown temperature, (2) the reactor coolant pump seals will not be damaged, and (3) the reactor coolant will not flash to steam when pressure is reduced. Letdown then goes through an expansion tank which causes the transport time to be delayed to allow for the decay of Nitrogen 16. This reduces the radiation levels outside of the reactor building.

Letdown then exits the reactor building and goes through Letdown Coolers Outlet (CV-1221). This valve serves as an automatic isolation valve upon ESAS actuation or a high letdown temperature condition. On the outlet of CV-1221, letdown passes through the Failed Fuel Monitor (RE-1237). This monitor, through the use of a spectrometer, can determine by detecting a specific isotope if there is fuel failure present in the core.

Letdown then goes through the flow control valves and to the demineralizers. Upstream of the demineralizers there is a tap-off for the makeup pre-filter and a tap-off for the Boronometer (AE-1290). The makeup pre-filter is designed to remove large particulate from the RCS (i.e. after a crud burst or during startup following an outage. The boronometer is normally not in-service but if used, will give the operator an indication of the approximate boron concentration in the RCS.

The purification demineralizers are mixed-bed demineralizers that remove ionic impurities. However, the beds also do a minimal amount of mechanical filtration. The normal system alignment is to have only one demineralizer in-service at a time with the other one in standby. The standby demineralizer is placed into service as needed to reduce the lithium concentration of the RCS.

After the demineralizers, letdown passes through the Letdown 3-Way Valve (CV-1248). This valve directs letdown to the clean liquid radwaste system or to the makeup tank. Normally it is lined up to the makeup tank. Prior to reaching the makeup tank, letdown passes through the makeup filters. These filters are used to remove suspended material that passes through the demineralizer and to prevent resin fines from entering the remainder of the purification system and the RCS. Just upstream of the filters, makeup condensate, boric acid, hydrazine, and lithium hydroxide are added to the system as needed.

Next letdown enters the makeup tank. This tank works as an expansion or surge volume tank for the maximum expected expansion and contraction of the RCS during power transients. Hydrogen is introduced into the letdown system via the makeup tank. Hydrogen is used to scavenge oxygen from the RCS and as a means to pressurize the makeup tank to provide NPSH to the makeup pumps which come next in the letdown flow path.

Exhibit A displays the upper pressure limit for the makeup tank pressure/level relationship. Assuming the makeup tank vent on low level interlock fails, the limit conservatively allows 20 minutes to isolate the makeup tank upon initiation of HPI. Exceeding this limit reduces the time margin, raising the risk of injecting hydrogen into the HPI pump suction.

There are three primary makeup pumps that are used to supply water for various functions under normal and abnormal conditions. Under normal conditions, only one pump is required to maintain normal makeup. However, under various conditions, more than one pump may be operating. The makeup pumps may take a suction from the borated water storage tank (BWST T-3) during testing or during an accident. The pumps can also take a suction from the discharge of the decay heat pumps if, after a small break LOCA, RC pressure is still high and it is necessary to take a suction on the reactor building sump. This is referred to as the piggyback mode.

After letdown goes through the makeup pumps it can go to any of the following locations:

- Normal makeup through the "D" HPI nozzle
- High pressure injection nozzles
- Reactor coolant pump seal injection

- High pressure Aux spray
- Makeup to the core flood tanks
- Recirc back to the makeup tank

3.4 During normal operation, a signal generated from pressurizer level controls the amount of makeup supplied to the RCS. However, even when the pressurizer level is such that no makeup is required, the throttle valve bypass around CV-1235 (MU-32) is throttled to provide 10 gpm at normal system pressure through the "D" HPI nozzle to keep the nozzle temperature warm enough to prevent thermal shock to the nozzle.

Upon ESAS actuation of channels 1 & 2, until operator action is taken, flow is sent into the RCS through each HPI nozzle at ~ 250 gpm per nozzle. During normal operation there is no flow through the A, B,& C nozzles.

Reactor coolant pump seal injection comes from the makeup pump, through a control valve, seal injection filter, reactor building isolation, and then to each seal injection line. RC Pump Seals Total Injection Flow (CV-1207) controls the total amount of seal injection flow to the seals. The seal injection filter removes small particulate matter that if sent to the seals could cause seal damage.

High pressure Aux spray is used to cool down and depressurize the pressurizer when reactor coolant pumps are not available and RC temperature/pressure is still too high for decay heat Aux spray.

The makeup pumps can be used to supply a source of makeup water to the core flood tanks. This operation is covered in the core flood tank operating procedure.

The makeup pumps each have a recirc line coming off the discharge of the pump with an orifice that will pass > 65 GPM back to the make up tank. This is to prevent overheating of the pumps if all pump discharge flow paths are secured. Actual required minimum pump flow is 55 gpm.

3.5 The Operating and the Standby HPI pumps should be aligned to the Operable Emergency Diesel whenever one Emergency Diesel Generator is inoperable, for longer than one hour. This is to make it easier to recover makeup and seal injection on a loss of offsite power.

## 4.0 REFERENCES

### 4.1 REFERENCES USED IN PROCEDURE PREPARATION

4.1.1 Unit 1 Technical Specifications

4.1.2 Unit 1 Safety Analysis Report

4.1.3 STM 1-4

4.1.4 Setpoint Index (M-517)

4.1.5 Bechtel Functional Description and Logic Diagram M-417

4.1.6 P&IDs M-230 and M-231

4.1.7 Gas Binding Of HPI Pumps (ANO-88-07958)

4.1.8 Makeup & Purification System HPI Flow Test & Check Valve Verification (Special Work Plan 1409.129)

4.1.9 Safety System Diagnostic Instrumentation (DCP 86-1006)

4.1.10 RCP Controlled Bleedoff Check Valve (DCP 86-1070)

4.1.11 HPI Pump Minimum Flow Requirements (Calc. 88E-0086-04 Rev. 1)

4.1.12 HPI Pump Disch Check Valves Back Flow Test (ANO-90-04790).

4.1.13 Vlv Stroke Requirements During CSD or Refueling SD (ANO-90-04790)

4.1.14 ESF Cooler Operability Requirements (PEAR 90-0989)

4.1.15 Pressurizer Surge Line Thermal Stratification BAW-2127

4.1.16 Power Operated Valve Safety Analysis Stroke Times Calc.-89-E-0040-01, Calc.-V-1063-00, and Calc.-V-1047-00

4.1.17 HPI Pump $\Delta P$ limits (ANO-91-00665)

4.1.18 Evaluation of Makeup Tank Pressure Limit Curve (CR-91-0035-04)

# 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:** Yes **Difficulty:** 2

**Group:** 1 **SRO Imp:** 2.5 **SRO Select:** Yes **Taxonomy:** C

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

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-5

**History:**

New for 2011 RO Exam.

## 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 SG press is ≤ 600 psig**
   **AND**
   **other SG press is > 600 psig OR ΔP between SGs is > 150 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:** 0662 **Rev:** 0 **Rev Date:** 12/15/06 **Source:** Direct **Originator:** Possage

**TUOI:** A1LP-RO-EOP04 **Objective:** 15 **Point Value:** 1

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

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

**Description:** Knowledge of the reasons for the following responses as they apply to Loss of Main Feedwater (MFW): Actons contained in EOPs for Loss of MFW.

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

**Tier:** 1 **RO Imp:** 4.4 **RO Select:** Yes **Difficulty:** 3

**Group:** 1 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** Ap

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

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, RT-16, Chg 009
1202.004 Chg 006

**History:**

New for 2007 RO Exam, K/A 054 AK1.02
Selected for 2011 RO Exam.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| **11. <u>WHEN</u> CET temps begin to drop, <u>THEN</u> perform the following:** | |
| A. <u>IF</u> CET SCM is adequate, <u>THEN</u> maintain RCS cooldown rate ≤100°F/hr by throttling HPI and Letdown flow. | A. <u>IF</u> CET SCM is less than adequate, <u>THEN</u> maintain full HPI flow. |
| **12. <u>IF</u> EFW restoration is imminent, <u>THEN</u> GO TO step 13.** | 12. Perform one of the following:<br>A. <u>IF</u> MFW or AUX Feedwater pump becomes available, <u>THEN</u> refill dry SG (RT 16).<br>1) **GO TO step 14.**<br>B. <u>IF</u> MFW and AUX Feedwater pumps are <u>not</u> available, **<u>THEN</u> GO TO 1202.011, "HPI COOLDOWN"** procedure. |
| **13. <u>WHEN</u> EFW becomes available, <u>THEN</u> refill dry SG (RT 16).** | 13. **RETURN TO step 12.** |

**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 |
|---|---|
| **14. Check adequate primary to secondary heat transfer established <u>AND</u> perform the following:** | 14. **GO TO 1202.011, "HPI COOLDOWN"** procedure. |
| A. Check Letdown in service. | A. Unless fuel damage or RCS to ICW leak is suspected, restore Letdown flow (RT 13). |

(14. CONTINUED ON NEXT PAGE)

## FEEDING INTACT SG

1. **IF feed source is EFW,**
   **THEN perform the following:**

   A. Place affected SG(s) EFW CNTRL valves in HAND **AND** verify closed:

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

   B. Check EFW Pump (P7A or P7B) running.

      1) **IF** EFW Pump P7A is available,
      **THEN** open either EFW Pump Turbine K3 Steam Admission valve (CV-2613 or CV-2663).

      2) **IF** EFW Pump P7B is available,
      **THEN** perform the following:

         a) Verify EFW Pump P7B handswitch removed from PULL-TO-LOCK.

         b) Start EFW Pump P7B.

   C. Place affected SG(s) EFW CNTRL valves in VECTOR OVERRIDE:

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

   D. Place affected SG(s) EFW ISOL valves in MANUAL **AND** verify open (modulating valves):

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

**(1. CONTINUED ON NEXT PAGE)**

FEEDING INTACT SG

**1. (Continued)**

**CAUTION**

Excessive EFW flow can result in loss of RCS pressure control due to RCS shrinkage.

**NOTE**

Table 1 contains examples of less than adequate/excessive EFW flow.

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

1) Select Reflux Boiling setpoint for the following:

- Train A
- Train B

2) **IF** both SGs are available,
**THEN** slowly adjust EFW CNTRL valve(s) in HAND to establish ≥ 340 gpm until SG level is 370 to 410".

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

a) **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".

3) **IF** only one SG is available,
**THEN** slowly adjust associated SG EFW CNTRL valve(s) in HAND to establish ≥ 570 gpm. Do **not** throttle below 570 gpm until SG level is 370 to 410".

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

**(1. CONTINUED ON NEXT PAGE)**

**FEEDING INTACT SG**

**1. (Continued)**

4) **IF** any good 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

5) **IF** SCM becomes adequate prior to establishing SG level of 370 to 410",
**THEN GO TO step 1.F**.

6) **WHEN** SG level is 370 to 410",
**THEN** place associated EFW CNTRL valves in AUTO:

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

a) Verify SG level maintained 370 to 410".

b) Place associated EFW ISOL valves in AUTO:

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

7) 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 $\Delta T$ stable or dropping

**(1. CONTINUED ON NEXT PAGE)**

## FEEDING INTACT SG

1. (Continued)

8) **WHEN** primary to secondary heat transfer is established,
**THEN** adjust associated EFW CNTRL valve as necessary to maintain the following:

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

- Adequate SCM
- ≤100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)
- Desired cooldown rate

| **Table 1** |
|---|
| **Examples of Excessive EFW Flow Indications** |
| • SG press drops ≥ 100 psig due to EFW flow induced overcooling<br>• SCM approaching minimum adequate due to EFW flow induced overcooling<br>• EFW CNTRL valve open with associated SG level > applicable setpoint level band |

(1. CONTINUED ON NEXT PAGE)

### FEEDING INTACT SG

**1. (Continued)**

> **CAUTION**
>
> Excessive EFW flow can result in loss of SCM due to RCS shrinkage.

F. **IF** SCM is adequate,
**THEN** adjust affected SG(s) EFW CNTRL valve(s) in HAND as necessary to maintain the following:

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

- EFW flow within the following limits,
  until primary to secondary heat transfer is restored:

| Condition | Flow |
|---|---|
| Any RCP running | ≤ 450 gpm |
| All RCPs off | ≤ 200 gpm |

- Adequate SCM
- ≤ 100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)

1) 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

(1. CONTINUED ON NEXT PAGE)

**FEEDING INTACT SG**

**1. (Continued)**

2) **WHEN** primary to secondary heat transfer is established,
**THEN** adjust affected SG(s) EFW CNTRL valve as necessary to maintain the following:

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

- Adequate SCM
- ≤100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)
- Desired cooldown rate

G. **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 |

H. **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<br>CV-6690 | **TURB BYP valves** | CV-6687<br>CV-6688 |

**(1. CONTINUED ON NEXT PAGE)**

**FEEDING INTACT SG**

**1. (Continued)**

I. **WHEN** associated SG level is near mid point of applicable level band, **THEN** place associated EFW CNTRL valve(s) in AUTO:

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

1) Verify SG level maintained within applicable band:

| Condition | Level Band |
|---|---|
| Any RCP running | 20 to 40" |
| All RCPs off<br>**AND**<br>Natural Circ selected | 300 to 340" |

2) Place associated EFW ISOL valves in AUTO.

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

J. Monitor EMERGENCY FEEDWATER and EFIC alarms on K-12.

| Table 1 |
|---|
| **Examples of Excessive EFW Flow Indications** |
| • SG press drops ≥ 100 psig due to EFW flow induced overcooling<br>• SCM approaching minimum adequate due to EFW flow induced overcooling<br>• EFW CNTRL valve open with associated SG level > applicable setpoint level band |

**FEEDING INTACT SG**

2. **IF feed source is MFW or AFW,**
   **THEN perform the following:**

   A. Verify affected SG(s) Main and Low Load Feedwater Block closed:

| SG A | | SG B |
|---|---|---|
| CV-2625 | **Main Feedwater Block** | CV-2675 |
| CV-2624 | **Low Load Feedwater Block** | CV-2674 |

   B. Place affected SG(s) Startup valve in HAND **AND** close.

| SG A | | SG B |
|---|---|---|
| CV-2623 | **Startup** | CV-2673 |

   C. Verify at least one Condensate pump running.

   D. Verify affected SG(s) Main Feedwater Isolation open:

| SG A | | SG B |
|---|---|---|
| CV-2680 | **Main Feedwater Isolation** | CV-2630 |

   E. Verify Feedwater Pumps DISCH Crosstie open (CV-2827).

   F. IF AUX Feedwater Pump (P75) is available,
      THEN verify Aux Feedwater Pump (P75) running.

      1) **GO TO step 2.H.**

   G. IF MFW pump is available,
      THEN verify MFW pump running.

**(2. CONTINUED ON NEXT PAGE)**

**FEEDING INTACT SG**

**2. (Continued)**

H. **IF** SCM is **not** adequate,
**THEN** establish **AND** maintain SG levels 370 to 410" within 25 minutes of SCM loss using Startup valve H/A stations in HAND.

1) **IF** SCM becomes adequate prior to establishing 370 to 410",
**THEN GO TO step 2.I.**

2) **IF** any good SG press drops below 720 psig,
**THEN** perform the following:

a) Bypass MSLI by momentarily placing SG Bypass toggle switch on each EFIC cabinet Initiate module in BYPASS.

- C37-3
- C37-4
- C37-1
- C37-2

3) **WHEN** SG level is 370 to 410"
**THEN** check primary to secondary heat transfer in progress indicated by all of the following:

- T-cold tracking associated SG T-sat (Fig. 2)
- T-hot tracking CET temps
- T-hot/T-cold $\Delta$T stable or dropping

4) **WHEN** primary to secondary heat transfer is established,
**THEN** adjust affected SG(s) Startup valve(s) to maintain the following:

| **SG A** | | **SG B** |
|---|---|---|
| CV-2623 | **Startup** | CV-2673 |

- Adequate SCM
- $\leq$100°F Tube-to-Shell $\Delta$T (tubes colder)
- $\leq$ 60°F Tube-to-Shell $\Delta$T (tubes hotter)
- Desired cooldown rate

**(2. CONTINUED ON NEXT PAGE)**

**FEEDING INTACT SG**

**2. (Continued)**

| **CAUTION** |
|---|
| Excessive FW flow can result in loss of SCM due to RCS shrinkage. |

l. **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 ≤ $0.2 \times 10^6$ lbm/hr
- Adequate SCM
- ≤ 100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)

1) 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) **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

**(2. CONTINUED ON NEXT PAGE)**

**FEEDING INTACT SG**

**2. (Continued)**

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<br>CV-6690 | **TURB BYP valves** | CV-6687<br>CV-6688 |

5) **WHEN** SG level is ≥ 20",
**THEN** perform the following:

a) **IF** any RCP is running,
**THEN** perform the following:

(1) **WHEN** SG level is near midpoint for 20 to 40",
**THEN** place associated Startup valve(s) in AUTO:

| **SG A** | | **SG B** |
|---|---|---|
| CV-2623 | **Startup** | CV-2673 |

(2) Verify SG level maintained 20 to 40".

**(2. CONTINUED ON NEXT PAGE)**

**FEEDING INTACT SG**

**2. (Continued)**

b) **<u>IF</u> <u>no</u>** RCPs are running,
**<u>THEN</u>** continue to raise SG level as necessary to establish and maintain 300 to 340", while maintaining the following:

- MFW Loop flow ≤ $0.2 \times 10^{6}$ lbm/hr until primary to secondary heat transfer is established
- Adequate SCM
- ≤ 100°F Tube-to-Shell ΔT (tubes colder)
- ≤ 60°F Tube-to-Shell ΔT (tubes hotter)
- Desired cooldown rate

**END**

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

**QID:** 0552 **Rev:** 0 **Rev Date:** 3-30-05 **Source:** Direct **Originator:** S.Pullin

**TUOI:** A1LP-RO-EOP08 **Objective:** 11 **Point Value:** 1

**Section:** 4.2 **Type:** Generic EPEs

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

**Description:** Knowledge of EOP entry conditions and immediate action steps.

**K/A Number:** 2.4.1 **CFR Reference:** 41.7 / 45.7 / 45.8

**Tier:** 1 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 2

**Group:** 1 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** K

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

Which of the following would cause entry into 1202.008, Blackout?

A. All 6900V busses de-energized, 4160V busses A1 and A2 de-energized

B. All 4160V busses de-energized

C. All 6900V busses de-energized, all 4160V busses de-energized except A4 bus

D. All 6900V busses de-energized

**Answer:**

B. All 4160V busses de-energized

**Notes:**

"B" is correct, this is the entry condition for 1202.008.
"A", "C" and "D" are incorrect, these are not entry conditions for 1202.008. .

**References:**

1202.008, Chg. 010

**History:**

New for 2005 RO exam
Selected for 2011 RO Exam.

## ENTRY CONDITIONS

**NOTE**

Throughout this procedure, harsh containment values in brackets [ ] shall be used, where provided, if either of the following criteria are met:

- Average RB Temp >200°F
- RB Radiation Level >$10^5$ R/hr

- All 4160V buses de-energized

**QID:** 0553 **Rev:** 0 **Rev Date:** 3/3/05 **Source:** Direct **Originator:** J.Cork

**TUOI:** A1LP-RO-EOP07 **Objective:** 9 **Point Value:** 1

**Section:** 4.2 **Type:** Generic APEs

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

**Description:** Knowledge of the parameters and logic used to assess the status of safety functions, such as Reactivity control, Core cooling and heat removal, Reactor coolant system integrity, Containment conditions, Radioactivity release control, etc.

**K/A Number:** 2.4.21 **CFR Reference:** 41.7 / 43.5 / 45.12

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

**Group:** 1 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** Ap

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

A Degraded Power event occurred.
Both EDG's are supplying associated ES buses.
You are directed to actuate MSLI for both SGs and verify proper EFW actuation and control per RT-6.

Which of the following would be a verification of primary to secondary heat transfer per RT-6?

A. Core exit temperature 600 °F and rising slowly.

B. T-hot/T-cold delta T 55°F and rising slowly.

C. T-cold 545°F dropping slowly and SG pressures 990 psig dropping slowly.

D. Core exit temperature 595 °F rising slowly with T-hot 580°F dropping slowly.

**Answer:**

C. T-cold 545°F dropping slowly and SG pressures 990 psig dropping slowly.

**Notes:**

"C" has the correct relationship of parameters per RT-6, all others have incorrect parameters.

**References:**

1202.012, Chg. 009, RT-6
1202.013, Rev. 4, Fig. 2

**History:**

New for 2005 RO exam. K/A 056 AK1.01
Selected for 2011 RO Exam.

## VERIFY PROPER MSLI AND EFW ACTUATION AND CONTROL

**8. <u>IF</u> SCM is adequate,**
**<u>THEN</u> perform the following:**

| **<u>CAUTION</u>** |
|---|
| Excessive EFW flow can result in loss of SCM due to RCS shrinkage. |

| **<u>NOTE</u>** |
|---|
| Table 2 contains examples of less than adequate/excessive EFW flow. |

A. Verify EFW CNTRL valves operate to establish and maintain applicable SG level band per Table 1.

1) **<u>IF</u>** EFW flow is less than adequate
**<u>OR</u>**
EFW flow is excessive,
**<u>THEN</u>** 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 SG level band reached.
- Maintain sufficient EFW flow to ensure SG level is either stable **<u>OR</u>** rising until applicable level band is reached.

**9. <u>IF</u> a Condensate pump is running,**
**<u>THEN</u> dispatch an operator to close bad SG Startup FW Control Valve Bypass while continuing:**

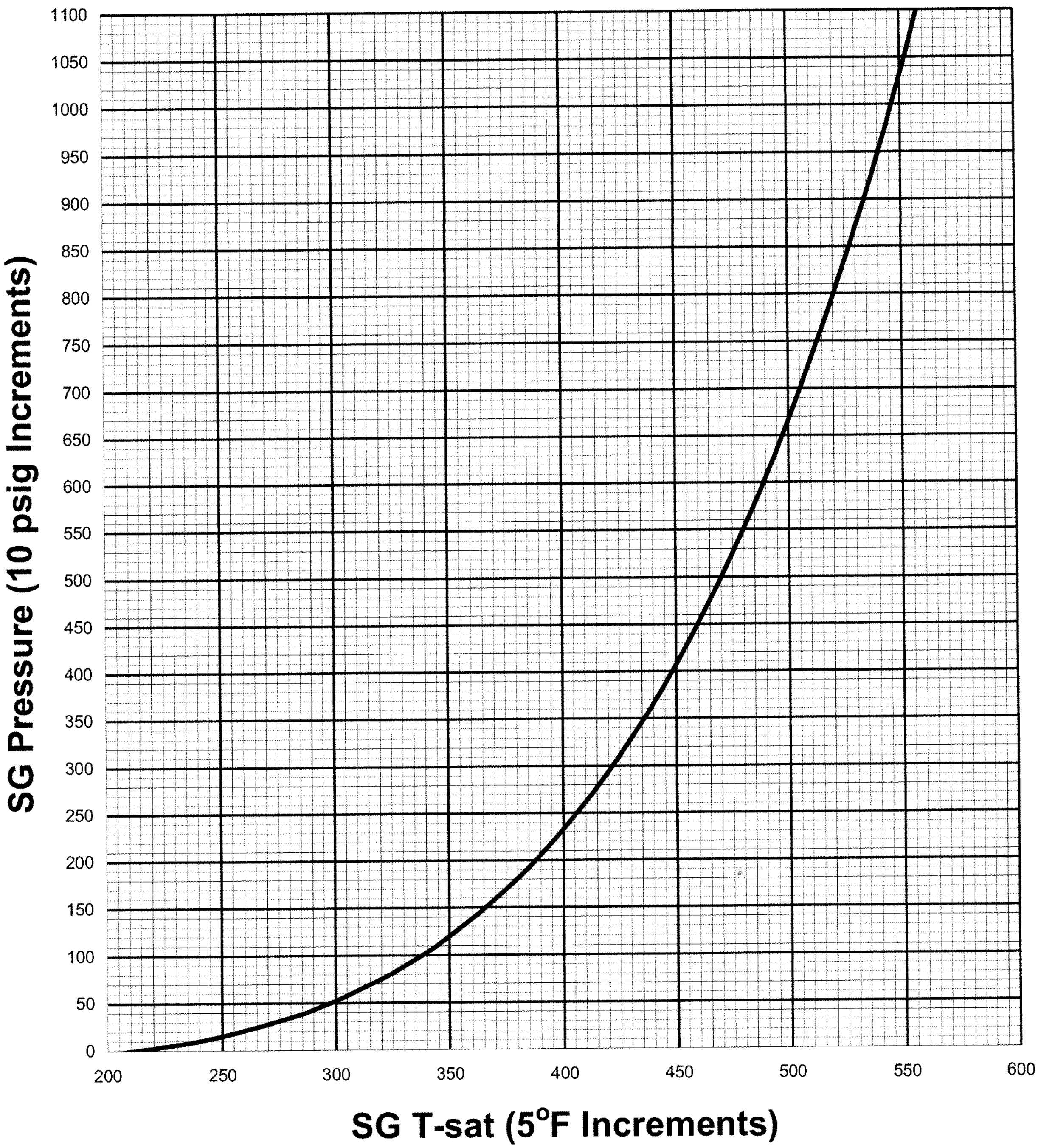
| **<u>SG A</u>** | **<u>SG B</u>** |
|---|---|
| FW-19 | FW-20 |

**10. <u>IF</u> all RCPs are off,**
**<u>THEN</u> 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 $\Delta$T stable or dropping

**11. Monitor EMERGENCY FEEDWATER and EFIC alarms on K-12.**

# FIGURE 2
# SG Pressure vs T-sat



**QID:** 0850 **Rev:** 0 **Rev Date:** 06/25/11 **Source:** Direct **Originator:** J.Cork

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

**Section:** 4.2 **Type:** Generic AOP's

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

**Description:** Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: That a loss of AC has occurred.

**K/A Number:** AA2.15 **CFR Reference:** 43.5 / 45.13

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 3.8 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 1 | **SRO Imp:** | 4.1 | **SRO Select:** | Yes | **Taxonomy:** | C |

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

Inverters are aligned with Inverter Y-25 supplying 120 Volt Vital AC Panel RS-4 and Inverter Y-22 supplying 120 Volt Vital AC Panel RS-2.

What would shifting the manual output transfer switch (S-2) on the Y-25 Inverter to the "System Output To Y-22" position do?

A. Power RS-2 from Y-25

B. De-energize RS-4

C. Parallel RS-2 and RS-4

D. Cause Y-25 to shift to alternate source

**Answer:**

B. De-energize RS-4

**Notes:**

B is correct, Y-25 can only power one Vital AC panel at a time. Shifting the manual output transfer switch on Y-25 to the Y-22 position will remove power from RS-4.
A is incorrect, although this sounds logical, this will only occur if the output transfer switch on Y-24 is also selected to Y-25.
C is incorrect, although this sounds possible, the output transfer switches on Y-22, Y-24, and Y-25 are "break before make" and thus it is not possible to parallel the panels.
D is incorrect, there is another manual transfer switch which transfers Y-25 to alternate source but it is not the system output transfer switch.

**References:**

1107.003, Chg. 020

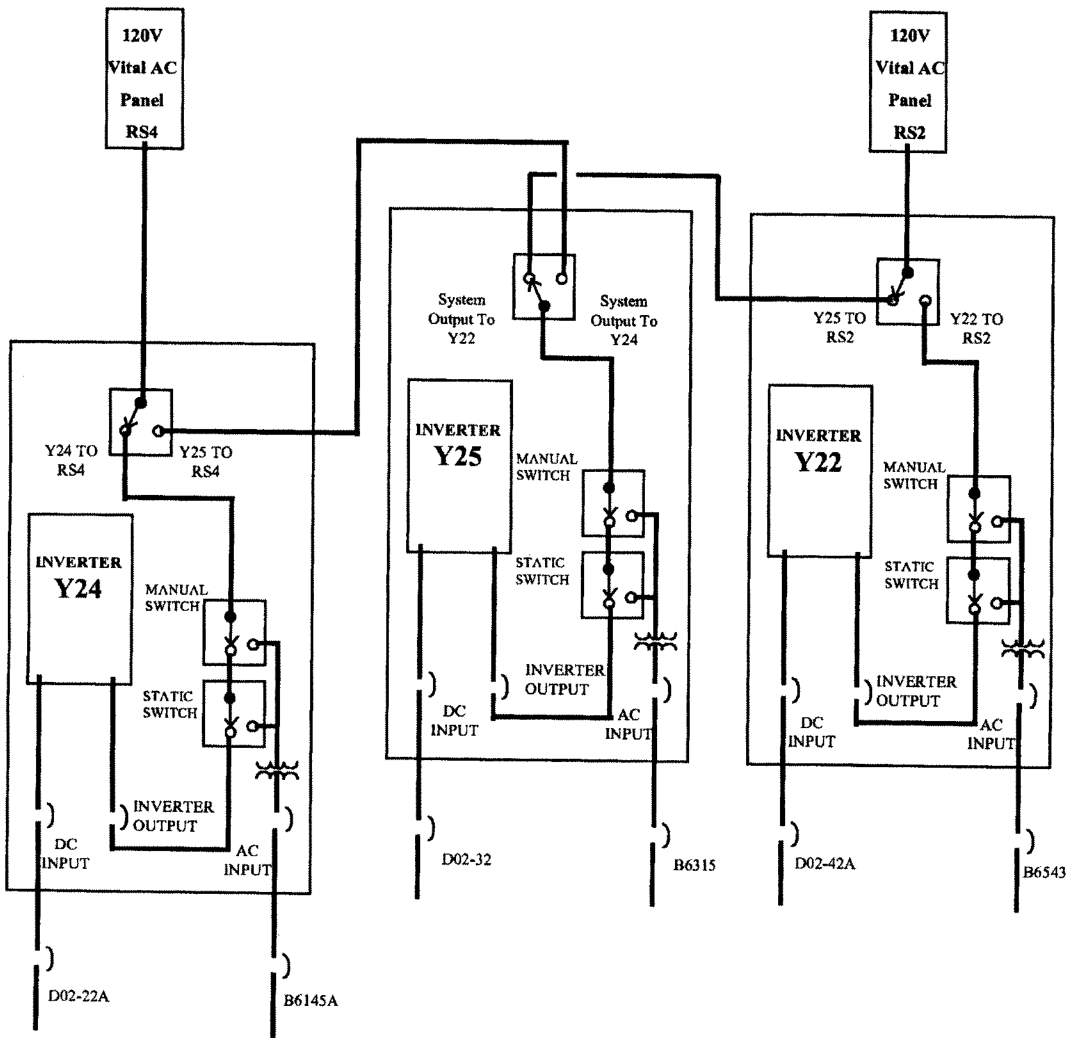
**History:**

Direct from regular exambank QID ANO-OPS1-4510.
Selected for 2011 RO Exam.

1107.003
EXHIBIT L

## Green Train Inverter One Line Diagram

Revised 11/25/96



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

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

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

**Section:** 4.2 **Type:** Generic APEs

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

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

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 3.1 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 1 | **SRO Imp:** | 3.1 | **SRO Select:** | Yes | **Taxonomy:** | C |

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

Given:
- ESAS actuation has occurred on channels 1 through 6
- SW Pump P-4A does not start, P-4B is aligned to the other SW loop
- HPI Pump P-36B is aligned to the A4 bus

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

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

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

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

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

**Answer:**

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

**Notes:**

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

**References:**

1203.030, Chg. 017
STM 1-42, Rev. 20

**History:**

New for 2005 RO re-exam.
Selected for 2011 RO Exam.

**CAUTION**

Maximum Service Water Pump flow is 8000 gpm.

5. **IF only one SW pump is operating**
**AND second SW pump can NOT be restored within ~5 minutes**
**OR as directed by CRS/SM,**
**THEN perform the following:**

A. Trip the reactor **AND** perform Reactor Trip (1202.001) in conjunction with this procedure.

B. Closely monitor SW loop pressures.

1) Verify the following control valves ~100% closed:

- Generator $H_2$ Temp Control (CV-4018)
- Main Lube Oil Temp Control (CV-4026)

C. Monitor temperatures of the following operating components/systems cooled by SW and ACW at a minimum:

- Circulating Water Pumps
- High Pressure Injection Pumps
- Condensate Pumps
- Non-Nuclear/Nuclear Intermediate Cooling Water Systems

D. Notify SM to perform the following:

1) Declare one loop of SW inoperable AND cascade Tech Specs, as described later in this procedure.

2) Review EALs.

E. Identify available SW pump:

- IF P-4A running
  **OR** P-4B running on A-3,
  **THEN GO TO step 5.F**
- IF P-4C running
  **OR** P-4B running on A-4,
  **THEN GO TO step 5.G**

(continued)

controlled by handswitch (HS-3643) located on panel C-16. HS-3643 positions are Open and Close, spring return to center type handswitch.

Valve position indication is provided above its associated HS. Power to CV-3643 is provided from vital MCC B56 breaker B-5653. B-56 can be powered from EDG1 bus B5 or EDG2 bus B6 ensuring the ACW isolation valve can be closed in the event of a single failure.

CV-3643 is provided with a "manual" / "auto" pushbutton used for valve control during an ESAS actuation. When an ESAS actuation occurs, valve position will be controlled by its associated ES signal in "auto". For valve control using their associated handswitch during an ES event, requires the "manual" pushbutton to be pushed along with the "manual" pushbutton for the SW crosstie interlocked with the isolation valve CV-3643. If only one channel is actuated, then the associated crosstie valves must be selected to "manual" for ACW isolation valve operation from its handswitch. For valve operation during an ESAS event (channels 1 & 2) all four SW Loop crossties must be selected to "manual" for valve operation from its associated handswitch. ACW valve CV-3643 will remain open when operating in a 3 pump configuration. For additional information on cross tie and ACW isolation valve controls refer to 1305.006 Integrated ES System Test Attachments 3 and 4.

Located directly down stream of the loop isolation valves are pressure transmitters, which provide pressure indication in the control room on panel C-09 and on the SPDS computer. SW Loop I pressure indication is provided by PI-3608, SW Loop II indication by PI-3607 and ACW Loop pressure by PI-3606 on C09.

### 2.3.10 Lube Water Supply to CW Pumps

(Refer to Figure 42.11)

Each SW loop provides cooling water to the Circ Water Pump lube water system through a two-inch line located downstream of the loop isolation valves. Each line is provided with a manual isolation valve, SW-3A (loop I) and SW-3C (loop II) for isolating flowpath to their associated CW pumps. SW Loop I provides cooling water to P-3A and P-3B, while SW Loop II supplies P-3C and P-3D.

Each line is equipped with a flow orifice (FO-3675 for loop I and FO-3676 for loop II) to limit flow to the CW pump lube water system following a seismic event, where the potential for this line to rupture or break could occur. In addition to the flow orifices these lines have been seismically qualified to address a NRC Service Water diagnostic team's loss of service water concern.

System cross connect valves, SW-4 and SW-96 allow for the two systems to be cross-connect when operating with only one SW Loop in service. This condition will normally occur during a refueling outage when SW Loop maintenance is required. During normal plant operation, SW-4 is open and SW-96 is closed.

Downstream of the cross connect piping the cooling water supply line branches into two separate flow paths (one for each CW pump).

These chillers are required to be operable in accordance with Tech Spec 3.7.10. T.S. 3.3.16 and T.S. 3.7.10 state that two independent circuits of the Control Room Emergency Air Conditioning and Isolation system shall be operable whenever reactor building integrity is required.

For additional information on 2VE-1A/B refer to Unit 2 STM 2-47-3, "Control Room Ventilation.

### 2.3.21 M/U / HPI Pump Lube Oil and Room Coolers

(Refer to Figures 42.01A & 42.01B and Table 42.1).

The service water system provides cooling water to the Makeup / HPI pump lube oil coolers, E-39A/B/C and room coolers, VUC-7A/B/C. HPI denotes High Pressure Injection mode.

Lube oil provides both cooling and lubrication to the HPI pump bearings and gearbox assembly. Oil is circulated through the lube oil cooler by an attached lube oil pump, (P-88A/B/C) during pump operation or using a motor driven lube oil pump, (P-64A/B/C) during pump startup or shutdown. SW flowing through the tubes of the coolers removes heat from the lube oil as it flows through the shell of the cooler.

Cooling the lube oil provides two benefits, which improve the operation of the makeup pumps. These benefits are:

* Ensures an adequate oil film between the bearings and journals by maintaining oil viscosity. Cooler oil will have a higher viscosity, which improves this oil film thus reducing bearing wear.
* Allows cooler lube oil to serve as a cooling medium in removing the heat from the pump gears and bearings.

During refueling outage 1R8 all three lube oil coolers were replaced with larger, stainless steel, horizontal shell and tube type heat exchangers. The new coolers were installed due to Bio-macro fouling in the old coolers, which reduced the thermal performance to a failure point. This change was accomplished by DCP-87-1016. For additional information on cooler change out, see generic letter 89-13.

SW loop I supplies P-36A lube oil cooler, E-39A and room cooler VUC-7A. SW loop II supplies P-36C lube oil cooler, E-39C and room cooler VUC-7C. SW loop I or loop II can supply cooling water to P-36B lube oil cooler, E-39C and room cooler VUC-7B.

Service water to P-36B components is provided through a four-valve arrangement.

Dual isolations are provided for each loop with the SW supply for P-36B in between the two set of isolation valves. SW to P-36B will be aligned to the SW loop associated with the operating or standby makeup pump.

Example: P-36A is the standby makeup pump with P-36B in service. SW will be aligned to P-36B lube oil cooler and room cooler from SW loop I, as shown on figure provided below.

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

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

**Section:** 4.2 **Type:** Generic APEs

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

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

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

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

**Group:** 1 **SRO Imp:** 2.8 **SRO Select:** Yes **Taxonomy:** C

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

Given:
- Unit One Instrument Air pressure has been degrading for approximately 30 minutes.
- Unit Two Instrument Air pressure is steady.
- Inst. Air is being used to supply contractors using air line hoods.

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

Which of the following actions are required at this pressure?

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

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

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

D. Place RCP Seal Injection Block, CV-1206, in OVERRIDE.

**Answer:**

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

**Notes:**

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

**References:**

1203.024, Chg. 012

**History:**

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

SECTION 1 -- LOW INSTRUMENT AIR PRESSURE (≤75 PSIG)

DISCUSSION

Low IA pressure can be caused by numerous conditions. This section assumes a gradual loss of air pressure with no major malfunction of air operated equipment. Expeditious action is required to minimize the impact on air operated systems and components. For additional discussion, see Attachment D.

1.0 SYMPTOMS

1.1 IA header pressure dropping.

1.2 INST AIR HEADER PRESS LO (K12-B3) alarm

1.3 INST AIR COMPRESSOR TROUBLE (K12-C3) alarm

1.4 M-1/F-8 ΔP (K21-5) alarm

1.5 BREATHING AIR COMPRESSOR AUTOSTART (K15-B4) alarm

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

**NOTE**

- IA HDR Pressure can be monitored using PMS point P5409.
- ANO2 Instrument Air Header Supply Pressure is PMS point P3013.

3.1 Verify standby Instrument Air Compressor(s) (C-28A/B, C-2A/B) running.

3.2 Dispatch an operator to determine specific compressor, air dryer, and filter condition.

3.3 IF IA is supplying respirable air,
THEN inform RP of loss of IA pressure, and that workers must back out of work in progress and isolate the IA supply.

3.4 IF low IA header pressure is due to loss of IA on Unit 2
AND IA is crossconnected,
THEN perform the following:

3.4.1 IF Unit 1 IA header pressure drops below 60 psig,
THEN direct Unit 2 control room operators to terminate crossconnection.

3.4.2 **GO TO** step 3.7.

**QID:** 0614 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** Direct **Originator:** Cork/Pullin

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

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

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

**Description:** Knowledge of the interrelations between the (Inadequate Heat Transfer) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

**K/A Number:** EK2.2 **CFR Reference:** 41.7 / 45.7

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 4.2 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 1 | **SRO Imp:** | 4.2 | **SRO Select:** | Yes | **Taxonomy:** | k |

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

An Overheating event is in progress.
Subcooling margin remains adequate.

The running RCP should be tripped when tube to shell delta T reaches:

A. 40°F (tubes hotter)

B. 60°F (tubes hotter)

C. 40°F (tubes colder)

D. 60°F (tubes colder)

**Answer:**

B. 60°F (tubes hotter)

**Notes:**

"B" is correct per 1202.004.
The other answers are incorrect values.

**References:**

1202.004, Chg. 006

**History:**

Direct from regular exambank, QID#1610.
Selected for use in 2005 RO exam, replacement question. K/A B&W E04 EA1.2
Selected for use in 2011 RO Exam.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| 4. (Continued). | D. **IF** SG Tube-to-Shell ΔT reaches 60°F (tubes hotter)<br>**AND**<br>CET SCM is adequate,<br>**THEN** trip the running RCP.<br><br>1) Do **not** restart an RCP until SG Tube-to-Shell ΔT is ≤50°F (tubes hotter).<br><br>E. Continue efforts to restore FW/EFW per step 3 **AND** continue with this procedure. |
| **5. Check ESAS actuation alarms clear on K11.** | 5. Verify proper ESAS actuation (RT 10). |
| **6. Check adequate CET SCM.** | **CAUTION**<br>Tripping all RCPs >2 minutes after loss of adequate SCM could cause Rx core to become uncovered.<br><br>6. Check elapsed time since loss of adequate SCM<br>**AND**<br>perform the following:<br><br>A. **IF** ≤ 2 minutes have elapsed,<br>**THEN** trip running RCP.<br><br>1) **IF** adequate CET SCM is restored,<br>**THEN** restart an RCP (RT 11).<br><br>B. **IF** > 2 minutes have elapsed,<br>**THEN** leave currently running RCP on.<br><br>1) **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure.<br><br>C. Verify full HPI (RT 3).<br><br>D. Verify proper EFW actuation and control (RT 5).<br><br>E. Close both RC to Letdown Coolers E29A and E29B on C04 (CV-1213 and 1215) |

(6. CONTINUED ON NEXT PAGE)

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

**QID:** 0827 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** New **Originator:** D. Thompson

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

**Section:** 4.2 **Type:** Generic APE's

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

**Description:** Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electrical Grid Disturbances: Over-excitation.

**K/A Number:** AK1.02 **CFR Reference:** 41.4 / 41.5 / 41.7 / 41.10 / 45.8

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 3.3 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 1 | **SRO Imp:** | 3.4 | **SRO Select:** | Yes | **Taxonomy:** | K |

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

OPEN REFERENCE

Due to a grid distrubance an over-excitation condition exists on the Main Generator.
Main Generator load is 800 Mwe
Main Generator hydrogen pressure is 75 psig

What is the main operational concern with continued operation in this condition?

A. Excessive field heating

B. Excessive armature heating

C. Excessive core end heating

D. Slipping generator poles

**Answer:**

A. Excessive field heating

**Notes:**

"A" is correct, at this load over-excitation results in too much current passing through the field windings.
"B" is incorrect, excessive armature heating is a result of too much torque on the machine.
"C" is incorrect, this would be the result of under-excitation.
"D" is incorrect, this is the result of too weak of a field to maintain grid synchronization.

**References:**

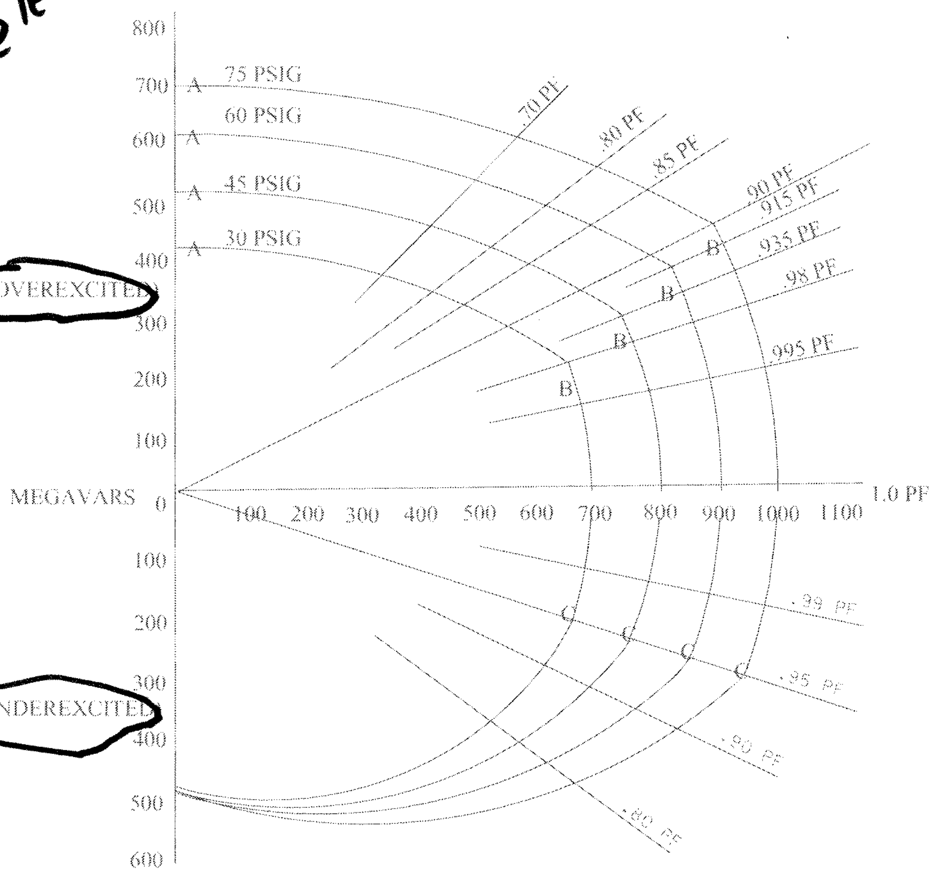
STM 1-30, Rev. 22

**History:**

New for 2011 RO Exam.

Delete for handout

FIGURE 30.14: MAIN GENERATOR CAPABILITY CURVE




HYDROGEN INNER-COOLED TURBINE GENERATOR

1002.6 MVA, 0.90 PF, 22 KV, 3 PHASE, 60 HZ

1800 RPM, 0.58 SCR, 75 PSIG

CALCULATED CAPABILITY CURVE

(AT RATED VOLTAGE)

CURVE AB LIMITED BY FIELD HEATING

CURVE BC LIMITED BY ARMATURE HEATING

CURVE CD LIMITED BY ARMATURE CORE END HEATING

# ANO 2011
# RO Questions
# Tier 1 Group2

**QID:** 0494 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Direct **Originator:** NRC
**TUOI:** A1LP-RO-NI **Objective:** 12 **Point Value:** 1

**Section:** 4.2 **Type:** Generic APE

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

**Description:** Knowledge of the operational implications of the following concepts as they apply to Inoperable / Stuck Control Rods: Flux Tilt

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

**Tier:** 1 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3
**Group:** 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** C

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

Unit 1 is at 25% power and power escalation is in progress.

An asymmetric rod alarm comes in and control rod 7-3 API indicates 9% lower than the group average.

You are the CBOR and are required to monitor Quadrant Power Tilt (QPT) which has exceeded the allowed value by Technical Specifications and 1015.003A "Unit 1 Operations Logs".

Select the equipment listed in the order from MOST accurate to LEAST accurate to obtain QPT.

A. Excore NI, Full Incore System (PMS), Minimum/Backup Incore Recorders

B. Full Incore System (PMS), Excore NI, Minimum/Backup Incore Recorders

C. Full Incore System (PMS), Minimum/Backup Incore Recorders, Excore NI

D. Minimum/Backup Incore Recorders, Full Incore System (PMS), Excore NI

**Answer:**

B. Full Incore System (PMS), Excore NI, Minimum/Backup Incore Recorders

**Notes:**

"B" is the correct answer, Full Incore (PMS) QPT is the most accurate for QPT, followed by Excore NI's for QPT, then last are the Minimum/Backup Incore Recorders.

**References:**

1102.004 "Power Operation", Chg. 050.
1015.003A "Unit 1 Operations Logs", Chg. 073.

**History:**

Developed by NRC for 2004 Exam.
Used on 2004 RO/SRO Exam.
Selected for 2011 RO Exam.

5.7 Quadrant power tilt (QPT) values from the full incore system (from PMS) are most accurate and preferred. Limits are contained in Unit 1 Operations Logs (1015.003A).

- If incore QPT from PMS is unavailable, QPT can be calculated using the equation supplied in Unit 1 Operations Logs (1015.003A), in section "Instructions for Logsheets."

- Excore QPT values are less accurate (NI calibration nulls any difference existing between excore detectors), but they are used and their associated limits applied if incore QPT from PMS is unavailable. All four excore NIs must be operable for valid readings.

- QPT values from the minimum/backup incore recorders (Form 1103.017B) are least accurate, but are used and their associated limits applied if incore QPT from PMS is unavailable and any excore NI is unavailable.

5.8 During extended reduced power operations, check valves may be subjected to damage due to disc rotation or chattering. Check valves in the feedwater, condensate, heater drain and extraction steam systems should be monitored for unusual noise. Any unusual indications should be reported to the Shift Manager.

5.9 Maximum FW flow to a steam generator during 3 RCP operations is $5.7 \times 10^6$ lbm/hr due to flow induced vibration concerns.

5.10 Up until end-of-cycle APSR withdrawal, the APSRs may be positioned as necessary for transient imbalance. After end-of-cycle APSR withdrawal, APSRs shall not be reinserted. (Ref. COLR) The normal steady state operating position for the gray APSRs is 30.4% withdrawn unless otherwise specified by Reactor Engineering.

5.11 Under non-steady state operating conditions, NI calibration is required daily (SR 3.3.1.2). A 50% power level change could cause nuclear instrumentation to be inaccurate by as much as 5% due to change in cold leg temperature.

5.12 Maximum steady state $\Delta$Tc with 4 RCPs running is 3°F to minimize effects on core and fuel performance and prevent exceeding steam generator feedwater and steam flow rates.

5.13 Power maneuvering requires close attention to prevent exceeding Control Rod Index Limits.

5.14 The main turbine generator shall be operated in the automatic voltage control mode when connected to the interconnected transmission system. If voltage control is removed from automatic, then the TOC and SOC shall be notified within thirty minutes of the change in voltage control and the expected duration of the change in status. (Ref. LO-HQNLO-2008-0014 CA#4)

ATTACHMENT D



| POINT | TECH SPEC LIMIT/ REFERENCE | MIN | MAX | FREQUENCY | LOGSHEET |
|---|---|---|---|---|---|
| **CONTROL ROD POSITION** | | | | | |
| NOTE<br>The Tech Spec rod limits are contained in the COLR. | | | | | |
| ROD INDEX | TS 3.2.1 | | | HOURLY | RX POWER (A8)/ALT (A21) |
| **IMBALANCE (>40%)** | | | | | |
| NOTE<br>The Tech Spec imbalance limits are contained in the COLR. | | | | | |
| INCORE IMBALANCE | SR 3.2.3.1 | -15 | +10 | 8 hours | RX POWER (A8) |
| EXCORE IMBALANCE NI-5 | SR 3.2.3.1 | -15 | +10 | 8 hours | ALT RX POWER (A21) |
| EXCORE IMBALANCE NI-6 | SR 3.2.3.1 | -15 | +10 | 8 hours | ALT RX POWER (A21) |
| EXCORE IMBALANCE NI-7 | SR 3.2.3.1 | -15 | +10 | 8 hours | ALT RX POWER (A21) |
| EXCORE IMBALANCE NI-8 | SR 3.2.3.1 | -15 | +10 | 8 hours | ALT RX POWER (A21) |
| **QUADRANT POWER TILT (>20%)** | | | | | |
| NOTE<br>• "Instructions for Logsheets" section contains contingencies for loss of the Plant Computer and loss of Power Range NIs.<br>• When QPT exceeds the steady state limits, then QPT is to be logged hourly until 12 consecutive hours have been logged within limits (ref. SR 3.2.4.1). | | | | | |
| MAX POSITIVE INCORE TILT VALUE (FULL INCORE SYSTEM) | >60%: 4.35<br>≤60%: 6.83<br>SR 3.2.4.1, COLR | | >60%: 3.9<br>≤60%: 6.7 | 8 hours | RX POWER (A8) |
| MAX POSITIVE INCORE TILT VALUE (MINIMUM/BACKUP INCORE) | >60%: 1.90<br>≤60%: 2.78<br>SR 3.2.4.1, COLR | | >60%: 1.8<br>≤60%: 2.7 | 8 hours | ALT RX POWER (A21) |
| QUADRANT WITH THE MAXIMUM POSITIVE INCORE TILT | | | | 8 hours | RX POWER (A8)/ALT (A21) |
| NOTE<br>Excore QPT limits apply only when incore QPT from PMS is unavailable. | | | | | |
| MAX POSITIVE EXCORE TILT VALUE | >60%: 1.96<br>≤60%: 4.05<br>SR 3.2.4.1, COLR | | >60%: 1.9<br>≤60%: 3.9 | 8 hours<br>hourly | RX POWER (A8)<br>ALT (A21) |
| QUADRANT WITH THE MAXIMUM POSITIVE EXCORE TILT | | | | 8 hours<br>hourly | RX POWER (A8)<br>ALT (A21) |

ATTACHMENT A



UNIT 1 OPERATIONS LOG FREQUENCY AND REQUIRED MODE

| Log | Title | Frequency | Modes Required |
|---|---|---|---|
| OPS-A1 | Unit 1 Inside AO Logsheet | 12 Hours | All |
| OPS-A2 | Unit 1 Outside AO Logsheet | 12 Hours | All |
| OPS-A3 | Unit 1 WCO Logsheet | 12 Hours | All |
| OPS-A4 | Unit 1 Weekly Miscellaneous Equipment Logsheet | Weekly | All |
| OPS-A5 Deleted (formerly Unit 1 Switchboard Logsheet) | | | |
| OPS-A6 | CBO Turbine Logsheet | 12 Hours | Modes 1—4 |
| OPS-A7 | CBO Reactor Logsheet | 12 Hours | Modes 1—4 |
| OPS-A8 | Reactor Power History, Tilt, and Imbalance Log | Hourly | Modes 1 and 2 |
| OPS-A9 | RCP Drinking Bird Log | 12 Hours | Modes 1—4 |
| OPS-A10 Deleted (formerly Reactor Building Sump Drain Log) | | | |
| OPS-A11 | Quench Tank Fillrate Log | At each drain | Modes 1—4 |
| OPS-A12 | Startup Boiler Logsheet | 12 Hours [1] | Startup Boiler in operation |
| OPS-A13 | CBO Cold Shutdown Log | 12 Hours | Modes 5 and 6 |
| OPS-A14 | Deleted (formerly SPING Process Monitor Log) | | |
| OPS-A15a | DG1 Logsheet | See below. [2] | DG is Running |
| OPS-A15b | DG2 Logsheet | | |
| OPS-A16 | RCS Level and Temperature Contingency Log | 15 Minutes | See below[3] |
| OPS-A17 | Chronological Logsheet | N/A | N/A |
| OPS-A18 Deleted (formerly AO Cold Shutdown) | | | |
| OPS-A19 | Cold Shutdown RCS Level Log | Hourly | Modes 5 and 6 |

[1] Startup Boiler parameters must be checked at least every 3 hours.

[2] At 30 and 60 minutes after start, and hourly thereafter, and as specified by the operating procedure.

[3] RCS drained ≤376.5 ft., <u>and either</u> LT-1195 <u>or</u> LT-1198 is OOS,
<u>or</u>
RV head is on <u>and</u> CET indication not available in Control Room.

ATTACHMENT A



| Log | Title | Frequency | Modes Required |
|---|---|---|---|
| OPS-A20 | CBO Weekly Logsheet | Weekly | Modes 1—4 and see below.[4] |
| OPS-A21 | Alternate RX Power History Tilt and Imbalance Logsheet | Hourly | Modes 1 and 2 and Plant Computer (PMS) OOS. |
| OPS-A22 | EFIC S/G Level and Pressure Crosscheck (PMS) | 12 Hours | Modes 1—4 take OPS-A22, or OPS-A23. If Plant Computer (PMS) is OOS, then must take OPS-A23. |
| OPS-A23 | Alternate EFIC S/G Level and Pressure Crosscheck | | |
| OPS-A24 | Tech Spec Channel Crosscheck (PMS) | 12 Hours | Modes 1—4 and in Mode 5 with any control rod drive trip breaker in the closed position and CRD System capable of rod withdrawal[5], take OPS-A24 or OPS-A25. If Plant Computer (PMS) is OOS, then must take OPS-A25. |
| OPS-A25 | Alternate Tech Spec Channel Crosscheck | | |
| OPS-A26 | Control Room Checklist - RCS ≥200°F | N/A | Modes 1—4 |
| OPS-A27 | Control Room Checklist - Cold Shutdown | N/A | Modes 5 and 6 |
| OPS-A28 Deleted (formerly Freeze Protection) | | | |
| OPS-A29 Deleted (formerly Unit 1 Switchyard) | | | |
| OPS-A30 | Alarm Panel Check | Weekly | All |
| OPS-B45 | Switchyard | Weekly [6] | All |

[4] Modes 1—4 <u>and</u> within 7 days prior to entering Mode 4 during plant startup.

[5] In Mode 5 only certain parameters need be taken. See Attachment E for details.

[6] Take the Switchyard Log on the same day of the week as much as practical. May be postponed due to inclement weather. Unit 1 and Unit 2 alternate taking OPS-B45.

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

**QID:** 0665 **Rev:** 0 **Rev Date:** 12/15/06 **Source:** Direct **Originator:** Possage

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

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

**System Number:** 028 **System Title:** Pressurizer Level Control Malfunction

**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:** | 1 | **RO Imp:** | 4.4 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 4.7 | **SRO Select:** | Yes | **Taxonomy:** | Ap |

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

Given:

Plant operating at 100% power.
The following annunciators come into alarm simultaneously.
PZR LEVEL LO LO (K09-A3)
PZR LEVEL LO (K09-D3)
LRS-1001 on C04 indicates 0 inches.
LIS-1002 on C04 indicates 0 inches.
Pressurizer levels on SPDS also indicates 0 inches (L1001 and L1002).

Which action should be taken for the given condition?

A. Isolate letdown by closing CV-1221, Letdown Coolers Inlet Isolation valve.

B. Adjust CV-1235, Pzr Level Control Valve, in hand to maintain previous slope of MUT level recorder.

C. Verify proportional Pressurizer Heater Controls hand/auto station in hand and full on to maintain RCS pressure.

D. Verify at least one Makeup Pump recirc valve CV-1300 or CV-1301 is open to ensure minimum recirc flow for the M/U Pump.

**Answer:**

B. Adjust CV-1235, Pzr Level Control Valve, in hand to maintain previous slope of MUT level recorder.

**Notes:**

"B" is correct. A loss of PZR level indication will effect the level control valve, CV-1235. The operator will adjust CV-1235 to maintain the same makeup flow that was present before the event. However the only chart recorder that is available is the MUT level recorder so the operator will maintain the previous slope that was present on the recorder.
"A" is incorrect. The operator should not make any adjustment to letdown because there is no way to monitor pressurizer level. The goal is to keep all flow stable so as not to cause an upset that cannot be monitored.
"C" is incorrect. Pressurizer heaters may or maynot be cutoff on low level interlock of 55 inches, but in either case there is no need to fire the heaters at full on.
"D" is incorrect. Although MUP flow rate changed as a result of the pressurizer level failure, there is no reason for minimum flow rates to be approached due to this condition, and therefore no need to be concerned with the recirc flow path.

**References:**

1203.015 Chg. 016

SECTION 7 -- LOSS OF ALL PRESSURIZER LEVEL INDICATION

## INSTRUCTIONS

1. **Take manual control of CV-1235.**

2. **Adjust makeup flow with CV-1235 to maintain the previous slope of Makeup Tank Level recorder (LR-1248).**

3. **Contact maintenance to attempt to restore any Pressurizer level indication possible.**

4. **Refer to TS 3.3.15.**

5. **Refer to "RCS Pressure, Temperature and Flow DNB Surveillance Limits" of the ANO1 COLR (TS 3.4.1).**

6. **Contact Ops Manager.**

> **CAUTION**
>
> Pressurizer heater cutoff interlock at 55" will not work if all Pressurizer level transmitters are failed. A reactor trip can uncover Pressurizer heaters.

7. **Within one hour, commence a slow power reduction (~10%/hr) per Power Reduction and Plant Shutdown (1102.016), "Power Reduction to ≥25%" section.**

8. **IF Pressurizer heaters are interlocked off due to Pressurizer level failed low, THEN contact maintenance to defeat heater cutout interlock.**

   A. Contact System Engineering AND refer to Temporary Modifications EN-DC-136.

9. **Monitor for indications of a low Pressurizer level.**

   A. IF a Pressurizer heater breaker trips
   OR annunciator PZR HEATER GROUND FAULT (K09-E3) alarms,
   THEN raise makeup flow at CV-1235 to raise Pressurizer level.

      1) For any tripped breaker, refer to "Reclosing Tripped Individual Load Supply Breakers" section of Electrical System Operations (1107.001).

(continued)

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

**QID:** 0819 **Rev:** 0 **Rev Date:** 5/16/2011 **Source:** New **Originator:** J. Cork

**TUOI:** A1LP-RO-EOP06 **Objective:** 10 **Point Value:** 1

**Section:** 4.2 **Type:** APEs

**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: Pressure at which to maintain RCS during S/G cooldown.

**K/A Number:** AA2.16 **CFR Reference:** 43.5 / 45.13

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 4.1 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 4.3 | **SRO Select:** | Yes | **Taxonomy:** | Ap |

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

OPEN REFERENCE

Given:
- A OTSG has a 20 gpm tube leak
- Reactor was tripped in accordance with 1203.023, Small Steam Generator Tube Leak
- Per 1202.001, CRS has transitioned to 1202.006, Tube Rupture
- RCS pressure 1700 psig
- RCS Tavg 540°F
- PZR level 65"

RCS cooldown is in progress and the CRS has directed you to control RCS pressure in accordance with 1202.006.

Which of the following actions should you take?

A. Turn on PZR heaters to raise Subcooling Margin.

B. Open PZR Spray to lower RCS pressure.

C. Open Turbine Bypass Valves to raise cooldown rate.

D. Initiate HPI to raise PZR level.

**Answer:**

B. Open PZR Spray to lower RCS pressure.

**Notes:**

The trainee should know that RCS pressure is maintained low within the limits of Figure 3 during a cooldown with a OTSG tube leak to reduce primary-to-secondary leakage.
"B" is correct since RCS pressure could be reduced by 300 psig and still maintain adequate SCM per EOP Figure 3.
"A" is incorrect, although to the untrained eye it appears SCM is close to the limit, RCS pressure should be reduced to limit leakage.
"C" is incorrect, this will reduce RCS temperature but it will raise SCM and lead to increased leakage.
"D" is incorrect, although the EOP directs initiation of HPI to maintain PZR level, this is only done if level is below 55".

**References:**

1202.006, Chg. 011
1202.013, Rev. 4, Fig. 3

**History:**

New for 2011 RO Exam.

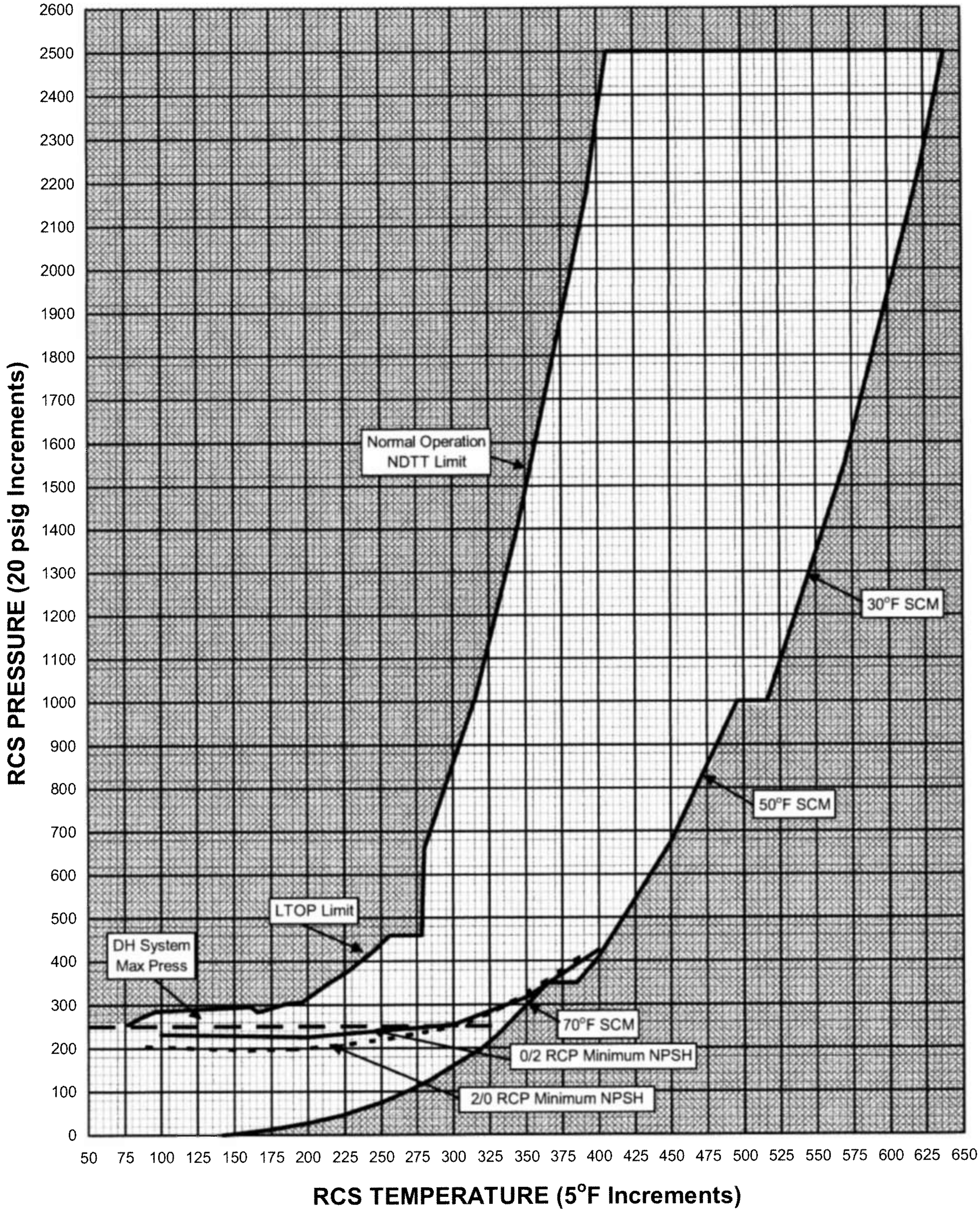
| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|

**NOTE**

PZR cooldown rate limits **do not** apply during SGTR.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| **14. Operate Pressurizer Heaters AND Pressurizer Spray valve (CV-1008) to maintain RCS press low within limits of Figure 3.** | 14. Verify ERV Isolation open (CV-1000) **AND** cycle ERV (PSV-1000). |
| A. **WHEN** RCS press is <1700 psig, **THEN** bypass ESAS. | |
| **15. Stabilize PZR level ≥55" as follows:** | |
| A. Adjust Pressurizer Level Control setpoint to 100". | |
| B. **IF** HPI is in service, **THEN** adjust HPI flow as necessary to maintain PZR level ≥55" **AND** RCS press low within limits of Figure 3. | B. **IF** necessary to maintain PZR level ≥55", **THEN** initiate HPI (RT 2). |
| **16. Verify OTSG N-16 monitors selected to GROSS.** | |

# FIGURE 3
## RCS Pressure vs Temperature Limits



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

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

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

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

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

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

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 3.2 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 2 | **SRO Imp:** | 3.5 | **SRO Select:** | Yes | **Taxonomy:** | K |

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

Reactor power is 90% and generated megawatts is 800.

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

A. 360 Mwe

B. 340 Mwe

C. 45% Reactor Power

D. 50% Reactor Power

**Answer:**

A. 360 Mwe

**Notes:**

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

**References:**

1105.004, Chg. 023

**History:**

Taken from Exam Bank QID # 4
Used in 98 RO Re-exam
Selected for use in 2005 RO exam, replacement question. K/A A01 AK2.2
Selected for 2011 RO Exam.

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

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

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

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

6.21 UNIT MASTER IN TRACK (K07-A1) alarms on any of the following conditions:

- Reactor trip
- Runback in effect
- Cross limits in effect
- Breakers 5114 and 5118 open (generator output)
- SG/RX Demand in HAND
- Reactor Demand in HAND
- Diamond Panel in MANUAL
- Turbine Control in TURBINE MANUAL or OPER AUTO.
- Both Feedwater Demand Loop A and Feedwater Demand Loop B in HAND.

**QID:** 0828 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** New **Originator:** J. Cork

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

**Section:** 4.2 **Type:** B&W EPEs/APEs

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

**Description:** Knowledge of the operational implications of the following concepts as they apply to the (Loss of NNI-X): Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of NNI-X).

**K/A Number:** AK1.3 **CFR Reference:** 41.8 / 41.10 / 45.3

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

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

Plant is at 100% power and the ATC notices a loss of NNI indication on C13.

Which of the following NNI power losses requires a manual reactor trip?

A. NNI-X AC

B. NNI-X DC

C. NNI-Y AC

D. NNI-Y DC

**Answer:**

A. NNI-X AC

**Notes:**

Answer "A" is correct. In accordance with Loss of NNI Power AOP, the only single loss of NNI power requiring a manual Reactor Trip is NNI-X.
The other choices contain actions within the AOP and some combinations require a manual Reactor Trip but only the loss of NNI-X alone requires a trip.

**References:**

1203.047, Chg. 000-01-0

**History:**

New for 2011 RO Exam.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|

**NOTE**

- MU Tank level recorder is inoperable on loss of NNI X AC.
- Pressurizer Level Control valve (CV-1235) and RC Pump seals Total INJ Flow valve (CV-1207) fail as follows:
  - * both fail to 50% on loss of NNI X AC and NNI X DC
  - * both fail to 50% on loss of NNI X DC only
  - * CV-1207 fails closed on loss of NNI X AC only
  - * CV-1235 failure position is indeterminate on loss of NNI X AC only
- Automatic Pressurizer Heater, Spray, and ERV controls are inoperable.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| **6. <u>IF</u> only NNI X power is lost, <u>THEN</u> perform the following:** | 6. **RETURN TO step 1.** |
| A. <u>IF</u> NNI X AC is lost, <u>THEN</u> trip the Rx <u>AND</u> perform **1202.001, "REACTOR TRIP"** in conjunction with this procedure. | |
| B. Verify NNI Y-powered instruments selected on C03, C04, and C13. | |
| C. Open BWST Outlet to OP HPI pump (CV-1407 or 1408). | |
| D. Close RCS Makeup Block (CV-1233 or 1234). | |
| E. Operate HPI Block (CV-1220 or 1285) associated with OP HPI pump to maintain PZR level 90 to 110". | |

(6. CONTINUED ON NEXT PAGE)

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

**QID:** 0829 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** Modified **Originator:** S. Pullin

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

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

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

**Description:** Knowledge of the interrelations between the (Emergency Diesel Actuation) 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:** | 4.0 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 3.8 | **SRO Select:** | Yes | **Taxonomy:** | C |

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

Diesel Generator #2 is running to supply A4 for maintenance on breaker A409, A2-A4 Feeder Breaker.
Low reactor coolant system pressure causes a reactor trip and ESAS actuation.

What will the ES Electrical response be?

A. A-3 and A-4 powered from SU #1, both diesel generators running unloaded.

B. A-3 and A-4 powered from SU #1, Diesel Generator # 2 tripped, Diesel Generator # 1 running unloaded.

C. A-4 powered from Diesel Generator #2, A-3 powered from SU #1, Diesel Generator # 1 running unloaded.

D. A-3 powered from Diesel Generator #1, and A-4 powered from Diesel Generator #2.

**Answer:**

C. A-4 powered from Diesel Generator #2, A-3 powered from SU #1, Diesel Generator # 1 running unloaded.

**Notes:**

"C" is correct, the #2 EDG output breaker will remain closed due to maintenace on A409.
"A" is incorrect, electrical response should not be normal response for an ESAS, #2 EDG will be powering A4.
"B" is incorrect, nothing should trip #2 EDG.
"D" is incorrect, this is normal response for a LOOP but A3 will be powered from SU#1.

**References:**

STM 1-32-2, Rev. 0

**History:**

Modified QID #349
Modified for 2011 RO Exam.

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

**QID:** 0349 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** D. Slusher

**TUOI:** ANO-1-LP-RO-ELEC **Objective:** 11J **Point Value:** 1

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

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

**Description:** Knowledge of the interrelations between the (Emergency Diesel Actuation) 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:** CFR: 41.7 / 45.7

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

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

Diesel Generator #1 is running for a surveillance test.
Low reactor coolant system pressure causes a reactor trip and ESAS actuation.

What will the ES Electrical response be?

PARENT

A. A-3 and A-4 powered from SU #1, both diesel generators running unloaded.

B. A-3 and A-4 powered from SU #1, Diesel Generator # 1 tripped, Diesel Generator # 2 running unloaded.

C. A-3 powered from Diesel Generator #1, A-4 powered from SU #1, Diesel Generator # 2 running unloaded.

D. A-3 powered from Diesel Generator #1, and A-4 powered from Diesel Generator #2.

**Answer:**

A. A-3 and A-4 powered from SU #1, both diesel generators running unloaded.

**Notes:**

"A" is correct, electrical response should be the normal response for an ESAS.
"B" is incorrect, nothing should trip #1 EDG.
"C" is incorrect, the #1 EDG output breaker should open on an ES signal.
"D" is incorrect, both busses should be powered from SU #1.

**References:**

STM 1-32, Rev. 33

**History:**

Used in 1999 exam.
Modified from ExamBank, QID# 453.
Selected for 2010 RO/SRO exam

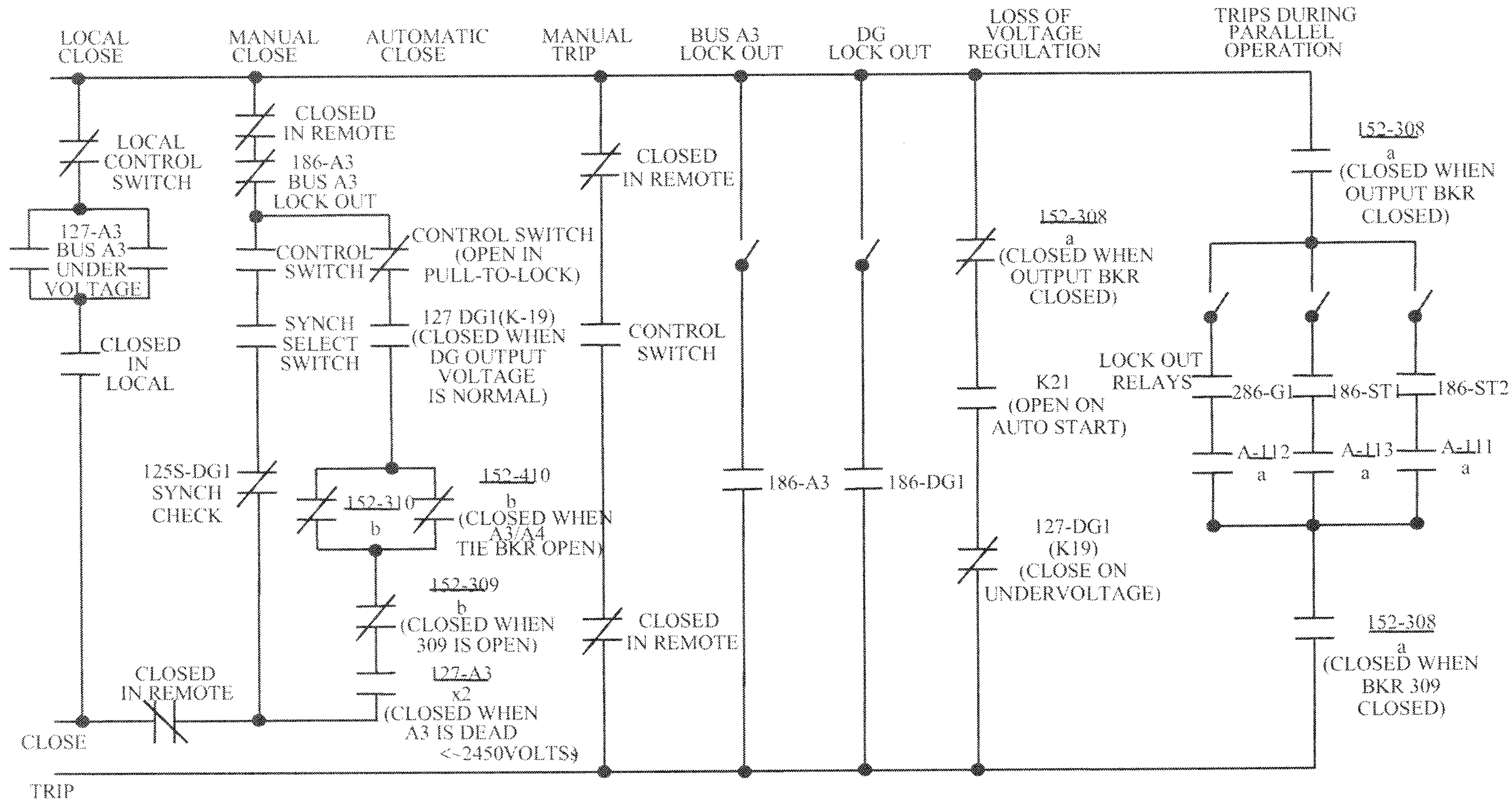



FIGURE 32-2-26: DIESEL GENERATOR OUTPUT BREAKER A308/A408

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

**QID:** 0290 **Rev:** 0 **Rev Date:** 9/4/99 **Source:** Direct **Originator:** E Wentz

**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:** Knowledge of the interrelations between the (Inadequate Subcooling Margin) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

**K/A Number:** EK2.2 **CFR Reference:** 41.7 / 45.7

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 4.3 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 1 | **SRO Imp:** | 4.3 | **SRO Select:** | Yes | **Taxonomy:** | K |

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

During a large break LOCA, RCP's have been secured due to loss of Subcooling Margin.

Which one of the following indicates that the reactor core is covered?

A. SPDS automatically switches from the ATOG display to the ICC display.

B. The A Hot Leg temperature indicator indicates superheated conditions.

C. The RCS is saturated as indicated by Core Exit Thermocouples.

D. ICCMDS display indicates voids in the Reactor Vessel head and hot legs.

**Answer:**

C. The RCS is saturated as indicated by Core Exit Thermocouples.

**Notes:**

"C" is correct, the RCS is saturated as indicated by Core Exit Thermocouples. "A" and "B" are incorrect because they both indicate core uncovery based on superheated conditions. "D" is incorrect because fuel level is below the level displays. If a level existed then maybe the fuel is covered.

**References:**

1202.002, Chg. 006

**History:**

Used in 1999 exam. Direct from ExamBank, QID# 2259.
Used on 2004 RO/SRO Exam.
Selected for 2011 RO Exam.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| **16. Check RCS temp remains either:**<br><br>**<580°F T-hot with any RCP on**<br>**<u>OR</u>**<br>**<610°F CET temp with all RCPs off.** | 16. **GO TO 1202.004, "OVERHEATING"** procedure. |
| **17. <u>IF</u> CET temps are superheated**<br>**<u>AND</u>**<br>**moving away from the saturation line, <u>THEN</u> GO TO 1202.005, "INADEQUATE CORE COOLING" procedure.** | |
| **18. <u>IF</u> cause of loss of adequate SCM is corrected**<br>**<u>AND</u>**<br>**CET SCM is adequate**<br>**<u>AND</u>**<br>**primary to secondary heat transfer is in progress, <u>THEN</u> GO TO 1202.001, "REACTOR TRIP" procedure.** | 18. Perform the following:<br><br>A. <u>IF</u> an uncontrolled RCS cooldown is occurring due to HPI/break flow, regardless of SG status, <u>**THEN**</u> **GO TO Small Break LOCA Cooldown (1203.041)**.<br><br>B. <u>IF</u> primary to secondary heat transfer is **<u>not</u>** established, <u>**THEN**</u> **RETURN TO step 9.**<br><br>C. <u>IF</u> PZR level is rising without a corresponding rise in RCS temp or press, <u>**THEN**</u> perform the following:<br><br>1) <u>IF</u> all RCPs are off, <u>**THEN**</u> perform the following:<br><br>a) <u>IF</u> RCPs are available, <u>**THEN**</u> start one RCP in each loop (RT 11).<br><br>b) <u>IF</u> RCPs are **<u>not</u>** available, <u>**THEN**</u> GO TO Natural Circulation Cooldown (1203.013), Section 2, "Offsite Power Available".<br><br>2) <u>IF</u> RCPs are running, <u>**THEN**</u> GO TO Forced Flow Cooldown (1203.040). |

(18. CONTINUED ON NEXT PAGE)

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

**QID:** 0172 **Rev:** 1 **Rev Date:** 05/13/2011 **Source:** Modified **Originator:** J. Cork

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

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

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

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

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 3.0 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 3.8 | **SRO Select:** | Yes | **Taxonomy:** | C |

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

Which of the following conditions would procedurally require that a Natural Circulation Cooldown be performed?

A. A plant cooldown is required and normal offsite power is NOT available.

B. A steam generator steam leak exists following a reactor trip and P-7B EFW pump is NOT available.

C. EFW CST (T-41B) level has dropped to less than 3' during a Blackout condition.

D. A loss of offsite power has occurred and both DGs are tied to their respective busses with an adequate SCM and SU Xfmr #1 becomes available.

**Answer:**

A. A plant cooldown is required and normal offsite power is NOT available.

**Notes:**

(a.) is correct. These are the entry conditions for 1203.013 for a natural circulation cooldown.
(b.) is incorrect. These conditions do not require performing a natural circulation cooldown.
(c.) is incorrect. These conditions would require that Hot Shutdown be maintained.
(d.) is incorrect. These conditions would require that forced flow be restored for a cooldown.

**References:**

1203.013. Chg. 018

**History:**

Developed for use in 98 RO Re-exam
Selected for 2011 RO Exam.

SECTION 1 - Degraded Power

## ENTRY CONDITIONS

- Cooldown is required and normal offsite power is not available.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
| --- | --- |
| **71. Check off-site power available <u>AND</u> ≥22KV on SU1.** | 71. Perform the following: |

A. <u>**IF**</u> SU1 or SU2 L.O. **RELAY TRIP** is alarming on K02, <u>**THEN**</u> before energizing bus from affected X-FMR, correct cause IAW Annunciator K02 Corrective Action (1203.012B) while continuing with this procedure.

B. <u>**IF**</u> power is available in the switchyard

**<u>AND</u>**

SU1 is de-energized,

**<u>AND</u>**

switchyard breaker 0125 is open, <u>**THEN**</u> have dispatcher close 0125.

1) <u>**IF**</u> SU1 is available with voltage ≥22KV, <u>**THEN**</u> **GO TO step 72.**

C. <u>**IF**</u> SU1 is <u>**not**</u> available with voltage ≥22KV, <u>**THEN**</u> perform the following:

**<u>NOTE</u>**

SU2 is considered available if <u>all</u> the following conditions are met:

- SU2 voltage ≥158KV
- AUTO X-FMR energized from 500KV
- AUTO X-FMR aligned to SU2
- <u>**No**</u> Unit 2 buses powered from SU2
- SU2 V REG 3% reduction disabled

1) <u>**IF**</u> SU2 is available, <u>**THEN**</u> **GO TO step 72.**

(71. CONTINUED ON NEXT PAGE)

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

**QID:** 0237 **Rev:** 0 **Rev Date:** 11/11/98 **Source:** Direct **Originator:** JCork

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

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

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

**Description:** Ability to operate and/or monitor the following as they apply to the (EOP Rules): 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:** 3.4 **RO Select:** Yes **Difficulty:** 2

**Group:** 2 **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** K

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

It is known that elevated containment temperatures and radiation levels can affect instrument readings.

How are allowances for these effects made when parameter values are required in EOP steps?

a. Transmitters inside containment are environmentally qualified for these effects.

b. The SE uses a set of instrument tables during accident conditions.

c. SPDS displays alternate values when these conditions are reached.

d. EOP parameter values for these conditions are inside brackets.

**Answer:**

d. EOP parameter values for these conditions are inside brackets.

**Notes:**

A note above the entry conditions to the Overheating EOP states what harsh containment conditions are and that under these conditions the values in brackets [] should be used. [d] is the correct answer.
[a] Certain transmitters inside containment are environmentally qualified but this only means they will continue to function under harsh conditions, they will still be prone to instrument error.
[b] In the distant past, the SE did possess a notebook containing tables to be used during accident conditions but no longer.
[c] is a nice idea but a falsehood, SPDS does not display alternate values when harsh containment conditions are reached.

**References:**

1202.004, Chg. 006

**History:**

Developed for use in 98 RO Re-exam.
Selected for 2011 RO Exam.

## ENTRY CONDITIONS

**NOTE**

Throughout this procedure, harsh containment values in brackets [ ] shall be used, where provided, if either of the following criteria are met:

- Average RB Temp >200°F
- RB Radiation Level $10^5$ R/hr

- RCS temp rising above either:

  580°F T-hot with any RCP on

  **OR**

  610°F CET temp with all RCPs off, following a Reactor trip.

- CET temp rising above 610°F

  **AND**

  all MFW and EFW is lost during loss of adequate SCM.

- Loss of all feedwater (MFW and EFW) following a Reactor trip.

# ANO 2011
# RO Questions
# Tier 2 Group1

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

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

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

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

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

**Description:** Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation.

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

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

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

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

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

A. Loss of nuclear ICW to RCP

B. Main steam line rupture inside RB

C. Loss of RCP seal injection

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

**Answer:**

B. Main steam line rupture inside RB

**Notes:**

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

**References:**

1202.012, Chg. 009, RT-10

**History:**

Used in 1998 initial RO exam
Selected for 2005 RO re-exam.
Selected for 2011 RO Exam.

## VERIFY PROPER ESAS ACTUATION

**NOTE**

Obtain Shift Manager/CRS permission prior to overriding ES.

1. **Verify BWST T3 Outlets open:**
   - CV-1407
   - CV-1408

   A. **IF** BWST T3 Outlet (CV-1407 or CV-1408) fails to open,
   **THEN** override **AND** stop associated HPI, LPI, and RB Spray pumps until failed valve is opened.

2. **Verify SERV WTR to DG1 and DG2 CLRs open:**
   - CV-3806
   - CV-3807

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
      - P32B
      - P32C
      - 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
         - P32B
         - P32C
         - P32D

**(3. CONTINUED ON NEXT PAGE)**

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

**QID:** 0834 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** New **Originator:** S. Pullin

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

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

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

**Description:** Ability to monitor automatic operation of the RCPs, including: Seal Injection flow.

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

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

**Group:** 1 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** K

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

The ATC notices that the pushbutton green light above HS-1206 on C-04 for Seal Injection Flow Block valve is no longer illuminated.

What be the expected response of Seal Injection flow?

A. Seal injection flow will control at setpoint.

B. Seal injection flow will drop to zero.

C. Seal injection flow will rise due to flow lower than setpoint.

D. Seal injection flow will lower due to flow higher than setpoint.

**Answer:**

B. Seal injection flow will drop to zero.

**Notes:**

"B" is correct, the green indicating light being illuminated means that seal injection flow is greater than 22 gpm. CV-1206 will automatically close when flow drops below this value to prevent shock to the seals upon seal injection restoratin.
The other answers are possible indications for standard control loop operations.

**References:**

STM 1-04, Rev. 10

**History:**

New for 2011 RO Exam.

### 2.20.1 Seal Injection Flow Control Valve (CV-1207)

Seal injection flow to the reactor coolant pumps is established and maintained utilizing an air operated angle valve (CV-1207). CV-1207 is positioned by a signal derived from demanded flow (in auto) or manual valve position demand (in manual). A toggle switch is provided on the operator control station for manual valve positioning. CV-1207 fails open on loss of instrument air. This valve is provided with a manual bypass valve for situations when CV-1207 is not available.

### 2.20.2 Seal Injection Filter F-2

Close tollerances in the RCP seals necessitate the use of filtered water to prevent damage/premature failure of the seals. This filter is in service to remove particles from the seal injection prior to entering the seals.

The Seal Injection Filter is designed for 3050 psig, 200°F, 60 gpm flow with a 2 psid clean pressure drop at design flow. Normal conditions for the filter are, 2600 psig, 130°F, 32 gpm with 0 psid to 5 psid (pressure drop at 32 gpm).

Limits and precautions for the seal injection filter are: Max -ΔP 150 psid, (Replace at 25 psid), and maximum flow of 60 gpm.A manual bypass valve is provided for seal injection flow while the filter is being replaced.

The Seal Injection Filter (F-2) has a differential pressure detector (PDIS-1206) in parallel that drives an alarm "REACTOR COOLANT PUMP'S SEAL INJECTION FILTER DIFFERENTIAL PRESSURE HIGH" (K08 A-6). It alarms at greater than 25 psid.

### 2.20.3 Seal Injection Flow Block Valve (CV-1206)

Seal injection flow to the reactor coolant pumps also passes through a motor operated isolation valve (CV-1206). This valve is located in UNPPR. The valve originally was intended for closure on ES actuation. Since water is going into RB is considered "safe" it is no longer ES actuated. CV-1206 will automatically close if seal injection flow (as measured by FE-1239 located upstream of CV-1206) falls below 22 gpm. This auto closure anticipates a loss of the operating Makeup Pump. If the operating Makeup Pump fails, seal injection flow will be terminated and CV-1206 closes. This action prevents stressing the RCP seals when a Makeup Pump is restarted.

The low flow interlock can be overridden using PB-1206 which is located directly above HS-1206 on C-04. PB-1206 is an alternating action pushbutton with two backlit indications "OVRD" (override; amber) and "FLOW" (green). The "FLOW" indicator is backlit when total seal flow is greater than 22 gpm. If seal injection flow is lost, the "FLOW" indicator will be extinguished. The "OVRD" indicator is backlit if the switch is in the override condition. If in the override position, CV-1206 will not auto-close. CV-1206 can still be reopened and/or closed using HS-1206 while PB-1206 is in "OVRD". Since it is not possible to tell the state of PB-1206 except by the indicating lights, it is recommended that the button be pressed periodically to test the override lamps. This

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

**QID:** 0657 **Rev:** 1 **Rev Date:** 6/20/11 **Source:** Direct **Originator:** Cork

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

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

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

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

**K/A Number:** K2.02 **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:** C

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

"A" HPI pump is operating.
"C" HPI pump is in ES Standby.
"B" HPI pump MOD is closed on A-4.

Which of the following best describes "B" HPI pump following ES actuation?

A. "B" HPI pump will start on A-3 if "A" HPI pump fails.

B. "B" HPI pump will start on A-4 if "A" HPI pump fails.

C. "B" HPI pump will start on A-4 if "C" HPI pump fails.

D. "B" HPI pump will not auto start on either "A" or "C" HPI pump failure.

**Answer:**

C. "B" HPI pump will start on A-4 if "C" HPI pump fails.

**Notes:**

[C] is correct. With the MOD closed on A-4, the logic is looking for a failure of the "C" HPI Pump if ES actuates.

[A] is incorrect. With A-4 MOD closed the pump cannot start on A-3.
[B] is incorrect. The logic is based on "C" pump not the "A" pump.
[D] is incorrect. A failure of "C" HPI pump will cause auto start of "B" pump.

**References:**

STM 1-04, Rev. 10

**History:**

Used on 2007 RO Exam. Direct from exam bank QID# ANO-OPS1-3208.
Selected for 2011 RO Exam.

P-36B. The MOD is located in the Lower South Electrical Equipment Room. Operation of the motor operated disconnects is covered in STM 1-32 "Electrical Distribution". If the selector switch is selected to bus A-3, then P-36B is controlled on panel C-18 using HS-1242. If it is selected to bus A-4, P-36B is controlled using handswitch HS-1262 on C-16.

On panels C-16 and C-18, directly below the handswitch for P-36B, a pair of indicating lights (one red, one green) shows the status of the MOD associated with that power supply to P-36B. The red light indicates that the MOD associated with that power supply is closed and the green light indicates that it is open. However, if the MOD is open it does not mean that the circuit breaker cannot be shut. The MOD should be verified closed before the breaker is shut and the MOD should not be opened unless the breaker is opened.

### 2.15.2 HPI Pump Automatic Actuation

The "A" ("C") HPI pump is actuated by an ES Channel 1 (Channel 2) signal and will start after a 5 second time delay provided:

- "B" HPI pump is not running on A3 (A4)
- A3 (A4) is powered from either A1 (A2) or DG#1 (DG#2) and the voltage on A3 (A4) is normal

The "B" HPI pump can be actuated by either an ES Channel 1 or an ES Channel 2 signal. It will start on A3 (A4) if MOD is closed on A3 (A4) provided:

* "A" ("C") HPI pump is not running
* "B" HPI pump is not running on A4 (A3)
* Normal voltage exists on A3 (A4)
* A3 (A4) is powered by A1 (A2) or DG#1 (DG#2)
* Starts after a 7 second time delay

### 2.15.3 Primary Makeup Pumps Classification

The Makeup/HPI Pumps are 9 stage, horizontal, double volute, axial split casing design. Refer to table 4.5 for pump design and operating data. The suction and discharge of the pump are integral with the bottom half of the casing. The top case and rotating element can be removed without disturbing the main piping. The pump has specially designed bearing housings which contain a thrust bearing in the outboard end and a radial bearing in the coupling end. The radial bearing is a split sleeve journal bearing. The thrust bearing combines a split sleeve journal bearing with a pivot shoe bearing, a thrust disc and a locating ring. (Fig. 4.14)

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

**QID:** 0820 **Rev:** 0 **Rev Date:** 5/17/11 **Source:** New **Originator:** J. Cork

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

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

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

**Description:** Knowledge of the operational implications of the following concepts as they apply to the RHRS: Dilution and boration considerations.

**K/A Number:** K5.09 **CFR Reference:** 41.5 / 45.7

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

**Group:** 1 **SRO Imp:** 3.4 **SRO Select:** Yes **Taxonomy:** C

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

The plant is in Mode 5, Cold Shutdown.
Both DH loops are operable.
No RCPs are running.
The running DH pump is stopped in preparation for swapping to the other loop.

In accordance with Technical Specifications for the Reactor Coolant System, which of the following are NOT allowed at this time?

A. Allowing RCS temperature to drift up to 190°F.

B. Starting a Reactor Coolant Pump.

C. Raising MUT level using DI water.

D. Raising OTSG levels using EFW.

**Answer:**

C. Raising MUT level using DI water.

**Notes:**

Answer "C" is correct, per TS 3.4.7 the running DH pump may be stopped for less than or equal to one hour providing no addition is made to the RCS with boron concentration less than that required to maintain SDM.
Answer "A" is incorrect but plausible, with the DH pump stopped RCS temperature will drift up but this is allowed as long as it is not within 10°F of saturation temperature.
Answer "B" is incorrect but plausible, if the trainee confuses starting a RCP an idle loop with water less than SDM.
Anser "D" is incorrect but plausible, if the trainee confuses raising OTSG levels with cooling down the RCS and intrducing positive reactivity via the moderator temperature coefficient.

**References:**

Technical Specifications 3.4.7

**History:**

New for 2011 RO exam.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.7 RCS Loops – MODE 5, Loops Filled

LCO 3.4.7 One decay heat removal (DHR) loop shall be OPERABLE and in operation, and either:

a. One additional DHR loop shall be OPERABLE; or

b. The secondary side of each steam generator (SG) shall be $\geq$ 20 inches.

--------------------------------------------NOTES--------------------------------------------

1. The DHR pump of the loop in operation may be removed from operation for $\leq$ 1 hour provided:

    a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

    b. Core outlet temperature is maintained at less than or equal to a temperature which is 10°F below saturation temperature.

2. One required DHR loop may be inoperable for $\leq$ 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation.

3. All DHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

---------------------------------------------------------------------------------------------------

APPLICABILITY: MODE 5 with RCS loops filled.

LCO (continued)

The LCO provides for either SG heat removal or DHR System heat removal. In this MODE, reactor coolant pump (RCP) operation may be restricted because of net positive suction head (NPSH) limitations, and the SG will not be able to provide steam for the turbine driven feed pumps. However, to ensure that the SG(s) can be used as a heat sink, a motor driven feedwater pump is needed, because it is independent of steam. Condensate pumps, the auxiliary feedwater pump, or the motor driven emergency feedwater pump can be used. If RCPs are available, the steam generator level need not be adjusted. If RCPs are not available, the water level must be adjusted for natural circulation. The high entry point in the generator should be accessible from the feedwater pumps so that natural circulation can be stimulated. The SGs are primarily a backup to the DHR pumps, which are used for forced flow. By requiring the SGs to be a backup heat removal path, the option to increase RCS pressure and temperature for heat removal in MODE 4 is provided.

Note 1 permits the DHR pumps to be stopped for up to 1 hour. The circumstances for stopping both DHR trains are to be limited to situations where: (a) Pressure and temperature increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits; and (b) no operations are in process that would cause reduction of the RCS boron concentration.

The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when DHR forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained below saturation temperature by $\geq$ 10°F so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the steam generators are used as a backup for decay heat removal and, to ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation. For example, this may be necessary to change operation from one DHR train to the other, perform surveillance or startup testing, perform the transition to and from the DHR System, or to avoid operation below the RCP minimum NPSH limit. The time period is acceptable because the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one required DHR loop to be inoperable for a period of $\leq$ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of DHR loops from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops.

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

**QID:** 0266 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

**TUOI:** A1LP-RO-ESAS **Objective:** 20 **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 monitor automatic operation of the ECCS, including: Safety Injection Pumps.

**K/A Number:** A3.05 **CFR Reference:** 41.7 / 45.5

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

**Group:** 1 **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** C

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

Given:

- All controls are in automatic
- RCS pressure 135 psig, slowly dropping
- Reactor Building pressure is 28 psia
- "A" and "B" OTSG levels at 415 inches

Which pair of pumps should be pumping fluid as designed (not recircing)?

A. HPI pumps and LPI pumps

B. RB spray pumps and LPI pumps

C. RB spray pumps and EFW pumps

D. HPI pumps and EFW pumps

**Answer:**

A. HPI pumps and LPI pumps

**Notes:**

"A" is correct because RCS pressure is less than the shutoff head of the LPI pumps.
"B" and "C" are incorrect because RB pressure has not reached the RB spray actuation setpoint of 30 psia.
"D" is incorrect because OTSG level is above the level for reflux boiling.

**References:**

1105.003, Chg. 013
1104.004, Chg. 092

**History:**

Used in 1999 exam.
Direct from ExamBank, QID# 1780 used in class exam
Selected for 2005 RO exam, modified later as a replacement. ESFAS K/A 013 K1.06
Selected for 2011 RO Exam.

3.5 Summary of ESAS trips and functions:

| Channel No. | Action | Trip Condition | Trip Point |
|---|---|---|---|
| 1 & 2 | HP injection & diverse containment isolation | Low RCS pressure<br>High RB pressure | <1590 psig<br>>4 psig (18.7 psia) |
| 3 & 4 | LP injection, diverse containment isolation & EFW | Low RCS pressure<br>High RB pressure | <1590 psig<br>>4 psig (18.7 psia) |
| 5 & 6 | RB isolation & RB cooling | High RB pressure | >4 psig (18.7 psia) |
| 7 & 8 | RB spray | High RB pressure | >30 psig (44.7 psia) |
| 9 & 10 | RB Spray NAOH Addition | High RB pressure | >30 psig (44.7 psia) |

4.0 REFERENCES

4.1 REFERENCES USED IN PROCEDURE PREPARATION

4.1.1 Unit 1 Technical Specifications

4.1.2 ESAS Technical Manual, Volume 3 (M-1-29)

4.1.3 Engineered Safeguards Actuation System (STM-1-65)

4.1.4 Safety Analysis Report, sections 6, 7.1.1, 7.1.3, 7.4.7 and 14.2.2.5

4.1.5 Integrated ES System Test (1305.006)

4.1.6 CR-1-96-0359 ESAS Setpoint Change

4.1.7 DCP 95-5010-D101 SW Valve Replacement

4.1.8 CR-1-02-0410 ESAS Digital Channel Logic Test Module

4.1.9 CR-ANO-1-2003-0373 Digital Channel failure during testing

4.2 REFERENCES USED IN CONJUNCTION WITH THIS PROCEDURE

4.2.1 Plant Startup (1102.002)

4.2.2 Plant Shutdown and Cooldown (1102.010)

4.2.3 Decay Heat Removal Operating Procedure (1104.004)

4.2.4 Emergency Feedwater Initiation and Control (1105.005)

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.

| P-34A | | P-34B |
| --- | --- | --- |
| CV-1436 | Decay Heat P-34 Suction from BWST | CV-1437 |
| CV-1407 | 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.

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 $\Delta T$ is >200°F.

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

**QID:** 0303 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J. Cork

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

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

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

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

**K/A Number:** K1.08 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

| | | | |
|---|---|---|---|
| **Tier:** 2 | **RO Imp:** 3.6 | **RO Select:** Yes | **Difficulty:** 2 |
| **Group:** 1 | **SRO Imp:** 3.9 | **SRO Select:** Yes | **Taxonomy:** K |

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

Which of the following occurs when HPI is automatically actuated on low RCS pressure?

a. RCP Seal INJ Block Valve CV-1206 receives an open signal.

b. Makeup Tank Outlet Valve automatically closes.

c. Decay Heat Cooler Outlet to HPI pump suction CV-1276 automatically opens.

d. RCS Makeup Block Valve CV-1234 receives a close signal.

**Answer:**

d. RCS Makeup Block Valve CV-1234 receives a close signal.

**Notes:**

"D" is correct.
"A", "B", and "C" are associated with ECCS components but are incorrect because they do not receive an ES signal.

**References:**

STM1-65, Rev. 6

**History:**

Used in 1999 exam. Direct from ExamBank, QID# 979
Selected for 2011 RO Exam.

different color border. This serves to make the components associated with one functional area more identifiable.

### 4.11.2 Annunciators

Human factors enhancements have also been applied to the control room annunciators for ESAS. The annunciators are grouped according to function. These functions are actuation, bypass, and information.

There is one actuation annunciator for each ESAS channel. By their grouping, tripping of ESAS will be readily apparent by the annunciators.

## 4.12 SUMMARY OF ACTUATION OF ENGINEERED SAFEGUARDS BY ESAS

### 4.12.1 High Pressure Injection and Diverse Containment Isolation

Upon 1590 psig RCS or 4 psig RB pressure, HPI and diverse containment isolation is actuated. (Channels 1 and 2).

- High Pressure Injection Pumps start with a design pressure and flow of 3000 psig and 300 gpm. The auxiliary oil pumps will run for only approximately 20 seconds after an ES actuation to minimize oil system over-pressurization and leaking out of the oil.
- Two Diesel Generators start and come up to rated speed (900 RPM). If needed can supply 2.7 MWe to each 4160 ES bus. Electrical buses align to ensure separation of vital buses.
- HPI Block Valves open, CV-1228, 1227, 1219, 1284, 1285, 1278, 1279 and 1220, to supply the RCS with water.
- The Letdown Coolers are isolated by the closing of CV-1214, 1216 Letdown Cooler Isolation valves and CV-1221 Letdown Isolation.
- MU Pump Recirc Valves, CV-1301 and 1300 close to allow full flow to the RCS.
- The BWST Outlet Valves, CV-1407 and 1408, open to supply the MU Pumps.
- BWST Recirc Isolation Valves CV-1441 and CV-1438 will receive a close signal from their associated BWST Isolation.

- The MU Block valves; CV-1234 and CV-1233 get a close signal to isolate normal MU.
- Service Water Valves, CV-3640 3642, 3644 and 3646 will either open or close to give two independent loops, CV-3643 closes to isolate the ACW System.
- CV-1270, 1271, 1272, 1273, and 1274 close to isolate the Reactor Coolant Pumps Seal Returns.
- CV-3820 and 3811 SW Supply to ICW Coolers close to isolate and give 2 independent Service Water loops.
- CV-4803 and 4804 close the RB vent.
- CV-4446 closes to prevent the RB Sump from draining to the Auxiliary Building Sump.
- CV-1052 closes to isolate the Quench Tank and CV-1845 and 1054 close the Quench Tank sample isolations.

### 4.12.2 Low Pressure Injection and Diverse Containment Isolation

Low Pressure Injection is also initiated by the 1590 psig low RCS pressure and the 4 psig high RB pressure, these signals actuate the following equipment: (Channels 3 & 4)

- Both P34A and B start (DH Pumps).
- The LPI Block Valves open, CV-1400 and 1401.
- CV-1407 and 1408, BWST Outlet Valves open.
- BWST Recirc Isolation Valves CV-1441 and CV-1438 will receive a close signal from their associated BWST Isolation.
- CV-1053 closes to isolate the Quench Tank.
- CV-5612 and 5611 close to isolate the RB from the Fire Water System.
- CV-7403, CV-7404 and CV-7401 and CV-7402, RB purge and isolations close.
- CV-7454 and 7453, RB Air Particulate Monitor isolation is closed.
- CV-4400, RB Sump drain to the Auxiliary Sump is closed.
- CV-1667 isolates nitrogen to the Quench Tank. See Note 1 below.

NOTE 1: Credit is no longer taken for the ES function for CV-1667. N2-47 performs the function of containment isolation.

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

**QID:** 0821 **Rev:** 0 **Rev Date:** 5/17/11 **Source:** Direct **Originator:** J. Cork

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

**Section:** 3.5 **Type:** Containment Integrity

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

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

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

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

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

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

What would be the PRIMARY concern if the Quench Tank level was NOTmaintained >4000 gallons but <8300 gallons per 1103.005, Pressurizer Operations?

A. Inadequate NPSH for the Quench Tank Transfer Pump, P-44, causing damage to pump from cavitation.

B. Too much gas space in the Quench Tank would increase the potential for buildup of explosive hydrogen.

C. If the ERV opened during a transient, the rupture disk could rupture prematurely releasing steam to the Rx Bldg.

D. There would be an inadequate water trap for the Rx Bldg vent header during cold shutdown.

**Answer:**

C. If the ERV opened during a transient, the rupture disk could rupture prematurely releasing steam to the Rx Bldg.

**Notes:**

"C" is correct. Per 1103.005 quench tank level must be maintained between 4000 and 8300 gallons with a steam bubble in the pressurizer to provide sufficient quench - cooling volume for pressurizer transients.
"A" is incorrect, although this would provide NPSH this is not the reason listed in the procedure.
"B" is incorrect, although it sounds logical, this is the reason for maintaining a nitrogen atmosphere in the QT, not for level.
"D" is incorrect, although the QT is used for this reason during shutdown, this is not the reason listed in the procedure.

**References:**

1103.005, Chg. 037

**History:**

Same as QID 658, different method of asking to meet random KA.
Selected for 2011 RO exam.

5.14 With a steam bubble in the pressurizer, Quench Tank level shall be maintained > 4000 gallons and < 8300 gallons to provide sufficient quench-cooling volume for pressurizer transients.

5.15 Do not allow Quench Tank pressure to approach bursting point of its Rupture Diaphragm (PSE-1051), 100 psig.

5.16 Opening the ERV causes a localized steam release at the pilot vent valve. This is a radiation and safety hazard.

5.17 Adding $N_2$ to the Quench Tank via RB $N_2$ Supply Penetration Isol ($N_2$-47) circumvents the double-valve isolation design of the penetration. When RCS is ≤ 200°F, Containment Closure Control, Attachment G of Decay Heat Removal and LTOP System Control (1015.002) applies.

5.18 To maintain the RCS in a non water solid condition, TS 3.4.11 Bases limits pressurizer level with RCS temperature <272°F and the reactor vessel head in place to the following:

- ≤105" at RCS pressures >100 psig
- ≤150" at RCS pressures ≤100 psig

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

**QID:** 0562 **Rev:** 0 **Rev Date:** 4/5/05 **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 the bus power supplies to the following: CCW pump, including emergency backup.

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

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

**Group:** 1 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** K

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

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

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

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

C. P33A, P33B and P33C are powered from B-12, B-22 and B32 respectively.

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

**Answer:**

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

**Notes:**

"B" lists the correct power supplies, the other choices do not.

**References:**

STM 1-43, Rev. 13

**History:**

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

* Automatic level control system with makeup from the Condensate Demineralized Water System (P-9).

Each ICW Surge tanks automatic level control system consists of the following components:

* Surge tank level indication and controlling signals -: PDIS-2228 for T-37A (Non-Nuc) and PDIS-2229 for T-37B (Nuc).
* A solenoid operated three-way valve positions air-operated level control valve. The solenoid valve receives an open or close signal from their associated PDIS. CV/SV-2228 controls T-37A level and CV/SV-2229 controls T-37B level. Control valves open when T-37 level drops to ~27 inches which corresponds to a ΔP reading of about 1 psid. LCV will closes when level reaches ~75 inches which corresponds to a ΔP reading of about 2.7 psid. Manual makeup can be accomplished using attached handwheel.
* Condensate makeup line is provided with a manual isolation valve and a flow totalizer. CS-66 isolates P-9 to T-37A and CS-67 isolates T-37B. The flow indicators FI-2228 (T-37A) and FI-2229 (T-37B) were added to assist in determining the amount of makeup water to the ICW systems for monitoring system leakage.

To allow P-33B to supply system flow for either loop, a suction and discharge cross-connect valve is provided for P-33B. System operating pressure differences will cause leakage through the closed discharge cross-connect. This would cause the surge tank with the lowest pressure to overflow and drain to the floor drain if the ICW Surge Tank Cross Connect Islolation Valve (ICW-165) was not opened. Additional information on P-33B suction and discharge cross connect valves will be covered in the following section.

## 2.4 ICW PUMPS

(Refer to Figure 43.01)

The three ICW pumps, P-33A/B/C are used to supply cooling water to components cooled by the ICW system. Net positive suction head is provided by the ICW surge tanks, discussed in the previous section. The ICW pumps are located in the Main Chiller room, elevation 354' of the Turbine Bldg.

Each ICW pump is a single-stage centrifugal pump rated for 2500 gpm with a discharge head of 125 feet at 1750 rpm. The ICW pumps are driven by a 100 HP motor. Power to the ICW pumps is provided from non-vital MCC's, B12 and B22. ICW pumps are controlled using handswitches located on panel C09. Associated MCC, breaker and HS for each pump are listed below:

| Pump | Handswitch | MCC | Breaker |
|---|---|---|---|
| P-33A | HS-2230 | MCC-B12 | B-1264 |
| P-33B | HS-2231 | MCC-B22 | B-2214 |
| P-33C | HS-2232 | MCC-B22 | B-2264 |

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

**QID:** 0822 **Rev:** 0 **Rev Date:** 5/18/11 **Source:** Modified **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

**Description:** Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of CCW Pump

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

| | | | | | |
|---|---|---|---|---|---|
| **Tier:** 2 | **RO Imp:** 3.3 | **RO Select:** Yes | **Difficulty:** 3 |
| **Group:** 1 | **SRO Imp:** 3.6 | **SRO Select:** Yes | **Taxonomy:** C |

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

Given:

- 100% power
- P33B and P33C ICW pumps in service
- P33A (ICW Pump) out of service

- P33B (ICW Pump) trips

What impact would this have on plant operations, and what actions are required per 1104.028, ICW System Operating Procedure?

A. Loss of Non-Nuc ICW, open ICW cross connect valves CV-2238, CV-2239, CV-2240 and CV-2241

B. Loss of Non-Nuc ICW, close "A" to "B" cross connect valves CV-2238 and CV-2240

C. Loss of Nuc ICW, open ICW cross connect valves CV-2238, CV-2239, CV-2240 and CV-2241

D. Loss of Nuc ICW, close "A" to "B" cross connect valves CV-2238 and CV-2240

**Answer:**

A. Loss of Non-Nuc ICW, open ICW cross connect valves CV-2238, CV-2239, CV-2240 and CV-2241

**Notes:**

"A" is correct P33A supplies the Non-Nuc ICW loads, OP-1104.028 has the operator open the suction and discharge cross connect valves to supply both loops with one pump prior to reducing loads.
"B" is incorrect due to procedure has you open the valves and not close them.
"C" is incorrect due to Nuc ICW loads were never lost.
"D" is incorrect due to Nuc ICW loads were never lost.

**References:**

1104.028, Chg. 029

**History:**

Modified version of QID 787 used in 2010 RO Exam.
Modified for 2011 RO Exam.

20.0 Contingency Actions for Loss of Two ICW Pumps

**CAUTION**

Operation of one ICW pump with the cross-connect valves open will result in pump operation at runout conditions. Pump cavitation can occur and there is elevated risk for motor breaker trip until ICW loads are reduced.

20.1 Place tripped ICW pump(s) in PULL-TO-LOCK.

20.2 Open the following valves:

- ICW Pump Suction Crossconnect CV-2240
- ICW Pump Suction Crossconnect CV-2241
- ICW Pump Discharge Crossconnect CV-2238
- ICW Pump Discharge Crossconnect CV-2239

20.3 Close the following valves to isolate letdown:

- Letdown Orifice Block Bypass (CV-1223)
- Letdown Orifice Block (CV-1222)

20.4 Isolate both Letdown Coolers (E-29A and E-29B) by closing the following valve pairs from C04:

- <u>E-29A</u> HS-2216 for Letdown Cooler Inlet Valve (CV-2216) and RC to Letdown Cooler E-29A (CV-1213)
- <u>E-29B</u> HS-2217 for Letdown Cooler Inlet Valve (CV-2217) and RC to Letdown Cooler E-29B (CV-1215)

20.5 Isolate both SFP Coolers (E-27A, E-27B) by closing the following:

- SFP Clr E-27A ICW Outlet (ICW-121A)
- SFP Clr E-27B ICW Outlet (ICW-121B)

20.6 Return one Letdown Cooler to service by opening one of the following valve pairs from C04:

- <u>E-29A</u> HS-2216 for Letdown Cooler Inlet Valve (CV-2216) and RC to Letdown Cooler E-29A (CV-1213)
- <u>E-29B</u> HS-2217 for Letdown Cooler Inlet Valve (CV-2217) and RC to Letdown Cooler E-29B (CV-1215)

20.7 Verify combined ICW flow is ≤ 3100 gpm.

20.8 Establish letdown by opening Letdown Orifice Block (CV-1222).

20.9 <u>IF</u> letdown isolated on high temperature,
<u>THEN</u> perform "Recovery of Letdown Following High Letdown Temperature" section of Makeup and Purification System (1104.002).

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

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

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

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

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

**Description:** Knowledge of the effect of a loss or malfunction of the following wll have on the PZR PCS: Pressure detection systems.

**K/A Number:** K6.01 **CFR Reference:** 41.7 / 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:** 37 **SRO:** 37

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

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

A. DH suction valve CV-1050 would close.

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

C. ERV (PSV-1000) would open.

D. DH suction valve CV-1410 would close.

**Answer:**

C. ERV (PSV-1000) would open.

**Notes:**

"C" is incorrect, the low range input to the ERV comes from WR PT-1020 (ESAS), however the high setpoint is always in effect and comes from "A" RPS.
"A" is incorrect, the inputs for valve closure come from PT-1020.
"B" is incorrect, a slow failure will not trigger SASS to swap channels.
"D" is incorrect, the input for valve closure comes from ESAS PT-1041.

**References:**

STM 1-69, Rev. 14

**History:**

New for 2005 RO re-exam. K/A 010 K1.01
Selected for 2011 RO Exam.

*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.

### 3.1.1 SASS Module Operation

The NNIX and the NNIY signals for each parameter (i.e., NNIX and NNIY signal for OTSG S/U level) are supplied to the SASS module and the transfer relay. Within the SASS module the signals input to the signal conditioning board.

Signal conditioning board filters the input signal to remove noise and converts the signal to a 0 to +5 volt signal. Each signal conditioning board can receive up to 8 input signals (four pairs). A low pass filter (4 hertz) removes noise from the NNI input signal. A voltage converter changes the ±10 volts input signal to a 0 to +5 volt signal that is required by the computer board. The resulting NNIX and NNIY signals are output to the computer board.

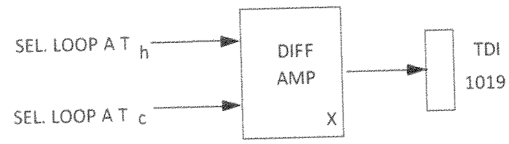
The computer board periodically (at a set scan rate) samples each input that is supplied from the signal conditioning board. When the computer is initialized (power on or reset), all inputs are set to zero and the program starts. Each input value is stored in memory on the computer board (provided no mismatch condition exists). The computer board determines if a mismatch or a trip condition is present. A mismatch condition is determined by comparing the NNIX and the NNIY signal. If the difference between the two signals exceeds a preset value, then a mismatch condition is triggered. When a mismatch condition exists, the computer performs the mismatch calculation but does not store new values of the input signal. A trip condition is determined by comparing the stored value of the NNIX signal (last value stored prior to the mismatch) to the present value of the NNIX signal. If the difference between the stored and present NNIX signals exceeds a preset value then a trip condition is triggered. The computer board outputs the trip signal, mismatch signal, and an Auto signal to the relay board. A reset button is located under the front panel. The reset switch is used to initialize the SASS modules. It is important to note that all inputs are reset and the SASS program starts. This will select all signals to the NNIX channel unless the NNIY signal is manually selected.

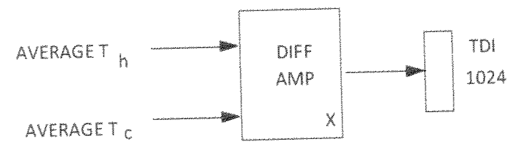
The relay board provides the relay interface between the transfer relay, the mismatch alarm and SASS automatic operation. The signals generated by the computer board are routed through the relay board to the front panel. Indications for mismatch, auto, trip X and trip Y are located on the front panel.

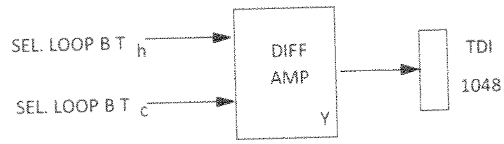
An alarm bypass module allows bypassing the mismatch alarm. Placing the switch to the bypass position will defeat the mismatch signal from the associated channel to the control room annunciator. All other functions of the SASS module remain functional. This allows another channel to actuate the control room mismatch annunciator.

### 3.1.2 SASS Logic

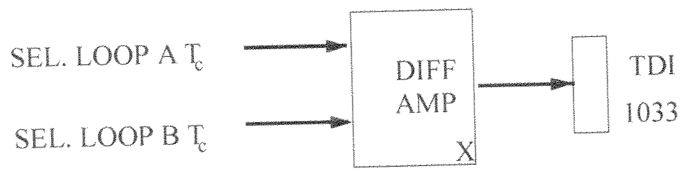
The NNI signals can fail in two general ways. The input signal can exhibit a step change or the NNIX and NNIY signals can slowly diverge. When the NNIX and NNIY signals slowly diverge, the SASS system cannot determine which signal is at fault. For this condition a mismatch alarm is generated and the operator will have to take the appropriate actions. When an input signals exhibits a step change, the SASS system will determine which signal is at fault and indicator located on C-13.




The SASS selected loop A and B cold leg temperatures supply a difference amplifier. The difference amplifier subtracts the loop B cold leg temperature from the loop A cold leg temperature. The resulting differential temperature is displayed on the cold leg differential located on C-13. The range of the indicator is -10 °F to +10 °F. The cold leg differential temperature also inputs into the ICS system. The input is used to re-ratio feedwater to the OTSG's in order to maintain the cold leg temperatures equal.




### 3.3.7 RCS Pressure Instruments

Ten pressure transmitters monitor RCS pressure. The pressure transmitters are located on instrument racks 1 and 2 inside the reactor building. The pressure taps for the pressure transmitters are located on the RCS hot leg piping on the vertical piping to the OTSGs. The pressure transmitters supply input to the Engineered Safeguards Actuation System (ESAS), Reactor Protection System (RPS), and EFIC instrument cabinets C-539 and C-540 (supplies inputs to SPDS).

Pressure transmitters PT-1021, PT-1023, PT-1038 and PT-1039 supply inputs to A, B, C, and D RPS channels, respectively. The pressure transmitters that supply RPS are Rosemount differential capacitance detectors. A and C RPS channels supply pressure recorders on C04. The range of indication is 1700 psig to 2500 psig. A and C RPS channels also supply inputs to NNIX for pressure control.

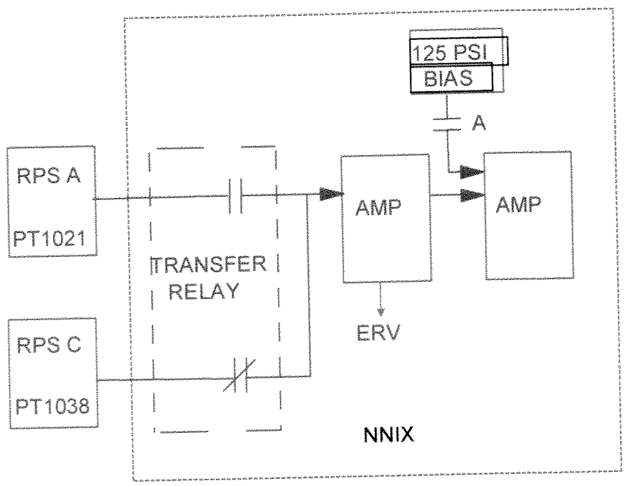
Pressure transmitters PT-1020, PT-1022, and PT-1040 provide input to A, B, and C ESAS analog channels, respectively. The pressure transmitters that supply ESAS are Rosemount differential capacitance detectors. ESAS analog channel A supplies indication on C-166 (Dasey Panel). The range of the indication is 0 psig to

2500 psig. ESAS analog channel A also inputs to NNIX for pressure control (ERV low setpoint at 400 psig). PT-1020 is also used for over pressure protection of the Decay Heat Removal System. CV-1050 will close if RCS pressure exceeds 320 psig. The interlock allows opening CV-1050 when RCS pressure is less than 290 psig.

Pressure transmitters PT-1041 and PT-1042 provide input to EFIC instrument cabinets C-540 and C-539, respectively. The pressure transmitters that supply C-539 and C-540 are Rosemount differential capacitance detectors. These transmitters satisfy REG. Guide 1.97 environmental qualification and Appendix R fire requirements (C-540). All outputs from C-539 and C-540 are buffered so that an output device failure will not affect the instrument string. C-540 supplies outputs to SPDS (Safe Shutdown), ICCMDS channel B, DROPS channel 2 and PI-1041 (located on C04). C-539 supplies outputs to SPDS (Alternate Shutdown), ICCMDS channel A, DROPS channel 1, and PR 1042 (located on C04). The range of indication is 0 psig to 3000 psig. C-540 also supplies an input to ESAS analog channel 2. The input is used for over pressure protection of the Decay Heat Removal System. CV-1410 will close if RCS pressure exceeds 385 psig. The interlock allows opening CV-1410 when RCS pressure is less than 290 psig.

### 3.3.8 NNIX pressure control

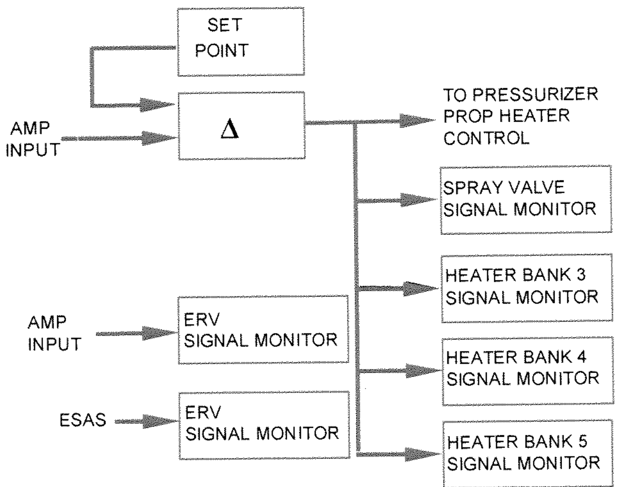
RPS channels A and C supply outputs from PT-1021 and PT-1038 to the NNIX instrument cabinets for RCS pressure control. A transfer relay selects which signal inputs to the NNIX pressure control channel. The relay is powered from the NNIX 120-volt AC bus. A three-position switch located on C04 controls the transfer relay. The switch positions are "A", "Auto", and "C".




In the Auto position SASS controls which signal inputs into NNIX. Normally, RPS channel A is selected for input. If RPS channel A signal fails, SASS would de-energize the transfer selecting the RPS channel C input. The A and C switch positions allow the operator to select RPS channel A or C independent of SASS (signal is hard selected and SASS cannot change it). The input scheme is shown below:

The SASS selected pressure signal inputs into an isolation amplifier. A 125 psi bias is input into the isolation amplifier when contact A closes. The bias is applied when either MFWP trips and reactor power is greater than 80%. This immediately opens the pressurizer spray valve to control RCS pressure. The output of the isolation amplifier is input to a difference amplifier and the ERV signal monitor.

The ERV signal monitor opens and closes the ERV in response to the input from the isolation amplifier. The signal monitor has two adjustable setpoints ( a high and a low setpoint). The signal monitor opens the ERV when RCS pressure reaches 2450 psig (high) and closes the ERV when RCS pressure reaches 2395 psig (low). ESAS analog channel 1 supplies wide range pressure input to a signal monitor. The ESAS input and associated signal monitor opens the ERV when RCS pressure is 400 psig and closes the ERV when RCS pressure reaches 350 psig.




Three switches are associated with the ERV, the ERV setpoint selector switch, HS-1013, and two auto/open switches, HS-1012 and HS-1014. HS-1013 (located on C-04) allows selecting either the high ERV setpoint (2450 psig) or the low ERV setpoint (400 psig). Hand switches HS-1012 (located in NNI cabinet C-47-2) and HS-1014 (located on C-04) allow manual opening of the ERV. Each handswitch has two positions; AUTO, and OPEN. With the handswitch in the AUTO position, the signal monitor opens and closes the ERV. When either handswitch is placed in the OPEN position, the ERV solenoid is energized and ERV is opened.

Four indicating lights are located on C-04 above HS-1014. The lower red indicating light is on when the ERV solenoid is energized. The lower white indicating light is on when handswitch HS-1012 is in the AUTO position. The upper green and red indicating lights are supplied from the relief valve monitor. They are on when the ERV is closed (green) and open (red).

The ERV automatic operation is interlocked with the NNIX positive and negative power supplies. The ERV will not open automatically if either of the positive or negative power supplies are lost. This prevents spurious opening of the ERV due to power supply losses.

The difference amplifier outputs pressure error signal. The pressure error is the difference between RCS pressure and the setpoint (2155 psig). The error signal inputs to the pressurizer proportional heater controller and the signal monitors for the pressurizer spray valve and pressurizer heater banks 3, 4, and 5. The signal monitors function is similar to that of the ERV discussed above. Setpoints for the pressurizer spray valve and heaters are:

| | open/on | close/off |
|---|---|---|
| spray valve | 2205 psig | 2155 psig |

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

**QID:** 0629 **Rev:** 0 **Rev Date:** 6/20/11 **Source:** Direct **Originator:** J.Cork
**TUOI:** A1LP-RO-RPS **Objective:** 17 **Point Value:** 1

**Section:** 3.7 **Type:** Instrumentation
**System Number:** 012 **System Title:** Reactor Protection System
**Description:** Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Sensors and detectors.

**K/A Number:** K3.01 **CFR Reference:** 41.7 / 45.7
**Tier:** 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3
**Group:** 1 **SRO Imp:** 2.8 **SRO Select:** Yes **Taxonomy:** C

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

I&C is performing a surveillance on the "A" RPS channel.
NI input to ICS is selected to "C" RPS channel.

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

Which of the following will occur?

A. Plant is stable since SASS will transfer to NNI-Y.

B. Control rods will start to insert.

C. Control rods will start to withdraw.

D. Feedwater flows will go down.

**Answer:**

B. Control rods will start to insert.

**Notes:**

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

**References:**

STM 1-67, Rev. 13
STM 1-64, Rev. 12

**History:**

New for 2005 RO re-exam. K/A 012 K3.01
Selected for 2011 RO Exam.

also a test jack and a meter zero adjust potentiometer on the front plate.

### 2.5.5 Summing and Difference Amplifiers

Refer to Figure 67.30

There are two Sum/Difference Amplifiers in each power range channel. One Sum/Difference Amplifier module sums the 2 linear amplifier signals. The other is used to take the difference between the two linear amplifiers and indicates the difference in flux between the top and bottom of the core. The outputs of this summing amplifier are:

a. Remote indications of total power(0-125%) on CO-3, (see Table 67.1) and to the plant computer.

b. Power level to auctioneer unit for ICS.

c. Power level signals to the following bistables:

   1) Reactor high power trip.

   2) Power to pumps trip.

   3) Power at 10% which feeds Source Range rod withdrawal interlock c/o and EFIC functions

   4) ARTS bypass bistables for main turbine and main feed pumps.

   5) Power, imbalance and flow trip.

   6) Shutdown bypass hi power trip.

The outputs of the difference amplifier are:

a. Remote indications (-62.5% to +62.5%) on CO-3 (see Table 67.5) and the plant computer for axial power distribution.

b. Difference signal to function generator. The function generator provides the delta flux portion of the trip signal to the flux /delta flux/flow trip bistable.

Refer to Figure 67.32 for face plate view.

### 2.5.6 Function Generator

The Function Generator is a Bailey E92358. It has two inputs and one output. One input is the difference amplifier output (Delta Flux) the other is RCS flow. The output of the Function Generator is the setpoint signal for the Power/Imbalance/Flow Bistable. (see STM-1-63 RPS for further explanation).

### 2.5.7 Auctioneer Unit

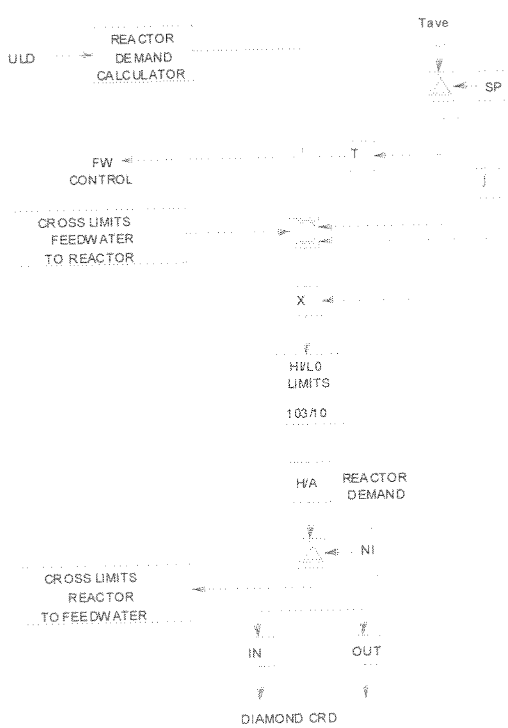
The power level of NI-5 and NI-6 are high auctioneered (highest signal is passed) in "B" RPS cabinet and sent as the "X" input to SASS for automatic signal selection for the ICS total power signal.

Power level of NI-7 and NI-8 are high auctioned in "C" RPS cabinet and sent as the "Y" input to SASS for automatic signal selection for the ICS total power signal.

"A" OTSG. $\Delta T_c$ will increase as power increases and will continue to increase until $T_{ave}$ reaches 579°F. During any further power increases, the operator will manually increase feedwater flow to the "A" OTSG to maintain Tave at 579°F. As "A" OTSG flow is increased the $\Delta T_c$ will move toward zero. At a $\Delta T_c$ of zero, the operator should increase both feedwater loop demands. When both OTSGs are above low level limit, the operator may place the ICS in auto. The total flow circuit will then be blocked until low level limit is reached while operating on 3 RCP's during a plant shut down or load reduction.

## 2.7 Reactor Demand Subsystem

FIGURE 64.28: REACTOR CONTROL




Refer to figure 64.28.

The megawatt demand signal that is received by the Reactor Demand Subsystem from the Integrated Master Subsystem has a Low Limit of 15%. It is undesirable to lower reactor power to less than 15% in automatic Therefore, the demand being sent to the reactor demand calculator will not be allowed to go below a value that is translated to 15% by the reactor demand calculator, should be about $.15 \times 902$ or ~135 megawatts. However, procedurally the operator will take manual control of the reactor when reduction of reactor power to < 20% is desired.

Since reactor power is not linear with generated megawatts, the reactor demand calculator changes the megawatt demand signal to a reactor demand signal equivalent to 0-125% power. The calculator output span then is 15% to 125% taking into account the low limit on the input.

The reactor demand signal is then modified as needed to keep $T_{ave}$ equal to setpoint. $T_{ave}$ is compared to the $T_{ave}$ setpoint which is controlled by the operator at the reactor demand H/A station. A 0% to 100% selection is possible. The 0% is equal to 520°F and 100% is equal to 620°F. Therefore, 59% (579°F) is the normal setpoint. If a $T_{ave}$ error exists it is used in both a proportional and integral action to adjust reactor demand.

The adjusted reactor demand signal is limited to between 10% and 103%. The low limit of 10% is there to allow $T_{ave}$ correction to decrease power a maximum of 5% when trying to establish $T_{ave}$ at setpoint. This could occur if low level limits are set too low. The high limit of 103% allows a $T_{ave}$ correction of 3% when reactor demand is 100%. However, the main purpose of the 103% limit is to prevent an automatic signal from raising power to its RPS trip setpoint.

The adjusted and limited reactor power demand signal is compared to the high auctioneered reactor power signal from the reactor protection system. The difference between the two signals is termed "Neutron Error". If actual power is greater than reactor demand, a positive neutron error results. If neutron error is > +1%, the control rods move into the core to reduce power until neutron error becomes < +.975%. If actual power is less than reactor demand, a negative neutron error results. If neutron error is > -1%, the control

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

**QID:** 0265 **Rev:** 0 **Rev Date:** 6/20/11 **Source:** Direct **Originator:** Cork

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

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

**System Number:** 013 **System Title:** Engineered Safety Features Actuation System(ESFAS)

**Description:** Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: RCP

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

**Tier:** 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3.5

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

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

Given:

- Reactor tripped due to low RCS pressure.
- ESAS has actuated due to high Reactor Building pressure.
- RCS pressure is 1600 psig.
- Core Exit Thermocouple temperature is 475 °F.

Which of the following actions should be taken for these conditions?

a. Restore RCP services.

b. Leave one RCP running in each loop.

c. Isolate RCP seal bleedoff.

d. Trip all running RCPs.

**Answer:**

d. Trip all running RCPs.

**Notes:**

Rising reactor building pressure will actuate channels 5 and 6 of ESAS which will isolate the reactor building and terminate cooling water flow to the RCPs. RCP temperatures will rapidly heat up unacceptably high level before the operator can take action to restore cooling water. Therefore "a" and "b" are incorrect. "c" is incorrect because seal bleedoff should not be isolated.
"d" is correct.

**References:**

1202.012, Chg. 009 (RT-10)

**History:**

Developed for 1999 exam.
Selected for 2011 RO Exam.

## VERIFY PROPER ESAS ACTUATION

> **NOTE**
> Obtain Shift Manager/CRS permission prior to overriding ES.

1. **Verify BWST T3 Outlets open:**
   - CV-1407
   - CV-1408

   A. IF BWST T3 Outlet (CV-1407 or CV-1408) fails to open,
   THEN override AND stop associated HPI, LPI, and RB Spray pumps until failed valve is opened.

2. **Verify SERV WTR to DG1 and DG2 CLRs open:**
   - CV-3806
   - CV-3807

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
      - P32B
      - P32C
      - P32D

   2) IF SCM is not adequate,
   THEN check elapsed time since loss of adequate SCM
   AND perform the following:

      a) IF $\leq$ 2 minutes have elapsed,
      THEN trip all RCPs:
         - P32A
         - P32B
         - P32C
         - P32D

**(3. CONTINUED ON NEXT PAGE)**

## VERIFY PROPER ESAS ACTUATION

**3. (Continued)**

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 or 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).

**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. On C19 verify:**

A. LPI (Decay Heat) Room Cooler running in each Decay Heat Room:

| P34A Room | P34B Room |
|---|---|
| VUC1A or B | VUC1C or D |

B. **IF** RB Spray has actuated,
**THEN** verify SW to RB Spray P35A and P35B LO CLRs open:

- CV-3804
- CV-3805

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

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

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

**Section:** 3.5 **Type:** Containment Integrity

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

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

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

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

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

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

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

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

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

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

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

**Answer:**

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

**Notes:**

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

**References:**

STM 1-05, Rev 17
1309.013, Chg. 020

**History:**

Used in 1999 exam.
Modified from ExamBank, QID# 1519.
Selected for 2005 RO re-exam. KA 022 A1.04
Selected for 2011 RO Exam.

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 "Process Monitor Radiation High" alerting the operator of this condition.

Operation without service water flow through the cooler is allowed with restrictions placed on pump operation. Refer to operability section of 1104.004 for additional information.

DHR Cooler outlet temperatures are indicated in the control room on panel C-14. The cooler outlet temperatures provide a high temperature alarm to the plant computer and SPDS. The high alarm setpoint is 280°F. "A" DHR cooler outlet temperature is provided by TE-1406, which reads out on TI-1433. "B" DHR cooler outlet temperature is provided by TE-1407 which reads out on TI-1432.

# Supplement 2



## THROTTLING OR BALANCING SERVICE WATER SYSTEM

1.0 PURPOSE

This Supplement provides instructions for throttling and balancing flows. Section 2.0 adjusts the flow to the Decay Heat Coolers to increase flow to the Reactor Building Coolers and other components. Section 3.0 adjusts the flow to the Makeup pump room cooler to increase the flow to the Makeup pump lube oil cooler or to decrease flows to the room cooler. Section 4.0 adjusts flow to the EDG heat exchangers to increase flow to balance of loop and 2VE-1A/1B chillers.

2.0 THROTTLING DECAY HEAT COOLERS

2.1 Procedure section 7.0 complete. ________

2.2 Procedure sections 8.1 and 8.2 complete. ________

**NOTE**

It is undesirable to throttle butterfly valves, SW-22A and SW-22B to more than 66% closed. If required to throttle to more than 66% closed than provide an engineering evaluation of this condition (Ref PEAR-88-3738).

2.3 **IF**, Loop 2 flow balancing is required,

**Then,** monitor flow to VCC-2C/D with SPDS point F3817. Throttle SW-22B until the desired flow (including uncertainties, applicable degradation factors, and margin) exceeding 1270 gpm is achieved, or valve is throttled to 66% closed, otherwise N/A. ________

2.4 Record indicated flow, F3817, if Loop 2 balancing was required ______gpm, otherwise N/A ________

2.5 **IF**, Loop 1 flow balancing is required,

**Then,** monitor flow to VCC-2A/B with SPDS point F3816. Throttle SW-22A until the desired flow (including uncertainties, applicable degradation factors, and margin) exceeding 1270 gpm is achieved or valve is throttled to 66% closed, otherwise N/A. ________

2.6 Record indicated flow, F3816, if Loop 1 balancing was required ______gpm, otherwise N/A. ________

2.7 If loop 1 flow balance is required, then measure loop 1 Decay Heat loop flow and record, otherwise N/A. ________
Loop 1 Decay Heat loop flow ________________gpm

2.8 Determine E-35A flow by subtracting VUC-1A, VUC-1B, E-47A, and E-50A flows, ________________________gpm ________

2.9 If Loop 2 flow balance is required, then measure loop 2 Decay Heat loop flow and record, otherwise N/A. ________
Loop 2 Decay Heat Loop flow ________________gpm

2.10 Determine E-35B flow by subtracting VUC-1C, VUC-1D, E-47B, and E-50B flows, ________________________gpm ________

2.11 Repeat steps 2.3 thru 2.10 until the desired flow distribution is established. ______

## Supplement 2



### THROTTLING OR BALANCING SERVICE WATER SYSTEM

> **NOTE**
> Positioning marks for butterfly valves, SW-22A and SW-22B shall be marked with a scribe marker. Using a mark with a fixed width significantly more than 1/64" will introduce additional measurement uncertainty in the flow to E-35A and E-35B heat exchangers.

2.12 Using a sharp metal scribe tool, mark the final throttled position of SW-22A and/or SW-22B on the valve position indicating plates. Scribed line shall be approximately 1/64" width. ______

2.13 **IF** position indicating arrow on SW-22A or B does not have a scribed center line, **THEN** scribe a line approximately 1" long, and approximately 1/64" wide at apex of arrow. ______

2.14 Remove previous throttle position markings by painting over, marking out or otherwise removing any doubt as to which mark remains the appropriate current marking. ______

2.15 Provide independent verification that the scribed marks installed in steps above are accurately placed and less than 1/64" in width and previous throttle position markings have been removed. ______

2.16 Return to step 8.3.3 or 8.4.7, as applicable, and continue/finish collecting flow and pressure data on desired components and record on Attachment 3. ______

3.0 THROTTLING MAKEUP ROOM COOLER

> **NOTE**
> The supply valve, SW-108, and Service Water ECP return header isolation valve, CV-3823, may be throttled to emulate the conditions of a simulated ES lineup, using the test pressure recorded in step 8.1.16.

3.1 Section 7.0 complete. ______

3.2 Section 8.1 complete. ______

3.3 Monitor flow to the Makeup pump lube oil cooler, E-39B. Throttle flow to the room cooler by closing SW-37B, keeping track of the number of turns from the open position. Throttle until a flow of ~4.5 gpm is achieved. ______

3.4 Record flow ______gpm ______

3.5 Verify flow to room cooler, VUC-7B exceeds 34.5 gpm. ______

3.6 Record flow ______gpm ______

3.7 Record final position of throttled valve ______turns (From open). ______

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

**QID:** 0830 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** New **Originator:** S.Pullin

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

**Section:** 3.5 **Type:** Containment Integrity

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

**Description:** Knowledge of bus power supplies to the following: Containment Spray pumps.

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

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

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

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

What are the power supplies for the Reactor Building Spray Pumps?

A. A3 and A4

B. A1 and A2

C. B5 and B6

D. B52 and B62

**Answer:**

A. A3 and A4

**Notes:**

Only "A" lists the correct power supplies, the other choices supply major loads but not the spray pumps.

**References:**

1107.002, Chg. 028

**History:**

New for 2011 RO Exam.

ATTACHMENT A



4160V ES SWITCHGEAR CHECKLIST

**CAUTION**

Do NOT close an open breaker until operator is sure of consequences.

**NOTE**

- This checklist assumes the unit is in Mode 5 or 6.
- Use of the local breaker status light to determine breaker position will also indicate control power availability.

1.0 Check each listed breaker for the following:

- Breaker in desired position.
- Breaker control power on.
- Breaker racked up.

- Breaker control selector in REMOTE.
- Breaker labeled properly.

1.1 Log any breaker that is danger tagged or NOT in desired position on Lineup Exception Sheet (E-doc 1015.001F).

1.2 Notify plant labeling of any label discrepancies.

| 4160V ES Bus A3 | | | | | |
|---|---|---|---|---|---|
| BREAKER NUMBER | DESCRIPTION | DESIRED POSITION | ACTUAL POSITION | TAG (✓) | INITIAL |
| A-301 | A3 Feed to B5 (E-104) | Closed | | | |
| A-302 | Service Water Pump P-4A (E-275) | - | | | |
| A-303 | Service Water Pump P-4B (E-276) | - | | | |
| A-304 | RB Spray Pump P-35A (E-241) | Note 1 - | | | |
| A-305 | LPI Pump P-34A (E-181) | - | | | |
| A-306 | HPI Pump P-36A (E-211) | Note 1 - | | | |
| A-307 | HPI Pump P-36B (E-212) | Note 1 - | | | |
| A-308 | DG1 Output (E-100) | Open | | | |
| A-309 | A1 Feed to A3 (E-97) | Closed | | | |
| A-310 | A3-A4 Crosstie (E-98) | Open | | | |
| A-311 | EFW Pump P-7B (E-294) | Open | | | |

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

ATTACHMENT A



| 4160V ES Bus A4 | | | | | |
|---|---|---|---|---|---|
| BREAKER NUMBER | DESCRIPTION | DESIRED POSITION | ACTUAL POSITION | TAG (✓) | INI-TIAL |
| A-401 | A4 Feed to B6 (E-104) | Closed | | | |
| A-402 | Service Water Pump P-4C (E-275) | - | | | |
| A-403 | Service Water Pump P-4B (E-276) | - | | | |
| A-404 | RB Spray Pump P-35B (E-241) | Note 1<br>- | | | |
| A-405 | LPI Pump P-34B (E-181) | - | | | |
| A-406 | HPI Pump P-36C (E-211) | Note 1<br>- | | | |
| A-407 | HPI Pump P-36B (E-212) | Note 1<br>- | | | |
| A-408 | DG2 Output (E-100) | Open | | | |
| A-409 | A2 Feed to A4 (E-97) | Closed | | | |
| A-410 | A4-A3 Crosstie (E-98) | Open | | | |

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

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

**QID:** 0831 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** Direct **Originator:** J. Cork

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

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

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

**Description:** Ability to manually operate and/or monitor in the control room: Emergency feedwater pump turbines.

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

**Tier:** 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2

**Group:** 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** K

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

What is the initial speed (in RPM) of the steam driven EFW pump after auto start and PRIOR to ramp to operating speed?

A. 825

B. 875

C. 910

D. 940

**Answer:**

C. 910

**Notes:**

"C" is correct, the others are possible answers.

**References:**

1106.006, Chg. 080

**History:**

Direct from regular exambank ANO-OPS1-1141
Selected for 2011 RO Exam.

**NOTE**

When handswitch for steam admission valves is placed to the OPEN position, the motor operated control valve will remain closed for a short time delay. Normally, this allows the bypass valve to bring turbine speed up to ~910 RPM. Only with the trip throttle valve (CV-6601A) on the 100% open limit switch, will the ramp circuit initiate acceleration past ~900 rpm.

21.3 IF using green-train steam admission valves, THEN perform the following:

21.3.1 Momentarily turn handswitch for EFW Pump Turbine K3 Steam Admission Valve (CV-2613) to OPEN.

21.3.2 Verify the following:

A. EFW Pump Turbine K3 Steam Admission Valve Bypass (CV-2615) opens.

B. CV-2613 opens.


**NOTE**

When handswitch for steam admission valves is placed to the OPEN position, the motor operated control valve will remain closed for a short time delay. Normally, this allows the bypass valve to bring turbine speed up to ~910 RPM. Only with the trip throttle valve (CV-6601A) on the 100% open limit switch, will the ramp circuit initiate acceleration past ~900 rpm.


21.4 IF using red-train steam admission valves, THEN perform the following:

21.4.1 Momentarily turn handswitch for EFW Pump Turbine K3 Steam Admission Valve (CV-2663) to OPEN.

21.4.2 Verify the following:

A. EFW Pump Turbine K3 Steam Admission Valve Bypass (CV-2665) opens.

B. CV-2663 opens.

21.5 Record start time for P-7A run: __________

21.6 Monitor turbine speed OR pump discharge pressure.

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

**QID:** 0063 **Rev:** 0 **Rev Date:** 7/12/98 **Source:** Direct **Originator:** GGiles

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

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

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

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

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

**Tier:** 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

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

**Question:** **RO:** 43 **SRO:** 44

Given:

- 100% power
- ICS in full automatic

The CBOR places the ICS Delta T-Cold Hand Auto Station meter selection switch in "POS" (position). The meter reads 54%.

What does this mean in terms of ICS control of main feed water?

A. The average of feedwater loop A and feedwater loop B demand is 54%.

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

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

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

**Answer:**

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

**Notes:**

A reading >50% indicates that loop A demand is > loop B demand, therefore (D) is the correct response.
(A) is incorrect because the meter does not indicate average demand.
(B) is an opposite response.
(C) applies to looking at the MV reading (for which it would still be incorrect).

**References:**

STM 1-64, Rev. 12, (2.6.2)

**History:**

Developed for the 1998 RO/SRO Exam.
Selected for use in 2002 RO/SRO exam.
Used on 2004 RO/SRO Exam.
Selected for 2011 RO Exam.

transferred into the "Tracking" mode. The occurrence of this limiting action indicates that the neutron power is not able to satisfy its demand. Therefore, by modifying the feedwater demand signal with the neutron error, feedwater is held to within 5% of reactor power. Since the ICS is in Track, the turbine merely controls header pressure and thus the load can be no greater nor less than 5% of the neutron power.

### 2.6.2 Load Ratio (ΔTc) Control

The total feedwater flow demand signal is split by the ICS into loop "A" and "B" feedwater demand signals by adjustment of the value of a multiplier controller. This controller sets the value of loop loop "A" feedwater demand by multiplying the total flow demand by the value of the multiplier. If the multiplier is set at .5, half of the total feedwater flow demand signal becomes loop "A" feedwater demand. The loop "B" feedwater demand is determined by subtracting the loop "A" demand from the total demand. Changing the multiplier value will change the value of both loop demand signals. The maximum loop feedwater demand signal is $6 \times 10^6$ pounds mass per hour.

The value of the multiplier is set by the value of a control signal. This signal is the algebraic summation of two other signals. One of these signals is the RCS flow mismatch signal and will be zero when all four RCP's are properly operating. This signal will be described under "Three Pump Operations". The other signal is the ΔTc correction signal.

The control of the ratio of feedwater to each OTSG will determine the amount of heat that will be removed from the primary water in the reactor coolant system (RCS) and the relative amount of loading that each OTSG will carry. Therefore, the loading of the OTSGs can be indicated by the relative RCS return temperatures to the reactor (Tc's). If the difference in the Tc's ($\Delta T_c$) is controlled near zero, then each OTSG will be loaded properly for the RCS flow through it. A trip of one RCP would give an immediate re-ratioing. An important benefit of keeping ΔTc low is that quadrant tilts within the reactor may be kept to a minimum.

The actual ΔTc is compared to the ΔTc setpoint. The difference (ΔTc Error) is used to generate the ΔTc correction signal. A zero ΔTc correction signal will split the signal equally between the loops.

The operator may choose to manually control the ΔTc correction signal by placing the Load Ratio Hand/Automatic Station in hand. The only difference between this station and the other feedwater hand/auto stations is the additional dial and knob located under the meter. This provides the ΔTc setpoint for automatic operation. The setpoint may be varied from 0% to 100% which corresponds to -10°F to +10°F. The normal value is 50% (0°F).

When position is selected on this station, the ΔTc correction signal is indicated on the meter. If the meter indicates 50%, the correction signal is zero (loop "A" multiplier set at .5) and loop demand signals are equal. If the meter indication is above 50%, then loop "A" demand is > loop "B" demand. The opposite is true if the indication is < 50%.

When measured variable is selected on this station, the difference between the actual ΔTc and the ΔTc setpoint (ΔTc error) is indicated on the meter. ΔTc = "A" Loop Tc - "B" Loop Tc. The meter scale is ±10°F. Positive reading means that "A" loop is hotter. A bumpless transfer from hand to auto may take place when the ΔTc error equals zero (50% on meter). If the ΔTc does not equal zero, adjustment to zero may be accomplished by adjusting the manual output of the station or by changing the ΔTc setpoint.

If both loop demand stations are placed in hand, this station rejects to hand and can not be placed in auto.

### 2.6.3 Feedwater Flow Control

The method of flow control used by the feedwater system is dependent upon the plant power level. (refer to figure 64.24) At low power feedwater flow is controlled by the startup and low load control valves with the main feedwater block valve shut and the feedwater pumps operating to maintain 70 psid across the feed valves. The valves are sequenced into operation so that the startup valve opens first followed by the low load control valves then the main FW block valves. As plant load is increased, feedwater flow control will be shifted from the valves to the pumps. This is accomplished by opening the main block valves and controlling the speed of the feedwater pumps to control flow.

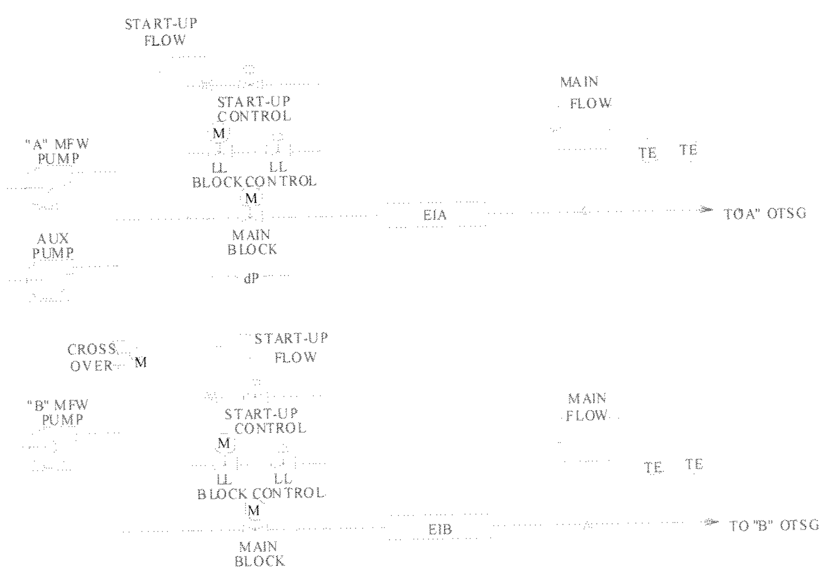


FIGURE 64.24: FEEDWATER FLUID SYSTEM

Below ~2% reactor power, the auxiliary feedwater pump has enough capacity to feed the OTSG's. Above ~2%, an indication that the auxiliary feedwater pump is not able to provide sufficient feedwater is, the startup and low load valves are full open and still not able to maintain OTSG level.

The flow control signal is obtained by comparing loop feedwater demand to loop feedwater flow. Actual flow is provided by the selected main FW flow instrument. The developed flow error signal is applied through proportional plus integral control to establish the valve demand signal for the startup and low load valves. The feedwater pump control signal is developed by having a feedforward control signal created by calculating a ball park base pump speed from each loop demand. The feedforward control signal will provide the coarse adjustment for the feed pump speed. ΔP or flow error will provide the fine tuning.

With the loop feedwater demand < 50%, the flow to each OTSG is controlled by modulation of startup and low load control valves. When the startup valve reaches 80% open, the low load block valve will

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

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

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

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

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

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

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

**Tier:** 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 4

**Group:** 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** An

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

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 280 gpm to each SG in HAND.

B. Throttle EFW to prevent overcooling.

C. Select Reflux Boiling setpoint.

D. Actuate MSLI on both OTSGs.

**Answer:**

B. Throttle EFW to prevent overcooling.

**Notes:**

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

**References:**

1202.012, Chg. 009, RT-5

**History:**

Created for 2002 RO/SRO exam.
Selected for 2005 RO re-exam.
Selected for 2011 RO Exam.

## 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. |

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 $\Delta$T stable or dropping

6. **Monitor EMERGENCY FEEDWATER and EFIC alarms on K-12.**

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

**QID:** 0538 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Direct **Originator:** NRC

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

**Section:** 3.4 **Type:** Heat Removal 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:** Yes **Difficulty:** 3

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

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

Plant conditions are:

- SCM is less than adequate.
- The RCS is in natural circulation mode.
- EFW pump P-7B failed to start.
- Both OTSG pressures are 900 psig.

The fill rate for the current condition is __________, and the OTSG level set point of 378" is selected by ____________.

A. 4.4 inch/min
depressing the Train A and Train B REFLUX BOILING pushbuttons on C09

B. 4.4 inch/min
EFIC automatically when RCP's are tripped

C. 4.0 inch/min
depressing the Train A and Train B REFLUX BOILING pushbuttons on C09

D. 4.0 inch/min
EFIC automatically when RCP's are tripped

**Answer:**

A. 4.4 inch/min
depressing the Train A and Train B REFLUX BOILING pushbuttons on C09

**Notes:**

"A" is correct, 4.4 inch/min, by depressing the Train A and Train B REFLUX BOILING pushbuttons on C09. (STM 1-27, rev 6, page 4). Candidate must use linear equation y=mx+b to solve for fill rate of 4.4 in/min, making "C" and "D" incorrect. By Procedure 1106.006, Emergency Feedwater Pump operation, the 378" set point must be selected on C09 by the operator, therefore "B" is incorrect.

**References:**

EOP 1202.012, RT5, Chg. 009

**History:**

Developed by NRC.
Used on 2004 RO Exam.
Selected for 2011 RO Exam.

## 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

**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:** 0413 **Rev:** 0 **Rev Date:** 12/06/00 **Source:** Direct **Originator:** S. Pullin

**TUOI:** A1LP-AO-ELECD **Objective:** 4 **Point Value:** 1

**Section:** 3.6 **Type:** Electrical

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

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

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

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

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

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

The plant is in Hot Shutdown with a normal electrical alignment.
Due to an electrical fault, K02-A6 "A1 L.O. RELAY TRIP" comes into alarm.

How will the electrical system respond?

a. #1 EDG will auto-start and will supply bus A3.

b. Bus A1 will auto-transfer to transformer SU#1.

c. Bus A1 will auto-transfer to transformer SU#2.

d. #1 EDG will auto-start, but will not supply bus A3.

**Answer:**

a. #1 EDG will auto-start and will supply bus A3.

**Notes:**

(a) is the correct answer, the A1 bus lock-out will cause a loss of A3, EDG#1 will auto-start and supply A3.
(b) and (c) are incorrect, the A1 lockout will prevent any transformer from supplying the bus.
(d) is incorrect, the A1 lockout will not prevent #1 EDG from supplying bus A3.

**References:**

1203.012B, Chg. 033

**History:**

New Question for 2001 RO/SRO NRC Exam. K/A 062 K4.01
Selected for 2011 RO exam.

Location: C10

Device and Setpoint: see next page.

A1
L.O. RELAY
TRIP

Alarm: K02-A6

1.0 OPERATOR ACTIONS

1. Verify the following breakers are tripped:

   - Startup Xfmr #2 Feed to A1 (A-111)
   - Unit Auxiliary Xfmr Feed to A1 (A-112)
   - Startup Xfmr #1 Feed to A1 (A-113)

2. IF A1 Feed to A3 (A-309) was powering bus A3,
   THEN verify DG1 auto start.

   A. Refer to EDG 1 AUTOSTART COMMAND (K01-A1).

3. Initiate steps to determine cause of lockout relay trip. Do NOT reset lockout relay until cause of trip has been determined and corrected.

4. Refer to Loss of Loadcenter (1203.046) due to loss of B-1,B-3, and B-7.

5. Refer to TS 3.8.1/3.8.2 for ES bus power source requirements.

6. WHEN cause of trip is known
   AND corrected,
   THEN at A112 reset 186-A1 Lockout relay by rotating handswitch clockwise.

7. Re-energize bus.

2.0 PROBABLE CAUSES

**NOTE**
This annunciator has multiple input without reflash.

1. Phase over-current on Startup Transformer 2
2. Ground over-current on Startup Transformer 2
3. Phase over-current on Startup Transformer 1
4. Ground over-current on Startup Transformer 1
5. Phase over-current on Unit Auxiliary Transformer
6. Ground over-current on Unit Auxiliary Transformer
7. Breaker A-112 failure to trip during auto transfer

**QID:** 0616 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** Direct **Originator:** S.Pullin
**TUOI:** A1LP-RO-ELECD **Objective:** 17 **Point Value:** 1

**Section:** 3.6 **Type:** Electrical
**System Number:** 062 **System Title:** AC Electrical Distribution
**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:** 2
**Group:** 1 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** K

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

The CBOT notices on C10 that all of the 161 KV ring bus breakers have opened.

Which of the following will be de-energized as a result of the above indications?

A. Startup #1 Transformer

B. Startup #2 Transformer

C. Auto transformer

D. Startup #3 Transformer

**Answer:**

B. Startup #2 Transformer

**Notes:**

Answer "B" is correct, the 161KV ring bus supplies the #2 SU transformer.
Answers "A", "C", and "D" are the remaining but incorrect transformers supplied by offsite power.

**References:**

STM 1-32-1, Rev. 1

**History:**

New for 2005 RO exam, replacement question.
Selected for 2011 RO Exam.

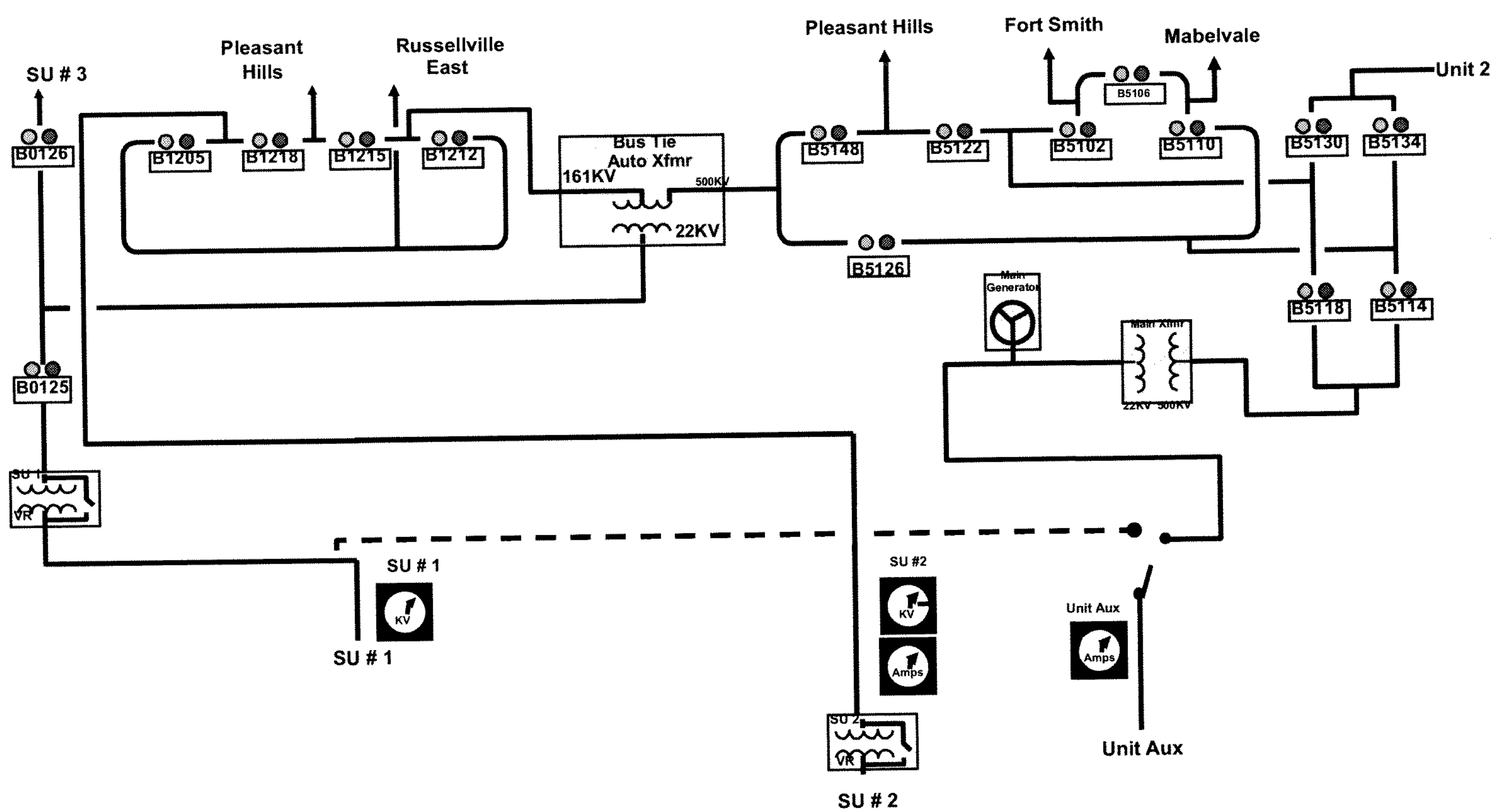



Figure 32-1-4, Switchyard Breaker Layout (C-10)

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

**QID:** 0087 **Rev:** 1 **Rev Date:** 11/4/98 **Source:** Direct **Originator:** JCork

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

**Section:** 3.6 **Type:** Electrical

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

**Description:** Knowledge of the effect that a loss or malfunction of the dc electrical system will have on the following: ED/G.

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

**Tier:** 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 3

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

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

During a Blackout condition, it may become necessary to perform an EDG start without DC control power. In this condition, how is the generator output voltage adjusted (battery chargers are not loaded)?

a. Adjusting engine speed via speed setting knob on governor.

b. By adjusting the setpoint for the automatic voltage adjuster.

c. Using the manual voltage adjuster on Exciter Control Panel.

d. Using voltage adjuster handswitch on C10.

**Answer:**

c. Using the manual voltage adjuster on Exciter Control Panel.

**Notes:**

Without DC control power the automatic voltage SETPOINT will not be present, This would then require the use of the manual adjuster as in answer "c".
The auto voltage adjuster will still function since it is powered directly from the generator's output but the setpoint requires DC power. Therefore "B" is an incorrect answer.
Answer "D" is incorrect, no DC power is available.
Answer "a" is incorrect, this will adjust frequency, not voltage.

**References:**

1104.036, Chg. 55

**History:**

Developed for 1998 SRO exam
Revised after 9/98 exam analysis review.
Selected for 2011 RO Exam.

13.0 DG1 START WITHOUT DC CONTROL POWER

**CAUTION**

If fault condition that caused loss of DC is NOT removed, be aware that a fault may still be present and will have to be dealt with when presented.

**NOTE**

Following sequence assumes no AC or DC is available.

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

13.2 Place DG1 Engine Control Selector switch (HS-5234) on C107 in MAINT.

**CAUTION**

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

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

- DG1 Local Field Flashing Power (D-1116A).
(inside voltage regulator cabinet E-11)

- DG1 Engine Control Power (D-1114A).
(inside engine control panel C107)

**NOTE**

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

- This is a serious condition and even if ESAS is required, ES signal must be overridden and de-energized.

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

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

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

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

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

13.6 Override governor run solenoid as follows:

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

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

13.6.3 Release pin AND verify knob stays in raised position.

**NOTE**

SW valve requires DC power to open automatically

13.7 Manually open Service Water to DG1 Coolers (CV-3806) to allow cooling water flow as soon as a service water pump is started.

**CAUTION**

- With loss of control power, the only functional DG protection is the mechanical overspeed device.
- The steps that follow the start of the DG up through closing the SW Pump breaker should be done without delay or interruption so that cooling will be established.

13.8 Start DG1 by overriding one of the starter solenoid valves with the override plunger.

13.8.1 WHEN engine starts,
THEN release solenoid.

13.8.2 IF DG1 does NOT start after ~5 seconds,
THEN repeat step 13.6.

13.8.3 IF it becomes necessary to shut down DG1, THEN perform ONE (A or B) of the following:

A. Perform the following:

1. Release the governor solenoid by lifting the knurled knob to release the locking pin.

2. Push the knurled knob down.

3. Ensure fuel rack lever returns to the shutdown position (black pointer moves to <1 on governor).

B. Pull the EMERGENCY FUEL CUTOFF (FO-66A) valve handle.

1. WHEN DG has coasted to a stop, THEN push EMERGENCY FUEL CUTOFF (FO-66A) valve handle in AND verify Limit Switch engaged.

13.9 Verify governor maintains engine speed at ~900 RPM speed and ~60 Hz.

13.9.1 IF necessary to adjust speed (frequency), THEN turn speed setting knob on governor.

**NOTE**

Loss of DC power to field flashing circuit will cause loss of automatic voltage regulator setpoint adjustment (loss of power to motor-operated rheostat). Automatic voltage regulation should still function.

13.10 Verify diesel generator output voltage ~ 4160 V.

13.10.1 IF voltage falls outside 3800-4500 volts, THEN perform the following:

A. place DG1 Excitation control selector switch on (E11) in MANUAL.

B. Adjust manual voltage adjust dial to maintain ~ 4160 V.

**CAUTION**

The next step will cause automatic loading of the diesel if 125V DC is restored through a battery charger. Be prepared to manually adjust the diesel voltage and frequency each time a load is started.

13.11 Manually close DG1 Output Breaker (A-308).

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

**QID:** 0088 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

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

**Section:** 3.6 **Type:** Electrical

**System Number:** 064 **System Title:** Emergency Diesel Generators (ED/G)

**Description:** Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Incomplete-start relay.

**K/A Number:** K4.05 **CFR Reference:** 41.7

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

**Group:** 1 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** An

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

The CRS directs you to perform Supplement 1 of 1104.036, #1 EDG Monthly Test.
You depress the start pushbutton on C10, nothing happens, then the "EDG 1 OVERCRANK" annunciator K01-B2 alarms.
The CRS directs the inside AO to check the EDG out and then depress the local RESET pushbutton.

Which of the following would occur after the AO depresses the RESET pushbutton?

a. EDG would be ready for another manual start.

b. EDG will not manually or automatically start.

c. EDG output breaker will be locked out.

d. EDG will immediately start cranking.

**Answer:**

d. EDG will immediately start cranking.

**Notes:**

Answer (d) is correct since the EDG will start after reset pushbutton is depressed if stop pushbutton is not depressed to reset the start logic.
Answer (a) is incorrect because the EDG start logic must be reset after an overcrank, (b) is incorrect since it will automatically start and (c) is incorrect since the breaker is not affected.

**References:**

STM 1-31, Rev. 12

**History:**

Developed for 1998 RO/SRO Exam.
Used in 98 RO Re-exam
Used in 2001 RO Exam.
Selected for 2011 RO Exam.

***NOTE:** These percentages of motor rated voltage are nominal. Refer to the Unit 1 Technical Specifications for range of permissible values.

When a start signal is received, the engine start pilot relay (K-8) is energized. An auto-start signal will also energize the engine auto-start relay (K-21). The engine auto-start relay starts the fuel oil priming pump and gives the EDG auto-start annunciator in the control room. The engine start pilot relay energizes the governor run solenoid (allows the governor to position the fuel racks to deliver fuel oil to the engine cylinders), starts the governor boost motor (provides high pressure oil for governor operation), and energizes the air start motor circuitry.

The air start motor circuitry consists of an alternator, a speed switch assembly, and time delay relays to disengage the air start motors and annunciate the failure of the EDG to start.

The alternator will energize the air start solenoid valve for air start system #1. The #1 air start system air start motors will begin cranking the engine. The engine speed should be >30 RPM in 2.5 seconds. If engine speed is at least 30 RPM, air start system #1 will stay in service. When the next start signal is received, the alternator will energize the air start solenoid valve for air start system #2.

If engine speed remains less than 30 RPM for 2.5 seconds, the alternator circuit will shift to air start system #2. The air start solenoid valve for air start system #2 is energized. The #2 air start system air start motors will begin cranking the engine. Alternation between the two air start systems will continue until engine speed is >30 RPM.

The air start motor circuit uses two parameters to indicate that the engine is running. When engine speed >300 RPM or the main oil pump discharge pressure is >20 psig, the air start solenoid valve is de-energized and the air start motors will disengage. When engine speed >300 RPM, the engine run relay energizes. The engine run relay gives the run indication in the control room, and de-energizes the governor boost motor and fuel oil priming pump.

In the event that the engine fails to start, two time delay relays are used to stop engine cranking and to annunciate the failure of the EDG to start. An eight second and a 45 second time delay relay are energized when the start signal is received. After the eight seconds, the air start solenoid valve is de-energized and engine cranking stops. After 45 seconds, the EDG overcrank alarm is annunciated in the control room. These relays are blocked when engine speed reaches 300 RPM. The engine reset switch resets the start logic.

The EDG will tie on to the respective bus if:

- the respective bus voltage is <75%
- the normal supply breaker (A309/A409) is open
- one of the cross tie breakers is open
- EDG voltage is normal
- the EDG output breaker is not in pull-to-lock

## 9.2 ENGINE STOP AND TRIP

(Refer to figures 44, 45, and 46)

Stopping the EDG, whether through trip devices or normal stop switches, is accomplished by de-energizing the governor run solenoid. De-energizing the governor run solenoid causes the governor to position the fuel racks so that the fuel injectors will not inject fuel oil into the cylinders. Any of the following will de-energize the start pilot relay (K-8) which will de-energize the governor run solenoid:

- Placing the engine control switch in the maintenance position
- Placing the normal/lockout switch on C-10 in the lockout position
- Energizing the emergency trip relay
- Energizing the EDG lockout relay
- Depressing the local stop switch (blocked when an auto-start signal is present)
- Depressing the C-10 stop switch (blocked when an auto-start signal is present)

### 9.2.1 Emergency Trip Relay

The emergency trip relay (K-11) is an energize to trip relay. If no DC power is available, the emergency trip relay will not trip the EDG if it is running (no DC start). Any of the following devices will energize the emergency trip relay:

- Local emergency stop push button
- Mechanical overspeed switch
- Low lube oil pressure (blocked on engine start until engine reaches 300 RPM +20 seconds)
- Positive crankcase pressure (trips the emergency trip relay through the low lube oil pressure switch)
- 

### 9.2.2 EDG Lockout Relay

The EDG lockout relay monitors the generator for trip conditions. The EDG lockout relay is located at the EDG output breaker (A-308/A-408). The lockout relay power supply is from the DC control power in the EDG output breaker cubicle. The following conditions will energize the EDG lockout relay.

- Generator Differential
- Overcurrent
- Loss of Generator Field
- Generator Motoring
- Engine Emergency Trip Relay (K-11)

When the EDG lockout relay is energized springs rotate the shaft of the relay. Contacts then close to perform the following functions:

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

**QID:** 0849 **Rev:** 0 **Rev Date:** 6/25/11 **Source:** Direct **Originator:** J. Cork

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

**Section:** 3.6 **Type:** Electrical

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

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

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

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

**Group:** 1 **SRO Imp:** 3.3 **SRO Select:** Yes **Taxonomy:** C

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

Given:
- #1 EDG is in the 8th hour of a 24 hour full load run after maintenance
- #2 EDG is in standby
- An AO has just completed a fuel oil tanker truck off-load
- Sediment from the tanker entered the Fuel Oil Bulk tank T-25 and caused the outlet filter F-27 to clog

What would be the result of this condition on the Unit 1 EDG Fuel Oil system?

A. Fuel Oil Day Tank T-30A level low alarm.

B. Fuel Oil Storage Tank T-57A implosion.

C. Fuel Oil Storage Tank T-57A level low alarm.

D. Fuel Transfer Pump P-16A D/P Hi alarm.

**Answer:**

C. Fuel Oil Storage Tank T-57A level low alarm.

**Notes:**

C is correct, with #1 EDG running fully loaded, T-57A would be starved for makeup and level would lower. T-57A is vented to atmosphere to prevent collapse. Components downstream of the storage tank would operate normally.

A is incorrect, although a T57A low alarm would result, the transfer pump would still be able to fill the Day Tank.
B is incorrect, although the strainer between T-25 and T-57A is clogged, T-57A is vented to atomsphere and thus a vacuum would not be created in T-57A.
D is incorrect, if the trainee believed the sediment in T-25 would be carried over to T-57A, then he would choose this answer.

**References:**

1203.012A, Chg. 042

**History:**

Direct from Unit 2 regular exam bank ANO-OpsUnit2-10285
Selected for 2011 RO Exam.

Location: C10

Device and Setpoint:
DG1 Fuel Tank (T-57A) level (PDIS-5211) ≤155 inches
DG2 Fuel Tank (T-57B) level (PDIS-5212) ≤155 inches
DG1 Fuel Transfer Pump (P-16A) pressure (PSH-5211) ≥30 psig
DG2 Fuel Transfer Pump (P-16B) pressure (PSH-5212) ≥30 psig

| EDG FUEL OIL TROUBLE |
| --- |

Alarm: K01-F4

**NOTE**

This annunciator has multiple inputs without reflash capability.

1.0 OPERATOR ACTIONS

1. IF Emergency Diesel Fuel Tank (T-57A or T-57B) level is low, THEN perform the following:

   A. Verify the following valves open:

      - FO-4, T-25 Outlet valve
      - FO-6A, T-25 to T-57A Inlet
      - FO-6B, T-25 to T-57B Inlet

   B. IF valves are aligned properly, THEN FO Fltr EDG Supp (F-27) could be fouled.

      1. IF F-27 fouled, THEN switch to idle F-27 filter element AND initiate prompt action to replace fouled cartridge.

   C. IF low level is due to empty Diesel Oil Storage Tank (T-25), THEN ensure fuel oil is ordered and expedited.

   D. IF necessary, THEN refer to "Emergency Crossconnect of DG1 and DG2 Fuel Oil Systems" section of Emergency Diesel Generator Operation (1104.036).

   E. Refer to TS 3.8.1, TS 3.8.2 and TS 3.8.3 for fuel oil supply requirements.

2. IF Fuel Transfer Pump (P-16A) or (P-16B) discharge pressure is high, THEN verify proper valve lineup using (1104.036).

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

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

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

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

**Section:** 3.7 **Type:** Instrumentation

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

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

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

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

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

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

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

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

B. Chemistry must obtain grab samples every hour during a release via this pathway.

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

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

**Answer:**

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

**Notes:**

The requirements for release when the Liquid Radwaste Process Monitor is inoperable are
  a. An independent sample and analysis of the tank contents
  b. Computer input data independently verified.
"A" is therfore the correct answer.
"B" is incorrect, although grab samples are required when other rad monitors are inoperable, they are not required here.
"C" is incorrect, this is done if the flow recorder is inoperable.
"D" is incorrect, although this detector also monitors the flume, it is not required.

**References:**

1104.020, Chg. 051

**History:**

Used in 1999 exam.
Direct from ExamBank, QID# 2765
Used in 2001 RO/SRO Exam.
Modified for 2005 RO re-exam.
Selected for 2011 RO Exam.

ATTACHMENT B1



1.5 Record the following:

- Number of CW Pumps running ____
- CW pump Disch Press ________ psig

1.6 Submitted to Chemistry for Analysis, Section 2.0:

Date ________________ Time ______________

Section 1.0 Performed By ________________________________________________

2.0 ANALYSIS (Chemistry)

2.1 Sample Tank T-16A for release analysis using Sampling Treated Waste Monitor Tank (T-16A/B) (1607.009).

Date/Time____________/___________

2.2 IF Liquid Radwaste Process Monitor (RI-4642) is inoperable
OR unavailable as identified in either "Request", or "Verification of Pre-Release Requirements" sections of this permit,
THEN obtain independent sample of tank contents for analysis.

Date/Time____________/___________

2.3 Record selected tank pH _____________.

2.4 Review gamma spectroscopy report and Tritium analysis.

2.5 IF release is radioactive
AND release desired,
THEN generate Preliminary Release Report.

2.6 Check sample results indicate that release of total tank contents will not violate ANO radioactive effluent discharge limit.

2.7 IF Liquid Radwaste Process Monitor (RI-4642) is inoperable
OR unavailable as identified in either "Request", or "Verification of Pre-Release Requirements" section of this permit,
THEN perform independent analysis of computer data input.

Date/Time____________/___________

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

**QID:** 0832 **Rev:** 0 **Rev Date:** 5/23/11 **Source:** New **Originator:** D. Thompson

**TUOI:** AQLP-RO-SW **Objective:** 6 **Point Value:** 1

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

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

**Description:** Ability to explain and apply system limits and precautions.

**K/A Number:** 2.1.32 **CFR Reference:** 41.10 / 43.2 / 45.12

**Tier:** 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2

**Group:** 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** K

**Question:** **RO:** 52 **SRO:** 52

Which of the following is the reason SW bay level is verified to be at least 332 ft for SW pump operation?

A. Prevent water column separation

B. Ensure adequate NPSH

C. Prevent excessive load on pump thrust bearing

D. Prevent overheating of motor windings

**Answer:**

B. Ensure adequate NPSH

**Notes:**

"B" is correct, pump NPSH is assured by this level.
The remaining choices are possible pump problems.

**References:**

1104.029, Chg. 086

**History:**

New for 2011 RO Exam.

5.0 LIMITS AND PRECAUTIONS

5.1 Loss of Service Water (1203.030) lists the Unit 1 TS associated with an inoperable service water loop. Loss of Service Water cooling must be evaluated under the Safety Function Determination Program.

5.2 When switching service water bays from the lake to the emergency pond, or from the pond to the lake, service water pump discharge pressure and SW bay level should be monitored to ensure proper suction transfer.

5.3 Minimum SW bay level for continuous SW pump operation is 332′.

5.4 Maximum SW pump flow is 8000 gpm.

5.5 Except under emergency conditions, observe the following motor starting limits for P-4A, P-4B, and P-4C:

- Two starts from ambient temperature.
- One start from rated temperature.
- For starts in excess of these, allow the pump to run for thirty minutes, or sit idle for sixty minutes prior to restarting.

5.6 If auxiliary cooling water is in-service, emergency pond level will drop when SW suction is aligned to pond. ACW returns to discharge flume regardless of SW discharge alignment.

5.7 Actual SW/ACW loop flows of less than 1500 gpm will cause SPDS flow indication to indicate 0 gpm.

5.8 Operating a SW pump without packing leakoff may result in scoring of the pump shaft.

5.9 Foreign material inadvertently dropped in the Service Water Bays may be pulled into Service Water Pump suction resulting in pump damage or discharge pressure degradation.

5.10 With SW Bay Level starting at ≤ 332 feet and SW Flow ≥ 8000 gpm, SW pump NPSH limits could be exceeded during suction transfer from Lake to Pond. This could result in pump damage.

5.11 When SW Pump Suction is aligned to the ECP, ECP level shall remain ≥ 5.5 ft., and ECP temp shall remain <95°F. It is desirable to maintain ECP level >5.5 ft. to supply additional inventory margin.

5.12 Time spent with SW Pump suctions aligned to the ECP with the SW Pump in operation, other than "Loss of Dardanelle Reservoir" or "Excessive debris accumulation at the Intake Structure", or "Controlled Conditions" to support Maintenance or Testing, should be minimized.

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

**QID:** 0046 **Rev:** 1 **Rev Date:** 11/4/98 **Source:** Direct **Originator:** JCork

**TUOI:** A1LP-RO-EOP07 **Objective:** 3 **Point Value:** 1

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

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

**Description:** Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures.

**K/A Number:** A1.02 **CFR Reference:** 41.5 / 45.5

**Tier:** 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 4

**Group:** 1 **SRO Imp:** 2.6 **SRO Select:** Yes **Taxonomy:** Ap

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

Given:

- Degraded Power
- Both EDGs operating
- ESAS has NOT actuated
- P4C failed to start
- P4B out of service

Which of the following actions should be accomplished?

A. Close SW Loop II Isolation Valve (SW-10C).

B. Open SW Loop I & II Crossconnects (SW-5 and SW-6).

C. Close ACW Loop Isolation (CV-3643).

D. Cross-tie SW Loops at Makeup Pump (SW-14 thru SW-17).

**Answer:**

C. Close ACW Loop isolation (CV-3643).

**Notes:**

Answer "C" is correct per the Degraded Power EOP, this action is taken to reduce SW system loads.
Answer "A" is incorrect since this will only isolate the loop flow from P4C with the SW loops still cross-tied at the ICW coolers.
Answers "B" and "D" responses are taken for other SW operations but they would not reduce SW loads.

**References:**

1202.007, Chg. 009

**History:**

Developed for 1998 RO/SRO Exam.
Revised after 9/98 exam analysis review.
Used in 98 RO Re-exam
Selected for use in 2002 RO exam. Was KA A4.02
Selected for use on 2007 RO Exam
Selected for 2011 RO Exam.

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| **2. Verify SW to DG1 and DG2 CLRs open to operating EDGs (CV-3806 and 3807).** | |
| **3. Verify <u>OR</u> start a Service Water pump on <u>each</u> operating DG, after 15-second time delay (P4A, B, C).** | 3. <u>IF</u> both EDGs are operating<br>**<u>AND</u>**<br>only one Service Water pump can be started<br>**<u>AND</u>**<br>ESAS has <u>not</u> actuated,<br>**<u>THEN</u>** perform the following:<br><br>A. Close ACW Isolation (CV-3643).<br><br>B. Verify both Service Water to ICW Coolers Supply valves open (CV-3811 and 3820). |
| **4. Actuate MSLI for both SGs <u>AND</u> verify proper actuation and control of EFW and MSLI (RT 6):**<br><br>A. Operate ATM Dump CNTRL valves in HAND to minimize cycling and conserve Instrument Air.<br><br>B. **<u>IF</u>** Instrument Air to ATM Dump CNTRL valves is lost,<br>**<u>THEN</u>** perform the following:<br><br>1) Dispatch an operator with a radio to place ATM Dump CNTRL valves on hand jack **<u>AND</u>** fully open (Refer to Alternate Shutdown (1203.002), Exhibit A)<br><br>2) Establish SG press control using ATM Dump ISOL valves in MANUAL from the Control Room. | 4. <u>IF</u> all EFW is lost,<br>**<u>THEN</u> GO TO step 53.** |
| **5. Check RCS press remains ≥1700 psig**<br>**<u>AND</u>**<br>**PZR level remains ≥30".** | 5. Initiate HPI (RT 2). |

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

**QID:** 0227 **Rev:** 1 **Rev Date:** 5/2/05 **Source:** Direct **Originator:** J. Cork

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

**Section:** 3.8 **Type:** Plant Services Systems

**System Number:** 078 **System Title:** Instrument Air System (IAS)

**Description:** Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems.

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

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

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

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

Instrument Air pressure has dropped to 68 psig.
Field operators can not find an Inst. Air leak on Unit One.

Which of the following is the appropriate response for the given plant conditions to restore or conserve Instrument Air pressure?

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

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

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

D. If ICW available, isolate Seal Injection by closing CV-1206.

**Answer:**

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

**Notes:**

Per 1203.024, the U1 to U2 cross connect should be closed first, so [b] is correct.
[a] is incorrect, this does not occur until pressure is at 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 unless necessary to maintain PZR level <290".

**References:**

1203.024, Chg. 012

**History:**

Developed for 1998 RO/SRO Exam QID 0102.
Modified for 98 RO Re-exam
Modified for 2005 RO exam.
Selected for 2011 RO Exam.

## SECTION 1 -- LOW INSTRUMENT AIR PRESSURE (≤75 PSIG)

DISCUSSION

Low IA pressure can be caused by numerous conditions. This section assumes a gradual loss of air pressure with no major malfunction of air operated equipment. Expeditious action is required to minimize the impact on air operated systems and components. For additional discussion, see Attachment D.

1.0 SYMPTOMS

1.1 IA header pressure dropping.

1.2 INST AIR HEADER PRESS LO (K12-B3) alarm

1.3 INST AIR COMPRESSOR TROUBLE (K12-C3) alarm

1.4 M-1/F-8 ΔP (K21-5) alarm

1.5 BREATHING AIR COMPRESSOR AUTOSTART (K15-B4) alarm

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

**NOTE**

- IA HDR Pressure can be monitored using PMS point P5409.
- ANO2 Instrument Air Header Supply Pressure is PMS point P3013.

3.1 Verify standby Instrument Air Compressor(s) (C-28A/B, C-2A/B) running.

3.2 Dispatch an operator to determine specific compressor, air dryer, and filter condition.

3.3 IF IA is supplying respirable air,
THEN inform RP of loss of IA pressure, and that workers must back out of work in progress and isolate the IA supply.

3.4 IF low IA header pressure is due to loss of IA on Unit 2
AND IA is crossconnected,
THEN perform the following:

3.4.1 IF Unit 1 IA header pressure drops below 60 psig,
THEN direct Unit 2 control room operators to terminate crossconnection.

3.4.2 **GO TO** step 3.7.

**QID:** 0716 **Rev:** 2 **Rev Date:** 4/10/2008 **Source:** Repeat **Originator:** David Thompson

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

**Section:** 3.5 **Type:** Containment Integrity

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

**Description:** Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Containment System controls including: Containment Pressure, Temperature, and Humidity.

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

**Tier:** 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

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

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

Given:

Plant power 100%

Rising Reactor Building temperature
Slowly rising Reactor Building pressure

Rising Reactor Building Dew Point
Reactor Building Leak Detector rising activity

Per 1203.039, "Excess RCS Leakage", what action addresses these conditions?

A. Initiate Reactor Building Purge.

B. Start the Hydrogen Recombiners.

C. Maximize Reactor Building Cooling.

D. Place a second Main Chiller in service.

**Answer:**

C. Maximize Reactor Building Cooling.

**Notes:**

A is incorrect. Reactor Building Purge is not allowed at power due to loss of containment integrity.
B is incorrect. Hydrogen Recombiners would not correct this condition.
C is correct. Maximizing Reactor Building Cooling is a compensatory measure of 1203.039 to protect the Containment system for this condition.
D is incorrect. Due to system design, a second chiller would not result in appreciable improvement in Containment conditions.

**References:**

1203.039, Chg. 011

**History:**

New for the 2008 RO Exam.
Repeated in 2011 RO Exam.

## INSTRUCTIONS

**1. IF HPI is required to maintain RCS inventory**
**AND**
**SG tube leakage is not indicated,**
**THEN trip the reactor**
**AND perform Reactor Trip (1202.001), while continuing with this procedure.**

**2. IF desired,**
**THEN open BWST Outlet (CV-1407 or CV-1408) to the OP HPI pump.**

**3. IF desired,**
**THEN reduce OR isolate letdown flow.**

**4. IF leak is inside RB,**
**THEN consider performing Repetitive Tasks (1202.012), Maximize RB Cooling (RT 9).**

**NOTE**

- Steps 6 through 12 may be performed in any sequence.
- Step 14 provides guidance for plant shutdown.

**5. IF location of leak is known,**
**THEN perform the applicable following steps (6 through 12):**

- RCS Sample Lines — step 6
- Primary to Secondary Leakage — step 7
- RCP Seal Degradation — step 8
- Makeup & Purification System Leakage — step 9

- RB Sump Inleakage — step 10
- Quench Tank Inleakage — step 11
- RCS Leakage into ICW System — step 12

A. Otherwise, perform steps 6 through 12 until the leak is isolated or leak location is identified.

**6. Verify RCS sample isolation valves closed.**

- Letdown Sample (SV-1841)
- Pressurizer Sample Valves - Steam Space (CV-1814)
- Pressurizer Sample Valves - Water Space (CV-1816)
- Hot Leg Sample (SV-1840)

# ANO 2011
# RO Questions
# Tier 2 Group2

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

**QID:** 0220 **Rev:** 0 **Rev Date:** 11/12/98 **Source:** Direct **Originator:** J. Cork

**TUOI:** A1LP-RO-NNI **Objective:** 19 **Point Value:** 1

**Section:** 3,2 **Type:** Reactor Coolant System Inventory Control

**System Number:** 011 **System Title:** Pressurizer Level Control System (PZR LCS)

**Description:** Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: Impact of high/low PZR level on interrelated system.

**K/A Number:** K5.13 **CFR Reference:** 41.5 / 45.7

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

**Group:** 2 **SRO Imp:** 3.4 **SRO Select:** Yes **Taxonomy:** An

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

Given:

- Reactor power steady at 80%.
- PZR Level Control selected to LT-1001 on C04
- A break in LT-1001's reference leg causes indicated level to rise.

How will Makeup Tank (T-4) level be affected by this failure?

A. Makeup flow drops, Makeup Tank level will rise.

B. No effect, SASS will auto select LT-1002.

C. Makeup flow rises, Makeup Tank level will drop.

D. Letdown flow drops, Makeup Tank level also drops.

**Answer:**

A. Makeup flow drops, Makeup Tank level will rise.

**Notes:**

With LT-1001's reference leg empty, the differential pressure transmitter will sense a minimum DP which results in maximum level indication. Unlike most of the NNI system SASS has no effect on PZR level instruments (b is thus incorrect), therefore the makeup valve, CV-1235, will close due to the indicated high level. Letdown flow will remain constant (making distracter "d" incorrect) and therefore Makeup Tank level will increase as in answer [a].
If the trainee determines indicated PZR level will decrease, then he will probably choose [c].

**References:**

1105.006, Chg. 010
STM 1-69, Rev. 14

**History:**

Developed for use in 98 RO Re-exam
Selected for 2005 RO re-exam. K/A 011 K3.01
Selected for 2011 RO Exam.

3.13 After a SASS trip has occurred, the AUTO pushbutton must be pressed to return the channel to AUTO. Transfer to AUTO is inhibited if a mismatch exists.

3.14 The Mismatch Alarm Bypass Switch is used to bypass a channel's input to SASS MISMATCH (K07-B4).

3.15 Pressurizer Level Transmitter HS on C04 selects either of two compensated level signals (LT-1001 or LT-1002) as input to the following:

- Pressurizer Level Control Valve (CV-1235) H/A station
- Pressurizer Lo-Lo Heater Cutoff (LS-1001)
- Pressurizer Hi/Hi-Lo/Lo Alarm
- Dasey Panel PZR LVL (LI-1000)

The compensated Pressurizer Level recorder and indicator on C04 are totally independent of the NNI X/Y systems and the Pressurizer Level Transmitter HS on C04.

3.16 Pressurizer Temperature Transmitter HS on C04 selects either of two temperature elements (TE-1001A or TE-1002A) to feed the Pressurizer Temp indicator on C04. The signal not selected is sent to the plant computer.

Temperature compensation of pressurizer level signals is accomplished independent of the NNI X/Y systems. Each level signal is compensated by a specific temperature signal at EFIC Signal Conditioning Cabinet (C539 or C540).

3.17 RC Pressure RPS A RPS C HS on C04 is a SASS selector switch which selects input from RPS A (PT-1021) or RPS C (PT-1038) for control of the following systems:

- Pressurizer Heater Control.
- Pressurizer Spray Valve Control
- Electromatic Relief Valve Control (high pressure setpoint)

In SASS ENABLE position, RPS A (PT-1021) is selected as the preferred input.

3.18 The three-position Cntrlg T-Hot HS on C03 selects T-hot of loop "A", T-hot of loop "B", or the average of loops "A" and "B" (marked UNIT, from RC Loop A/B Hot Leg T-ave TY-1023 in C47). The selected signal is used by the ICS for control. This signal is also used by Reactor Coolant T-hot (TR-1023) on C13 and the recorder's HI alarm contact, RC Loop A/B Hot Leg (TS-1023).

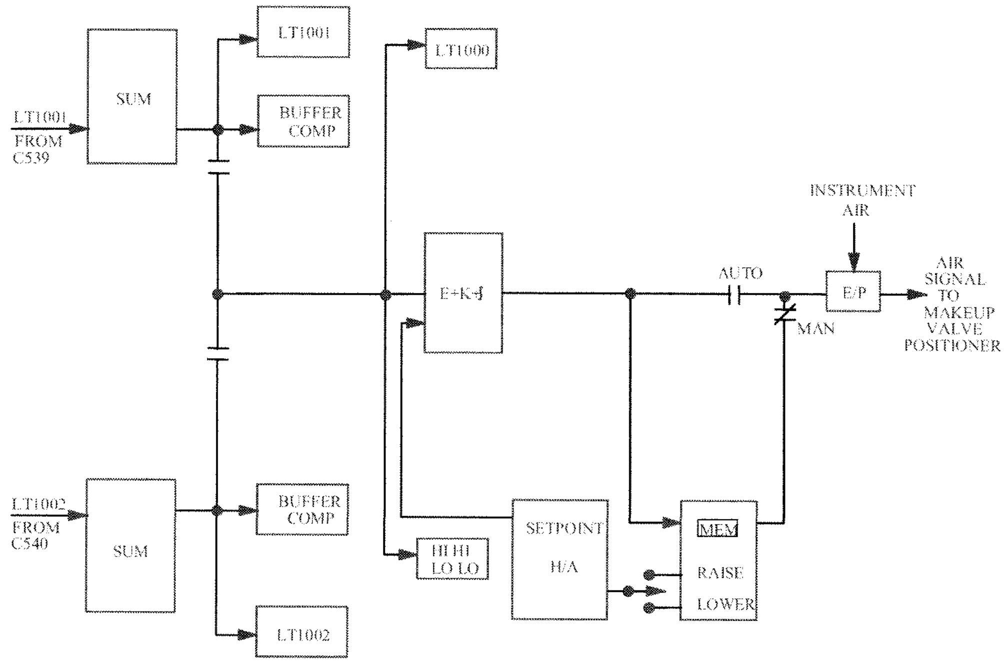



FIGURE 69.20: NNIX PRESSURIZER LEVEL CONTROL

The pressurizer level controller consists of a summing junction, a proportional amp, and an integral amp. The summing junction takes the difference between the pressurizer level signal and the pressurizer level setpoint. The setpoint is located on the pressurizer control station (C03) and is operator adjustable. The normal setpoint is 220". The resulting output error is supplied to a proportional/integral amplifier. The proportional amplifier provides quick response to a level error. The integral amplifier will raise or lower the output until the input error is reduced to zero. The output of the level controller is supplied a voltage to pneumatic converter (via auto/man contacts) and the memory unit of the pressurizer level control station. The output of the memory unit tracks the output of the proportional/integral amplifier when the controller is in automatic. When manual is selected, a bumpless transfer occurs. The position of the makeup valve is then controlled through the use of the toggle switch on the pressurizer level control station.

An indicator on the pressurizer level control station shows the condition of the control loop. A two-position switch allows selection of "measured variable" or "position". The following table shows the meter indication inputs.

| | POSITION | MEASURED VARIABLE |
|---|---|---|
| Indication with controller in AUTO | Prop/Int. amp output to E/P | Auto/Manual difference |
| Indication with controller in MANUAL | Analog memory output to E/P | Prop/Int. amp output |

**QID:** 0833 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** Direct **Originator:** D. Thompson

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

**Section:** 3.1 **Type:** Reactivity Control

**System Number:** 014 **System Title:** Rod Position Indication System

**Description:** Knowledge of the RPIS design feature(s) and/or interlock(s) which provide for the following: Zone reference lights.

**K/A Number:** K4.04 **CFR Reference:** 41.5 / 45.7

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 2.6 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 2 | **SRO Imp:** | 2.9 | **SRO Select:** | Yes | **Taxonomy:** | K |

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

As indicated in the CRD system logic cabinets, each CRDM has __________ absolute zone reference switches, NOT counting the "out limit" and an "in limit" switches.

A. 5

B. 7

C. 9

D. 11

**Answer:**

A. 5

**Notes:**

"A" is correct, the others are odd numbered choices.

**References:**

STM 1-02, Rev. 9

**History:**

Direct from regular exam bank QID ANO-OPS1-721.
Selected for 2011 RO exam.

a square fiberglass housing that is mounted on the outside of the upper motor tube. Each API assembly contains 79 reed switches and a voltage dividing resistor network.

The reed switches are normally divided into four groups.

The in-limit switch and 0% switch. The in-limit switch is actuated at approximately 1.5 inches above the tripped position. The 0% switch is actuated at approximately 2.25 inches above the tripped position.

The out-limit switch and 100% switch. The out-limit switch is actuated at approximately 138.5 inches. The 100% limit switch is set at approximately 137.25 inches.

Three switches, actuated at 25, 50, and 75% travel, along with the 0% and 100% switches, make up the zone reference indication. These switches are actuated at 34.75 ± 1.50, 69.50 ± 1.50, and 104.25 ± 1.50.

An analog output corresponding to leadscrew position over the full travel is provided. This indication is derived from the sequential closing and opening of 72 reed switches that are connected to a resistor network. Each API assembly consists of two individual position indication circuits (an "A" and "B" circuit). Each circuit contains a string 36 reed switches at 4 inch intervals and a voltage dividing resistance network. The strings are staggered by 2 inches so that there is a reed switch every 2 inches of rod travel. The reed switch closes when the permanent magnet located on the torque taker is in the immediate vicinity of the reed switch. The reed switch returns to its normally open position when the magnet continues past the reed switch. (Refer to figure 2.37) The effect is an increasing output voltage from the reed switch/resistor network as the control rod is withdrawn. The two strings are averaged to provide one output. Switches located on the individual buffer amplifier cards can be used to switch either string off in the event of a failure in one of the strings. Accuracy of the system ± 2.5 inches.

As previously mentioned, two switches are used at both extremes of rod travel. On the upper end, one switch (100%) indicates that a particular control rod is nearing its full out position. The other, "out limit", stops further out-travel. The first rod in any group to reach the second switch (out limit) will stop further outward travel of all rods in that group. The same type of setup is used to limit movement in the "in" direction. There is a 0% switch and an "in limit" switch. The first rod in a group to reach the "in limit" will stop any further motion in the "in" direction.

### 2.6.2 Relative Position Indication System

The relative position indication receives its signal by monitoring the input pulses to the CRDM motor. Refer to figure 2.38. Every other phase of the six-phase input to the CRDM is used to drive a stepping motor which operates a rheostat. Changing the rheostat

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

**QID:** 0464 **Rev:** 0 **Rev Date:** 5/21/2002 **Source:** Direct **Originator:** J.Cork

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

**Section:** 3.7 **Type:** Instrumentation

**System Number:** 015 **System Title:** Nuclear Instrumentation System

**Description:** Ability to monitor automatic operation of the NIS, including: Verification of proper functioning/operability.

**K/A Number:** A3.03 **CFR Reference:** 41.7 / 45.5

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 3.9 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 1 | **SRO Imp:** | 3.9 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** 58 **SRO:** 58

Given:

- Reactor startup is in progress.
- Count rate on NI-1 & NI-2 is 1E3 CPS.

During the next rod pull, the HI SUR rod hold alarms.

The most likely cause is due to SUR exceeding:

a. 1 DPM on the source range monitors.

b. 2 DPM on the source range monitors.

c. 2 DPM on the intermediate range monitors.

d. 1 DPM on the intermediate range monitors.

**Answer:**

b. 2 DPM on the source range monitors.

**Notes:**

b. is the correct answer, 2 DPM on the source range monitors will cause the HI SUR alarm .
a. is an incorrect answer, 1 DPM on the source range monitors will not cause the HI SUR alarm to come in.
c. is an incorrect answer, 2 DPM on the intermediate range monitors will not cause the HI SUR alarm to come in. Setpoint for the IR is 3 DPM .
d. is an incorrect answer, 1 DPM on the intermediate range monitors will not cause the HI SUR alarm to come in. Setpoint for the IR is 3 DPM .

**References:**

1105.009, Chg. 036

**History:**

Direct from regular exambank QID 1789.
Selected for use in 2002 RO/SRO exam.
Selected for 2011 RO Exam.

3.5 Group 8 (APSR) rods are used to shape core axial flux distribution. This group has manual control only. Like regulating groups, Group 8 has a regulating power supply and can be transferred to the auxiliary power supply. APSRs are mechanically held by contact buttons on the bottom portion of the segment arms to prevent insertion when drive power is de-energized. They do not drop on reactor trip.

3.6 Diamond Panel

TRIP CONF lamp, when on, indicates that control rod drives are de-energized and, except for Group 8, should be fully inserted into the core. All trip CRDM breakers should be open at this time.

ASYMM FAULT lamp, when on, indicates that any rod's API position is >6.5% from its API group average position.

- Individual fault lamps on the PI Panel indicate a rod is >5% out of alignment with its group average position.

OUT INHIBIT lamp, when on, indicates that control rods will not respond to out commands. Control rod out inhibits:

- Source range SUR >2 DPM and reactor power <10% and IR <10-9 amps.
- IR range SUR >3 DPM and reactor power <10%.
- Loss of any safety group (1-4) out limit and reactor power >40%.
- Any rod group asymmetric fault (any rod >6.5% from group average) and reactor power >40%.

SEQUENCE INHIBIT lamp, when on, indicates that regulating groups cannot be withdrawn in sequence. A sequence monitor provides control input for this indication. The lamp will come on if regulating groups are operated in any of the following conditions.

- Group 5 less than 80% and Group 6 greater than 5%.
- Group 5 less than 95% and Group 6 greater than 20%.
- Group 6 less than 80% and Group 7 greater than 5%.
- Group 6 less than 95% and Group 7 greater than 20%.
- Group 5 less than 95% and Group 7 greater than 5%.

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

**QID:** 0403 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Direct **Originator:** D.Slusher

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

**Section:** 3.7 **Type:** Instrumentation

**System Number:** 016 **System Title:** Non-nuclear Instrumentation

**Description:** Ability to manually operate and/or monitor in the control room: NNI channel select controls.

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

**Tier:** 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 4

**Group:** 2 **SRO Imp:** 2.8 **SRO Select:** Yes **Taxonomy:** An

**Question:** **RO:** 59 **SRO:** 59

Initial Conditions:

- SASS Mismatch Annunciator
- Rx is Feedwater Limited Annunciator
- Feedwater is Rx Limited Annunciator
- Unit Tave TI-1032 is 583 degrees F
- Loop A Tave TI-1020 is 588 degrees F
- Loop B Tave TI-1043 is 578 degrees F

Which of the following actions would clear the cross limits and return temperature indications to normal?

A. Select the NNI-Y signal for RCS loop A hot leg temperature.

B. Place the Controlling Tave Selector Switch in the Loop B position.

C. Place Loop A Feedwater Demand in hand and raise Loop A feedwater flow.

D. Select the NNI-Y signal for RCS loop B cold leg temperature.

**Answer:**

a. Select the NNI-Y signal for RCS loop A hot leg temperature.

**Notes:**

Answer [a] is correct, because the annunciator for SASS mismatch points to an instrument failure. The "Feedwater is Rx Limited" annunciator indicates a problem with a false high temperature feeding into the unit Tave since this annunciator alarms when FW is less than demand by 5% or more. This points to a failed high Thot instrument since a failed Tcold instrument would cause the affected loop to be artificially low and the "Feedwater is Rx Limited" annunciator would not be in.
Answer [b] is incorrect, due to instrument failure, the controlling Tave would not be selected until ICS components were placed in Manual.
Answer [c] is incorrect, ICS would be trying to raise A FW flow due to the cross limits.
Answer [d] is incorrect, "B" T cold is not the problem, the loop with the higher temperature has the failed instrument.

**References:**

STM 1-64, Rev. 12
STM 1-69, Rev. 14

**History:**

Direct from exambank 4574, used in 2001 SRO Exam.
Selected for 2005 RO re-exam.
Selected for 2011 RO Exam.

The megawatt demand signal is converted to a total feedwater demand by a feedwater demand calculator. This conversion is needed because feedwater flow is not linear to generated megawatts (Figure 64.22).

Feedwater demand is then modified by feedwater temperature error which is developed by comparing measured feedwater temperature to a feedwater temperature program developed from the demand signal. (Refer to figure 64.23) This characterization of the feedwater demand signal is required to compensate for the change in Btu input from feedwater. If feedwater temperature is low, then feedwater flow should be decreased. The primary purpose of this function is to maintain a constant Btu/lb of steam for large temperature errors such as might be experienced on a bypass of feedwater heaters.

FIGURE 64.22: TOTAL FEEDWATER FLOW vs. POWER

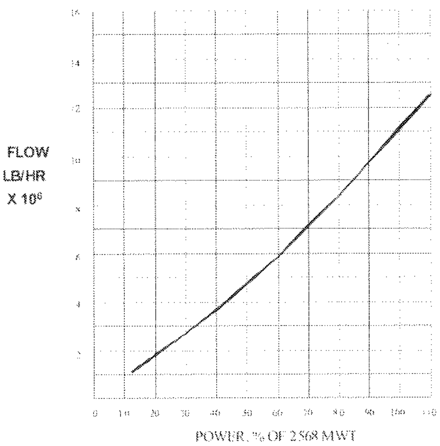
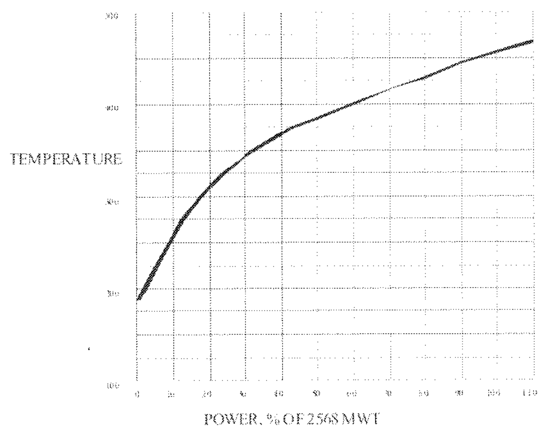



FIGURE 64.23: FEEDWATER TEMPERATURE vs. POWER LEVEL




## 2.6.1 Cross Limiting

One requirement for proper steam production is that the feedwater flow/neutron power ratio must never exceed predetermined limits. Whenever the feedwater control is on automatic, a set of limits is imposed on the feedwater demand to maintain feedwater flow within 5% of the neutron power. The cross limit of feedwater is taken from neutron error in the reactor control subsystem. Greater than a ±5% neutron error will modify the feedwater demand signal. If you assume the feedwater demand and the reactor demand signals are together then if power is less than demand by more than 5%, the amount of error greater than 5% will decrease feedwater by that amount. For example, the demand has increased, the reactor is not responding, thus hold back the feedwater demand in order to keep the reactor and feedwater within 5% of each other. Power greater than demand by more than 5%, will increase feedwater demand.

If either limiting action on feedwater does occur, "Feedwater is Reactor Limited" annunciator will alarm and the ICS will be transferred into the "Tracking" mode. The occurrence of this limiting action indicates that the neutron power is not able to satisfy its demand. Therefore, by modifying the feedwater demand signal with the neutron error, feedwater is held to within 5% of reactor power. Since the ICS is in Track, the turbine merely controls header pressure and thus the load can be no greater nor less than 5% of the neutron power.

### 2.6.2 Load Ratio (ΔTc) Control

The total feedwater flow demand signal is split by the ICS into loop "A" and "B" feedwater demand signals by adjustment of the value of a multiplier controller. This controller sets the value of loop loop "A" feedwater demand by multiplying the total flow demand by the value of the multiplier. If the multiplier is set at .5, half of the total feedwater flow demand signal becomes loop "A" feedwater demand. The loop "B" feedwater demand is determined by subtracting the loop "A" demand from the total demand. Changing the multiplier value will change the value of both loop demand signals. The maximum loop feedwater demand signal is 6 x $10^6$ pounds mass per hour.

The value of the multiplier is set by the value of a control signal. This signal is the algebraic summation of two other signals. One of these signals is the RCS flow mismatch signal and will be zero when all four RCP's are properly operating. This signal will be described under "Three Pump Operations". The other signal is the ΔTc correction signal.

The control of the ratio of feedwater to each OTSG will determine the amount of heat that will be removed from the primary water in the reactor coolant system (RCS) and the relative amount of loading that each OTSG will carry. Therefore, the loading of the OTSGs can be indicated by the relative RCS return temperatures to the reactor (Tc's). If the difference in the Tc's ($\Delta T_c$) is controlled near zero, then each OTSG will be loaded properly for the RCS flow through it. A trip of one RCP would give an immediate re-ratioing. An important benefit of keeping ΔTc low is that quadrant tilts within the reactor may be kept to a minimum.

The actual ΔTc is compared to the ΔTc setpoint. The difference (ΔTc Error) is used to generate the ΔTc correction signal. A zero ΔTc correction signal will split the signal equally between the loops.

The operator may choose to manually control the ΔTc correction signal by placing the Load Ratio Hand/Automatic Station in hand. The only difference between this station and the other feedwater hand/auto stations is the additional dial and knob located under the meter. This provides the ΔTc setpoint for automatic operation. The setpoint may be varied from 0% to 100% which corresponds to -10°F to +10°F. The normal value is 50% (0°F).

When position is selected on this station, the ΔTc correction signal is indicated on the meter. If the meter indicates 50%, the correction signal is zero (loop "A" multiplier set at .5) and loop demand signals are equal. If the meter indication is above 50%, then

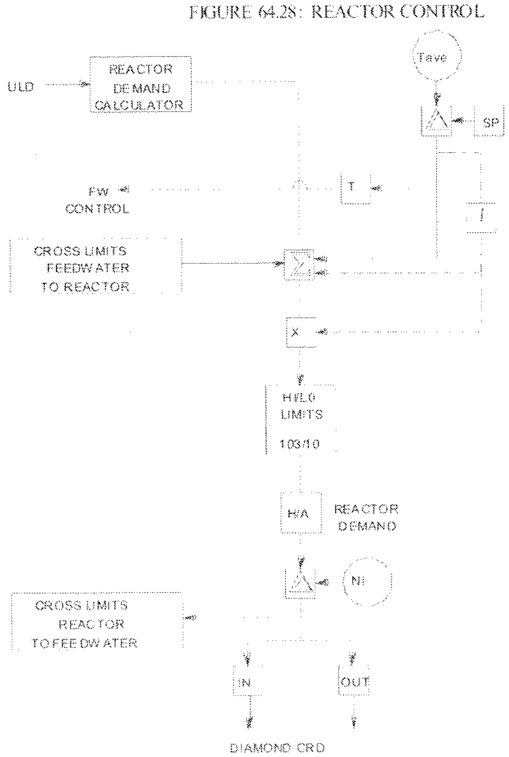
"A" OTSG. $\Delta T_c$ will increase as power increases and will continue to increase until $T_{ave}$ reaches 579°F. During any further power increases, the operator will manually increase feedwater flow to the "A" OTSG to maintain Tave at 579°F. As "A" OTSG flow is increased the $\Delta T_c$ will move toward zero. At a $\Delta T_c$ of zero, the operator should increase both feedwater loop demands. When both OTSGs are above low level limit, the operator may place the ICS in auto. The total flow circuit will then be blocked until low level limit is reached while operating on 3 RCP's during a plant shut down or load reduction.

## 2.7 Reactor Demand Subsystem

Refer to figure 64.28.

The megawatt demand signal that is received by the Reactor Demand Subsystem from the Integrated Master Subsystem has a Low Limit of 15%. It is undesirable to lower reactor power to less than 15% in automatic Therefore, the demand being sent to the reactor demand calculator will not be allowed to go below a value that is translated to 15% by the reactor demand calculator, should be about .15 × 902 or ~135 megawatts. However, procedurally the operator will take manual control of the reactor when reduction of reactor power to < 20% is desired.

FIGURE 64.28: REACTOR CONTROL




Since reactor power is not linear with generated megawatts, the reactor demand calculator changes the megawatt demand signal to a reactor demand signal equivalent to 0-125% power. The calculator output span then is 15% to 125% taking into account the low limit on the input.

The reactor demand signal is then modified as needed to keep $T_{ave}$ equal to setpoint. $T_{ave}$ is compared to the $T_{ave}$ setpoint which is controlled by the operator at the reactor demand H/A station. A 0% to 100% selection is possible. The 0% is equal to 520°F and 100% is equal to 620°F. Therefore, 59% (579°F) is the normal setpoint. If a $T_{ave}$ error exists it is used in both a proportional and integral action to adjust reactor demand.

The adjusted reactor demand signal is limited to between 10% and 103%. The low limit of 10% is there to allow $T_{ave}$ correction to decrease power a maximum of 5% when trying to establish $T_{ave}$ at setpoint. This could occur if low level limits are set too low. The high limit of 103% allows a $T_{ave}$ correction of 3% when reactor demand is 100%. However, the main purpose of the 103% limit is to prevent an automatic signal from raising power to its RPS trip setpoint.

The adjusted and limited reactor power demand signal is compared to the high auctioneered reactor power signal from the reactor protection system. The difference between the two signals is termed "Neutron Error". If actual power is greater than reactor demand, a positive neutron error results. If neutron error is > +1%, the control rods move into the core to reduce power until neutron error becomes < +.975%. If actual power is less than reactor demand, a negative neutron error results. If neutron error is > -1%, the control the control rods move out of the core to increase reactor power until neutron error becomes < -.975%.

### 2.7.1 Cross limits

The purpose of crosslimits is to keep the heat production (the reactor) and the heat removal (feedwater) within 5% of each other. In accomplishing this purpose, ICS assumes that reactor demand and feedwater demand are matched. Therefore, if actual reactor power is out from demanded reactor power, it is also out from demanded feedwater flow.

The first of the two crosslimits concerns reactor power which was discussed earlier but will be repeated here. If reactor power is > ± 5% out from reactor demand, then it is out from feedwater demand by > ± 5%. If actual feedwater flow is equal to its demand, then actual reactor power is > ± 5% mismatched to feedwater flow. To correct this problem, the amount of mismatch greater than ± 5% is calculated and sent to adjust total feedwater demand by that amount. An alarm "Feedwater is Reactor Limited" is given. This means that the feedwater demand is being limited by the reactor mismatch (neutron error).

The second crosslimit has to do with feedwater flow. We could have a crosslimit setup identical to the one for the reactor. However, this could put us in the condition of having rods being pulled to raise reactor power when a feedwater flow mismatch occurred, this was determined to be undesirable.

If total feedwater demand is 5% greater than total feedwater flow, then the excess above 5% is used to correct (lower) reactor demand. The basis for this crosslimit is that, if for some reason feedwater flow is not able to keep up with demand, then hold back (decrease) reactor power to keep reactor and feedwater together. An alarm "Reactor is Feedwater Limited" will be received and the unit will go into track.

When a crosslimit occurs, the unit will go into track. As previously described, this means that actual generated megawatts now becomes unit load demand instead of the previous setting on the unit master.

### 2.7.2 Reactor Control In Manual

Manual control of the reactor can be taken from either the Reactor Demand Hand/Auto Station or the Diamond Control Rod Station. The following description will assume that the rest of ICS is is in full automatic control.

When either one of these stations is placed in manual, the ICS goes into the tracking mode of operation. Any increases in reactor power will increase the steam production of the OTSG. This in turn causes the turbine governor valves to be opened to keep turbine header pressure equal to it's setpoint. The increase in steam flow to the turbine will generate more electricity. The increase in megawatt production results (because of track) in feedwater flows being increased to match the reactor power. Decreases in reactor power result in the same adjustments in the opposite direction. This mode of operation may be thought of as the "Reactor Leading" mode.

#### *2.7.2.1 Reactor Demand in Manual Only*

The operator initiates changes at this station by moving a toggle switch on the station face. A meter on the station indicates the transfer to the instrument AC when the vital power source is lost. The ABT will shift back to the vital source 10 minutes after vital power is restored to the ABT.

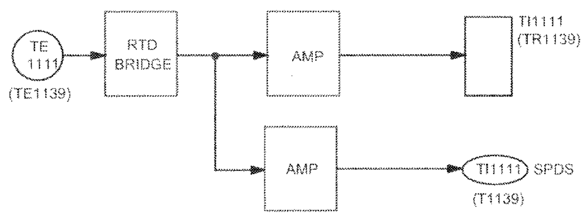
## 3.3 Reactor Coolant System Instrumentation

### 3.3.1 RCS Hot Leg (Th)

Three dual element RTDs are located on each RCS hot leg on the vertical piping at the outlet of the reactor vessel. Hot leg RTD locations are as follows:

| RCS A Loop | RCS B Loop |
|---|---|
| TE-1011 | TE-1039 |
| TE-1012 | TE-1040 |
| TE-1013 | TE-1041 |
| TE-1014 | TE-1042 |
| TE-1111 | TE-1139 |
| TE-1112 | TE-1140 |

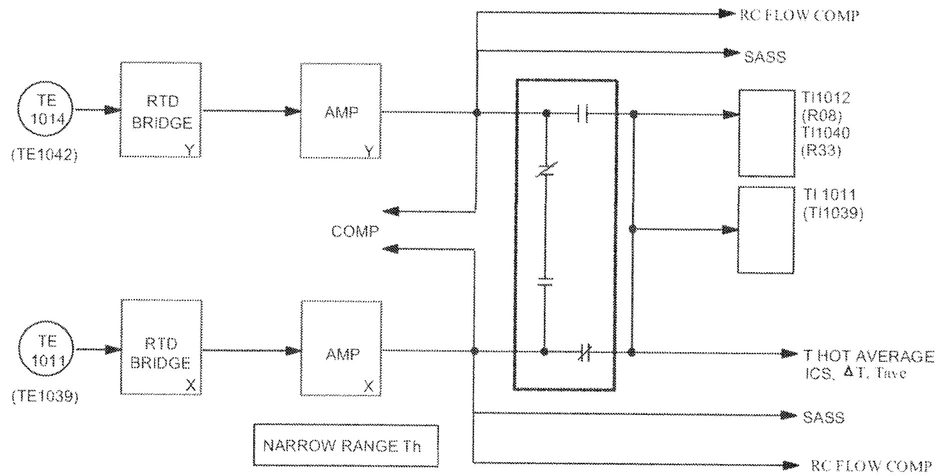
TE-1139 and TE-1112 provides an input into C-540B. TE-1111 inputs into C-539B. The temperature elements input into an RTD bridge that converts the resistance of the RTD to a corresponding output voltage. The output then goes to isolation amplifiers. The isolation amplifiers supply outputs to the SPDS computer and hot leg temperature indication on C03. The range of temperature indication is 50 °F to 700°F.




WIDE RANGE Th

TE-1011 and TE-1039 input into the NNIX cabinet. TE-1014 and TE-1042 input into the NNIY cabinet. The temperature elements input into an RTD bridge, which converts the resistance of the RTD to a corresponding output voltage. The output then goes to amplifiers (millivolt to volt converters) which increase the voltage to a usable level (-10v to +10v). The output from the amplifiers inputs to the transfer relay, Smart Automatic Signal Selection System, RCS flow instrument for density compensation, and plant computer. The transfer relay selects the temperature that is displayed on C03, remote shutdown instruments, and used to compute RCS average and differential temperatures. The NNI hot leg temperature range is 520 °F to 620 °F.

A three-position selector switch is located on C03. The switch positions are "AUTO", "X", and "Y". With the selector switch in the "AUTO" position, SASS provides automatic signal auctioning. Normally, the NNIX signal is selected by the SASS system. When the NNIX signal fails (as outlined in the SASS section), SASS de-energizes the transfer relay to select the NNIY temperature. The "X" and "Y" switch positions allow the operator to select the NNIX or NNIY signal independent of the SASS system. The remainder of the hot leg temperature elements input into the RPS cabinets.




## 3.3.2 RCS Wide Range Cold Leg Temperature

Two dual element RTDs are located at the suction of each RCP. RCS Cold temperature elements are as follows:

| RCS Loop "B" | | RCS LOOP "A" | |
|---|---|---|---|
| "A" RCP | "B" RCP | "C" RCP | "D" RCP |
| TE-1047 | TE-1044 | TE-1017 | TE-1015 |
| TE-1048 | TE-1045 | TE-1018 | TE-1016 |
| TE-1147 | TE-1144 | TE-1117 | TE-1115 |
| TE-1148 | TE-1145 | TE-1118 | TE-1116 |

The wide range RCS cold leg temperature indication is supplied from C-539B and C-540 B. TE-1144 and TE-1117 input into C-539B while TE-1148 and TE-1115 input into C-540B. TE-1147, TE-1145, TE-1118, and TE-1116 are spares. The temperature elements input into an RTD bridge, which converts the resistance of the RTD to a corresponding output voltage. The output then goes to isolation amplifiers. The isolation amplifiers supply outputs to the SPDS computer and cold leg temperature indication on C03. The range of temperature indication is 50 °F to 700°F.

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

**QID:** 0723 **Rev:** 1 **Rev Date:** 5/1/2008 **Source:** Repeat **Originator:** Steve Pullin
**TUOI:** A1LP-RO-RBVEN **Objective:** 5 **Point Value:** 1

**Section:** 3.5 **Type:** Containment Integrity
**System Number:** 028 **System Title:** Hydrogen Recombiner and Purge Control System
**Description:** Knowledge of bus power supplies to the following: Hydrogen Recombiners

**K/A Number:** K2.01 **CFR Reference:** 41.7
**Tier:** 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3
**Group:** 2 **SRO Imp:** 2.8 **SRO Select:** Yes **Taxonomy:** K

**Question:** **RO:** 60 **SRO:** 60

Which MCC supplies power to the Hydrogen Recombiners, M-55A and M-55B?

A. M-55A B-53;
M-55B B-61.

B. M-55A B-6;
M-55B B-5.

C. M-55A B-61;
M-55B B-53.

D. M-55A B-5;
M-55B B-6.

**Answer:**

A. M-55A B-53; M-55B B-61

**Notes:**

A is the correct vital power supplies
B is incorrect but a credible distracter since the amperage of these units border on load center requirements.
C is incorrect. They are the correct load centers but for the wrong units.
D is incorrect but a credible distracter since the amperage of these units border on load center requirements.

**References:**

STM 1-09, Rev. 11

**History:**

New for the 2008 RO Exam.
Repeated for 2011 RO Exam.

(chromel-alumel) thermocouples welded to the heater sheaths. These thermocouples are monitored on C-26, are non-Q and are not required for post-LOCA operability.

The heated air rises to the mixing chamber at the top of the unit and blends with cooler containment air drawn through its allotted section of louvers. Air from the upper section then exhausts to the containment atmosphere. Each unit is completely enclosed for protection against RB Spray.

| M-55A/B | |
|---|---|
| Mfg. | Westinghouse |
| Flow | 100 SCFM natural convection |
| Heater Banks | 75 kW total, 60 htr/bank, 480 vac<br>#1 (Bottom) - 30 kW,<br>#2-21 kW<br>#3-14 kW<br>#4-10 kW |

The power supply sections are located in LSEPR and LNEPR. M-55A is powered from B5333A and M-55B is powered from B61104A. Controls and Indications for the recombiners are located on C-26 and consist of the following for each:

OFF/ON hand switch
Power Adjust Potentiometer
Thermocouple Selector (three inputs)/Off
Temperature Indication (Dual indicator with wattmeter)
Wattmeter
Hydrogen Concentration Meter

## 2.4 RX-7460 (RCS) Leak Detection Monitor

RX-7460 (Installed by DCP-89-1021) takes suction from the RB through containment isolation valves, CV-7453 (interior) and SV-7454 (exterior). Sample airflow is routed to the unit located on the 354' elevation of the Aux. Bldg. near the elevator. Air return may either be processed back to the RB or passed to the Waste Gas Collection Header for RB pressure reduction. Discharge of RX-7460 normally returns to RB through SV-7456 powered by RA-2 and SV-7459 powered by RA-1, outside isolation, and CV-7446 powered by B-5322, inside isolation.

CV-7453 is powered from MCC B5326 through hand switch HS-7453 on panel C-26. An ES actuation signal from ES Ch. 3 shuts this valve for containment isolation. SV-7454 is powered from RA-1 Brkr. 12 through hand switch HS-7454 on panel C-26. An ES actuation signal from ES Ch. 4 shuts this valve for containment isolation. Both valves have position indication on panel C-26. The discharge header may be aligned to the Gas Collection Header.

CV-7455 is a motor operated discharge valve used to isolate the discharge of RE-7461 from the gas collection header. CV-7455 is

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

**QID:** 0200 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** B. Short

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

**Section:** 3.8 **Type:** Plant Services Systems

**System Number:** 033 **System Title:** Spent Fuel Pool Cooling System

**Description:** Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Spent Fuel Pool Cooling System controls including: Spent fuel pool water level.

**K/A Number:** A1.01 **CFR Reference:** 41.7 / 45.7

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 2.7 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 2 | **SRO Imp:** | 3.3 | **SRO Select:** | Yes | **Taxonomy:** | K |

**Question:** **RO:** 61 **SRO:** 61

A break has occurred on the discharge line downstream of the discharge valve of the in service Spent Fuel Cooling Pump (P-40A).
The pump is stopped and the discharge valve is closed.

What would you expect to happen with Spent Fuel Pool level?

A. The SFP will drain to ~ 2 feet above the spent fuel assemblies.

B. Emergency makeup from service water will be needed to prevent the SFP level from reaching the spent fuel assemblies.

C. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

D. The SFP level will drop ~3 feet to the bottom of the suction pipe.

**Answer:**

C. The SFP level will stay relatively constant due to siphon holes in the discharge piping.

**Notes:**

(a.) & (b.) are incorrect. With no operator action at all, the lowest the level would go is ~3 feet to the bottom of the suction pipe. This is still ~ 20 feet above the fuel.
(c.) is correct. The discharge pipe has the siphon break holes located at normal pool level.
(d.) is incorrect. The suction pipe bottom is at ~3 feet, however, with the discharge valve closed the pool will stop draining out the break at the normal pool level due to the siphon holes on the discharge pipe.

**References:**

STM 1-07, Rev. 8

**History:**

Developed for use in 98 RO Re-exam
Selected for use in RO/SRO exam. 2002
Selected for 2005 RO re-exam.
Selected for 2007 RO Exam.
Selected for 2011 RO Exam.

Pool. During refueling the minimum boron concentration is determined by RCS requirements. Additional shutdown margin is required during core reload assuming the most reactive fuel assembly is placed in the worst core location.

Cooling and purification of the pool water is accomplished by recirculating it through heat exchangers, with a bypass flow through the lead Spent Fuel Pool Filter (F-4A), to Spent Fuel Pool Demineralizer (T-5), and then through the lag filter (F-4B). Water enters the pool at two separate nozzles, each equipped with a manually-operated globe valve for flow control. One nozzle discharges at the bottom of the pool and the other at the surface. The bottom discharge impacts turbulence to the pool and tends to keep particles in suspension, thereby increasing the likelihood of their being recirculated and removed by the SF filters. In addition, the bottom discharge promotes pool circulation through the stored spent fuel assemblies to improve cooling. The suction nozzle for the spent fuel pool circulating pumps is located at the opposite end of the pool from the discharge nozzles and is near the surface. This arrangement provides thermal mixing and insures uniform water temperature.

The SF Pool Skimmer is not normally used and requires installation of equipment before use. The skimmer system is designed to remove floating debris from the surface of the SF pool. Skimmer suction is provided to either the Borated Water Recirc Pump (P-66) or the header for the Spent Fuel Pool Circulating Pumps (P-40A or B).

To prevent inadvertent draining of the SF Pool below the stored fuel elements, pool drains are not provided and the suction line for the pumps is located three feet below the normal water level (centerline of the suction line is at elevation 397′0″). The two discharge lines have siphon breaker holes drilled into them at the normal pool level to prevent a break in the return piping from siphoning the water from the pool. A portable pump is required for complete draining of the SF pool (no fuel elements in pool).

### 2.1.2 Spent Fuel Racks

The assemblies are stored in stainless steel racks arranged in 44 rows of 22 elements, for a capacity of 968 spent fuel assemblies. One space is available for storage of a failed fuel assembly and its special container. The parallel rows are designed to ensure a center to center distance of 10.65 inches between fuel assemblies is maintained in all directions. This spacing is sufficient to maintain a Keff of less than 1.00 even if the pool is flooded with unborated water. A fuel rack has been fitted with a fuel assembly upper end fitting for load testing of the bridge grapple and/or storage of a control component.

The spent fuel rack designs described employ three separate and different arrays which will be considered as three separate spent fuel racks. All three storage arrays are designed on the basis of the currently accepted NRC guidance on spent fuel rack design, with consideration of the changes in fuel and fission product inventory resulting from depletion in the reactor core. Although all three storage racks types differ in design, they take credit for the reduction in reactivity associated with fuel burnup. Criticality safety is assured

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

**QID:** 0434 **Rev:** 0 **Rev Date:** 4/30/2002 **Source:** Direct **Originator:** J.Cork

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

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

**System Number:** 056 **System Title:** Condensate System

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

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 4.1 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 4.1 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** 62 **SRO:** 62

The plant is operating steady state at 100% power when Bus A1 is deenergized.

What will cause one of the main feedwater pumps to trip?

a. Low bearing oil pressure

b. High discharge pressure

c. Low suction pressure

d. High vibrations

**Answer:**

c. Low suction pressure

**Notes:**

"C" is correct due due to two condensate pumps being deenergized and P8A Heater Drain Pump also deenergized.
The other choices are other MFW pump trips or former trips.

**References:**

1203.012F, Chg. 028

**History:**

Direct from regular exambank QID 3715.
Selected for use in 2002 RO/SRO exam.
Selected for 2011 RO Exam.

Location: C13

Device and Setpoint: see next page.

A MFP
TURBINE
TRIP

Alarm: K07-A7

1.0 OPERATOR ACTIONS

1. IF Main Feedwater Pump "A" was supplying feedwater, THEN GO TO Loss of Steam Generator Feed (1203.027).

2.0 PROBABLE CAUSES

**NOTE**
This annunciator has multiple input without reflash.

1. MFWP suction pressure low
2. MFWP discharge pressure high
3. Bearing oil pressure low
4. Overspeed
5. Manual trip

3.0 REFERENCES

1. Condensate & Feedwater System Logic Diagram (M-402, sheet 6)
2. Main FW Pump Turbine Trip and Reset (E-285)
3. Condensate Feedwater P&ID (M-204, Sheet 2)
4. Main Feedwater Pump Turbine & Gland Seal Steam P&ID (M-202, Sheet 2)
5. Seal Oil, Main Turbine & FW Pump Bearing Lube Oil Sys. P&ID (M-226, Sheet 2)
6. Schematic Diagram Annunciator K07 (E-457)
7. ER-ANO-2002-0170-000, U1 MFWP Low Brg Oil Press Trip Time Delays

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

**QID:** 0205 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** B. Short

**TUOI:** A1LP-RO-MSSS **Objective:** 1.4 **Point Value:** 1

**Section:** 3.8 **Type:** Plant Services System

**System Number:** 075 **System Title:** Circulating Water System

**Description:** Knowledge of Circulating Water System design feature(s) and interlock(s) which provide for the following: Heat sink.

**K/A Number:** K4.01 **CFR Reference:** 41.7

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 2.5 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 2.8 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** 63 **SRO:** 63

During 3 circulating water pump operation, the 'A' circ water pump trips.
The standby circ pump was started and plant conditions have stabilized.
It is noticed that the condenser waterbox discharge temperature is 10 degrees higher and condenser vacuum has dropped.

Which of the following is the cause of this condition?

a. The stopping and starting of a circ pump caused fouling to be removed from the tube sheet promoting better heat transfer capabilities.

b. The discharge valve on the tripped pump did not go completely closed and circulating water is short cycling.

c. The debris on the bar grates of the circulating water bays was stirred up during the circ pump swap causing reduced flow.

d. These are normal conditions following rotation of circulating pumps and temperatures will return to normal within 30 minutes.

**Answer:**

b. The discharge valve on the tripped pump did not go completely closed and circulating water is short cycling.

**Notes:**

(a.) is incorrect. Although some fouling can be removed during pump rotations, it should not result in a 10 degree change in waterbox discharge temperature.
(b.) is correct. The discharge valve on an idle pump can allow a significant amount of backflow from the operating pumps if it is not closed completely.
(c.) is incorrect. This condition is normal for a circ pump swap and may contribute to waterbox fouling, however, the service water system would be affected by this condition as well.
(d.) is incorrect. There should not be such a large temperature difference even if only 3 CW pumps are in service.

**References:**

1104.008, Chg. 028

**History:**

Developed for use in 98 RO Re-exam
Used in 2001 RO Exam.
Selected for use in 2002 RO exam.
Selected for 2011 RO Exam.

5.11 Use of hoses as jumpers to cross-connect systems is a potential design change or Temporary Modification. Controls exist in this procedure for making temporary connections and for independent verification of connection and removal of temporary hoses.

5.12 Failure of pump discharge CV to close upon stopping of pump will result in short cycling of circ water back to lake which can cause pump reverse rotation and lowering of condenser vacuum.

5.13 When starting Circ Water pumps from the control room hand switch, the hand switch must be placed firmly in the START position (for ~1-2 seconds) to enable pump start relays to pick up. Past experience has shown instances of CW pump start failures when this was not done.

5.14 Prior to personnel entering waterbox, cathodic protection system shall be de-energized to prevent accidental shock.

5.15 Removing E-11A South Water Box from service can result in a lower vacuum sensed at the Main Turbine Trip Block. The Low Vacuum Trip pressure indicator (PI-8554) should be monitored when removing <u>AND</u> periodically while E-11A South Water Box is removed from service. CR-ANO-1-2006-00509.

5.16 Low circulating water flow velocities in the condenser water boxes have resulted in excessive silting in the outlet water boxes. Excessive silting can impact condenser efficiency such that reactor power would not be able to be maintained at 100% during summer months. It is desired to maintain 4 Circ Water pumps running when Lake temperature is $\geq 57°F$. When condenser vacuum is not being maintained it is recommended that at least 2 Circ Water pumps be operating with 2 to a maximum of 3 water boxes in service. CR-ANO-1-2010-02479

6.0 SETPOINTS

6.1 Interlocks necessary for starting Circ Water Pump:

6.1.1 Cooling water flow from lube oil reservoir, 2 gpm.

- P-3A Lube Oil Cooling Flow Switch (FIS-3628)
- P-3B Lube Oil Cooling Flow Switch (FIS-3624)
- P-3C Lube Oil Cooling Flow Switch (FIS-3620)
- P-3D Lube Oil Cooling Flow Switch (FIS-3616)

6.1.2 Cooling/lubrication flow to individual Circ Water Pump, 13 gpm.

- P-3A Cooling Water Flow Switch (FIS-3617A)
- P-3B Cooling Water Flow Switch (FIS-3618A)
- P-3C Cooling Water Flow Switch (FIS-3620A)
- P-3D Cooling Water Flow Switch (FIS-3616A)

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

**QID:** 0273 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

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

**Section:** 3.7 **Type:** Instrumentation

**System Number:** 072 **System Title:** Area Radiation Monitoring System

**Description:** Knowledge of the physical connections and/or cause-effect relationships between the ARM system and the following systems: Control room ventilation.

**K/A Number:** K1.04 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 3.3 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 2 | **SRO Imp:** | 3.5 | **SRO Select:** | Yes | **Taxonomy:** | K |

**Question:** **RO:** 64 **SRO:** 64

The in-service Unit 1 Control Room Supply Vent Radiation Detector, 2RITS-8001A, detects a high radiation condition.

Which of the following will occur?

A. Control Room Air Supply Fan (VSF-8A or 8B) starts.

B. Control Room Chiller Unit (VCH-2A or 2B) trips.

C. Normal ventilation ducts are isolated automatically.

D. 2VSF-9 (CR Emerg. A/C Fan) starts.

**Answer:**

C. Normal ventilation ducts are isolated automatically.

**Notes:**

"A" is incorrect because VSF-8A and VSF-8B are stopped.
"B" is incorrect because VCH-2A or 2B are unaffected.
"C" is correct the high radiation isolates the control room and starts emergency ventilation fan VSF 9.
"D" is incorrect because VSF-9 is started.

**References:**

2104.007, Chg. 053

**History:**

Direct from ExamBank, QID# 1131
Used in 1999 exam.
Used on 2004 RO Exam.
Selected for 2011 RO Exam.

7.2 IF ANY of the following monitors/detectors are actuated:

- Unit 1 Control Room Area Radiation monitor (RI-8001)
- Unit 1 Control Room Supply Vent Radiation monitor (2RITS-8001A)
- Unit 1 Control Room Supply Vent Radiation monitor (2RITS-8001B)
- Unit 1 Control Room Supply Vent Chlorine detector (2CLS-8760-2)
- Unit 2 Control Room Supply Vent Chlorine detector (2CLS-8762-2)

THEN perform the following:

* 7.2.1 IF actual high chlorine condition exists,
THEN don SCBAs within two minutes of determination.
(This applies to all Control Room Operators.)

7.2.2 Verify the following isolation dampers closed:

A. Unit 1 dampers:

- CV-7905, Unit 1 Control Room Supply damper (Green light ON at C141A/B in Unit 1 Computer Room)
- CV-7907, Unit 1 Control Room Return damper (Green light ON at C141A/B in Unit 1 Computer Room)

B. Unit 2 Dampers:

- Place Unit 2 Control Room Supply damper 2UCD-8683 (2HS-8683-1) on 2C33 in CLOSE.
  1. Check Green light ON above 2HS-8683-1 on 2C33.
- Place Unit 2 Control Room Return damper 2PCD-8685 (2HS-8685-2) on 2C33 in CLOSE.
  1. Check Green light ON above 2HS-8685-2 on 2C33.

7.2.3 Perform the following:

A. Verify Unit 1 Control Room Supply fans stopped:

- VSF-8A
- VSF-8B

B. IF actual high chlorine or high radiation condition exists,
THEN verify the following fans stopped AND place HS in OFF:

- Unit 1 Control Room Supply fans:
  - VSF-8A (HS-7820 on B1544)
  - VSF-8B (HS-7821 on B4161)
- Unit 2 Control Room Supply fans (non-actuated):
  - 2VSF-8A (2HS-8601 on 2C22)
  - 2VSF-8B (2HS-8602 on 2C22)
- Unit 2 Control Room Exhaust fans (non-actuated):
  - 2VEF-43A (2HS-8693 on 2C22)
  - 2VEF-43B (2HS-8694 on 2C22)
- Control Room Smoke Exhauster VEF-43, (non-actuated) (HS-7840) on C19

**NOTE**

If both Control Room isolation channels tripped, then VSF-9 may already be manually secured per step 7.1. In this case, step 7.2.4 may be N/A.

7.2.4 Verify Unit 1 Emergency Fan/Filter (VSF-9) running as follows:

- VSF-9 fan running.
- VSF-9 Outside Air damper CV-7910 open.
- VSF-9 Air flow by ONE of the following:
  - Red light ON above (FS-7806) on C19.
  - Air flow from duct in Unit 1 Control Room.

7.2.5 IF ANY of the following conditions exist:

- VSF-9 can NOT be started.
- VSF-9 Air flow NOT adequate with fan in service.
- VSF-9 NOT operable.

THEN perform the following:

A. Close VSF-9 Outside Air damper CV-7910 by placing (HS-7910) on C19 in RESERVE.

B. Obtain 2nd person verification.

C. Verify VSF-9 secured by placing (HS-7806) on C19 in OFF.

D. IF Unit 2 Emergency Fan/Filter **2**VSF-9 NOT running, THEN start **2**VSF-9 by placing (2HS-8603-1) on 2C33 in START AND releasing.

E. Verify Unit 2 Emergency Fan/Filter (**2**VSF-9) running as follows:

- **2**VSF-9 fan running.
- **2**VSF-9 Outside Air damper (2PCD-8607B) open.
- **2**VSF-9 Air flow by ONE of the following:
  - Red light ON above (2FS-8603-1) on 2C33 OR Red flow light ON above (HS-7852) on C19.
  - Air flow from duct in Unit 1 Control Room.

F. Obtain 2nd person verification of the above three conditions.

G. Initiate Condition Report.

H. Verify entry into appropriate Technical Specifications for VSF-9 inoperability (failure to start OR low flow). Refer to Attachment B, Component/Tech Spec Cross-Reference.

**NOTE**

- When Control Room in Emergency Recirc, ONR Emergency Filtration fan may be placed in STOP or OFF AND associated outside air damper placed in RESERVE, without affecting operability (CR-ANO-C-2001-0271)
- If BOTH Control Room isolation channels tripped, then EITHER step 7.1.6 or 7.2.6 is performed. N/A steps for Emergency Filtration fan left running.

7.2.6 IF VSF-9 running properly, THEN perform the following to secure **2**VSF-9:

A. Close **2**VSF-9 Outside Air damper 2PCD-8607B by placing (2HS-8607B-1) on 2C33 in RESERVE.

B. Verify **2**VSF-9 secured by placing (2HS-8603-1) on 2C33 in STOP.

7.3 Perform ONE of the following to start the Control Room Emergency coolers:

7.3.1 IF 2VUC-27A or 2VUC-27B aligned to Unit 2 power, THEN verify ONE of the following is in service:

- 2VUC-27A (place 2HS-8665-1 on 2C33 to ON and release)
- 2VUC-27B (place 2HS-8666-2 on 2C33 to ON and release)

7.3.2 IF 2VUC-27A or 2VUC-27B aligned to Unit 1 power, THEN refer to "OPERATION OF CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM WHEN POWERED FROM UNIT 1", section of this procedure.

7.4 Check associated Emergency Control Room chiller running:

- 2VE-1A
- 2VE-1B

7.5 IF necessary to secure CR Extension HVAC for Control Room pressurization, THEN depress CREF HVAC Shutdown switch for 2VUC-42A and 2VUC-42B (2PB-8000) located adjacent to Door 341 (U2 SM Office).

7.6 Verify the following doors closed:

- Unit 1 Restroom (Door 65)
- Unit 1 S/M office (Door 64)
- Unit 2 S/M office (Door 450)
- Unit 2 Control Room Foyer (Door 342)

7.7 IF there is NO seismic event, THEN maintain one door open between Control Rooms to equalize pressure.

7.8 IF Automatic Control Room Isolation caused by any of Unit 1 and/or Unit 2 Radiation monitor(s) failure (going inoperable), THEN perform the following:

7.8.1 Verify affected Unit 2 monitor (NOT both) placed in BYPASS on 2C25A.

7.8.2 Verify affected Unit 1 monitor (NOT both) placed in BYPASS on 2C474.

7.8.3 Verify Unit 1 Status Board updated for 2RITS-8001 present configuration.

7.9 Verify entry into appropriate TRM/TS. Refer to Attachment B, Component/Tech Spec Cross-Reference and Attachment C, Chlorine Detector Matrix.

7.10 Shutdown and de-energize any laptop computers in BOTH Control Rooms. (Emergency ventilation not analyzed for additional heat load per ER-ANO-2004-0412-001.)

**NOTE**

If operating Control Room Chiller (VCH-2A/2B) trips on underload the ICS Relay Room may experience high temperature/high humidity.

* 7.11 IF Control Room expected to be isolated for more than 12 hours, THEN Unit 1 should monitor Control Room chillers (VCH-2A/2B) operation at least every 2 hours. (CR-ANO-1-2002-0194)

**NOTE**

If operating Control Room Chiller (2VCH-2A/2B) trips on underload, then CEDMCS and MG Set Rooms may experience high temperature/high humidity.

* 7.12 WHEN Control Room isolated, THEN Unit 2 should monitor Control Room chillers (2VCH-2A/2B) operation at least every 2 hours.

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

**QID:** 0151 **Rev:** 1 **Rev Date:** 4/23/2002 **Source:** Direct **Originator:** JCork
**TUOI:** A1LP-RO-FPS **Objective:** 9 **Point Value:** 1

**Section:** 3.8 **Type:** Plant Service Systems
**System Number:** 086 **System Title:** Fire Protection
**Description:** Knowledge of the effect of a loss or malfunction of the Fire Protection System will have on the following:
Fire, smoke, and heat detectors.

**K/A Number:** K6.04 **CFR Reference:** 41.7 / 45.7

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 2.6 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | 2 | **SRO Imp:** | 2.9 | **SRO Select:** | Yes | **Taxonomy:** | K |

**Question:** **RO:** 65 **SRO:** 65

ONLY Fire Zones with "cross-zoned" detection have INHIBIT switches on their C463 modules.

When is the INHIBIT switch used?

A. To prevent automatic actuation if one of the detector strings has a fault.

B. To manually actuate the system from C463 vs. locally.

C. To prevent nuisance alarms from one of the detector strings due to a fault.

D. To prevent manual actuation of the system on C463 from a faulty solenoid.

**Answer:**

A. To prevent automatic actuation if one of the detector strings has a fault.

**Notes:**

Answer "a" is correct, the inhibit switch is used when one of the two detector strings is malfunctioning and it is desired to prevent an automatic actuation caused by a single failure of the remaining detector string.
Answers "b" and "d" are incorrect since the Inhibit switch does not have anything to do with any manual actuations.
Answer "c" is incorrect, the Inhibit switch works with the actuation circuit only and not with detection trouble alarms.

**References:**

1104.032, Chg. 066

**History:**

Modified for use in 2002 RO/SRO exam.
Selected for 2011 RO Exam.

18.0 Inhibiting Deluge and Halon Actuation Systems at Fire Panel C463

**NOTE**

The inhibit function prevents automatic deluge and halon system actuation driven from the associated detectors. The UAV can still be tripped by using the Man Trip Switch (module left-hand switch) or using the local break glass station at the UAV.

18.1 Inhibit the system as follows:

18.1.1 Place inhibit (right-hand) switch in OPERATED.

18.1.2 Verify the following:

- The module's yellow inhibit lamp illuminates.
- Fire Protection System Trouble (K12-D1) alarms.

18.1.3 Perform the following:

- Place Trouble Silence rocker switch on C463-1 in silence (rocker to right).
- Perform required actions of Unit 1 TRM 3.7.9 or 3.7.10 AND report fire system impairment if required.

18.2 Remove inhibit (restore auto capability) as follows:

18.2.1 Verify no fire alarm (red led) on area's fire module.

18.2.2 Place inhibit (right-hand) switch to NORMAL.

18.2.3 Place Trouble Silence rocker switch in normal (rocker to left), unless other trouble conditions exist.

18.2.4 Verify the following:

- The module's yellow inhibit lamp extinguishes.
- Fire Protection System Trouble (K12-D1) clears, unless alarming for another cause.

18.2.5 Perform the following:

- Reference Unit 1 TRM 3.7.9 or 3.7.10 AND stop any unneeded compensatory actions.
- Record impairment restoration if previously reported.

# ANO 2011
# RO Questions
# Tier 3

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

**QID:** 0851 **Rev:** 0 **Rev Date:** 06/24/11 **Source:** New **Originator:** J. Cork

**TUOI:** ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

**Section:** 2.0 **Type:** Generic Knowledge and Abilities

**System Number:** 2.1 **System Title:** Conduct of Operations

**Description:** Ability to make accurate, clear, and concise verbal reports.

**K/A Number:** 2.1.17 **CFR Reference:** 41.10 / 45.12 / 45.13

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 3 | **RO Imp:** | 3.9 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | | **SRO Imp:** | 4.0 | **SRO Select:** | Yes | **Taxonomy:** | K |

**Question:** **RO:** 66 **SRO:** 66

When communicating verbally to other personnel, which of the following should be a vital component of that communication?

A. Including multiple instructions to prevent multiple communications.

B. Making a verbatim repeat back for precise communications.

C. Use standard terminology with a minimal use of acronyms for clarity.

D. Minimize repeat backs to clear the airways of unnecessary noise.

**Answer:**

C. Use standard terminology with a minimal use of acronyms for clarity.

**Notes:**

C is correct, the other choices are items which should NOT be done in verbal communications per COPD001.

**References:**

COPD001, Chg. 053

**History:**

New for 2011 RO Exam.

ATTACHMENT G



COMMUNICATION AFFECTING PLANT OPERATION

The purpose of this attachment is to provide a standard for all plant personnel for verbal communication methods to ensure information is effectively transmitted and received; verbal communication that is operational in nature and establishes a methodology to ensure information is transmitted and received effectively, and describes the use of plant communication systems including radio communication, inside the plant communication and inside the plant paging.

1.0 Definitions

1.1 Communication

Communication is the process of sending and receiving information.

1.2 Verbal Communications

Verbal communications are transmitted by voice and heard by ear.

1.3 Operational Communications

Operational communications are messages concerning operation of the plant.

1.4 Face-to-Face Communications

Face-to-Face communications are directly communicating verbally without the aid of mechanical or electrical communications equipment.

1.5 Portable Radios

Portable radios consist of any radio communications system where at least one station is remote and portable.

1.6 Point-to-Point

Point-to-point communications consist of those that are primarily between two individuals, as in a telephone call.

2.0 General Requirements for Communications Regarding Plant Equipment or Operation

2.1 Personnel answering a telephone should identify their location, followed by their name or title (e.g. Unit One Control Room, Jane Doe).

2.2 Written notes or other physical aids may be used in conjunction with verbal communication to aid in remembering the specific details of the message.

2.3 Sections 6.0 and 7.0 of this attachment should be used during verbal communications.

2.4 When communicating alphanumeric information the phonetic alphabet should be used (Attachment C of this directive).

2.5 Verbal instructions should be clear and concise. Orders involving the operation of plant equipment will be repeated back to the extent necessary to allow the sender to ensure orders are correctly understood. Verbatim repeat backs are not necessary. To avoid confusion, specific details should be used. Generalities and vague terms should not be used.

2.6 Multiple actions should not be contained in a verbal instruction where control or coordination concerns exist. It is preferable to report back for additional instructions.

2.7 The completion of Control Room ordered actions should be reported back to the Control Room.

2.8 The Control Room may need to be informed of interdepartmental verbal communications concerning plant operation, condition or status. For instance Chemistry personnel may inform Radiation Protection of an adverse radiological control condition. This communication should also be reported to the Control Room.

2.9 Some verbal instructions may need to be supplemented with written guidance to ensure important information is not forgotten. For example, a Fire Watch Supervisor may write down instructions when multiple items are requested and formal written guidance is not available, such as perform the following patrols in the following order. This ensures actions are correct and in the desired sequence (in this instance the most logical and time saving order).

2.10 Refer to "Communication Examples", section 8.0 of this attachment for preferred communication techniques.

2.11 It is the message sender's responsibility to ensure that the message is concise and that the receiver understands the message prior to proceeding.

3.0 Verbal Messages

3.1 Clarity - The message should be free of ambiguity. Personnel should avoid the use of slang terms and words that sound alike, such as increase and decrease. Messages are unclear when they contain similar words with different meanings. Messages should be clear and short enough that repeat backs are not burdensome.

3.2 Specificity - The message should be specific to ensure the correct unit or component (Alpha or Bravo, 1 or 2) is identified. It is preferred that noun names, equipment numbers, units of measure and observed trends as required are used together to ensure the message is properly transmitted and received. Standard terminology should be used with minimal use of acronyms. Messages must be of sufficient detail to convey information accurately and precisely (e.g. not like the following: An operator was directed via radio to check out the P-8s and called back that he was ready to swap Isophase Bus Cooling Fans. P-8A and P-8B are Heater Drain Pumps and the Isophase Bus Cooling Fans are C-8A and C-8B).

3.3 Acknowledgement - the receiver will acknowledge all operational communications. The receiver may paraphrase the message, provided that all important points of the message are included.

3.4 Precautions - The message should contain precautions necessary to successfully achieve the desired results. Remote operations can be controlled by such requirements as: requiring continuous communications, requiring specific conditions prior to performance of an activity, or specifying expected communications following the requested action.

3.5 If the receiver repeats or paraphrases the message incorrectly, the sender should immediately correct the receiver by saying "wrong" or "no" or something similar. Then, the sender should repeat or rephrase the message.

3.6 Repeat backs should be confirmed by the sender to ensure the message is understood. Listening to the message repeated back is very important because studies have shown that people hear what they want to or expect to hear and frequently do not hear what they should hear.

3.7 The sender is responsible for miscommunication if an incorrect repeat back is accepted.

3.8 The message should be spoken slowly and plainly in a normal tone of voice.

3.9 A message that is not understood should be questioned before any action is taken.

3.10 If the receiver does not understand the message, the receiver should ask the sender to repeat or rephrase the message.

3.11 Message Format

3.11.1 The typical message format consists of these parts in the following order:

A. Name or work station of called individual (required for initial contact).

B. Title or work station of calling individual (required for initial contact).

C. Message text.

"Shift Chemist, this is the Unit 1 Control Room. The Inside Auxiliary Operator is waiting in Corridor 98 to assist you."

3.11.2 The basic acknowledgment consists of the following parts in the following order:

A. Originating individual's name or work station.

B. Acknowledging individual's name or work station.

C. Paraphrase or explanation of the instructions or message in the employee's own words.

"Unit 1, this is Max, I understand that the AO is waiting to assist me in getting an emergency switchgear chiller sample."

3.11.3 Confirmation of the acknowledgment from the originating individual.

"That's correct."

3.11.4 After the initial contacts are made, it is not required that parties identify themselves if all of the following apply:

A. The parties are certain that they are speaking with the correct person.

B. There are no other evolutions in progress which may cause confusion.

3.12 In meetings, three part communication should be used when plant related items are initially assigned or communicated. The action should be specific and the due date should be communicated.

3.13 Face to face communications should cease when a plant announcement or radio communication is taking place during steady state conditions.

4.0 Communication Methods

4.1 Face-to-Face Communication

4.1.1 The use of sign language is not permitted, except in accepted industry practices, such as among crane operators.

4.1.2 The sender's name/title identifier may be omitted from the message after the initial contact is made.

**NOTE**

Prior to beginning any evolution that affects plant operation, careful consideration should be given to the communication method selected. During a plant transient, operators on shift may not be able to respond rapidly to telephone calls.

4.2 Plant Telephone

4.2.1 Once initial identification of the sender and receiver has been established, it need not be repeated with each message, since this is a point-to-point system.

**NOTE**

During personnel emergencies the Control Room dedicated emergency phone numbers should be used: 1911 for Unit One, 2911 for Unit Two and 3911 to ring both units.

4.2.2 Routine communications with Operations during standard working hours (dayshift Monday through Thursday) should be made to the Control Room Supervisor-Administrative (CRSA) in the Work Management Center (WMC): 4201 for Unit 1 and 3191 for Unit 2.

A. IF the control room must be contacted, THEN the sender should hang up when the Control Room fails to answer after the fourth ring.

1. The phone may be allowed to ring until answered if the information is vital to the Control Room.

4.2.3 To gain access on a party line in the event of an emergency, the phrase "Silence on the line, this is an emergency" should be used to clear the line for priority communications.

4.2.4 Repeat backs and three part confirmations should be used to ensure the message is understood.

**NOTE**

It is necessary to pause approximately 15 seconds after dialing 199 before speaking.

4.2.5 Paging is accomplished by accessing the plant Gai-Tronics system by dialing 197 or the RERTC/EOF by dialing 199 then speaking a message.

4.2.6 When a plant page is heard, all personnel should stop their ongoing conversations in order to listen to the plant page.

4.2.7 The plant paging system is intended for non-routine use for those situations where personnel cannot be contacted via other means or to alert personnel of changing equipment status, overall plant status, etc.

4.3 Plant Gai-Tronics

4.3.1 Gai-Tronics phones are available for surveillance testing, post maintenance testing or special operational situations that require coordination through continuous communication which precludes radio or telephone party line use.

A. To operate, select party line, depress paging push button to page, then release for private conversation.

4.3.2 If a point-to-point circuit is used, the sender and receiver identification need only be established on the initial contact.

4.3.3 If a party line circuit is used, the sender and receiver identification should be included in each message.

4.3.4 Repeat backs and three part confirmations should be used to ensure the message is understood.

**CAUTION**

Do not key portable radios in posted areas in order to prevent Radio Frequency Interference (RFI). RFI has resulted in equipment malfunctions and reactor/turbine trips. Unit 1/Unit 2 Trip Sensitive Area Designation (COPD-010) contains a listing and description of radio sensitive locations.

4.4 Portable Radios

4.4.1 The nature of radio transmissions is such that virtually all communications should be three part, regardless of whether orders, information or briefings are exchanged.

4.4.2 Portable radio communications should be tested prior to conducting a procedure that requires radio communication.

4.4.3 Radio communication equipment used in normal plant operations will be used in an emergency to communicate with fire brigade personnel.

4.4.4 Minimize use of portable radio for messages that are long or that require discussion. These messages should be sent via telephone or Gai-Tronics phone system.

4.4.5 The use of the unit one or unit two radio channels should be limited to operators unless control room permission has been given (non-operator examples could include equipment hatch watch, waterbox personnel watches, etc.).

5.0 Communication Systems

5.1 ANO Public Address System (Gai-Tronics)

The Gai-Tronics system is an industrial communication system independent of offsite communications and is designed to provide voice communication between two or more locations.

5.2 ANO Radio System

5.2.1 The ANO radio system consists of UHF repeaters, multi-channel control consoles and portable and vehicle radios. The ANO radio system also provides for interconnections with the Arkansas Department of Emergency Management (ADEM) and the Sheriff's frequencies.

5.2.2 The following channels are provided:

- Channel 1 Maintenance/In-Plant Emergency Teams
- Channel 2 Security
- Channel 3 Offsite Monitoring/Dardanelle Dam
- Channel 4 Operations, Unit 1
- Channel 5 Operations, Unit 2
- Channel 6 Arkansas Department of Emergency Management
- Channel 7 Sheriff's Office
- Channel 8 Spare

6.0 Abbreviations and Acronyms

6.1 Acronyms should only be used when interfacing with Operations Department personnel. Individual using acronyms should evaluate the audience and qualification level of the audience prior to using any acronyms.

6.2 Increase and decrease should not be used in verbal communications directing operation of plant equipment.

7.0 Preferred Action Verbs

7.1 The following table provides the preferred action verbs to be used in verbal communications.

| Preferred Verb | Definition |
|---|---|
| Actuate | To bring to a specified position or state.<br>Example: Manually actuate EFW. |
| Add | To make an addition.<br>Example: Add boric acid to the MU tank. |
| Adjust | To change to a different state or to regulate.<br>Example: Adjust Pressurizer Level Control setpoint to 100". |
| Advise | To give information.<br>Example: Advise Shift Manager to implement Emergency Action Level Classification (1903.010). |
| Align | To arrange components into desired configuration.<br>Example: Manually align valves. |
| Allow | To permit a stated condition to be achieved prior to proceeding.<br>Example: Allow RCS pressure to rise to 1000 psig. |
| Announce | To make known publicly.<br>Example: Attention all personnel, the Unit 2 Reactor is critical. |
| Attempt | To make an effort to accomplish.<br>Example: Attempt to determine affected SG. |
| Balance | To make two or more things equal.<br>Example: Balance HPSI flow in BOTH headers. |
| Batch | To add to the makeup tank by the batch controller.<br>Example: Batch in existing boron concentration. |
| Begin | To start performing an action. Implies action will continue while performing subsequent steps.<br>Example: Begin batching equilibrium boron concentration. |
| Bleedoff | To emit or lose a liquid or gas.<br>Example: Bleedoff IA pressure from the valve operator. |
| Borate | To inject boric acid into the RCS.<br>Example: Borate RCS to 1000 ppm. |
| Bump | To operate a component briefly.<br>Example: Bump RCP. |
| Bypass | To avoid the usual route, or to prevent an automatic actuation.<br>Example: Bypass ESAS. |
| Calculate | To perform a mathematical calculation.<br>Example: Calculate actual amount of boric acid required. |

| Preferred Verb | Definition |
|---|---|
| Check | To inspect for satisfactory condition, accuracy, safety, or performance without adjusting.<br>Example: Check Secondary Radiation monitors. |
| Close | For valves, positioning to completely stop flow. For breakers, positioning to make an electrical connection with a power supply.<br>Example: Close MSIVs. |
| Commence | To begin a task or evolution.<br>Example: Commence plant shutdown. |
| Compensate | To counterbalance.<br>Example: Add boric acid as necessary to compensate for rod motion. |
| Complete | To accomplish specific requirements.<br>Example: Complete checklist in Attachment 1. |
| Confirm | To determine the validity of something.<br>Example: Confirm diagnosis of event. |
| Consider | To take into account.<br>Example: Consider terminating CNTMT Spray. |
| Consult | To ask for advice or opinion.<br>Example: Consult with Operations Manager. |
| Contact | To communicate with another group or individual.<br>Example: Contact Chemistry. |
| Continue | To maintain without interrupting course, condition, or action.<br>Example: Continue with this procedure. |
| Control | To operate equipment as necessary to maintain procedural requirements of process parameters: pressure, temperature, level, flow, etc.<br>Example: Control EFW flow to maintain proper SG level. |
| Cooldown | To reduce the temperature by heat removal.<br>Example: Cooldown RCS to less than 540°F. |
| Coordinate | To bring into a common action, movement, or condition.<br>Example: Coordinate with Chemistry. |
| Correct | To alter or adjust so as to bring into accordance with a standard or some required condition.<br>Example: Correct SG level using Figure 4. |
| Crack | To slightly open as necessary to establish procedural requirements of process parameters: pressure, temperature, level, flow, etc.<br>Example: Crack open CV-2623. |

ATTACHMENT G



| Preferred Verb | Definition |
|---|---|
| Crosstie | To establish connection between two systems.<br>Example: Crosstie A3 and A4. |
| Cycle | To place a system or device in a position or condition, and then subsequently reverse the position or condition.<br>Example: Cycle Pressurizer heaters to control RCS pressure 2000 to 2200 psig |
| Decrease | "Decrease" should not be used. See LOWER, DROP, REDUCE. |
| De-energize | To remove voltage from, disconnect from power supply.<br>Example: De-energize Decay Heat Suction valves. |
| Defeat | To override.<br>Example: Defeat Undervoltage trip. |
| Depress | To press down, as a pushbutton.<br>Example: Depress the Reset pushbutton. |
| Depressurize | To reduce pressure in any stated component.<br>Example: Depressurize CFTs to atmospheric pressure. |
| Determine | To calculate or evaluate.<br>Example: Determine affected SG. |
| Diagnose | To recognize by signs or symptoms.<br>Example: Diagnose event using Attachment 28, Diagnostic Actions. |
| Direct | To tell or instruct; to impart orally.<br>Example: Direct CBOs to acknowledge alarms. |
| Dispatch | To send an operator outside the control room.<br>Example: Dispatch an operator to manually Close Letdown Coolers Outlet valve. |
| Disregard | To ignore.<br>Example: Disregard the alarm. |
| Don | To put on.<br>Example: Don SCBA prior to entry. |
| Drain | To reduce liquid inventory by draining.<br>Example: Drain RB sump to 10%. |
| Drop | To lower.<br>Example: Allow PZR level to drop to 55". |
| Energize | To apply voltage, connect to power supply.<br>Example: Energize 4160 Volt bus A4. |
| Engage | To come together and interlock, be or become in gear.<br>Example: Engage Turning gear. |

| Preferred Verb | Definition |
|---|---|
| Ensure | To make an outcome certain.<br>Example: Ensure the Fire Brigade responds. |
| Enter | To write down in a record.<br>Example: Enter results in the Station log.<br>To put data into a computer system.<br>Example: Enter current EFPD in the Plant Computer. |
| Establish | To take the necessary steps to obtain the desired condition.<br>Example: Establish communication with Control Room. |
| Estimate | To make an educated guess.<br>Example: Estimate required amount of boric acid. |
| Evacuate | To remove personnel.<br>Example: Evacuate unnecessary personnel prior to initiating RB Sump recirc. |
| Evaluate | To determine the value or significance of.<br>Example: Evaluate the need for plant cooldown based on the following: |
| Exit | To remove oneself or leave the area or procedure.<br>Example: Exit this procedure. |
| Go | To assume another state or condition.<br>Example: Allow PZR to go solid. |
| Go To | To discontinue performing this instruction (step) and continue where directed.<br>Example: GO TO step "X" of this procedure |
| Hand Jack | To take local manual control of a valve.<br>Example: Hand jack ATM Dump CNTRL valves open. |
| Hold | To maintain.<br>Example: Hold Hand switch in close position. |
| Identify | To establish the identity.<br>Example: Identify affected SG(s). |
| Index | To align a piece of equipment on index marks.<br>Example: Index the bridge and trolley over the Dummy Fuel assembly. |
| Implement | To begin a stated objective concurrently with continuing the performance of the present instruction.<br>Example: Implement Emergency Action Level Classification (1903.10) |
| Increase | "Increase" should <u>not</u> be used. See RAISE. |
| Inform | To impart information or knowledge.<br>Example: Inform the TSC. |

ATTACHMENT G



| Preferred Verb | Definition |
|---|---|
| Initiate | To begin a stated objective.<br>Example: Initiate HPI. |
| Inject | To force or drive (a fluid) into something.<br>Example: Inject water into RCS. |
| Insert | To put into the body of something.<br>Example: Insert Control rods. |
| Investigate | To look into or attempt to determine cause of.<br>Example: Investigate cause of annunciator alarm. |
| Isolate | To set apart or separate from a system in use.<br>Example: Isolate Main steam to MSRs. |
| Install | To connect or set in position and prepare for use.<br>Example: Install Caution Card on component. |
| Leave | To avoid changing or interfering with.<br>Example: Leave one RCP running in each loop. |
| Limit | To control within parameters.<br>Example: Limit RCS pressure to <2300 psig. |
| Lineup | To align: See align.<br>Example: Lineup to add hydrogen to VCT. |
| Lock | To fix in a given position.<br>Example: Open and lock CFT Outlet Valve breakers. |
| Log | To enter in a record.<br>Example: Log in Operator rounds. |
| Lower | To reduce in value or amount.<br>Example: Lower SG pressure to <700 psig. |
| Maintain | To keep in a certain condition or position.<br>Example: Maintain RCS pressure <1000 psig. |
| Maximize | To raise to the greatest quantity or value attainable.<br>Example: Maximize RB cooling. |
| Minimize | To reduce to the least quantity or value attainable.<br>Example: Minimize secondary contamination. |
| Monitor | To observe and evaluate at a frequency sufficient to remain appraised of the value, trend, and rate of change of the identified plant parameter.<br>Example: Monitor RCS subcooling margin. |
| Note | To make a brief record.<br>Example: Note the reading. |
| Notify | To give formal notice to a position or group.<br>Example: Notify Chemistry to sample SGs for activity. |

| Preferred Verb | Definition |
|---|---|
| Observe | To adhere to.<br>Example: Observe applicable limits and precautions. |
| Obtain | To gain or attain by planned action.<br>Example: Obtain Operations Manager approval. |
| Open | For valves, To position to allow flow. For breakers, disconnecting from a power supply.<br>Example: Manually open breaker. |
| Operate | To turn ON and OFF or to adjust as necessary to achieve the stated objective.<br>Example: Operate Pressurizer Spray valve as necessary to limit RCS pressure <2450 psig. |
| Override | To neutralize the action of an automatic control.<br>Example: Override ES signal. |
| Perform | To do, carry out, or bring about; to accomplish, or affect; to reach an objective.<br>Example: Perform plant cooldown. |
| Place | To put in a specified position. See SELECT.<br>Example: Place Meter Input Select switch to OFF. |
| Plot | To locate by means of coordinates as on a chart.<br>Example: Plot pressurizer cooldown rate. |
| Prepare | To make ready.<br>Example: Prepare to limit RCS pressure as PZR goes solid. |
| Purge | To make free of something unwanted.<br>Example: Purge Main generator. |
| Raise | To make greater in size, amount, or intensity.<br>Example: Raise Tank pressure to 400 psig. |
| Record | To write down manually or via an automatic device.<br>Example: Record Pressurizer level. |
| Re-diagnose | To diagnose again.<br>Example: Re-diagnose event and GO TO appropriate EOP. |
| Re-energize | To return to an energized state.<br>Example: Re-energize A2 from DG2. |
| Reduce | To lower.<br>Example: Reduce RCS pressure to <300 psig. |
| Re-establish | To restore a previous condition.<br>Example: Re-establish Letdown flow. |
| Refer | To use a supplemental information source.<br>Example: Refer to Plant Startup (1102.002). |

| Preferred Verb | Definition |
|---|---|
| Relax | To make lax or loose.<br><br>Example: Relax Flash Protection boundary. |
| Remove | To change location or position of.<br>Example: Remove Danger Tag. |
| Repeat | To say or state again.<br>Example: Repeat the announcement. |
| Report | To present oneself.<br>Example: Report to the Control Room. |
| Request | To ask for.<br>Example: Request SG sample. |
| Reset | To reposition a breaker or control circuit to reactivate a device or logic scheme.<br>Example: Reset MSLI actuation signal |
| Restart | To restore to an operating condition.<br>Example: Restart an RCP. |
| Restore | To bring back to or put back into a former or original state.<br>Example: Restore Pressurizer level to 100". |
| Resume | To recommence a suspended process.<br>Example: Resume plant cooldown. |
| Return | To restore to a previous state or go to a previous step.<br>Example: Return handswitches to Normal-After-Stop.<br>Example: RETURN TO step 23. |
| Re-verify | To verify again.<br>Example: Re-verify ECP. |
| Review | To re-familiarize.<br>Example: Review system operating procedure. |
| Rotate | To change direction about an axis.<br>Example: Rotate RCP by hand. |
| Sample | To take a sample of or from.<br>Example: Direct Chemistry to sample RCS for activity |
| Sat | To receive satisfactory verification of expected component response.<br><br>Example: Pump post-start checks sat. |
| Secure | To stop in a controlled manner.<br>Example: Secure remaining RCPs. |
| Select | To choose.<br>Example: Select the desired sample point. |
| Set | To physically adjust an adjustable feature to a specified value.<br>Example: Set PZR Level Control setpoint to 100". |
| Shift | To change status from one configuration from one component to another (synonymous with "swap").<br>Example: Shift running pumps from "A" to "B". |

ATTACHMENT G



| Preferred Verb | Definition |
|---|---|
| Shutdown | To cease or suspend an operation.<br>Example: Shutdown DGs. |
| Stabilize | To hold steady, to limit fluctuations.<br>Example: Stabilize RCS pressure. |
| Start | To originate motion of an electrical or mechanical device directly or by remote control.<br>Example: Start RCP P-32A. |
| Steam | To withdraw heat from, by the removal of steam.<br>Example: Steam bad SG to maintain SG pressure <970 psig. |
| Stop | To cause a cessation of operation in a controlled manner.<br>Example: Stop RCP P-32A. |
| Strip | To open all breakers on a bus.<br>Example: Strip 4160v Vital bus 2A3. |
| Suspend | To cause to stop for a period; interrupt.<br>Example: Suspend core alterations. |
| Swap | See "shift". |
| Synchronize | To make synchronous in operation. To match voltage and frequency and parallel power sources.<br>Example: Synchronize Vital 4160V bus A3 with A1. |
| Take | To undertake to do.<br>Example: Take handswitch to pull-to-lock. |
| Throttle | To regulate or adjust flow by incremental positioning of a valve.<br>Example: Throttle HPI flow. |
| Transfer | To change electrical or fluid supply from one source to another.<br>Example: Transfer P-36B MOD to A4. |
| Trip | To activate a mechanical or electrical device in order to prevent a device or system operation.<br>Example: Manually trip reactor. |
| Turn | To move to or from a position.<br>Example: Turn on SYNC switch for A-309. |
| Type | To write on a typewriter or keyboard.<br>Example: Type in Point ID at the prompt. |

| Preferred Verb | Definition |
|---|---|
| Unlatch | To remove from a latched state.<br>Example: Unlatch RB Cooler Bypass dampers. |
| Unlock | To remove from a locked state.<br>Example: Unlock and close CFT Outlet Valve breakers. |
| Update | To provide current information.<br>Example: Update the Status Board. |
| Use | To put into action or service.<br>Example: IF normal spray not available, THEN use Auxiliary spray. |
| Vent | To relieve gas or liquid pressure.<br>Example: Vent Quench tank. |
| Verify | To confirm that a condition exists and if it does not to take the necessary action to establish that condition.<br>Example: Verify Turning gear in AUTO. |

8.0 Communication Examples

8.1 Face-to-Face Direct Order (three part communications required)

"Mark, trip the reactor and perform your immediate actions."

"Understand trip the reactor and perform immediate actions."

"That's correct."

8.1.1 The acknowledgment must be given before the action is taken.

8.2 Face-to-Face Information (three part communications required, but request need not be repeated).

"Steve, verify the turbine is tripped."

"Ken, the turbine is tripped, all throttle and governor valves are closed."

"Understand the turbine is tripped."

"That's correct."

8.2.1 The following conversation is an example of unnecessary, over-communications.

"Ernie, verify the turbine is tripped."

"Verify the turbine is tripped."

"That's correct."

"Dennis, the turbine is tripped, all throttle and governor valves are shut."

"I understand that the turbine is tripped."

"That's correct."

ATTACHMENT G



8.3 Telephone Communication with the Control Room

Operator to Shift Chemist to sample the Boric Acid Addition Tank for boron concentration.

"James, request a boron sample on the Boric Acid Addition Tank."

"Sample the Boric Acid Addition Tank for boron."

"That's correct."

8.4 Radio Communications

E-11B Main Condenser Water Box Watch reporting to the Unit One Control Room.

"Unit One Control Room, this is the E-11 Bravo Condenser Water Box Watch."

"E-11B Main Condenser Water Box Watch, this is Unit One Control Room. Go ahead."

"Unit One, all personnel have exited E-11 Bravo East Water Box."

"Understand all personnel have exited the E-11 Bravo East Water Box."

"That's correct."

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

**QID:** 0837 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** New **Originator:** D. Thompson

**TUOI:** ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

**Section:** 2.0 **Type:** Generic Knowledge and Abilities

**System Number:** 2.1 **System Title:** Conduct of Operations

**Description:** Knowledge of 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:** 2

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

**Question:** **RO:** 67 **SRO:** 67

Which of the following is NOT a valid reason to override any automatically actuated safety system per EN-OP-115?

A. Required by procedures.

B. Subcooling margin is adequate.

C. Adequate core cooling is assured by at least two independent indications.

D. Mis-operation in automatic mode is confirmed by at least two independent indications.

**Answer:**

B. Subcooling margin is adequate.

**Notes:**

"B" is correct as SCM alone is not a valid reason to override a safety system per EN-OP-115, the other choices are specifically listed in the procedure

**References:**

EN-OP-115, Rev. 010

**History:**

New for 2011 RO Exam.

5.3 MANUAL CONTROL OF AUTOMATIC SYSTEMS

[1] Do not override an automatic initiation of a safety function unless one of the following conditions exists:

- Adequate core cooling is assured by at least two independent indications.
- Mis-operation in automatic mode is confirmed by at least two independent indications.
- Required by procedures.

[2] If an automatic control malfunctions, immediately place that control in manual.

[3] If an operator cannot be dedicated to monitor systems placed in the manual mode, frequently check the system for proper operation and system response.

[4] When manual operation is no longer required, return systems to automatic or standby mode.

[5] When practicable, before placing controls in manual for activities which require manual control, review system response and actions to be taken during potential off-normal events.

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

**QID:** 0838 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** New **Originator:** J. Cork

**TUOI:** ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

**Section:** 2.0 **Type:** Generic Knowledge and Abilities

**System Number:** 2.4 **System Title:** Conduct of Operations

**Description:** Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

**K/A Number:** 2.1.4 **CFR Reference:** 41.10 / 43.2

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 3 | **RO Imp:** | 3.3 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | | **SRO Imp:** | 3.8 | **SRO Select:** | Yes | **Taxonomy:** | K |

**Question:** **RO:** 68 **SRO:** 68

For the purpose of maintaining an NRC operator's license, which of the following should be reported to the NRC?

A. A change in marital status.

B. A traffic citation for speeding.

C. A new prescription for high blood pressure.

D. A summons for grand jury duty in federal court.

**Answer:**

C. A new prescription for high blood pressure.

**Notes:**

Only "C" is requred to be reported per EN-NS-112 and 1063.008.
The others are situations that are not required to be reported as part of an operator's license.

**References:**

1063.008, Chg. 039

**History:**

New for 2011 RO Exam.

C. A licensed individual shall, as soon as possible, notify the Manager, Operations if during the term of the license the individual develops a physical or mental condition that could adversely affect the performance of assigned operator duties or cause operational errors. The facility shall notify the NRC within thirty (30) days of learning of the diagnosis.

The individual should then be directed to the Medical Review Officer for evaluation. Based on this evaluation, or items identified during normal license physical examinations, the Medical Review Officer should make any required restrictions known to the Superintendent, Operations Training, for submittal to the NRC for evaluation. This restriction should be reported to the regional office on NRC form 396, "Certification of Medial Examination by Facility Licensee" for review. This submittal will include a copy of all supporting medical information and recommended wording for the conditional license to be issued to the affected operator.

D. A licensed individual shall notify the NRC within 30 days about a conviction for a felony

E. An individual whose SRO license has become inactive may reactivate that license for the purpose of supervising refueling operations by completing all the requirements identified on form 1063.008B.

6.10.9 Monitoring for marginal performance

A. Is a described within EN-TQ-114, Licensed Operator Requalification Training Program Description.

{3.2.20} B. Guidance (derived from TQ-201, Academic Review Board):

**1. Student's test score average is < 85% AND > 5% below the group average score for the same period (minimum 5 test points for data comparison). Monitor over a rolling two year period.**

**2. Student fails 3 tests during a rolling two year period.**

**3. Student fails a remedial examination.**

**B. If a student is identified by any of these criterion:**

**1. Conduct Academic Review Board.**

**2. Operations and Training Management will determine course of action to upgrade student. Reference TQF-201-IM05 as applicable**

4.7 Supervisors of Licensed Nuclear Operators or designee are responsible for:

[1] Providing the Medical Examiner or designee a report on each employee referred for a Licensed Operator medical examination prior to the conduct of the examination. The report shall address work performance, attendance and behavioral changes noted since the previous review. Incidents of ineptness, poor judgment and lack of physical or emotional stamina should be noted.

[2] Ensuring that Licensed Nuclear Operator candidates and Licensed Nuclear Operators are made available to complete the scheduled medical examination.

[3] Notifying medical services personnel of any physical or mental condition that may limit the performance of Licensed Nuclear Operators.

[4] Notifying licensed operator's of their responsibility to report all medication and any changes in their health status to their supervisor and medical services personnel.

4.8 Licensed Nuclear Operators are responsible for:

[1] Complying with the conditions of licenses in accordance with 10CFR55.

[2] Notifying their Supervisor and appropriate medical services personnel of any changes in their health status (e.g., hospitalization or any physical or mental condition treated by a personal physician, which may limit the performance of his or her duties).

[3] Reporting all prescription medications at the time of their physical for review by medical services personnel/Medical Examiner to determine if notification to the NRC per 10CFR regulations and ANSI standards is required.

[4] Using the medication reporting form to document any new medications, medications currently being taken and any changes in medication dosage. Reporting of medications should occur on the first day of work following the medication being taken/administered.

4.9 Employees or designee are responsible for:

[1] Scheduling their physicals giving sufficient time for all results to be returned prior to the expiration date of their physical.

[2] Reporting to their direct management and medical services personnel, any changes in physical or mental conditions that could impact their current medical qualifications.

4.7 Supervisors of Licensed Nuclear Operators or designee are responsible for:

[1] Providing the Medical Examiner or designee a report on each employee referred for a Licensed Operator medical examination prior to the conduct of the examination. The report shall address work performance, attendance and behavioral changes noted since the previous review. Incidents of ineptness, poor judgment and lack of physical or emotional stamina should be noted.

[2] Ensuring that Licensed Nuclear Operator candidates and Licensed Nuclear Operators are made available to complete the scheduled medical examination.

[3] Notifying medical services personnel of any physical or mental condition that may limit the performance of Licensed Nuclear Operators.

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[3] Reporting all prescription medications at the time of their physical for review by medical services personnel/Medical Examiner to determine if notification to the NRC per 10CFR regulations and ANSI standards is required.

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[1] Scheduling their physicals giving sufficient time for all results to be returned prior to the expiration date of their physical.

[2] Reporting to their direct management and medical services personnel, any changes in physical or mental conditions that could impact their current medical qualifications.

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

**QID:** 0458 **Rev:** 0 **Rev Date:** 5/6/2002 **Source:** Direct **Originator:** S.Pullin

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

**Section:** 2 **Type:** Generic Knowledges and Abilities

**System Number:** 2.1 **System Title:** Conduct of Operations

**Description:** Ability to determine Technical Specification Mode of Operation.

**K/A Number:** 2.2.35 **CFR Reference:** 41.7 / 41.10 / 43.2 / 45.13

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

**Group:** **SRO Imp:** 4.5 **SRO Select:** Yes **Taxonomy:** K

**Question:** **RO:** 69 **SRO:** 69

Which one of the following conditions is required by Unit 1 Technical Specifications in order to consider the reactor in Mode 4?

A. The reactor must be subcritical by at least 1.5% Delta k/k.

B. RCS T average must be between 200 °F and 280 °F.

C. The neutron chain reaction is self sustaining and K eff = 1.0.

D. RCS temperature is no more than 200 °F.

**Answer:**

B. RCS T average must be between 200 °F and 280 °F.

**Notes:**

"B" is correct for Mode 4.
The other choices are correct for other modes.

**References:**

Technical Specifications 1.1

**History:**

Direct from regular exambank QID 39.
Selected for use in 2002 SRO exam.
Selected for 2011 RO/SRO Exam.

Table 1.1-1

MODES

| MODE | TITLE | REACTIVITY CONDITION ($K_{eff}$) | % RATED THERMAL POWER [(a)] | AVERAGE REACTOR COOLANT TEMPERATURE (°F) |
|---|---|---|---|---|
| 1 | Power Operation | ≥ 0.99 | > 5 | NA |
| 2 | Startup | ≥ 0.99 | ≤ 5 | NA |
| 3 | Hot Standby | < 0.99 | NA | ≥ 280 |
| 4 | Hot Shutdown [(b)] | < 0.99 | NA | $280 > T_{avg} > 200$ |
| 5 | Cold Shutdown [(b)] | < 0.99 | NA | ≤ 200 |
| 6 | Refueling [(c)] | NA | NA | NA |

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

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

**QID:** 0233 **Rev:** 0 **Rev Date:** 11/25/98 **Source:** Direct **Originator:** B. Short

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

**Section:** 2.0 **Type:** Generic K/As

**System Number:** 2.2 **System Title:** Equipment Control

**Description:** Knowledge of surveillance procedures.

**K/A Number:** 2.2.12 **CFR Reference:** 41.10 / 45.13

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 3 | **RO Imp:** | 3.7 | **RO Select:** | Yes | **Difficulty:** | 3 |
| **Group:** | G | **SRO Imp:** | 4.1 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** 70 **SRO:** 70

An NI calibration was performed yesterday.

Today due to a problem with a Condenser Vacuum pump, reactor power had to be lowered to 89% and has subsequently been returned to 100%.

When is the next NI calibration required to be performed?

A. Within the next 7 days.

B. Within the next 4 days.

C. Within the next 36 hours.

D. Within the next 24 hours.

**Answer:**

D. Within the next 24 hours.

**Notes:**

Per TS 3.3.1 NI calibration shall be performed twice weekly during steady state operations and daily during non-steady state operations. The power change that occurred stipulates a non-steady state condition and therefore a calibration is required within the next 24 hours.(D.)
(A.) (B.) & (C.) are incorrect.

**References:**

Technical Specifications 3.3.1

**History:**

Developed for use in 98 RO Re-exam
Selected for use in 2002 RO exam.
Selected for 2007 RO Exam.
Selected for 2011 RO Exam.

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.3.1.2 | -----NOTES----- <br> 1. Adjust power range channel output if the absolute difference is > 2% RTP. <br> 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. <br> ----- <br> Compare results of calorimetric heat balance calculation to power range channel output. | 96 hours <br> <u>AND</u> <br> Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP |
| SR 3.3.1.3 | -----NOTES----- <br> 1. Adjust the power range channel imbalance output if the absolute value of the imbalance error is ≥ 2% RTP. <br> 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. <br> ----- <br> Compare results of out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE. | 31 days |
| SR 3.3.1.4 | Perform CHANNEL FUNCTIONAL TEST. | 31 days |
| SR 3.3.1.5 | -----NOTE----- <br> Neutron detectors are excluded from CHANNEL CALIBRATION. <br> ----- <br> Perform CHANNEL CALIBRATION. | 18 months |

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

**QID:** 0231 **Rev:** 1 **Rev Date:** 11/20/00 **Source:** Direct **Originator:** J.Cork

**TUOI:** ASLP-RO-OPSPR **Objective:** 4 **Point Value:** 1

**Section:** 2.0 **Type:** Generic K/As

**System Number:** 2.2 **System Title:** Equipment Control

**Description:** Knowledge of tagging and clearance procedures.

**K/A Number:** 2.2.13 **CFR Reference:** 41.10 / 45.13

**Tier:** 3 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 2

**Group:** G **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** K

**Question:** **RO:** 71 **SRO:** 71

Which of the following conditions is correct with regard to preparation and installation authorization of a common unit tagout?

A. Installation may be authorized by either the Unit 1 or the Unit 2 Operations Supervisor.

B. Preparers and reviewers from both units must be licensed operators.

C. Preparer and reviewer may be non-licensed if authorized by both Unit Operations Supervisors.

D. Preparer may be non-licensed as long as the opposite unit reviewer is licensed.

**Answer:**

B. Preparers and reviewers from both units must be licensed operators.

**Notes:**

Answer [b] is correct, procedure requires both the preparer and the reviewer on the unit preparing the tagout have to be licensed.
Answer [a] is incorrect, a common unit tagout requires both Unit's Operations Supervisors to approve it.
Answer (c) is incorrect, both Unit Ops Supervisors must approve but the preparation & review must be done by licensed operators.
Answer [d] is incorrect, the preparation & review must be done by licensed operators on their respective units.

**References:**

EN-OP-102, Rev. 13

**History:**

Developed for use in 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.
Selected for use on 2007 RO Exam.

8.0 **Section 8.0 is applicable at ANO only**

**Common Tagouts (ANO 1 and ANO 2)**

**NOTE**

The determination of whether a TAGOUT should be considered a COMMON TAGOUT is based upon whether the SSC may normally be operated by either unit (both units train and qualify on the system). Examples are not all inclusive.

- The following examples should be considered a COMMON TAGOUT based upon common operator qualification:
  - Primary Hydrogen System
  - Generator Hydrogen System
  - Liquid Nitrogen System
  - T-41B
  - Cardox System (components tagged on both units)
  - Vendor Supplied Demineralized Water Trailers
- The following examples by the shared nature of the systems should be considered a COMMON TAGOUT:
  - Instrument Air cross-ties
  - Turbine Building Crane
  - Fuel Handling Crane (L-3 and 2L-35)
  - MCC B81 (power to both units condensate vacuum degasifiers

8.1 IF a Tagout is determined to be Common,
THEN Respond Yes to the COMMON TAGOUT Attribute

8.2 IF a Tagout is determined to be Common,
THEN a LICENSED OPERATOR from each unit shall review it. (The preparer and reviewer shall be Licensed Operators on their respective units).

**AND**

Both unit OPERATIONS SUPERVISORS must authorize installation. Opposite Unit Supervisors Should sign into the eSOMS Clearance module and select their name from list in the Opposite Unit Supervisor Attribute

8.3 IF a Common Tagout not utilized, but common components are affected,
THEN the authorizing OPERATIONS SUPERVISOR is responsible for ensuring that equipment status is communicated to the other unit.

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

**QID:** 0817 **Rev:** 0 **Rev Date:** 2/5/2010 **Source:** New **Originator:** S.Pullin

**TUOI:** A1LP-WCO-CZ **Objective:** 11 **Point Value:** 1

**Section:** 2.2 **Type:** Generic K&A

**System Number:** 2.3 **System Title:** Radiation Control

**Description:** Ability to control radiation releases.

**K/A Number:** 2.3.11 **CFR Reference:** 41.13 / 43.4 / 45.10

**Tier:** 3 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2

**Group:** **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** K

**Question:** **RO:** 72 **SRO:** 72

Which document does Chemistry use to obtain the dose calculation for Radiation Monitor high setpoints during gaseous radioactive releases?

A. Technical Specifications

B. ODCM

C. Chemistry Procedures

D. TRM

**Answer:**

B. ODCM

**Notes:**

B. is correct the Offsite Dose Calculation manual is the document that is used

A.,C., and D are incorrect

**References:**

Offsite Dose Calculation manual Rev 017

**History:**

New for 2010 RO exam, not used
New for 2011 RO Exam.

$U_a$ = annual usage factor that specifies the intake rate for an individual of age group "a", in liters/year. The program selects this usage factor according to the user-specified controlling age group "a";

F = average flow rate in $ft^3$/sec; this value is based on total dilution volume for one quarter divided by time into the quarter;

$Q_i$ = number of curies of nuclide "i" in the release; and

$D_{aij}$ = ingestion dose factor, for age group "a", nuclide "i", and organ "j", in mrem per ρCi ingested. The program selects the ingestion dose factor according to the user-specified controlling age group "a" and controlling organ "j".

### 2.3 Liquid Projected Dose Calculation

The quarterly projected dose is based upon the methodology of Section 2.2 and is expressed as follows:

$$D_{QP} = 92*(D_{QC} + D_{RP})/T$$

where:

$D_{QP}$ = quarterly projected dose (mrem);

92 = number of days per quarter;

$D_{QC}$ = cumulative dose for the quarter (mrem);

$D_{RP}$ = dose for current release (mrem); and

T = current days into quarter;

## 3.0 GASEOUS EFFLUENTS

### 3.1 Gaseous Monitor Setpoints

> NOTE
>
> Sections 3.1.1 and 3.1.2 below detail two methods of calculating setpoints at ANO. These methods cover two different sets of monitors of which only one will be in-service at any one time.

3.1.1 Batch Release Setpoint Calculations

3.1.1.a This section applies to the following gaseous radiation monitors (These releases are also monitored by the SPING monitors in Section 3.1.2):

| | |
|---|---|
| <u>ANO-1</u> | |
| RE-4830* | Waste gas holdup system monitor |
| RX-9820 | Unit 1 Containment Purge SPING |
| <u>ANO-2</u> | |
| 2RE-8233 | Containment purge |
| 2RX-9820 | Unit 2 Containment Purge SPING |
| 2RE-2429* | Waste gas holdup system monitor |

* These monitors provide automatic isolation for the waste gas holdup systems.

The setpoints to be used during a batch type of release (i.e., reactor building [containment] purge, release from the waste gas holdup system or any other non-routine release) will be calculated for each release before it occurs.

3.1.1.b The basic methodology for determining a monitor setpoint is based upon the expected concentration at the monitor ($C_M$). This is in turn based upon the fraction of an MPC assigned to this release point. Batch releases are maintained below the assigned MPC fraction by controlling the release rate. The calculated value of S may not exceed the equivalent of 1 MPC at site boundary. If value of S for RX(2RX)-9820 is less than SPING channel 5 high alarm setpoint, then high alarm setpoint may be used as a default value. If value of S for RE-4830 and 2RE-2429 is less than 50,000 counts/min, then 50,000 counts/min may be used as a minimum setpoint. If value of S for 2RE-8233 is less than 1,000 counts/min, then 1,000 counts/min may be used as a minimum setpoint.

$$S = 1.2 * (C_M * K) + (2.0 * B)$$

where:

S = monitor setpoint (counts/min);

$C_M$ = Xe-133 equivalent concentration at the monitor (μCi/ml);

K = conversion factor determined from response curve of monitor (counts/min per μCi/ml). This value is 1.0 when calculating S for RX(2RX)-9820.

2.0 = factor to accommodate random count rate fluctuations;

B = background count rate at the monitor (counts/min).

1.2 = Safety Factor to correct for instrument uncertainties.

# 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:** Yes **Difficulty:** 3

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

**Question:** **RO:** 73 **SRO:** 73

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.

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

**History:**

New for 2011 RO Exam

**QID:** 0848 **Rev:** 0 **Rev Date:** 06/24/11 **Source:** New **Originator:** J. Cork

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

**Section:** 2 **Type:** Generic K/A's

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

**Description:** Knowledge of fire protection procedures.

**K/A Number:** 2.4.25 **CFR Reference:** 41.10 / 43.5 / 45.13

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

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

**Question:** **RO:** 74 **SRO:** 74

You are standing watch and it is now 1500 on 08/26/2011.

Which of the following would NOT be considered a fire system impairment in accordance with 1003.002, Insurance Impairment Reporting?

A. P-6A Electric Fire Pump inoperable due to corrective maintenance at 0400.

B. A smoke detector string in Corridor 98 defeated for PMs at 0900.

C. Leaking fire hose station in Aux Bldg 335 elevation isolated at 0530.

D. Control Room Halon System #3 failed surveillance at 0145.

**Answer:**

B. A smoke detector string in Corridor 98 defeated for painting at 0900.

**Notes:**

B is correct since this detector string was taken out of service during the current shift. Per 1003.002 any systems out of service for less than one shift for surveillances, corrective maintenance, or PMs are not considered an impairment.
All of the others are impairments since they occurred on the previous shift.

**References:**

1003.002, Chg. 004

**History:**

New question created for 20011 RO Exam.

4.5 IMPAIRMENT LOG

A record containing fire systems that have been removed or returned to service.

5.0 <u>RESPONSIBILITY AND AUTHORITY</u>

5.1 MANAGER, Engineering Programs & Components

5.1.1 Ensures that the requirements of this procedure are met.

5.2 SUPERVISOR, Fire Protection

5.2.1 Notifies the Insurer of fire system impairments and their restoration.

5.2.2 Completes fire system impairment reports (1003.002A) as required by our Insurer.

5.2.3 Maintain impairment log on fire systems.

5.2.4 Updating Plant Effect code in INDUS Asset Suite (IAS) for impairments that require a system listed in section 6.0 to be inoperable.

6.0 <u>INSTRUCTIONS</u>

**<u>NOTE</u>**

Fire Suppression including suppression subsystems or Detection Systems that are out of service for less than one shift due to routine surveillances, corrective maintenance or PMs are not considered to be an impairment. <u>Failure</u> of any of the components or support systems listed below constitutes an impairment.

A. Fire Pump(s)

B. Sprinkler Systems (wet and dry)

C. Deluge Water Spray Systems

D. Loop Sectionalizing Valve

E. Halon, CO2 and FM200 Systems

F. Fire Hydrants

G. Fire Hose Stations

H. Fire Protection System Piping

I. Any Detection System That Actuates a Suppression System

6.1 Fire Protection Engineering shall ensure that fire system impairments are promptly identified by checking the 24-hour phone number weekly during normal working hours.

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

**QID:** 0835 **Rev:** 0 **Rev Date:** 5/24/11 **Source:** New **Originator:** D. Thompson

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

**Section:** 2.0 **Type:** Generic Knowledge and Abilities

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

**Description:** Knowledge of crew roles and responsibilities during EOP usage.

**K/A Number:** 2.4.13 **CFR Reference:** 41.10 / 45.12

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 3 | **RO Imp:** | 4.0 | **RO Select:** | Yes | **Difficulty:** | 2 |
| **Group:** | | **SRO Imp:** | 4.6 | **SRO Select:** | Yes | **Taxonomy:** | K |

**Question:** **RO:** 75 **SRO:** 75

Who performs the Transient Checklist during a transient per 1015.037, Post Transient Review?

A. Shift Manager

B. Control Rooom Supervisor

C. Shift Technical Advisor

D. Control Board Operator Turbine

**Answer:**

C. Shift Technical Advisor

**Notes:**

"C" is correct per 1015.037.

**References:**

1015.037, Chg. 013

**History:**

New for 2011 RO Exam.

6.4 The Post Transient Review Shift Manager is responsible for the following:

- Performance of the Post Transient Review
- Assembling the Post Transient Review Team and directing the investigation
- Determining if additional personnel are needed for the Post Transient Review Team
- Requesting and obtaining statements from plant personnel involved in the events related to the transient using Plant Personnel Statement (Attachment P)
- Ensuring Plant Personnel Statements are collected and given to the Shift Engineer/Shift Technical Advisor before personnel leave the plant site after the transient
- Determining the actions necessary for safe restart of the reactor and progress through defined operational restraints.
- Define the operational restraints associated with the actions required for restart and power escalation (Attachments C and D).

6.5 The Shift Engineer/Shift Technical Advisor is responsible for the following:

- Notifying the members of the Post Transient Review Team when the Post Transient Review Team is required to be activated
- Ensuring the information and physical evidence that may be useful for transient investigation is promptly gathered and retained. Photographs of any equipment damage in "as found" condition should be included when possible.
- Ensuring the accurate and timely completion of data collection. Operations personnel, Computer Support, etc. may be utilized to assist in collecting and documenting data.
- Notifying Chemistry of Steam Releases via MSSVs or ADVs
- Verifying that an evaluation of Rosemount Transmitter response is initiated per Attachment H and L of this procedure

# ANO 2011
# SRO Questions
# Tier 1 Group 1

**QID:** 0741 **Rev:** 0 **Rev Date:** 6/3/2008 **Source:** Direct **Originator:** Steve Pullin

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

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

**System Number:** E10 **System Title:** Post Trip Stabilization

**Description:** Ability to determine and interpret the following as they apply to the (Post Trip Stabilization): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

**K/A Number:** EA2.2 **CFR Reference:** 43.5 / 45.13

**Tier:** 1 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 3

**Group:** 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

**Question:** **RO:** **SRO:** 76

Given:

- Plant Power 48%
- Low Pressure Turbine Boot fails causing vacuum to rapidly fall to 24 in Hg.

Which procedure should be used to stabilize the plant?

A. 1203.016, "Loss of Condenser Vacuum."

B. 1203.045, "Rapid Plant Shutdown."

C. 1203.012D, "K05-B3 Vacuum Pump Auto-start" ACA.

D. 1202.001, "Reactor Trip."

**Answer:**

D. 1202.001, "Reactor Trip."

**Notes:**

A, B, and C are incorrect. If the student fails to recognize that they are below the automatic turbine and therefore Reactor trip setpoint, these procedures would be used to stabilize the plant but do not provide the required actions to stabilize the plant post trip.

D is correct as "Reactor Trip" contains the actions to stabilize the plant after an automatic Reactor Trip due to >43% power and a Turbine trip which would occur due to exceeding the automatic low vacuum trip setpoint.

**References:**

1202.001, "Reactor Trip" Chg 031
1203.016, "Loss of Condenser Vacuum" Chg 014

**History:**

New for the 2008 SRO Exam.
Selected for 2011 SRO Exam.

## INSTRUCTIONS

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

   - IF MWe is >270 and vacuum is <24.5" Hg,
     THEN verify the turbine has tripped.

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

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

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

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

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

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

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

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

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

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

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

**16. (cont)**

B. Open Polisher Bypass (CS-27).

1) Make a Station Log entry.

C. While maintaining MFWP suction pressure (PMS P2842 or P2830) >260 psig, isolate in-service polishers per Condensate Demineralizer System Operation and Regeneration (1106.024).

- As polishers are isolated, continue to reduce power as needed to maintain >260 psig MFWP suction pressure.

D. **IF** at any time secondary chemistry quality degrades per any of the following, **THEN** consult with Chemistry to determine acceptability of continuing power operations.

- Feedwater sodium rises by 1.0 ppb
- Condensate sodium rises by 1.0 ppb
- MSR sodium rises by 4.0 ppb

**17. IF main turbine trips at <43% reactor power, THEN refer to Turbine Trip Below 43% Power (1203.018) in conjunction with this procedure.**

**18. IF reactor trips during transient, THEN perform Reactor Trip (1202.001) in conjunction with this procedure.**

**19. IF Main Condenser vacuum degrades to ~23" Hg, AND the local Condenser Vacuum Pump AUTO-HOG switches (HS-3636 and HS-3638) are in AUTO, THEN verify operating Condenser Vacuum Pump(s) automatically switch to the hogging mode.**

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

**QID:** 0839 **Rev:** 0 **Rev Date:** 5/25/11 **Source:** Modified **Originator:** S. Pullin

**TUOI:** A1LP-RO-EOP5 **Objective:** 11 **Point Value:** 1

**Section:** 4.2 **Type:** APE

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

**Description:** Ability to determine and intepret the following as they apply to the Pressurizer Vapor Space accident: RCS pressure and temperature indicators and alarms.

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

**Tier:** 1 **RO Imp:** 3.9 **RO Select:** No **Difficulty:** 3

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

**Question:** **RO:** **SRO:** 77

Given:

- Reactor tripped due to a Pressurizer Vapor Space leak.
- K09-B1 "RCS PRESSURE LO-LO" alarm is in.
- CET temperatures 560°F and rising.
- 'A' & 'B' OTSG levels 13" and rising slowly.
- RCS pressure 1200 psig and stable.
- Both EDGs are in emergency standby.

For these conditions, which operating procedure should be used?

A. 1202.002, "Loss of Subcooling Margin"

B. 1202.004, "Overheating"

C. 1202.005, "Inadequate Core Cooling"

D. 1203.015, "Pressurizer Systems Failure"

**Answer:**

A. 1202.002, "Loss of Subcooling Margin"

**Notes:**

A is correct. A LOSM exists due to the temperature and pressure givens.
B is incorrect. Although CET temperatures are rising, Loss of SCM takes precedence.
C is incorrect. Although SCM has been lost, conditions are not yet in the ICC realm.
D is incorrect. Although a Pressurizer failure exists, the EOPs takes precedence.

**References:**

1202.002, Loss of Subcooling Margin, Chg. 006

**History:**

Modifed QID 753 for 2011 SRO Exam.

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

**QID:** 0753 **Rev:** 0 **Rev Date:** 11/10/2009 **Source:** Direct **Originator:** Steve Pullin

**TUOI:** A1LP-RO-EOP5 **Objective:** 11 **Point Value:** 1

**Section:** 4.2 **Type:** APE

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

**Description:** Ability to determine and intepret the following as they apply to the Pressurizer Vapor Space accident: Inadequate Core Cooling.

**K/A Number:** AA2.30 **CFR Reference:** 43.5/45.13

**Tier:** 1 **RO Imp:** 4.3 **RO Select:** No **Difficulty:** 3

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

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

Given:

- Reactor tripped due to a Pressurizer Vapor Space leak.
- CET temperatures 590 degrees and rising.
- 'A' & 'B' OTSG levels 13".
- RCS pressure 1200 psig and stable.
- Both EDGs are in emergency standby.

Parent

For these conditions, which operating procedure should be used?

A. 1202.002, "Loss of Subcooling Margin"

B. 1202.004, "Overheating"

C. 1202.005, "Inadequate Core Cooling"

D. 1203.015, "Pressurizer Systems Failure"

**Answer:**

C. 1202.005, "Inadequate Core Cooling"

**Notes:**

A is incorrect. Although a LOSM exists, the ICC takes precedents.
B is incorrect. Although an Overheating exists, the ICC takes precedents.
C is correct. ICC conditions exist.
D is incorrect. Although a Pressurizer failure exists, the ICC takes precedents.

**References:**

1202.005, "Inadequate Core Cooling" Chg 006

**History:**

New for the 2009 Retake SRO Exam

## ENTRY CONDITIONS

- Loss of adequate SCM following a Reactor trip,
- Loss of adequate SCM while attempting to correct overcooling,
- Loss of adequate SCM following ESAS actuation
  **<u>AND</u>**
  RCS press stabilizes >150 psig (LPI pump discharge pressure).

# Floating Steps

## RCS Inventory/Press

- **IF** SCM is less than adequate,
  **THEN GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure.

- **IF** PZR level approaches 55", **THEN** perform step 26.

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

- **WHEN** PZR level is 90 to 110" **AND** RCS temp is stable, **THEN** check MU Tank and BWST level stabilize.

## RCS TEMP

- **IF** RCS temp is rising above:

  580°F T-hot with any RCP on
  **OR**
  610°F CET temp with all RCPs off,

  **THEN GO TO 1202.004, "OVERHEATING"** procedure.

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

## ESAS

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

## Electrical

- **IF** all 4160V buses are de-energized, **THEN GO TO 1202.008, "BLACKOUT"** procedure.

- **IF** ONLY EDG power is supplying 4160V buses,
  **THEN GO TO 1202.007, "DEGRADED POWER"** procedure.

- **IF** any NNI AC or DC Instrument Power Supply is lost,
  **THEN** perform Loss of NNI Power (1203.047).

- **IF** any ICS AC or DC Instrument Power Supply is lost, **THEN** perform step 13.

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

**QID:** 0639 **Rev:** 0 **Rev Date:** 10/18/2006 **Source:** Direct **Originator:** Cork/Possage

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

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

**System Number:** 025 **System Title:** Loss of Residual Heat Removal System (RHRS)

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

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 4.5 | **RO Select:** | No | **Difficulty:** | 2 |
| **Group:** | 1 | **SRO Imp:** | 4.7 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** **SRO:** 78

Given:
- Plant is in Mode 5
- "A" Decay Heat Removal is in service
- P-34A Decay Heat Removal flow is steady at 1900 gpm.
- K10-B2 "PROCESS MONITOR RADIATION HIGH" in alarm
- RI-3809, Loop A DH Process Rad Monitor in alarm

What procedure should be used to address the above conditions?

A. 1203.014, Control of Secondary System Contamination

B. 1203.028, Loss of Decay Heat Removal

C. 1203.030, Loss of Service Water

D. 1203.039, Excess RCS Leakage

**Answer:**

B. 1203.028, Loss of Decay Heat Removal

**Notes:**

Answer "B" is correct Loss of Decay Heat Removal procedure will address the cooler leak into Service Water.
Answer "A" is incorrect, although a leak of RCS via the DH cooler would cause a contamination concern 1203.014 deals with a SG tube leak.
Answer "C" is incorrect, although the leak will be into the service water system, 1203.030 is not needed.
Answer "D" is incorrect, although RCS leakage is present and 1203.039 does deal with intersystem LOCA's, it does not address a decay heat cooler leak.

**References:**

1203.012I, Chg. 048

**History:**

New, created for 2007 SRO exam.
Selected for 2011 SRO Exam.

Location: C16

Device and Setpoint:

PROC MONITOR
RADIATION
HI

Alarm: K10-B2

Any process monitor in Radiation Monitoring System Panel (C25 Bays 1 thru 3)
or
Turb Bldg Drn Rad Monitor (RI-5641) .................... HIGH ALARM or loss of power
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 RE-4830,
         THEN perform step 14 to reset.

      2. IF any other alarm,
         THEN select "ALARM RESET" on the appropriate drawer and exit this procedure.

**NOTE**

Actions to correct this condition on next page.

**CAUTION**

A break in SW piping to RB cooling inside RB would dilute boron concentration in RB sump.

**NOTE**

Transmitters for loop I and loop II flow are not environmentally qualified for LOCA conditions.

C. Monitor service water flow to RB cooling loop and RB pressure on SPDS:

- RB Cooler SW Flow Loop I (F3816)
- RB Cooler SW Flow Loop II (F3817)

D. IF large SW line break is indicated by very low flow in loop II as compared to loop I,
THEN isolate VCC-2C and VCC-2D by closing the following valves from C16:

- RB Cooling Coils Inlet (CV-3813)
- RB Cooling Coils Outlet (CV-3815)
- CV-3813 Bypass (SV-3813)

E. Notify Chemistry that a potentially radioactive discharge to lake may have occurred.

6. IF Decay Heat Loop A (RI-3809, Bay 1) radiation is high,
THEN perform the following:

A. IF cooling in decay heat removal mode,
THEN GO TO Loss of Decay Heat Removal System (1203.028), "Loss of Inventory or DH Removal System Leak <20 gpm" section.

B. IF plant is in ES mode
AND DH Cooler (E-35A) is required to remain in-service,
THEN align SW system to take suction from and discharge to the emergency pond.

C. IF LPI loop "A" is NOT required to remain in-service,
THEN secure LPI Pump (P-34A) and cooler E-35A per "Decay Heat Removal Operating Procedure" (1104.004).

D. Initiate steps to have samples taken.

E. Initiate steps to have repairs made.

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

**QID:** 0731 **Rev:** 2 **Rev Date:** 5/20/2008 **Source:** Direct **Originator:** David Thompson

**TUOI:** A1LP-SRO-EOP **Objective:** 7 **Point Value:** 1

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

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

**Description:** Ability to determine and interpret the following as they apply to the "excessive heat transfer": adherence to appropriate procedures and operation within the limitations in the facilities license and amendments.

**K/A Number:** EA2.2 **CFR Reference:** 43.5 / 45.13

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 3.6 | **RO Select:** | No | **Difficulty:** | 3 |
| **Group:** | 1 | **SRO Imp:** | 4.0 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** **SRO:** 79

Given:

A steam line rupture has caused excessive overcooling.

Which of the conditions below would have invoked PTS limits AND violate Technical Specification limits?

A. A cooldown resulting in a step change of 51 °F in 10 minutes with Tcold at 360 °F.

B. A cooldown rate of 90°F/hr with Tcold at 295 °F.

C. A cooldown resulting in a step change of 26°F in 10 minutes with Tcold at 279°F.

D. A cooldown rate of 110°F/hr with Tcold at 330°F.

**Answer:**

D. A cooldown rate of 110°F/hr with Tcold at 330°F.

**Notes:**

A is incorrect. Although it violates TS it does not invoke PTS.
B is incorrect. Although Tcold is below the 300F limit, /hr.
C is incorrect. Although it violates TS it does not invoke PTS.
D is correct since it is above the 100F/hr limit and below the 355F Tcold limit.

**References:**

1202.012, "Repetitive Tasks" Chg 009

**History:**

New for the 2008 SRO Exam.
Selected for 2011 SRO Exam.

## CONTROL RCS PRESS

**NOTE**

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

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

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

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

   A. Throttle makeup flow.

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

      1) Verify both HPI Recirc Blocks open:

         - CV-1300
         - CV-1301

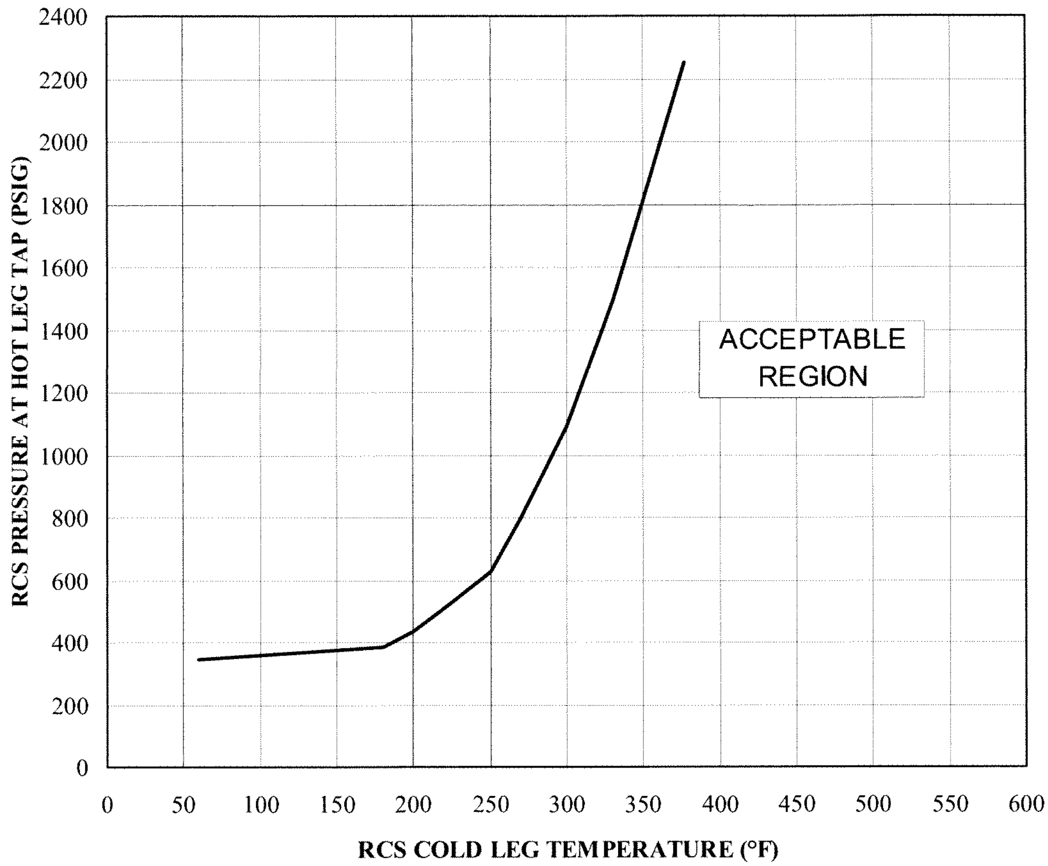
      2) Throttle HPI.

   C. Raise Letdown flow.

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

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

FIGURE 3.4.3-2
RCS Cooldown Limits to 31 EFPY




Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25°F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

| RCS TEMP | RCP RESTRICTIONS |
|---|---|
| T > 255°F | None |
| 150°F ≤ T ≤ 255°F | ≤ 2 (See Note 5) |
| T < 150°F | No RCPs operating |

4. Allowable Cooldown Rates:

| RCS TEMP | C/D RATE | STEP CHANGE |
|---|---|---|
| T ≥ 280°F | 100°F/HR | ≤ 50°F in any 1/2 HR |
| 280°F > T ≥ 150°F | 50°F/HR (Note 5) | ≤ 25°F in any 1/2 HR |
| T < 150°F | 25°F/HR | ≤ 25°F in any 1 HR |

5. If RCPs are operated < 200°F, then the RCS cooldown rate from 150°F ≤ T ≤ 180°F is reduced to 30°F in 15 hours.

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

**QID:** 0412 **Rev:** 1 **Rev Date:** 10/19/2006 **Source:** Direct **Originator:** S.Pullin

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

**Section:** 4.2 **Type:** Generic APEs

**System Number:** 057 **System Title:** Loss of Vital AC Inst. Bus

**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:** | No | **Difficulty:** | 3 |
| **Group:** | 1 | **SRO Imp:** | 4.2 | **SRO Select:** | Yes | **Taxonomy:** | An |

**Question:** **RO:** **SRO:** 80

Given:

- Plant at 100% power.
- RPS Channel "C" inoperable and in the tripped state.
- I&C is performing RPS Channel "B" monthly calibration.

Subsequently, RS1 INVERTER TROUBLE (K01-A5) alarms. The Inside AO reports that Inverter Y-11 has a fault and the static switch failed to operate.

Which of the following procedures contain the actions that should be in use?

A. Reactor Trip, 1202.001

B. Loss of NNI Power, 1203.047

C. Loss of 125 VDC, 1203.036

D. ESAS, 1202.010

**Answer:**

A. Reactor Trip, 1202.001

**Notes:**

Answer [A] is correct since a loss of Y11 will cause a loss of RS-1 which will trip "A" RPS channel. This will cause a reactor trip since "C" is in the tripped state. The Reactor Trip EOP contains the immediate actions that should be taken / verified.
Answer [B] is incorrect, NNI is powered from RS-1 but will transfer to alternate power.
Answer [C] is incorrect, although Y11 is normally powered from a DC source, the Rx Trip EOP has the highest priority.
Answer [D] is incorrect, although ESAS is partially powered from RS-1, no actuations will result.

**References:**

1203.012A, Chg. 041
STM 1-63, Rev. 9

**History:**

New created for 2001 SRO Exam.
Used on 2004 SRO Exam.
Selected for use on 2007 SRO Exam.
Selected for 2011 SRO Exam.

Location: C10

Device and Setpoint: N/A

RS1
INVERTER
TROUBLE

Alarm: K01-A5

**NOTE**
This annunciator has multiple inputs with reflash capability.

1.0 OPERATOR ACTIONS

1. Check SPDS ACDC display to determine RS1 bus voltage AND Inverter (Y11 or Y15) feeding it.
2. Dispatch Operator to local alarm panel (K1620 or K1654) on inservice Inverter (Y11 or Y15).
3. Refer to Attachment E for further instructions.
4. Refer to TS 3.8.7, TS 3.8.8, TS 3.8.9 and TS 3.8.10 for operability requirements.

2.0 PROBABLE CAUSES

- Low DC Input Voltage
- Hi DC Input Voltage
- Low Inverter Voltage
- Hi Inverter Voltage
- Inverter Failure
- Out of Sync
- Fan Failure
- Static Switch Transfer
- Hi Temp
- Alternate Source Trouble
- Low System Voltage
- Hi System Voltage

3.0 REFERENCES

- Schematic Diagram Annunciator K01 (E-451)
- Inverter Y11, Y13, Y22, Y24 Local Reflash Annunciator Schematic Diagram (E-418, Sheet 9)

The modules, logic and analog equipment associated with a single protection channel are contained within two reactor protection cabinets. Within these cabinets there is a meter for every analog signal employed by the protection channel and a visual indication of the state of every digital logic element. At the top of the left cabinet of each pair is a channel indicating panel easily visible at all times.

(Refer to Figure 63.09) Lamps on this panel give a quick visual indication of the trip status of the control rod drive breaker associated with that channel and the status of the reactor trip modules associated with all four channels. Lamps are also provided to indicate a fan failure in either cabinet and initiation of manual bypass feature.

## 2.2 Power Supply

Power supplied to each RPS cabinet is provided from 120 volt vital AC distribution panels (RS1, RS2, RS3 and RS4). The +15 and -15 volt system power supply modules are AC input, DC output, series regulated, constant voltage power sources. The supply provides a rated output of 15 volts DC with a current rating of 0 to 12 amps. Front plate components consists of an AC power on lamp, an "On-Off" circuit breaker, output voltage meter, output current meter, output voltage test jack and an output voltage adjustment.

Over current and high voltage protection is provided within the module. Low voltage will trip the associated power distribution panel breaker.

Refer to table provided below for associated channel power supply.

| RPS Cabinets | Power Supply |
|---|---|
| Channel “A” C41 | RS1 breaker 1 |
| Channel “B” C42 | RS2 breaker 1 |
| Channel “C” C43 | RS3 breaker 1 |
| Channel “D” C44 | RS4 breaker 1 |

## 1.6 System Logic

(Refer to Figure 63.01 and 63.08)

The trip logic system consists of four functionally identical reactor protection channels, each terminating in a channel trip relay within the reactor trip module. Reactor trip modules are the communications interface between the four channels and also between the reactor protection system and the Control Rod Drive system. The entire system functions as a "de-energize to trip" system. RPS logic requires any two of four channels to trip before associated CRD breakers will open.

### 1.6.1 Channel Trip Logic

When a variable monitored by RPS to cause a reactor trip exceeds acceptable values, contacts in the circuitry monitoring that value are opened by bistables or contact buffers. The contacts for the variables in each channel are connected in a series circuit, forming a contact string that terminates in a channel trip relay. When all variables are within acceptable values, all of the contacts in the contact string are closed maintaining the trip relay energized. If the circuit is broken by any one of the contacts opening, the channel trip relay is de-energized resulting in a channel trip. The channel trip relay for each channel is located in the reactor trip module (RT).

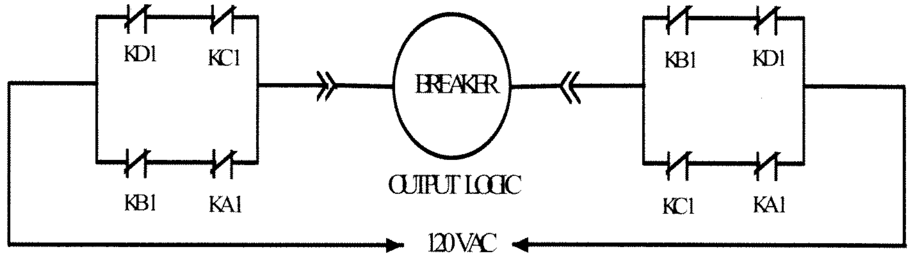
The RT relays are given the same letter designation as the protection channel in which they are physically located. Thus RPS channel "A" RT module trip relay is designated as KA, channel "B" KB, channel "C" KC and channel "D" KD. Four coincidence (output) logic relays are also located in each reactor trip module. The logic relays (one from each channel) are de-energized when contacts operated by the associated channel trip relay opens. The logic relays are designated based on cabinet location and associated trip relay. The logic relays located in "A" RPS cabinet are all designated as (1), channel "B" (2), channel "C" (3) and channel "D" (4). Example: the logic relay for "D" channel located in "B" reactor trip module would be designated as KD2.

### 1.6.2 Coincidence Logic

Coincidence Logic determines which RPS channel and which CRD breaker will trip. Each coincidence logic relay operates two contacts in the supply to the UV coils for the associated CRD breakers. A total of eight contacts are provided in each coincidence circuit, arranged in two groups of four such that any two logic relays being de-energized (four contacts open) will result in interruption of current to the UV coil and subsequent trip of the breaker(s). Refer to figure provided below or Figure 63.20.

One group of four contacts will cause the breaker(s) associated with that reactor trip module to open if called for by the combination of channel A or channel B and channel C or channel D. To complete the coincidence logic and assure a trip when called for by two of the four reactor protection system channels, a second set of four contacts is provided. This set of contacts will produce a trip when called for by channels A or C and channels B or D. Between the two sets of contacts, all possible combinations of two channels are covered and the demand by any two channels for a trip will result in a reactor trip.

When coincidence logic is satisfied the reactor trip module will trip the associated breaker(s) in the control rod drive power supply for that channel.




### 1.6.3 CRD Breaker Reactor Trip Logic

The under-voltage coils of the control rod drive breakers receive their power from the protection channel associated with each breaker. A redundant shunt trip coil will also receive a breaker trip input from its associated RPS channel. The manual reactor trip switch is interposed in series between each coincidence logic circuit and the assigned breakers under-voltage coil.

Each reactor trip module with trip a designated breaker removing power to the CRD system. The CRD breakers associated with each RPS channel reactor trip module is listed below and the associated function of each breaker.

* Channel A: Trips the main or primary power supply breaker designated as "A".
* Channel B: Trips the secondary power supply breaker designated as "B".
* Channel C: Trips control power breaker to rod hold circuitry for groups 1 through 4. Breaker designated as "C". In addition it will trip gating power supply breaker designated as "E". This removes power to controlling groups 5 through 8
* Channel D: Trips control power breaker to rod hold circuitry for groups 1 through 4. Breaker designated as "D". In addition it will trip gating power supply breaker designated as "F". This removes power to controlling groups 5 through 8.

The control rod drive circuit breaker trip and contactor combinations that initiate a reactor trip can best be noted by referring to "Hourglass Figure" provided below. Arrows show the trip logic combinations which when satisfied will cause a full Reactor trip. This is a one out of two logic used twice and is referred to as a 1-out-of-2 x 2 logic. When any two of the four reactor protection channels trip, all RT module logic's trip, commanding all control rod drive power supply breakers to trip.

## 2.0 Detailed System Description

This section will cover the cabinet design, system modules, power supply and normal operation of the Reactor Protection

**QID:** 0840 **Rev:** 0 **Rev Date:** 5/25/11 **Source:** New **Originator:** J. Cork

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

**Section:** 4.2 **Type:** Generic APEs

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

**Description:** Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

**K/A Number:** 2.2.25 **CFR Reference:** 41.5 / 41.7 / 43.2

**Tier:** 1 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3

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

**Question:** **RO:** **SRO:** 81

Unit One has entered Technical Specifications 3.8.4 due to a loss of one of the required DC electrical power subsystems.

Which of the following components comprise a DC subsystem in according with the Tech Spec bases?

1. 125V DC panel
2. Battery
3. Inverter
4. Cables
5. 120v Vital AC panel
6. Static Switch
7. Battery Charger

A. 1, 2, 4, 7

B. 1, 2, 3, 7

C. 2, 4, 6, 7

D. 2, 3, 5, 7

**Answer:**

A. 1, 2, 4, 7

**Notes:**

"A" is correct per the TS bases for 3.8.4.
"B", "C", and "D" contain incorrect components which are related to the DC system in that the DC system supplies power to the 120v AC vital inverters.

**References:**

Technical Specifications 3.8.4

**History:**

New for 2011 SRO Exam.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and 120 VAC vital bus power (via inverters). As required by SAR, Section 1.4, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Red Train and Green Train). Each subsystem consists of one 125 VDC battery, the associated battery charger for each battery, and all the associated control equipment and interconnecting cabling.

Additionally, there is one spare battery charger per subsystem, which provides backup service in the event that a battery charger is out of service. If the spare battery charger is substituted, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, each 125 VDC subsystem is powered from the inservice battery charger with the battery floating on the system. In case of a loss of normal power to the battery charger, the DC load is automatically powered from the station battery. This results in a discharge of the associated battery (and may affect both the system and cell parameters).

The Red Train and Green Train DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 480 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the 120 VAC vital buses.

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distributions System – Operating."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours in addition to supplying power for the operation of momentary loads during the 2 hour period as discussed in the SAR, Chapter 8 (Ref. 4).

# ANO 2011
# SRO Questions
# Tier 1 Group 2

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

**QID:** 0841 **Rev:** 0 **Rev Date:** 5/25/11 **Source:** New **Originator:** S. Pullin

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

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

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

**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:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 4.7 | **SRO Select:** | Yes | **Taxonomy:** | An |

**Question:** **RO:** **SRO:** 82

Given:

- Plant is at 40% power.
- A control rod (Grp 5 Rod 5) dropped at 1430.
- The control rod was declared inoperable.
- SDM was evaluated at 1500 and is NOTwithin limits of COLR.

Per Technical Specifications, which of the following actions are required?

A. Perform a SDM calculation by 0300.

B. Initiate shutdown to Mode 3 by 2300.

C. Initiate boration to restore SDM by 1600.

D. Restore control rod alignment by 1800.

**Answer:**

C. Initiate boration to restore SDM by 1600.

**Notes:**

"C" is correct per TS 3.1.4 action A.1.2 completion time.
"A" is incorrect, this would only be done if the SDM was within limits.
"B" is incorrect, this must be completed within 6 hours if all actions of condition A were not taken within the required time.
"D" is incorrect, this must be completed within 2 hours.

**References:**

Technical Specifications 3.1.4

**History:**

New for 2011 SRO Exam.

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.4 CONTROL ROD Group Alignment Limits

LCO 3.1.4 Each CONTROL ROD shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One CONTROL ROD inoperable, or not aligned to within 6.5% of its group average height, or both. | A.1.1 Verify SDM to be within the limit provided in the COLR. | 1 hour<br><br>AND<br><br>Once per 12 hours thereafter |
| | OR | |
| | A.1.2 Initiate boration to restore SDM to within limit. | 1 hour |
| | AND | |
| | A.2.1 Restore CONTROL ROD alignment. | 2 hours |
| | OR | |
| | A.2.2.1 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER. | 2 hours |
| | AND | |
| | A.2.2.2 Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis. | 72 hours |
| | AND | |

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

**QID:** 0590 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** Direct **Originator:** S.Pullin

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

**Section:** 4.2 **Type:** Generic APEs

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

**Description:** Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Tech-spec limits if both intermediate-range channels have failed.

**K/A Number:** AA2.10 **CFR Reference:** 43.5 / 45.13

**Tier:** 1 **RO Imp:** 3.1 **RO Select:** No **Difficulty:** 3

**Group:** 2 **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** An

**Question:** **RO:** **SRO:** 83

Given:

- Plant startup in progress.
- Channel 1 Source Range, NI-501 at 9 E4 cps
- Channel 2 Source Range, NI-502 at 1 E5 cps
- Reactor power Wide Range recorder, NR-502, is operable and at 5 E -2% power
- Intermediate Range Channel NI-3 at 2 E -11 amps
- Intermediate Range Channel NI-4 at 5 E -11 amps
- Power Range Channels NI-5 thru 8 at 0%

Which of the following actions are procedurally required and ensure compliance with Technical Specification 3.3.10, Intermediate Range Neutron Flux?

A. Trip the reactor immediately and refer to 1202.001, Reactor Trip.

B. Immediately suspend positive reactivity additions and initiate a shutdown so that all CRD breakers are open within one hour.

C. Lower power until Source Range is on scale and corrective maintenance is performed on appropriate IR channel(s).

D. Since NR-502 is operable, continue with plant operations until Power Range channels come on-scale.

**Answer:**

B. Immediately suspend positive reactivity additions and initiate a shutdown so that all CRD breakers are open within one hour.

**Notes:**

Answer "B" is IAW 1203.021 and TS 3.3.10 since both IR channels are inoperable.
Answer "A" is incorrect, this would be applicable only if no on-scale neutron flux existed.
Answer "C" is incorrect, this is the required action for improper overlap.
Answer "D" is incorrect, this would be the action if only one IR channel was inoperable.

**References:**

T.S. 3.3.10
1203.021, Chg. 010
STM 1-67, Rev. 13, Figure 67-1

**History:**

New for 2005 SRO exam.
Selected for 2011 SRO Exam

## 3.3 INSTRUMENTATION

### 3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One intermediate range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODE 2
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Required channel inoperable. | -------------------NOTE------------------- Plant temperature changes are allowed provided the temperature change is accounted for in the SDM calculations. --------------------------------------------- | |
| | A.1 Suspend operations involving positive reactivity changes. | Immediately |
| | <u>AND</u> | |
| | A.2 Open CRD trip breakers. | 1 hour |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.10.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.3.10.2 | Perform CHANNEL FUNCTIONAL TEST. | 31 days |

SECTION 2

LOSS OF ONE OR MORE INTERMEDIATE RANGE NI CHANNELS IN MODE 2
(Continued)

**NOTE**
Resetting intermediate range channel power supplies will cause voltage spikes and neutron power signal spikes, which may alarm annunciators CRD WITHDRAWAL INHIBITED (K08-A2) and HI SUR ROD HOLD (K08-B2).

C. Depress both toggle switches to RESET position.

D. IF intermediate range indication returns to normal,
THEN continue plant operations.

3.3 IF only one intermediate range channel is operable,
OR 2 of 4 power range instrument channels indicate >5% power,
THEN continue plant operations (TS 3.3.10).

3.4 IF all three of the following conditions are met,

- Both intermediate range channels have failed,
- 3 of 4 power range instruments indicate ≤5%,
- Reactor Power Wide Range Recorder (NR-502) is available,

THEN perform the following:

**NOTE**
Plant temperature changes which result in positive reactivity additions are allowed provided the temperature change is accounted for in the Shutdown Margin calculations (TS 3.3.10 Condition A).

- Refer to TS 3.3.10.
- Immediately suspend operations involving positive reactivity changes.
- Initiate a shutdown in order to have all CRD trip breakers open (Mode 3) within 1 hour.
- IF reactor power is ≥2%,
  THEN perform applicable steps of Power Reduction and Plant Shutdown (1102.016).
- Concurrently with reactor shutdown, monitor reactor power using NR-502.

  A. WHEN ~1E-4 log reactor power on NR-502 is reached,
  THEN observe that source range indicators come on scale.

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

STM 1-67

# Nuclear Instrumentation

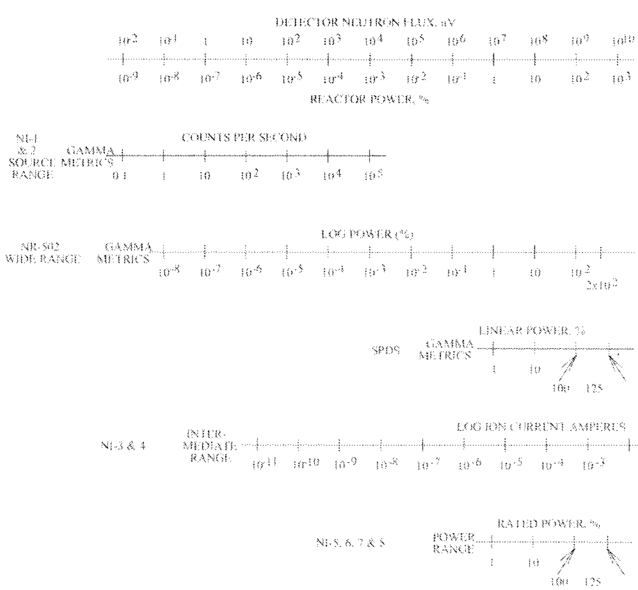
## 1.0 Introduction

This STM contains information on the Excore (Out of Core) Nuclear Instrument System (NIs) for ANO Unit 1. It includes operational theory of detectors, component locations in the plant and normal and abnormal operations and equipment conditions. The effect nuclear instruments have on plant operation, and the effect plant operations have on the Nuclear Instruments is discussed. Additional information on theory of detector operation is found in STM 1-62, Radiation Monitoring.

### 1.1 System Function

The Nuclear Instrumentation (NI) System is designed to measure over twelve decades of neutron flux using ten channels of out of core neutron detectors and instrumentation. (Refer to Figure 67.01) The full range of indications are displayed to the Reactor Operator and are supplied to the Reactor Protection and Integrated Control systems. Measurement ranges are designed to overlap to provide complete and continuous information of the full operating range of the reactor.

FIGURE 67.01: NUCLEAR INSTRUMENTATION FLUX RANGES



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

**QID:** 0347 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** G. Alden

**TUOI:** A1LP-RO-FH **Objective:** 1.4 **Point Value:** 1

**Section:** 4.2 **Type:** Generic APEs

**System Number:** 036 **System Title:** Fuel Handling Incidents

**Description:** Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurrence of a fuel handling incident.

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

**Tier:** 1 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 2

**Group:** 2 **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** C

**Question:** **RO:** **SRO:** 84

The main fuel bridge has a spent fuel assembly and is indexed over the core when an NI seal plate cover failure occurs.

Water level in the canal is falling at two inches per minute.

The main fuel bridge operator should:

A. Continue to the upender and place the assembly in the upender.

B. Leave the fuel assembly in the mast and evacuate the area.

C. Place the assembly in the fuel rack in the deep end of the canal.

D. Return the assembly to an available location in the reactor vessel.

**Answer:**

D. Return the assembly to an available location in the reactor vessel.

**Notes:**

In this scenario the fuel transfer canal level is dropping and the fuel assembly must be placed in an area that will remain covered with water after the canal is drained.
Therefore, "d" is the only correct answer. "a" is incorrect as this is a time consuming maneuver and the transfer tube should be isolated anyway to prevent losing level in the SFP. "b" is very incorrect since this will expose the assembly to atmosphere and dose rates will be lethal for quite some time. "c" is incorrect since the deep end will not contain enough water to keep the assembly covered.

**References:**

1203.042, Refueling Abnormal Operations, Chg. 007

**History:**

Used in 1999 exam.
Direct from ExamBank, QID# 4282 used in class exam
Used in 2001 RO/SRO Exam.
Selected for use in 2002 SRO exam.
Selected for 2011 SRO Exam.

SECTION 2 -- TRANSFER CANAL LEAK

**INSTRUCTIONS**

1. **Perform the following while continuing with this section:**

   - **Commence "Setting Containment Closure" Attachment K of Decay Heat Removal and LTOP System Control (1015.002). Utilize CRS Admin and Outage management when manned.**

   - **Perform "Control Room Actions For Containment Closure And Evacuation" Attachment G of Loss of Decay Heat Removal (1203.028).**

2. **Notify Shift Manager to perform the following:**

   - Implement Emergency Action Level Classification (1903.010).

   - Notify Operations Manager.

   - Notify Outage Desk, if manned.

   - Notify Reactor Engineering.

3. **IF fuel transfer mechanism is in the Reactor Building,**
   **THEN return fuel transfer mechanism to the Spent Fuel Pool.**

4. **Verify Fuel Transfer Tube Isolation (SF-45) is closed.**

5. **IF Fuel Transfer Canal level is lowering,**
   **THEN GO TO applicable section of Loss of Decay Heat Removal (1203.028) while continuing with this procedure.**

6. **IF Fuel Transfer Canal is being purified via the SF Pool Purification loop,**
   **THEN ensure that the purification loop is discharging to the DH pump suction as follows:**

   A. Verify F-4 A&B Discharge to SF Pool (SF-25) is closed.

   B. Verify SF to DH Suction Header (SF-20) is open.

7. **Return any assemblies or control components in the Main and Auxiliary Bridges and storage racks to an available position in the Reactor Vessel (Ref. Tech Spec 3.9.6).**

(continued)

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

**QID:** 0342 **Rev:** 1 **Rev Date:** 10/20/2006 **Source:** Direct **Originator:** J Haynes

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

**Section:** 4.2 **Type:** Generic APE's

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

**Description:** Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in the RCS.

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 1 | **RO Imp:** | 2.8 | **RO Select:** | No | **Difficulty:** | 4 |
| **Group:** | 2 | **SRO Imp:** | 3.4 | **SRO Select:** | Yes | **Taxonomy:** | Ap |

**Question:** **RO:** **SRO:** 85

Given:

- Reactor at 80% power.
- Failed fuel ratio, as indicated by the WCO logs, has dropped by 45%.

According to __________________, Reactor power must be reduced to ________ power.

A. 1203.045 Rapid Plant Shutdown 55%

B. 1203.019 High Activity In Reactor Coolant 55%

C. 1203.045 Rapid Plant Shutdown 40%

D. 1203.019 High Activity In Reactor Coolant 40%

**Answer:**

D. 1203.019 High Activity In Reactor Coolant 40%

**Notes:**

"D" is the correct response per 1203.019 which states to reduce Rx power by 50% of the current level with the given conditions.
"A" is incorrect, both an incorrect power level and incorrect procedure are listed.
"B" is incorrect, an incorrect power level is listed with the correct procedure.
"C" is incorrect, the correct power level with an incorrect procedure is listed.

**References:**

1203.019, Chg. 014

**History:**

Used in 1999 exam.
Direct from ExamBank, QID# 1816
Selected for use in 2002 SRO exam.
Modified for use in 2007 SRO exam.
Selected for use in 2011 SRO exam,

SECTION 2
FAILED FUEL

1.0 SYMPTOMS

1.1 PROC MONITOR RADIATION HI (K10-B2) alarm.

1.2 HIGH ALARM on Failed Fuel Iodine (RI-1237S) monitor.

1.3 Marked drop in gross/iodine ratio.

1.4 RCS sampling indicates rise in fission products.

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

**NOTE**

Selecting PDO will initiate a lengthy report that is used by Reactor Engineering. Selecting this option multiple times has the potential to lose the original report.

3.1 IF Plant Computer is available,
THEN from NASP menu initiate Plant Data Output program by selecting PDO.

**NOTE**

The following points are available on the Plant Computer:

- Failed Fuel Gross - R1237
- Failed Fuel Iodine - R1237S
- Calculated Failed Fuel Gross/Iodine Ratio - R1237R

3.2 IF failed fuel ratio drops by 40% as indicated by WCO Logsheet (OPS-A3) or Plant Computer,
THEN reduce reactor power by 50% of present power level as follows:

3.2.1 Commence power reduction per Power Reduction and Plant Shutdown (1102.016) using applicable section(s).

3.2.2 Contact the duty Reactor Engineer and Operations Manager.

# ANO 2011
# SRO Questions
# Tier 2 Group 1

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

**QID:** 0465 **Rev:** 0 **Rev Date:** 7/17/2002 **Source:** Direct **Originator:** G.Giles

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

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

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

**Description:** Ability to (a) predict the impacts of the following malfunction or operations on the CVCS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Isolation of letdown/makeup.

**K/A Number:** A2.07 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

**Tier:** 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

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

**Question:** **RO:** **SRO:** 86

Unit One is operating at 100% power.
An ICW problem causes Letdown temperature to rise to 148°F.

Which of the following provides specific guidance for recovery of the system lost due to this condition?

a. 1203.012I, Annunciator K10 Corrective Action, K10-A8 "LETDOWN TEMP HI"

b. 1104.002, Makeup and Purification System Operations

c. 1104.028, ICW System Operating Procedure

d. 1203.026, Loss of Reactor Coolant Makeup

**Answer:**

b. 1104.002, Makeup and Purification System Operations

**Notes:**

The condition given will cause a high temperature isolation of Letdown.
(b) gives the correct procedure for recovery of letdown. The other procedures listed are related to this event but do not contain actions to restore Letdown operation.

**References:**

1104.002, Makeup and Purification System Operations, Chg. 071

**History:**

Created for 2002 SRO exam.
Selected for 2011 SRO Exam.

15.0 Recovery of Letdown Following High Letdown Temperature

15.1 To recover from Letdown Coolers Outlet (CV-1221) closure, perform the following:

15.1.1 Close Letdown Block Orifice Bypass (CV-1223).

15.1.2 Close T-36A Purif Demineralizer Inlet (CV-1244) AND T-36B Purif Demineralizer Inlet (CV-1245).

15.1.3 Open T-36A & B Bypass (MU-9).

**CAUTION**

If T-36s are out of service for an extended time with Zinc Injection in-service, then a zinc transient can occur.

15.1.4 IF it is known Or expected, that both T-36A and T-36B will be out of service or bypassed for >4 hours THEN as soon as practical AND within 4 hours, perform "Securing Zinc Injection" section of Chemical Addition (1104.003).

15.1.5 Determine cause of high temperature.

**NOTE**

CV-1221 may re-close until warm water locked in letdown line is flushed past TE-1221. To restore letdown it may be necessary to cycle CV-1221.

15.1.6 WHEN cause is corrected, THEN open CV-1221.

A. Verify letdown temperature < 130°F indicated on TIS-1221 (locally) or TE-1237 (C04, SPDS or PMS).

B. IF CV-1221 closes automatically on high temperature, THEN cycle CV-1221 as necessary to lower temperature < 130°F on TE-1237 or TIS-1221.

C. IF temperature < 130°F indicated on TE-1237 AND CV-1221 closes automatically on high temperature, THEN verify < 130°F indicated on TIS-1221.

15.1.7 Open one or both of the following valves:

- CV-1244
- CV-1245

15.1.8 Close MU-9.

15.1.9 Re-establish desired letdown flow.

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

**QID:** 842 **Rev:** 0 **Rev Date:** 5/26/11 **Source:** Modified **Originator:** D Schaubroeck

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

**Section:** 3.5 **Type:** Containment Integrity

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

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

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

**Tier:** 2 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 3

**Group:** 1 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** C

**Question:** **RO:** **SRO:** 87

Given:

- All Pressurizer heaters ON.
- RCS pressure is 2050 psig and going down.
- Quench Tank (T-42) level is rising.
- Pressurizer Level Control Valve, CV-1235, full closed.

For the above conditions, which of the following actions is procedurally required to be performed?

A. Close the Spray Line Isolation Valve (CV-1009) per Section 6, Pressurizer Spray Valve (CV-1008) Failure, of OP-1203.015, Pressurizer Systems Failure.

B. Close the ERV Isolation Valve (CV-1000) per Section 1, Electromatic Relief Valve (PSV-1000) System Failure or Leak, of OP-1203.015, Pressurizer Systems Failure.

C. Take manual control of Pressurizer Level Control Valve (CV-1235) and open per OP-1203.026, Loss of Reactor Coolant Makeup.

D. Minimize letdown flow per OP-1104.002, Makeup & Purification System Operation.

**Answer:**

B. Close the ERV Isolation Valve (CV-1000) per Section 1, Electromatic Relief Valve (PSV-1000) System Failure or Leak, of OP-1203.015, Pressurizer Systems Failure.

**Notes:**

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

**References:**

1203.015, Chg. 016

**History:**

Developed for use in 98 RO Re-exam
Selected for 2005 RO re-exam.
Modified for 2011 SRO exam.

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

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

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

**Section:** 3.5 **Type:** Containment Integrity

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

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

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

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

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

Given:

PARENT

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

For the above conditions, which of the following actions is procedurally required to be performed?

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

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

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

D. Minimize letdown flow.

**Answer:**

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

**Notes:**

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

**References:**

1203.015, Chg. 016

**History:**

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

SECTION 1-- ELECTROMATIC RELIEF VALVE (PSV-1000) SYSTEM FAILURE OR LEAK

## INSTRUCTIONS

> **NOTE**
> CV-1000 torque switch can be overridden in the OPEN or CLOSE direction by holding the hand switch in the respective position.

**1. Close Pressurizer ERV Isolation Valve (CV-1000).**

**2. IF ERV leakage with CV-1000 closed exceeds capability to maintain RC pressure, THEN trip reactor and refer to Emergency Operating Procedure series (1202.XXX).**

**3. IF closing CV-1000 stops leak, THEN perform the following:**

A. Continue power operations with ERV isolated.

B. Notify Ops Manager.

C. Log in station log and on plant status board.

**4. IF closing CV-1000 does NOT stop leak, THEN perform the following:**

> **NOTE**
> - SPDS/PDS point L1051, Quench Tank volume (gallons) can help determine fill rate.
> - Steam space leakage of 0.1 gpm will raise heater load 8.75 KW.

A. Initiate leak-rate determination per "RCS Leakage Monitoring" section of RCS Leak Detection (1103.013).

B. IF total RC leakage exceeds limit allowed by Tech Spec, THEN perform Rapid Plant Shutdown (1203.045).

1) Refer to Emergency Action Level Classification (1903.010) to determine emergency class.

C. IF total PZR steam space leakage is >1 gpm, THEN initiate a Condition Report and perform an operability determination within 24 hours.

1) IF leakage is evaluated as unsafe, THEN commence plant shutdown per Power Reduction and Plant Shutdown (1102.016), and Plant Shutdown and Cooldown (1102.010).

D. Monitor Quench Tank (T-42) pressure, level, and temperature. Maintain quench tank parameters within limits described in Pressurizer Operation (1103.005).

(continued)

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

**QID:** 0843 **Rev:** 0 **Rev Date:** 5/26/11 **Source:** New **Originator:** S. Pullin

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

**Section:** 3.7 **Type:** Instrumentation

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

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

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 3.9 | **RO Select:** | No | **Difficulty:** | 3 |
| **Group:** | 1 | **SRO Imp:** | 4.5 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** **SRO:** 88

Given:

- Plant is at 100% power and stable.
- "A" RPS is inoperable due to RCS pressure transmitter failure.
- I&C has finished performing a surveillance on "C" RPS and the SM declares the channel inoperable.

Which of the following actions are required to be performed FIRST per Technical Specification 3.3.1?

A. Place both "A" and "C" RPS channels in bypass.

B Place "A" RPS channel in trip and prevent bypass of remaining channels.

C. Reduce thermal power to <10% of Rated Thermal Power.

D. Place "A" in channel bypass and trip "C" RPS channel.

**Answer:**

D. Place "A" in channel bypass and trip "C" RPS channel.

**Notes:**

"D" is correct, per Tech Spec 3.3.1 action B, one channel must be tripped and the other place in bypass.
"A" is incorrect, although 3.3.1 says to place inoperable channels in bypass or trip, two channels cannot be place in bypass.
"B" is incorrect, although 3.3.1 says to do this, this is for one inoperable channel only.
"D" is incorrect, 3.3.1 action statement G says to do this, this applies only when three channels are inoperable.

**References:**

Technical Specifications 3.3.1

**History:**

New for 2011 SRO Exam.

## 3.3 INSTRUMENTATION

### 3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 Four channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One channel inoperable. | A.1 Place channel in bypass or trip. | 1 hour |
| | OR | |
| | A.2 Prevent bypass of remaining channels. | 1 hour |
| B. Two channels inoperable. | B.1 Place one channel in trip. | 1 hour |
| | AND | |
| | B.2.1 Place second channel in bypass. | 1 hour |
| | OR | |
| | B.2.2 Prevent bypass of remaining channels. | 1 hour |
| C. Three or more channels inoperable.<br>OR<br>Required Action and associated Completion Time of Condition A or B not met. | C.1 Enter the Condition referenced in Table 3.3.1-1 for the Function. | Immediately |

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| D. As required by Required Action C.1 and referenced in Table 3.3.1-1. | D.1 Be in MODE 3.<br><br>AND<br><br>D.2 Open all control rod drive (CRD) trip breakers. | 6 hours<br><br><br><br>6 hours |
| E. As required by Required Action C.1 and referenced in Table 3.3.1-1. | E.1 Open all CRD trip breakers. | 6 hours |
| F. As required by Required Action C.1 and referenced in Table 3.3.1-1. | F.1 Reduce THERMAL POWER < 45% RTP. | 6 hours |
| G. As required by Required Action C.1 and referenced in Table 3.3.1-1. | G.1 Reduce THERMAL POWER < 10% RTP. | 6 hours |

SURVEILLANCE REQUIREMENTS

------------------------------------------------------------NOTE------------------------------------------------------
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.
-------------------------------------------------------------------------------------------------------------------------

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.3.1.1 Perform CHANNEL CHECK. | 12 hours |

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

**QID:** 0844 **Rev:** 0 **Rev Date:** 11/28/06 **Source:** Modified **Originator:** Cork/Possage

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

**Section:** 3.5 **Type:** Containment Integrity

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

**Description:** Ability to (a) predict the impacts of the following malfunctions or operations on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of Containment Spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature, exceeded cavitation, voiding, or sump level below cutoff (interlock) limit.

**K/A Number:** A2.07 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

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

**Question:** **RO:** **SRO:** 89

Given
A LOCA has caused ES actuation
Transfer to RB sump recirculation has been completed.
The following indications are observed:
- RB Sump level dropped
- RB Flood level is steady
- Both LPI Pump Discharge pressures dropping to 100 psig and rising up to 160 psig
- Both RB Spray P-35A/B ES Failure annunciators are coming in and subsequently clearing
- HPI Flow Indications are erratic
- Dose assessment reports there is no offsite release in progress.

Which of the following actions should be taken for these conditions?

A. Verify BWST Outlet Valves CV-1407 and CV-1408 open per RT-10, Verify Proper ESAS Actuation.

B. Override and throttle both LPI Pumps close to but not below 2800 gpm per pump per 1202.010 Att. 1, Shift to RB Sump Suction.

C. Override and stop one RB Spray Pump per 1202.010 Att. 1, Shift to RB Sump Suction.

D. Override and throttle both RB Spray Pumps to 1050 - 1200 gpm each per RT-10, Verify Proper ESAS Actuation

**Answer:**

B. Override and throttle both LPI Pumps close to but not below 2800 gpm per pump per 1202.010 Att. 1, Shift to RB Sump Suction.

**Notes:**

Answer "B" is correct IAW 1202.010, Att. 1, since the indications are of sump blockage and LPI flow must be throttle to the minimum required.
Answer "A" is incorrect, although initially open during ESAS, after transfer to RB sump recirc the BWST outlet valves are closed.
Answer "C" is incorrect, this action would be correct if there were indications of a CNTMT breach.
Answer "D" is incorrect, although this is for RB/LPI pump NPSH concerns, this action is performed prior to transfer to RB sump recirc.

**References:**

1202.010, Chg. 007, Att. 1

**History:**

New for the 2007 Exam
Modified for 2010 SRO Exam.

5. **IF RB sump blockage is indicated by fluctuations in LPI, HPI or RB Spray parameters below, THEN perform all the following:**

- RB Sump level dropping
- Fluctuations in LPI, HPI or RB Spray parameters below,
  - Discharge press, suct press or flow on dedicated SPDS displays
  - Flow on C16/C18
  - LPI discharge press, suct press, or motor amps on dedicated PDS/PMS displays.
  - Discharge press, suct press, flow or motor amps on the SPDS points listed below,

| **SPDS Points to Monitor for RB Sump Blockage** | | | | | | | |
|---|---|---|---|---|---|---|---|
| | **LPI** | | **RB Spray** | | **HPI** | | |
| | **P-34A** | **P-34B** | **P-35A** | **P-35B** | **P-36A** | **P-36B** | **P-36C** |
| **disch press** | P1404 | P1405 | P2426 | P2425 | P1241 | P1242 | P1243 |
| **suct press** | P1407 | P1408 | P2429 | P2428 | P1246 | P1247 | P1248 |
| **flow** | F1401 | F1402 | F2401 | F2400 | F1228, F1230, F1231, F1232 | F1209, F1210, F1211, F1212 | |
| **motor amps** | I1A305 | I1A405 | | | | | |

A. Re-verify suction flowpath properly aligned as follows:

1) Verify the following valves open:

| | **P-34A** | **P-34B** |
|---|---|---|
| **RB Sump Outlets (Outside RB)** | CV-1405 | CV-1406 |
| **RB Sump Outlets (Inside RB)** | CV-1414 | CV-1415 |
| **Suctions From BWST** | CV-1436 | CV-1437 |

2) Verify the following valves closed:

| | **P-34A** | **P-34B** |
|---|---|---|
| **BWST Outlets** | CV-1407 | CV-1408 |

B. Override **AND** throttle LPI to minimum flow listed below (CV-1400 & 1401):

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

(5. CONTINUED ON NEXT PAGE)

5. (Continued).

C. **IF both** trains of RB Spray are operating, **THEN** perform the following:

1) **IF** there is **no** evidence of Containment breach, **THEN** perform the following:

a) Override **AND** stop **both** RB Spray pumps (P-35A and B).

b) Override **AND** close **both** RB Spray Block valves:

| P-35A | P-35B |
|---|---|
| CV-2401 | CV-2400 |

c) **GO TO step E.**

2) **IF** there is evidence of Containment breach, **THEN** perform the following:

a) Override **AND** stop **one** RB Spray pump (P-35A or B).

b) Override **AND** close associated RB Spray Block valve.

| P-35A | P-35B |
|---|---|
| CV-2401 | CV-2400 |

c) **GO TO step E.**

D. **IF one** train of RB Spray is operating
**AND**
there is **no** evidence of Containment breach,
**THEN** perform the following:

1) Override **AND** stop RB Spray pump (P-35A or B).

2) Override **AND** close associated RB Spray Block valve.

| P-35A | P-35B |
|---|---|
| CV-2401 | CV-2400 |

E. Contact TSC for further direction.

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

**QID:** 0740 **Rev:** 2 **Rev Date:** 6/3/2008 **Source:** Direct **Originator:** Steve Pullin

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

**Section:** 3.4 **Type:** RCS Heat Transport

**System Number:** 061 **System Title:** Auxiliary / Emergency Feedwater (AFW System)

**Description:** Ability to (a) predict the impacts of the following malfunction 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: Loss of DC Power.

**K/A Number:** A2.03 **CFR Reference:** 41.5 / 45.5

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 3.1 | **RO Select:** | No | **Difficulty:** | 3 |
| **Group:** | 1 | **SRO Imp:** | 3.4 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** **SRO:** 90

Given:
- Unit Tripped on loss of D01
- EFW has actuated

What is the status of EFW control valve CV-2646, "P7B to A OTSG" and EFW control valve CV-2648, "P7A to B OTSG"; and which procedure should FIRST be used to address the EFW system?

A. Valves closed; 1202.003, "Overcooling"

B. Valves closed; 1203.036, "Loss of 125 V DC"

C. Valves open; 1202.003, "Overcooling"

D. Valves open; 1203.036, "Loss of 125 V DC"

**Answer:**

D. Valves failed open; 1203.036, "Loss of 125 V DC"

**Notes:**

A is incorrect with the incorrect failure mode and incorrect procedure.
B is incorrect with the incorrect failure mode and correct procedure.
C is incorrect with the correct failure and incorrect procedure.
D is correct with the correct failure mode and correct procedure.

**References:**

1203.036, "Loss of 125 V DC" Chg 009
1107.004 Chg 018

**History:**

New for the 2008 SRO Exam.
Selected for 2011 SRO Exam.

SECTION 1 -- **LOSS OF D01**

1.0 SYMPTOMS

1.1 Low DC voltage alarms:
(alarms inoperable if generator output breakers open).

- D01 UNDERVOLTAGE (K01-A7)
- D11 LOSS OF VOLTAGE (K01-B7)
- RA1 LOSS OF VOLTAGE (K01-C7)
- D01 TROUBLE (K01-D7)

- H1 DC CONTROL POWER OFF (K02-B4)
- A1 DC CONTROL POWER OFF (K02-C6)
- A3 DC CONTROL POWER OFF (K02-D6)
- SU 1 L.O. RELAY DC FAILURE (K02-D1)

- SU 2 L.O. RELAY DC FAILURE (K02-E3)
- GENERATOR L.O. RELAY DC FAILURE (K04-D8)
- TURBINE L.O. RELAY DC FAILURE (K04-B5)
- EOS SYSTEM TROUBLE (K04-C5)

1.2 Loss of breaker position indicator lights for plant buses on left side of C10.

1.3 "Trip Solenoid Power Available" light on C01 **NOT** lit.

2.0 IMMEDIATE ACTION

NONE

3.0 FOLLOW-UP ACTIONS

3.1 At C10, transfer D11 to EMERG SUPPLY D02.

3.2 IF reactor trips,
THEN perform the following:

3.2.1 IF SG pressure is <900 psig,
THEN perform the following on BOTH SGs:

- Actuate MSLI
- Actuate EFW

3.2.2 Perform RT-6.

3.2.3 Perform Emergency Operating Procedures (1202.XXX) in conjunction with this procedure.

| 125V DC Panel RA1 Breakers | | | |
|---|---|---|---|
| BREAKER NUMBER | DESCRIPTION | CONSEQUENCES OF OPENING | REQUIRED ACTION |
| RA1-1 | DG1 run/stop indication on C10 | Loss of DG1 run/stop indication on C10. | None |
| RA1-5 | U/V Relays for Load Center B5 | Loss of B5 UV protection | None |
| RA1-6 | CV-2646, EFW to SGA CV-2648, EFW to SGB | EFW control valves CV-2646 and CV-2648 fail full open. | Verify EFW shutdown or controlling on isolation valves. |
| RA1-8 | C-186 EFIC Main Steam Isol. | Loss of redundant trip solenoid valve for MSIVs. If MSIVs are closed, opening this breaker will open the MSIVs unless instrument air is isolated. | None if MSIVs are open. Verify instrument air is isolated if MSIVs are closed. |
| RA1-9 | RB Cooler SW Inlet Bypass SV-3812,RB Purge Iso Valves CV-7403 & CV-7404 and ES control for following valves: DH Cooler Bypass CV-1433 Letdown Coolers Outlets CV-1214 and CV-1216, and RCP Seal Bleedoff CV-1270 through CV-1273 C18 | RB Cooler SW Inlet Bypass SV-3812 fails open. RB Purge Iso Valves (CV-7403 and CV-7404) fail closed. Loss of ES function for following valves: DH Cooler Bypass CV-1433 Letdown Coolers Outlets CV-1214 and CV-1216, and RCP Seal Bleedoff valves CV-1270 through CV-1273 | If RB Purge in progress, stop RB Purge Fans. Address containment integrity associated with SV-3812. |
| RA1-10 | RCP Seal CLR/LTDN CLRS ICW Inlet RB Isol CV-2233, RCP MTR Air & LO CLR ICW Inlet RB Isol CV-2234, and RCP Seal CLRS/LTDN CLRS ICW Outlet RB Isol CV-2214 on C18 | Loss of redundant power supply to RCP Seal CLR/ LTDN CLRS ICW Inlet RB Isol CV-2233 and RCP MTR Air & LO CLR ICW Inlet RB Isol CV-2234. RCP Seal CLRS/LTDN CLRS ICW Outlet RB Isol CV-2214 fails open. | Address containment integrity associated with CV-2214. |

# ANO 2011
# SRO Questions
# Tier 2 Group 2

**QID:** 0570 **Rev:** 0 **Rev Date:** 5/3/05 **Source:** Direct **Originator:** J.Cork
**TUOI:** A1LP-RO-RCS **Objective:** 5 **Point Value:** 1

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

**System Number:** 002 **System Title:** Reactor Coolant System

**Description:** Ability to (a) predict the impacts of the following malfunctions or operations on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of forced circulation.

**K/A Number:** A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

**Tier:** 2 **RO Imp:** 4.1 **RO Select:** No **Difficulty:** 3

**Group:** 2 **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** An

**Question:** **RO:** **SRO:** 91

The plant was at 80% power when ONLY one failure caused the following:

- Plant running back
- Tavg Control swaps to Loop "A"
- "A" Main Feedwater flow rapidly rising
- "B" Main Feedwater flow rapidly dropping
- Feedwater pumps' discharge crosstie valve shut

Which one of the following procedures would be used in response to the above conditions?

A. 1203.022, Reactor Coolant Pump Trip

B. 1203.020, Load Rejection

C. 1203.047, Loss of NNI Power

D. 1203.027, Loss of Steam Generator Feed

**Answer:**

A. 1203.022, Reactor Coolant Pump Trip

**Notes:**

"A" is correct, indications are indicative of FW re-ratioing in response to an RCP trip.
"B" is incorrect, although the plant is running back, FW flows should be matched.
"C" is incorrect, a loss of NNI power would cause both FW flows to be rapidly rising.
"D" is incorrect, these could be the indications of a "B" MFW pump control problem but "A" FW would be dropping due to mis-matched T-cold temps.

**References:**

1203.022, Chg. 011
STM 1-69, Rev. 14

**History:**

New for 2005 SRO exam.
Selected for 2011 SRO Exam.

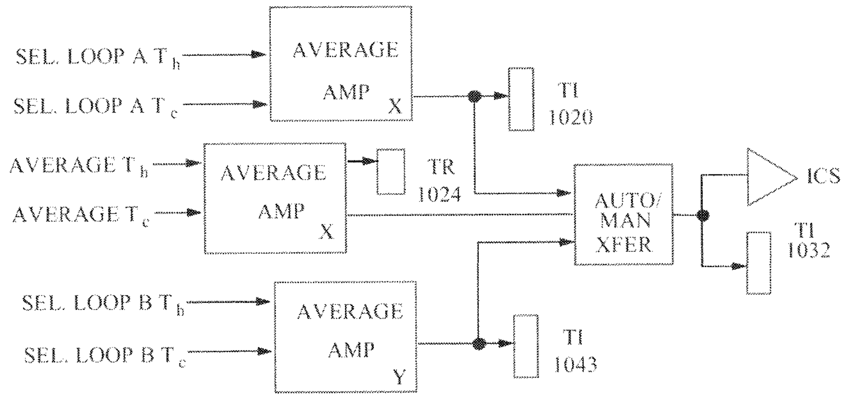
## ENTRY CONDITIONS

- **Annunciator Alarms**
  - RCP TRIP (K08-A6)
  - RCS FLOW LO (K09-D2)
  - UNIT MASTER IN TRACK (K07-A1)
  - LOSS OF RCP RUNBACK IN EFFECT (K07-B1)
  - FW RERATIO ON RC-FLOW ENABLED (K07-E1)
  - A OTSG BTU LIMIT (K07-E2)
  - B OTSG BTU LIMIT (K07-E3)
- **RCS flow dropping**
- **ICS running back**

is used for calculation of Unit RCS average temperature and Unit RCS differential temperature.

### 3.3.5 RCS Average Temperature

RCS loop "A", loop "B", and Unit Tave indications are calculated. The loop average temperatures are displayed on a 520 °F to 620 °F meter which is located on C03 (TI-1020/TI-1043). Unit Tave is displayed on a recorder located on C-13.




Average amplifiers average the SASS selected Th and Tc signals. RCS loop "A" average temperature is calculated by the NNIX channel. The NNIY channel calculates RCS loop "B" average temperature. Unit Tave is calculated from the average Th and Tc (loop "A" and loop "B" Th and Tc are averaged) signals.

Loop A average temperature, loop B average temperature, and Unit average temperature provide input to an auto/manual selector switch. The Tave selector switch output supplies the digital Tave indicator on C03 and ICS for temperature control. Unit Tave, loop A Tave, or loop B Tave may be selected by depressing the appropriate button. The selected average temperature will be backlighted. The Tave selector switch selects one of the inputs based on RCS flow. Normally Unit Tave is selected for display and control. Should either RCS loop flow drop below 95%, the opposite loop Tave is selected for output. For instance, if RCS loop A flow is less than 95% then loop B Tave is selected. In this case, the operator will not be able to select any other average temperature. If both loop flows are less than 95%, then any of the inputs may be selected.

### 3.3.6 Differential Temperatures

Loop A, loop B, and Unit differential temperatures are calculated from the hot leg and cold leg temperature inputs. The loop A SASS selected cold leg and hot leg temperatures are supplied to a difference amplifier and the loop B SASS selected cold leg and hot leg temperatures are supplied to a difference amplifier. The difference amplifiers subtract the cold leg temperature from the hot leg temperature. The resulting output is displayed on the loop A/B differential meter located on C-13. The range of indication is 0 °F to 70°F. The average hot leg and average cold leg temperatures (described above) also supply inputs a difference amplifier. The difference amplifier supplies the Unit differential temperature

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

**QID:** 0450 **Rev:** 1 **Rev Date:** 11/29/06 **Source:** Direct **Originator:** J.Cork

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

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

**System Number:** 034 **System Title:** Fuel Handling

**Description:** Ability to apply Technical Specifcations for a system.

**K/A Number:** 2.2.40 **CFR Reference:** 41.10 / 43.2 / 43.5 / 45.3

**Tier:** 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

**Group:** 2 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** C

**Question:** **RO:** **SRO:** 92

Given:

- Refueling is in progress.
- A fuel assembly is being moved toward the upender in the Fuel Transfer Canal.
- Source Range channel NI-502 power supply fails.

Which of the following actions is appropriate for these conditions?`

A. All refueling operations may continue as long as one Source Range channel is operable.

B. The fuel assembly may be placed in any alternate core location during repairs.

C. The fuel assembly may be moved to the Spent Fuel Pool but core alterations are not allowed.

D. The fuel assembly must be placed back in its original position in the core.

**Answer:**

C. The fuel assembly may be moved to the Spent Fuel Pool but core alterations are not allowed.

**Notes:**

Per 1506.001 and Tech Specs, answer "C" is correct. When changing core geometry, two neutron flux detectors are required.
Answer "A" is incorrect, one detector is ok if fuel is being shuffled from SFP to the RB, but not if the core is being altered.
Answer "B" is incorrect, the fuel assembly may not be placed back into the core, this is a core alteration.
Answer "D" is incorrect, the fuel assembly may not be placed back into the core, this is a core alteration.

**References:**

Technical Specification 3.9.2

**History:**

Created for 2002 RO exam.
Used in 2007 SRO exam. KA changed from 034 K1.04 to 2.2.27.
Selected for 2011 SRO Exam.

## 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. | Immediately |
| | 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 |
| B. No OPERABLE source range neutron flux monitor. | B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status. | Immediately |
| | AND | |
| | B.2 Perform SR 3.9.1.1. | Once per 12 hours |

**QID:** 0845 **Rev:** 0 **Rev Date:** 5/26/11 **Source:** New **Originator:** S. Pullin
**TUOI:** **Objective:** **Point Value:** 1

**Section:** 3.4 **Type:** Heat Removal from Reactor Core
**System Number:** 045 **System Title:** Main Turbine Generator System
**Description:** Ability to (a) predict the impacts of the following malfunctions or operations on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunction of electrohydraulic control.

**K/A Number:** A2.17 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 2 | **RO Imp:** | 2.7 | **RO Select:** | No | **Difficulty:** | 3 |
| **Group:** | 2 | **SRO Imp:** | 2.9 | **SRO Select:** | Yes | **Taxonomy:** | An |

**Question:** **RO:** **SRO:** 93

Given:

-Plant 100% power
-Dispatcher calls and requests Unit 1 to reduce power to 500 MWe in 10 minutes

During power reduction the following alarms come in:
-MAIN STEAM PRESSURE HI/LO (K07 C6)
-UNIT MASTER IN TRACK (K07 A1)

Which of the following procedures would be in use to mitigate this event?

A. ICS Abnormal Operation 1203.001 and Rapid Plant Shutdown 1203.045

B. ICS Abnormal Operation 1203.001 and Power Reduction and Plant Shutdown 1102.016

C. Load Rejection 1203.020 and Rapid Plant Shutdown 1203.045

D. Load Rejection 1203.020 and Power Reduction and Plant Shutdown 1102.016

**Answer:**

A. ICS Abnormal Operation 1203.001 and Rapid Plant Shutdown 1203.045

**Notes:**

"A" is correct, the symptoms meet the entry conditions for 1203.001 and the rate at which power must be reduced dictates that 1203.045 be used.
"B" is incorrect, the symptoms meet the entry conditions for 1203.001 but the rate for power reduction exceeds that where 1102.016 would be used.
"C" is incorrect, the symptoms meet the entry conditions for 1203.001 but 1203.020 would be used for other conditions.
"D" is incorrect, this combination makes the answers meet the logical combo of choices.

**References:**

1203.001, Chg. 011
1203.045, Chg. 007
1102.016, Chg. 016
1203.012F, Chg. 028

**History:**

New for 2011 SRO Exam.

SECTION 7 -- Selected Turbine Header Pressure High

## ENTRY CONDITIONS

**One or more of the following:**

- SASS MISMATCH (K07-B4)
- Affected loop Turbine Header Pressure SASS Enable Light OFF (± 18 psig)
- MAIN STEAM PRESSURE HI/LO (K07-C6)
- Turbine Reference/Setter rising
- Turbine Header Pressure rising
- OTSG Pressure Lowering
- EFIC SG Pressures Lowering

Location: C13

Device and Setpoint: see page 3 of 3.

UNIT MASTER
IN
TRACK

Alarm: K07-A1

1.0 OPERATOR ACTIONS

1. Determine cause of tracking.

2. IF Unit Master in track is a result an ICS failure
OR an ICS input signal failure,
THEN take manual control of affected ICS station(s)
AND return plant to steady-state condition.

   A. Refer to ICS Abnormal Operation (1203.001).

3. IF caused by EHC in manual,
THEN perform the following:

   A. Verify that NSSS stabilizes.

   B. Operate EHC system OR ICS in manual until problem has been corrected.

4. IF caused by ULD >40% (~360 MWe) and asymmetric rod condition,
THEN GO TO ASYM ROD RUNBACK IN EFFECT (K07-B3).

5. IF caused by loss of RC pump,
THEN GO TO LOSS OF RCP RUNBACK IN EFFECT (K07-B1).

6. IF caused by loss of MFWP (P-1A or P-1B),
THEN GO TO LOSS OF MFP RUNBACK IN EFFECT (K07-B2).

7. IF caused by ULD greater than runback limit OR maximum load limit,
THEN perform the following:

   A. Verify ICS running back to appropriate power level. Refer to HI LOAD LIMIT IN EFFECT (K07-C3).

   B. IF necessary,
   THEN operate affected portion of ICS in manual until problem is corrected.

8. IF caused by cross-limits,
THEN GO TO FEEDWATER IS REACTOR LIMITED (K07-C2) OR REACTOR IS FEEDWATER LIMITED (K07-C1).

9. IF caused by SG/RX Demand station in HAND,
THEN maintain desired unit demand by operating H/A station in HAND.

Location: C13

Device and Setpoint:
Stm Header A (PT-2683), high 935 psig, low 835 psig
Stm Header B (PT-2633), high 935 psig, low 835 psig
Stm Gen E-24B to HPT (PS-6676) 930 psig
Stm Gen E-24A to HPT (PS-6677) 930 psig

MAIN STEAM
PRESSURE
HI/LO

Alarm: K07-C6

1.0 OPERATOR ACTIONS

**NOTE**
Signal monitor checks header pressure selected at Steam Generator Header Pressure recorder on C03, either PT-2683 or PT-2633.

1. Check for the following conditions:

   - Turbine at valve position limit
   - Cross-limits in effect
   - Load limit in effect
   - Runback in effect

   A. IF none of the above conditions applies,
   THEN verify turbine has rejected to OPER AUTO.

2. IF alarm is result of an ICS input signal failure,
THEN GO TO ICS Abnormal Operation (1203.001).

3. Adjust either ICS or EH panel to obtain proper steam header pressure.

   A. Turbine should NOT be placed in INTEG CONTROL until cause of error is known and corrected.

4. Initiate steps to determine cause of pressure deviation.

2.0 PROBABLE CAUSES

1. Transmitter malfunction.

3.0 REFERENCES

Schematic Diagram Annunciator K07 (E-457)

SECTION 7 -- Selected Turbine Header Pressure High

**INSTRUCTIONS**

1. **Perform the following:**

   - Verify Turbine in OPER AUTO OR TURB MAN control.

     - IF Turbine in OPER AUTO,
       THEN perform the following:

       A. While monitoring SG pressure, lower SETTER as necessary to lower turbine load to stabilize RCS Pressure.

       B. Depress GO pushbutton AND release.

       C. Verify REFERENCE matches SETTER.

     - IF Turbine in TURB MAN,
       THEN while monitoring SG pressure operate GV Lower pushbuttons as necessary on C01 Lower Operator Console to lower Turbine Load to stabilize RCS pressure.

   - Verify SG/RX Demand H/A station in HAND.

   - Place BOTH TURB BYP Valve H/A stations in HAND.

2. **IF open,**
   **THEN while monitoring turbine load and SG pressure close LOOP A and LOOP B TURBINE BYPASS VALVES.**

3. **Select the good Turbine Header Pressure instrument for indication.**

4. **Lower SG/RX Demand H/A station as necessary to stabilize power < 100%.**

5. **Proceed as directed by CRS/SM.**

**END**

## INSTRUCTIONS

**NOTE**

- Shutdown rate shall be based on plant conditions and safety considerations. Rate may be raised or lowered at any time as plant conditions necessitate.
- Recommended shutdown rates for RCS leaks inside containment with no additional complications are as follows:
  - <50 gpm -- 0.5 to 5% per minute
  - ≥50 gpm -- 5 to 10% per minute
- Use of this procedure may be terminated at any point if a complete shutdown is not required.
- Net Generation can be monitored using PMS point JNETGEN.

1. **Commence a plant shutdown at 0.5 to 10% per minute.**

   - **IF** shutdown is due to RCS leak or steam leak inside RB,
     **THEN** maximize RB cooling per RT9 of 1202.012.

   - **IF** necessary to maintain Makeup tank level,
     **THEN** open BWST Outlet to Operating HPI Pump (CV-1407 or CV-1408)
     **AND**
     minimize or isolate Letdown.

   - **IF** shutdown due to Dispatcher request,
     **THEN** as time allows, refer to "Transmission Loading Relief Request", Attachment L of Electrical System Operations (1107.001).

     – Verify ANO2 informed of Dispatcher request.

2. **Monitor ICS and EHC subsystems for proper integrated response.** ←

   - Manual control of ICS or EHC subsystems may be initiated at any time deemed appropriate by the operator.

   - IF desired to restore automatic control of a subsystem,
     THEN refer to Integrated Control System (1105.004).

3. **IF plant has been stabilized with the reactor critical,** ←
   **THEN instruct At The Controls Operator to refer to "Contingency Reactivity Plans"**
   **AND**
   **Exhibit A (Operation of APSR Group) of Power Operation (1102.004)**
   **both of which are located in the Plant Data Book.**

**CAUTION**

As plant power is lowered, additional Pressurizer heaters will cycle on. Pressurizer heater bank 4 will add ~42 amps to Bus B2 load. Consideration should be given to shifting loads from Bus B2 to Bus B1 to allow adequate margin to B2 amperage limit.

8.3 Review anticipated Bus B2 loading and shift loads as necessary from Bus B2 to Bus B1 to ensure Bus B2 amperage limits are NOT exceeded during power reduction or subsequent Tave reduction after SG low level limits are reached. ______

8.4 Secure systems as follows:

- IF reducing power below Heater Drain Pump operation, THEN as time permits, perform "Removing MSR DI From Service" section of MSR Drain Demineralizer Operation (1106.031). ______
- IF reducing power <90% THEN perform "Securing Zinc Injection" section of Chemical Addition (1104.003). ______

**CAUTION**

Under non-steady state operating conditions, an NI calibration is required daily (SR 3.3.1.2). A 50% power level change could cause nuclear instrumentation to be inaccurate by as much as 5% due to change in cold leg temperature.

**NOTE**

- Maintaining reactor maneuvering rate of Attachment L, <30%/hr, also ensures the slightly less restrictive turbine maneuvering rates specified by Westinghouse are maintained.
- ULD rate of change of 0.45%/min (MWe), corresponds to 30%/hr (RX).
- Reactor Engineering can supply contingency or reactivity plans to aid reactor control.
- Between ~40% and ~60% power, plant response can cause oscillations in primary and secondary parameters including erratic +/-1% swings in neutron error. If Oscillations become excessive then consideration should be given to placing RX Demand H/A Station in manual per 1105.004 (Ref. CR-ANO-1-2005-1421).

8.5 Set desired ULD rate of change. Do NOT exceed reactor maneuvering limit of 30%/hr. ←

8.5.1 Set desired ULD load demand. ______

8.5.2 Adjust boric acid concentration as necessary per applicable sections of Soluble Poison Concentration Control (1103.004). ←

# ANO 2011
# SRO Questions
# Tier 3

**QID:** 0846 **Rev:** 0 **Rev Date:** 5/26/11 **Source:** Modified **Originator:** D. Thompson

**TUOI:** A1LP-SRO-FH **Objective:** 6 **Point Value:** 1

**Section:** 2.0 **Type:** Generic Knowledge and Abilities

**System Number:** 2.1 **System Title:** Conduct of Operations

**Description:** Knowledge of the fuel-handling responsibilities of the SRO.

**K/A Number:** 2.1.35 **CFR Reference:** 41.10 / 43.7

**Tier:** 3 **RO Imp:** 2.2 **RO Select:** No **Difficulty:** 2

**Group:** **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** K

**Question:** **RO:** **SRO:** 94

Given:
- Unit 1 refueling is in progress.
- Due to difficulties inserting a fuel assembly into the core, the Bridge Operator requests an alteration in the fuel load sequence.

Who must approve a change to fuel shuffle sequence?

A. SRO in Charge of Fuel Handling and Refueling Project Manager

B. Operatons Manager and Refueling Project Manager

C. SRO in Charge of Fuel Handling and Reactor Engineer

D. Operatons Manager and Reactor Engineer

**Answer:**

C. SRO in Charge of Fuel Handling and Reactor Engineer

**Notes:**

"C" is correct per 1502.004
The other choices are combos of correct personnel and incorrect personnel.

**References:**

1502.004, Chg. 48

**History:**

Modified QID 250 for 2011 SRO Exam.

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

**QID:** 0250 **Rev:** 0 **Rev Date:** 9-1-99 **Source:** Direct **Originator:** D. Slusher

**TUOI:** ANO-1-LP-RO-FH **Objective:** 14 **Point Value:** 1

**Section:** 2 **Type:** Generic

**System Number:** 2.1 **System Title:** Conduct of Operations

**Description:** Knowledge of the fuel-handling responsibilities of the SRO.

**K/A Number:** 2.1.35 **CFR Reference:** 41.10/43.7

**Tier:** 3 **RO Imp:** 2.2 **RO Select:** No **Difficulty:** 2

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

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

Movement of a fuel assembly that does NOT follow the sequence in the approved fuel shuffle procedure requires prior approval of:

Parent

A. SRO in Charge of Fuel Handling and Shift Manager

B. Shift Manager and Reactor Engineer

C. Reactor Engineer and SRO in Charge of Fuel Handling

D. SRO in Charge of Fuel Handling and Reactor Building Coordinator

**Answer:**

C. Reactor Engineer and SRO in Charge of Fuel Handling

**Notes:**

1502.004, Control of Unit 1 Refueling, requires any deviations from the fuel shuffle sequence to be approved by the SRO in Charge of Fuel Handling and the Reactor Engineer (NRC commitment). Therefore, "c" is the only correct response. Answers "a", "b" and "d" are incorrect due to providing the wrong combination supervisory personnel.

**References:**

1502.004, Rev. 040

**History:**

Used in 1999 exam.
Direct from ExamBank, QID# 4301
Used on 2004 SRO Exam.
Selected for the 2009 SRO Retake Exam.

4.2 References Used In Conjunction With This Procedure:

4.2.1 1022.012, Storage, Control and Accountability of Nuclear Fuel

4.2.2 1506.001, Fuel and Control Component Handling

4.2.3 EN-OP-116, Infrequently Performed Tests or Evolutions

4.2.4 1000.018, Housekeeping

4.2.5 1502.003, Refueling Equipment and Operator Checkouts

4.2.6 5120.416, In-Place Testing of the Unit 1 Reactor Building Purge Filtration System

4.2.7 5120.413, In-Place Testing of the Unit 1 Fuel Handling Area Filtration System

4.2.8 1025.019, System Cleanliness Controls During Modification and Maintenance

4.2.9 Control Room Emergency Air Conditioning and Ventilation (2104.007)

4.2.10 TS 3.9.3 Reactor Building Penetrations

4.3 NRC Commitments

4.3.1 P 205, Response to NRC Bulletin 89-03, Fuel in temporary core locations shall not reduce the shutdown margin below minimum required limit. Contained in Limits and Precautions and Initial Conditions sections.

4.3.2 P 9071, Emphasize housekeeping requirements. Contained in Limits and Precautions, and Initial Conditions sections.

4.3.3 P 12369, Caution tag source range power supplies. Contained in Initial Conditions section.

4.3.4 P 12368, Record neutron count rate with each fuel assembly. Contained in Instructions sections.

4.3.5 P 12366, Deviations from the fuel shuffle sequence require approval of SRO in Charge of Fuel Handling and Reactor Engineer. Contained in Limits and Precautions and Instructions sections.

4.3.6 P 14883, Ensure core offloads are performed after sufficient time for decay of fuel heat load, or when lake temperature is in range to assure existing SFP design temperature limits are not exceeded. Contained in Limits and Precautions, and in Initial Conditions sections.

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

**QID:** 0407 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** Direct **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:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** **SRO:** 95

The plat 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 0210.

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:**

A. 0400

**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.

# 5.0 ADMINSTRATIVE CONTROLS

## 5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power unit.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the unit specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Safety Analysis Report (SAR);

b. The Plant Manager Operations shall be responsible for overall safe operation of the unit and shall have control over those onsite activities necessary for safe operation and maintenance of the unit;

c. A specified corporate executive shall have corporate responsibility for overall unit nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the unit to ensure nuclear safety. The specified corporate executive shall be identified in the SAR; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

a. A non-licensed operator shall be on site when fuel is in the reactor and two additional non-licensed operators shall be on site when the reactor is in MODES 1, 2, 3, or 4.

b. The minimum shift crew composition for licensed operators shall meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

## 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. In MODES 1, 2, 3, or 4, an individual shall provide advisory technical support for the operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

(i) Each licensee shall meet the minimum licensed operator staffing requirements in the following table:

**Minimum Requirements[1] Per Shift for On-Site Staffing of Nuclear Power Units by Operators and Senior Operators Licensed Under 10 CFR Part 55**

| Number of nuclear power units operating[2] | Position | One Unit | Two units | | Three units | |
|---|---|---|---|---|---|---|
| | | One control room | One control room | Two control rooms | Two control rooms | Three control rooms |
| None | Senior Operator | 1 | 1 | 1 | 1 | 1 |
| | Operator | 1 | 2 | 2 | 3 | 3 |
| One | Senior Operator | 2 | 2 | 2 | 2 | 2 |
| | Operator | 2 | 3 | 3 | 4 | 4 |
| Two | Senior Operator | | 2 | 3 | [3]3 | 3 |
| | Operator | | 3 | 4 | [3]5 | 5 |
| Three | Senior Operator | | | | 3 | 4 |
| | Operator | | | | 5 | 6 |

[1]Temporary deviations from the numbers required by this table shall be in accordance with criteria established in the unit's technical specifications.

[2]For the purpose of this table, a nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling as defined by the unit's technical specifications.

[3]The number of required licensed personnel when the operating nuclear power units are controlled from a common control room are two senior operators and four operators.

(ii) Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. If a single senior operator does not hold a senior operator license on all fueled units at the site, then the licensee must have at the site two or more senior operators, who in combination are licensed as senior operators on all fueled units.

(iii) When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.

(iv) Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

(3) Licensees who cannot meet the January 1, 1984 deadline must submit by October 1, 1983 a request for an extension to the Director of the Office of Nuclear Regulation and demonstrate good cause for the request.

(n) The licensee shall not, except as authorized pursuant to a construction permit, make any alteration in the facility constituting a change from the technical specifications previously incorporated in a license or

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

**QID:** 0119 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork

**TUOI:** A1LP-RO-TS **Objective:** 5 **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:** No **Difficulty:** 3

**Group:** G **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** An

**Question:** **RO:** **SRO:** 96

Given the following "A" Core Flood Tank (CFT) parameters with the plant is at 100% power.

- Level transmitter out of service
- CFT level is 12.5 feet
- Boric acid concentration is 2280 ppm
- CFT pressure is 580 psig

Which of the above parameters makes the "A" CFT inoperable per Tech Specs?

A. Level transmitter

B. CFT level

C. Boric acid concentration

D. CFT pressure

**Answer:**

B. CFT level

**Notes:**

T.S. 3.5.1 requires the following:
1/2 level channels and 1/2 pressure channels
"B" is correct, since TS requires 970 cubic ft and bases states the minimum level required is 11.95 ft.
"A" is incorrect, level transmitters are addressed in TRM but only one is required.
"C" is incorrect, since TS requires 2270 ppm.
"D" is incorrect, CFT pressure is 5 psig above lower limit of 575 psig.

**References:**

Technical Specifications, 3.5.1 and Bases

**History:**

Used in 1998 SRO exam
Used in NRC developed SRO exam, no.23, 2/6/95
Selected for 2005 SRO exam
Selected for 2011 SRO Exam.

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One CFT inoperable due to boron concentration not within limits. | A.1 Restore boron concentration to within limits. | 72 hours |
| B. One CFT inoperable for reasons other than Condition A. | B.1 Restore CFT to OPERABLE status. | 1 hour |
| C. Required Action and associated Completion Time of Condition A or B not met.<br><br><u>OR</u><br><br>Two CFTs inoperable. | C.1 Be in MODE 3.<br><br><u>AND</u><br><br>C.2 Reduce RCS pressure to $\leq$ 800 psig. | 6 hours<br><br><br><br>12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.1 | Verify each CFT isolation valve is fully open. | 12 hours |
| SR 3.5.1.2 | Verify borated water volume in each CFT is $\geq$ 970 $ft^3$ and $\leq$ 1110 $ft^3$. | 12 hours |
| SR 3.5.1.3 | Verify nitrogen cover pressure in each CFT is $\geq$ 560 psig and $\leq$ 640 psig. | 12 hours |

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.4 | Verify boron concentration in each CFT is ≥ 2270 ppm. | 31 days<br><br><u>AND</u><br><br>----------NOTE--------<br>Only required to be performed for affected CFT<br>--------------------------<br><br>Once within 12 hours after each solution level increase of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration ≥ 2270 ppm |
| SR 3.5.1.5 | Verify power is removed from each CFT isolation valve operator. | 31 days |

APPLICABLE SAFETY ANALYSES (continued)

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to cover the core to the 3/4 point even assuming no liquid remains in the reactor vessel following a LOCA (Ref. 1). The downcomer then remains flooded until the HPI and LPI systems start to deliver flow for limiting large break LOCAs.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection and ensure the ability of the CFTs to fully discharge. The limiting safety analysis volume requirement is $1040 \pm 70$ ft$^3$. This volume corresponds to CFT levels of $\geq 11.95$ ft and $\leq 14.00$ ft. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The minimum nitrogen cover pressure requirement of 560 psig ensures that the contained gas volume will generate discharge flow rates during injection that satisfy the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The maximum nitrogen cover pressure limit of 640 psig will affect the amount and timing of CFT inventory discharged while the RCS depressurizes. Limiting the maximum pressure will therefore limit the CFT inventory lost through the break and assure that the CFT inventory injected into the RCS at the proper time is bounded by that predicted by the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum CFT level, the reactor will remain adequately shutdown in the cold condition following mixing of the CFT and Reactor Coolant System (RCS) water volumes. This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

In MODE 1, the CFTs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 2 and MODE 3 with RCS pressure > 800 psig, the CFTs satisfy Criterion 4 of 10 CFR 50.36.

Table 3.3.5-1

Miscellaneous Instrumentation

| | INSTRUMENT | TEST REQUIREMENTS | FREQUENCY |
|---|---|---|---|
| 1. | Decay Heat Removal System isolation valve automatic closure and interlock system[a] | TR 3.3.5.1[b]<br>TR 3.3.5.3 | 12 hours<br>18 months |
| 2. | Off-site power undervoltage and protective relaying interlocks and circuitry | TR 3.3.5.1 | 7 days |
| 3. | Sodium Hydroxide Tank Level | TR 3.3.5.3 | 18 months |
| 4. | Incore Neutron Detectors[c] | TR 3.3.5.1 | 31 days |
| 5. | Low Temperature Overpressure Protection Alarm Logic | TR 3.3.5.2<br>TR 3.3.5.3 | 18 months<br>18 months |
| 6. | Turbine overspeed trip mechanism | TR 3.3.5.2 | 18 months |
| 7. | CFT pressure and level instruments[d] | TR 3.3.5.1<br>TR 3.3.5.3 | 12 hours<br>18 months |

NOTES:

a. Surveillance testing required by Technical Specification SR 3.4.14.3 is performed with Reactor Coolant System (RCS) pressure ≥ 200 psig, but < 300 psig and includes RCS Pressure Analog Channel.

b. Includes RCS Pressure Analog Channel and Core Flood Tank Isolation Valve Position.

c. Check functioning. Not required to be met below 20% Rated Thermal Power.

d. Only one CFT pressure and one CFT level instrument required to be OPERABLE.

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

**QID:** 0852 **Rev:** 0 **Rev Date:** 06/26/2011 **Source:** New **Originator:** J. Cork

**TUOI:** ASLPP-SRO-MNTC **Objective:** 2 **Point Value:** 1

**Section:** 2 **Type:** Generic K&A

**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 / 43.3 / 45.13

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 3 | **RO Imp:** | 2.6 | **RO Select:** | No | **Difficulty:** | 2 |
| **Group:** | G | **SRO Imp:** | 3.8 | **SRO Select:** | Yes | **Taxonomy:** | Ap |

**Question:** **RO:** **SRO:** 97

OPEN REFERENCE

Unit One is at 100% power.

The following items are on the schedule to be worked this shift:
1. 1305.001, Supplement 6, Area Radiation Monitor Monthly Alarm Check
2. 1305.036, Unit 1Power Linear Amp Calibration At Power
3. WO#0175862, Corrective Maintenance on TR-9001, Gen & Coolers Gas Outlet Temperature Recorder
4. Swapping SW Pumps P4C to P4B for strainer cleaning
5. 1304.021, Unit 1 N-16 Monitor Calibration

In accordance with COPD013, Operations Maintenance Interface Standards and Expectations, which of the above items are allowed to be worked this shift?

A. 1, 3, 4

B. 2, 3, 4

C. 3, 4, 5

D. 1, 4, 5

**Answer:**

C. 3, 4, 5

**Notes:**

Per COPD013 Att. M the activities listed have the following point values:

| | |
|---|---|
| 1. 1305.001, Supplement 6, Area Radiation Monitor Monthly Alarm Check | 2 |
| 2. 1305.036, Unit 1Power Linear Amp Calibration At Power | 4 |
| 3. WO#0175862, Corrective Maintenance on TR-9001 | 1 |
| 4. Swapping SW Pumps P4C to P4B for strainer cleaning | 2 |
| 5. 1304.021, Unit 1 N-16 Monitor Calibration | 1 |

Per the same attachment the maximum sum of all activities must be 4 or less for normal control room shifts. Therefore only "C" meets this requirement with a total of 4.

**References:**

COPD013, Chg.034
Attachment M must be in SRO handout.
If asked by any candidate, inform all candidates that the Shift Manager has NOT granted approval to exceed control room activity limits.

**History:**

New for 2011 SRO Exam.

OPERATIONS DEPARTMENT DOCUMENT CHANGE REQUEST



Document Name: **COPD-013 OPERATIONS MAINTENANCE INTERFACE STANDARDS AND EXPECTATIONS**

Revision/Change Number: **034**

Purpose/Reason for Change: (May be handwritten, attach pages as required)

ATTACHMENT A: Changed Reactivity Management Program (EN-OP-103) to new procedure ANO Reactivity Management Program (COPD-030).

COPD013F CONTROL ROOM WORK ORDER AUTHORIZATION: Changed Reactivity Management Program (EN-OP-103) Att. 9.3 to new procedure ANO Reactivity Management Program (COPD-030) Risk Level appropriate attachment.

| | |
|---|---|
| Keith A Kennamore Keith A Kennamore<br>Originator | 5/13/2011<br>Date |
| Rick Crilty Rick Crilty<br>Unit 1 Ops. Independent Reviewer | 5-13-11<br>Date |
| [illegible]<br>Unit 2 Ops. Independent Reviewer | 5/13/2011<br>Date |
| Karl Jones 0887<br>Ops Manager or Unit 1 AOM | 5/13/2011<br>Date |
| Karl Jones 0887<br>Ops Manager or Unit 2 AOM | 5/13/2011<br>Date |

5/13/2011
Required Effective Date

| FORM TITLE : | FORM NO. | CHANGE |
|---|---|---|
| **OPERATIONS DEPARTMENT DOCUMENT CHANGE REQUEST** | **1015.004A** | **007** |

ATTACHMENT M



CONTROL ROOM SCHEDULING GUIDELINES

This is a guideline on Control Room scheduling aimed at minimizing online Control Room distractions. This guideline has been created in part based on INPO SOER 96-1 Control Room Supervision Operational Decision-Making and Teamwork which states: "Control Room activities must be coordinated and conducted in a professional manner that contributes to safe and reliable plant operation. It is important to manage control room activities so operators are not distracted from monitoring plant parameters properly. Excessive scheduled activities and other distractions and potential distractions to control room operators must be carefully managed and controlled." This guideline utilizes a base schedule written around standard activities with a point value assigned to maintenance affecting control room work load. The procedure assumes that a cumulative work load will not exceed a point value of 4 on-line or 8 during outages without Shift Manager approval.

1.0 Definitions

1.1 Control Room Activities: those activities that affect/distract control room personnel from routine monitoring of plant indications and controls.

1.2 Control Room Activity Grading: a point system that is assigned to control room activities based on the impact they have on control room personnel. The grading is a point value from 1 to 8. 1 is a minor impact such as a minor activity located in the control room or an activity that is limited to a single, expected control room alarm. A grade of 8 is a major impact to the control room such as a Reactor Startup.

2.0 Responsibilities

2.1 Operations Management

2.1.1 Oversees the implementation of control room scheduling and work activities.

2.2 Shift Manager

2.2.1 Reviews the schedule prior to T-6 to determine if control room work load is acceptable for the scheduled week.

2.2.2 Approves deviations for control room activities exceeding the normal 4 point on-line or 8 point outage scheduling grade.

2.2.3 When on-shift, retains the ultimate decision on whether a control room activity will be performed as scheduled.

2.2.4 Approves non-scheduled, control room activities (e.g. RP surveys, procedure changes, SCBA inspections, housekeeping activities) using Generic Control Room Activity Values for grading.

2.2.5 Ensures their respective crew reviews the schedule provided by the OWLs during the T-process By T-1. The crew will print/review impact statements and control room work authorization forms for scheduled control room activities on their upcoming dayshift (T-0). Crew will also print/review major PMTs, or PMTs associated with returning safety related equipment to service. These will be kept in files for each work day of the crews dayshift. If desired, the files may be maintained electronically.

2.3 Operations Work Control Supervisors

2.3.1 Assists Schedulers and Planners with control room activity scheduling.

2.3.2 Provides qualitative input to the Schedulers and Planners for the scheduling of simultaneous control room activities.

2.4 Scheduler

2.4.1 Ensures that the total number of scheduled control room activities at any time does not create an undue distraction to the control room operating personnel.

2.4.2 Screens scheduled activities for control room impact. If determined that an activity is a control room activity, then ensures that the work order is properly coded to reflect a control room activity.

2.4.3 Provides a schedule visually showing the aggregate control room activity levels for the following work management schedule meetings: T-11 and T-10.

2.5 Planner

2.5.1 Ensures that work orders are properly coded as a control room activity.

2.6 Work Week Manager

2.6.1 Reviews the control room activities at the appropriate T-minus meeting.

2.6.2 Ensures that the quantity and type of scheduled work activities do not create undue distractions to the control room operating personnel.

2.6.3 Assigns a scheduler to review the T-10 (scope freeze) schedule prior to the appropriate T-minus meeting to ensure control room activities meet the requirements of this guideline.

2.6.4 Obtains Shift Manager authorization if scheduled control room activities exceed the 4 point grade at any time on-line and 8 during outages.

2.6.5 Distributes a schedule which visually shows the aggregate control room activity when schedule is approved at T-10 and T-6.

2.6.6 Updates the Control Room Activity Schedule daily when control room activities are added, deleted or rescheduled to a different time in the work week. If required to update the control room activity schedule, provides updated Control Room Activity Schedule to the organization at the Plan of the Day meeting.

2.7 FIN Team

2.7.1 Uses this guidance and the Control Room Activity Schedule to identify when FIN control room activities can take place.

2.8 FIN SRO

2.8.1 Reviews the weekly schedule and ensures that FIN activities that impact the control room do not exceed the 4 point grade on-line or 8 point during outages without prior approval from the Shift Manager.

**NOTE**

Control room activity scheduling is accomplished by assigning a code to the activity that flags the activity as a control room impact. The activity is then assigned a control room distraction resource value, graded from 1 to 8. The schedule is then sorted to view only those activities that control the control room activity code and the resource view is used to see the aggregate effect.

3.0 Schedule Planning --- Basic Timeline for Scheduling of Control Room Activities

3.1 During work order planning the Planner ensures that any work orders with control room impact are coded as control room activities.

3.2 Scheduler schedules "base load" repetitive control room activities that have been previously approved by an Operations Work Control Supervisor.

3.3 T-20 to T-11, the Schedulers schedule additional control room activity work orders and activities using the guidelines of this attachment.

3.4 T-20 to T-6, the Schedulers ensure when moving or scheduling an activity, the control room activity level is acceptable.

3.5 At T-11 meeting, Schedulers, Discipline Schedulers and Operations Work Control Supervisors meet and agree on control room activity level. A copy of the aggregate Control Room Activity Schedule is provided by Schedulers for this meeting.

3.6 Prior to T-10 meeting, the Work Week Manager assigns a Scheduler to review control room activity level to verify levels are acceptable. Schedule activities should be moved as appropriate before the T-10 meeting.

3.7 At T-10 meeting, a copy of the aggregate Control Room Activity Schedule is provided for this meeting.

3.8 During T-10 meeting, control room activity levels are discussed.

3.9 At T-10 approval, scope is frozen. No changes are made to control room activities without prior approval through the scope change process.

3.10 At T-8, the aggregate Control Room Activity Schedule is provided to the implementing Operations crew to allow review as part of their normal schedule review.

3.11 T-7 to T-6, comments from the Operations crew reviews are incorporated into the schedule. Control Room Activities are rescheduled as appropriate.

3.12 At T-6 meeting, Schedulers provide a copy of the aggregate Control Room Activity Schedule to participants. Control room activity level is verified acceptable prior to T-6 approval.

3.13 When T-6 approved, aggregate Control Room Activity Schedule is distributed with approved color schedule.

3.14 During implementation week, Work Week Manager is responsible for updating aggregate Control Room Activity Schedule and providing the update daily to the organization if any control room activities are added, deleted, or rescheduled to a different time. This updated schedule is provided at the Plan of the Day meeting.

4.0 Control Room Activity Scheduling Guidance

- Control room activity should be normally limited to a value of 4 on-line and 8 during outages.
- On rare occasions, control room activity may be greater than 4 on-line or 8 during outages but requires prior approval by the Shift Manager.
- When the cumulative control room activity value is high, additional control room oversight should be scheduled during the activities.
- Schedulers and Shop Coordinators will use this guidance to schedule control room activities.
- The scheduling of control room activities should begin at T-20 and continue throughout work week development.
- Shift Managers will review the schedule for their assigned work week and approve control room activity prior to T-6.
- The on-shift Shift Manager retains the ultimate decision on whether a control room activity will be performed as scheduled.
- FIN team should use this guidance and the Control Room Activity Schedule to identify where FIN control room activities can take place.
- Operations Work Control Supervisors will provide qualitative input to the schedulers for the scheduling of simultaneous control room activities.

4.0 (continued)

- The Generic Control Room Activity Value tables in this attachment provide the normal quantitative control room activity values. The list is not all inclusive. Control room activity assessment requires both a quantitative and qualitative assessment to determine the overall control room activity grade.

- The Operations Work Control Supervisors will determine final control room activity grade for scheduled items.

- No additional control room activities are scheduled during plant maneuvers. Emergent failed equipment may necessitate the need for additional control room activities and will only be allowed with the approval of the on-shift Shift Manager.

ATTACHMENT M



5.0 Generic Control Room Activity Values (Unit 1)

| | |
|---|---|
| **Low Impact** | |
| Walkdowns / inspections | 1 |
| Print or procedure updates | 1 |
| Chart recorder maintenance | 1 |
| Sigma / Dixson maintenance | 1 |
| Maintenance that causes 1-2 control room alarm occurrences | 1 |
| Cabinet PM's | 1 |
| Main Steam Rad monitor monthly | 1 |
| Containment Hi Range Rad monitor monthly | 1 |
| RB Leak Detector paper change out | 1 |
| Control Room Rad Monitor monthly | 1 |
| Operations PMT valve strokes | 1 |
| Fire extinguisher checks | 1 |
| CSG minor computer (PMS / SPDS) maintenance | 1 |
| Starting/Stopping RB Purge | 1 |
| Refueling/Defueling | 1 |
| **Medium Impact** | |
| RPS Monthly Test (non-trip initiator portion) | 2 |
| EFIC Monthly Test | 2 |
| Maintenance that causes >2 control room alarm occurrences | 2 |
| Operations control channel changes | 2 |
| I&C DROPS/AMSAC calibrations | 2 |
| Fire detector testing that causes repeat control room alarms | 2 |
| Rad Monitor testing that causes repeat control room alarms | 2 |
| Operations RPS bypass operations | 2 |
| RCS Heatup | 2 |
| RCS Cooldown in Mode 5 | 2 |
| Starting/Stopping pumps (EFW, SW, HPI, etc.) | 2 |
| **High Impact** | |
| Operations pump surveillances | 3 |
| RPS Qrtly (Breaker trip test) | 3 |
| EDG Surveillance | 3 |
| Placing / removing control room on emergency recirc | 3 |
| Fire Panel maintenance | 3 |
| Fill Fuel Transfer Canal or RCS | 3 |
| Shifting Electrical Loads | 3 |
| RCS Cooldown to Mode 5 | 3 |
| RCS Dilution | 3 |
| Starting or Stopping DHR or RCPs | 3 |
| Undervoltage Monitor Relay Testing | 3 |
| NI Calibration | 4 |
| ICS to Auto/Manual | 4 |
| CRD exercise | 4 |
| I &C Semi-annual DROPS/AMSAC Testing | 4 |
| TV/GV Stroke Testing | 4 |
| Plant maneuver | 5 |
| Physics Testing | 5 |
| Reactor Startup | 8 |
| Draining the RCS with fuel in vessel | 8 |
| Integrated ES Test | 8 |

**No additional CONTROL ROOM ACTIVITIES are scheduled during Plant maneuvers. Emergent failed equipment may necessitate the need for additional CONTRL ROOM ACTIVITIES and will only be allowed with the approval of the on-shift Shift Manager.**

6.0 Generic Control Room Activity Values (Unit 2)

| | |
|---|---|
| **Low Impact** | |
| Chart recorder maintenance | 1 |
| Sigma / Dixson maintenance | 1 |
| Maintenance that causes 1-2 control room alarm occurrences | 1 |
| 2C31A/B EHC cabinet PM's | 1 |
| Main Steam Rad monitor monthly | 1 |
| Containment Hi Range Rad monitor monthly | 1 |
| Operations PPS bypass operations | 1 |
| CAMS paper changout | 1 |
| Control Room Rad Monitor monthly | 1 |
| Operations PMT valve strokes | 1 |
| CSG minor computer (PMS / SPDS) maintenance | 1 |
| Starting/Stopping Cntmt Purge | 1 |
| Refueling/Defueling | 1 |
| **Medium Impact** | |
| PPS Triannual (non-trip initiator portion) | 2 |
| Excore Monthly / Quarterly | 2 |
| Maintenance that causes >2 control room alarm occurrences | 2 |
| Operations control channel changes | 2 |
| Operations DSS/DEFAS operations | 2 |
| I&C DSS/ DEFAS calibrations | 2 |
| CPC Triannual | 2 |
| Fire detector testing that causes repeat control room alarms | 2 |
| Rad Monitor testing that causes repeat control room alarms | 2 |
| RCS Heatup | 2 |
| RCS Cooldown in Mode 5 | 2 |
| Starting/Stopping pumps (AFW, SW, HPSI, etc.) | 2 |
| **High Impact** | |
| Operations pump surveillances | 3 |
| CPC / CEAC operations (updating addressable constants) | 3 |
| PPS Triannual (TCB & Matrix testing) | 3 |
| EDG Surveillance | 3 |
| AAC Surveillance | 3 |
| Placing / removing control room on emergency recirc | 3 |
| 2C343 Fire Panel maintenance | 3 |
| COLSS addressable constant updates | 3 |
| Fill Refueling Canal or RCS | 3 |
| Shifting Electrical Loads | 3 |
| RCS Cooldown to Mode 5 | 3 |
| RCS Dilution | 3 |
| SDC or RCP Operations | 3 |
| ESF Response Time Testing | 3 |
| NI Calibration | 4 |
| CEA exercise | 4 |
| MSIV partial stroke | 4 |
| MTG Control Valve Stroke | 4 |
| Pump Refueling Canal | 4 |
| Moving CEAs | 4 |
| Plant maneuver * | 5 |
| Physics Testing | 5 |
| RCS Drain with TRVH installed | 5 |
| Reactor Startup | 8 |
| Draining the RCS with fuel in vessel | 8 |
| Integrated ESF Test | 8 |

**No additional CONTROL ROOM ACTIVITIES are scheduled during Plant maneuvers. Emergent failed equipment may necessitate the need for additional CONTRL ROOM ACTIVITIES and will only be allowed with the approval of the on-shift Shift Manager.**

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

**QID:** 0391 **Rev:** 0 **Rev Date:** 5/26/11 **Source:** New **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:** Yes **Taxonomy:** C

**Question:** **RO:** **SRO:** 98

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. The worker's Supervisor, 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. The worker's Supervisor, 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.

*5.3[3], continued*

- Lifetime greater than age = 1000 mrem onsite TEDE up to 2000 mrem TEDE for year.
- Declared Pregnant Woman TEDE = 50 mrem/month, 400 mrem/gestation period
- Minors TEDE – Minors are not allowed access to RCAs.
- Unmonitored Individuals TEDE = 50 mrem/month, 100 mrem/year
- Members of the General Public TEDE = 50 mrem

5.4 EXTENDING ADMINISTRATIVE DOSE GUIDELINES (ADG)

[1] Prior to dose extension, requesting supervisor should:

(a) Evaluate dose equalization in the department, **AND**

(b) Check other personnel qualifications to perform tasks, **AND**

(c) Check other means to reduce dose.

[2] Obtain verification of the worker's current year exposure prior to allowing a worker to exceed 2000 mrem TEDE for the year. Any of the following may be used for verification:

(a) An NRC Form 5 or equivalent provided by either the worker or the licensee(s) providing monitoring for each monitoring period, **OR**

(b) An NRC Form 4 or equivalent signed by the person, **OR**

(c) Electronic, telephone or facsimile transfer of exposure data provided by the licensee(s) providing the monitoring.

*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<br>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<br>Radiation Protection Manager approves<br>Plant General Manager approves |
| Greater than 4000 mrem and less than 4500 mrem per year for Radiation Workers.<br>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<br>Plant General Manager approves<br>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<br>Radiation Protection Manager approves |

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

**QID:** 0411 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** Repeat **Originator:** E-Plan

**TUOI:** ASLP-RO EPLAN **Objective:** 7 **Point Value:** 1

**Section:** 2 **Type:** Generic Knowledges and Abilities

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

**Description:** Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

**K/A Number:** 2.4.30 **CFR Reference:** 41.10 / 43.5 / 45.11

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Tier:** | 3 | **RO Imp:** | 2.7 | **RO Select:** | No | **Difficulty:** | 2 |
| **Group:** | G | **SRO Imp:** | 4.1 | **SRO Select:** | Yes | **Taxonomy:** | C |

**Question:** **RO:** **SRO:** 99

A fire was reported at 0844 in the vicinity of the Old Radwaste Building.
It is now 0920 and the fire is still burning.

Based on the above conditions what is the time requirement for notification to the NRC?

A. Notification to the NRC is required within 15 minutes of the declaration of an emergency class.

B. Notification to the NRC is required immediately following notification of the ADH and within 1 hour of the declaration of an emergency class.

C. Notification to the NRC is required immediately following declaration of an emergency class and notify the ADH within 1 hour.

D. Notification to the NRC is required within 4 hours of the declaration of an emergency class.

**Answer:**

B. Notification to the NRC is required immediately following notification of the ADH and within 1 hour of the declaration of an emergency class.

**Notes:**

Answer [B] is correct since this is the procedural requirement.
Answer [A], [C], [D] are incorrect, these are not in accordance with 1903.011.

**References:**

1903.011Y, Emergency Initial Notification Message, Chg 037

**History:**

Modified E-Plan exam bank QID#61 for use in 2001 SRO Exam.
Selected for use in 2002 SRO exam.
Selected for 2010 SRO exam
Repeated for 2011 SRO Exam.

## ACTIONS FOR INITIAL NOTIFICATION

The Arkansas Department of Health (ADH) **SHALL** be notified within **15 minutes** of an:

- Emergency Class Declaration
- Emergency Class Change (Upgrade or Downgrade)
- PAR Change

The Nuclear Regulatory Commission (NRC) **SHALL** be notified **immediately** following notification of the ADH and **SHALL NOT** exceed **1 hour** following the declaration of an emergency class.

ERDS must be started within 1 hour of the declaration of an **ALERT or higher** emergency class.

**NOTE**

- The material contained within the symbols (*) throughout this form is proprietary or private information.
- The Emergency Telephone Directory contains the emergency telephone numbers that you may need to complete this notification.
- Computer generated Form 1903.011-Y may be used for notifications. The computer generated form is not an identical copy to the hard copy form, but contains all necessary information.

**INSTRUCTIONS (circle/slash)**

**1.0** **Complete Initial Notification Message** in accordance with Step 1.1 Computerized Notification Method **OR** Step 1.2 Manual Notification Method. Computerized Notification Method preferred.

1.1. Computerized Notification Method

1.1.1. **IF** the Computerized Notification Method fails while performing notifications, **THEN** go to the "Manual Notification Method" Step 1.2.

1.1.2. Sign onto the computerized notification system computer using your Entergy logon ID and password. Control Room may use a generic ID and password.

1.1.3. Verify your computer is connected to a local or network printer in your area. [Start]→[Settings]→[Printers and Faxes]

1.1.4. On the desktop double click the "EP Notification" icon **OR** select [Start], [(All) Programs], [EP Notifications], [EP Notifications Version XXXX] to start notification program.

1.1.5. Enter the appropriate data into the data fields for the Initial Notification Message. Use the [Tab] key (preferred) or mouse to navigate through the form. Refer to Emergency Class Notification Instructions page 7 of this form as needed.

1.1.6. **WHEN** the data fields are populated, **THEN** press the [Create PDF only] button.

1.1.7. **IF** you receive an error message (i.e. "You have not correctly entered all the required data on Tab..."), **THEN** review the form and make corrections. Go to Step 1.1.6 above.

1.1.8. **WHEN** the PDF notification message is displayed on the computer screen, **THEN** print the message to a local printer.

1.1.9. Give the notification message to the person with ED&C for review and approval.

1.1.10. Once approval has been obtained, then close the PDF notification message on the computer screen by pressing [X] in upper right hand corner of PDF document.

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

**QID:** 0357 **Rev:** 0 **Rev Date:** 9/7/99 **Source:** Direct **Originator:** E Wentz

**TUOI:** ASLP-EP-A0082 **Objective:** 14 **Point Value:** 1

**Section:** 2 **Type:** Generic Knowledges and abilities

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

**Description:** Knowledge of the SRO's 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:** G **SRO Imp:** 4.5 **SRO Select:** Yes **Taxonomy:** K

**Question:** **RO:** **SRO:** 100

During a declared Alert emergency, the TSC Director may NOT assume responsibility for Emergency Direction and Control until:

A. The next shift's Shift Manager arrives for shift relief and receives a turnover.

B. The Operations Manager arrives in the Control Room and passes on responsibility for EDC.

C. The TSC Director receives a turnover from the SM and assumes responsibility for EDC.

D. The emergency is terminated by mutual agreement of the appropriate on and off site agencies.

**Answer:**

C. The TSC Director receives a turnover from the SM and assumes responsibility for EDC.

**Notes:**

"c" is the correct response in accordance with 1903.010.
"a" is incorrect, this will only serve to slow the turnover to the TSCD.
"b" is incorrect, the Ops Manager is not responsible for Emergency Direction and Control.
"d" is incorrect, if this were true, emergency direction and control would never be turned over to the TSCD.

**References:**

1903.064, Chg. 009

**History:**

Direct from ExamBank, QID# 2524
Used in 1999 exam.
Used on 2004 SRO Exam.
Selected for 2011 SRO Exam.

6.3 TURNOVER

6.3.1 Shift Manager

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 Director or TSC Director.

**[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 will normally be transferred from the Shift Manager to the EOF Director within 60 - 90 minutes of an Alert, or higher, emergency class. However, if the situation dictates, the TSC Director may relieve the Shift Manager of this responsibility.

E. The EOF Director 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 EOF Director.

G. The Shift Manager shall announce the turnover to the Initial Response Staff (IRS) personnel and report this turnover to the Support Manager located in the EOF.

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).