PWR Examination Outline

Facility: **Arkansas Nuclear One – Unit 1** Date of Exam: **3/5/2010**

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Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

RO Written Exam

Tier 1 Group 1

INSTRUCTIONS	CONTINGENCY ACTIONS
① **Manually Trip Rx.** A. Verify all rods inserted **AND** Reactor power dropping.	A. Perform the following: 1) **IF** Rx fails to trip, **THEN** depress CRD Power Supply Breaker Trip PBs on C03 (A-501 and B-631). a) **IF** A-501 or B-631 fails to trip, **THEN** manually insert rods at C03. **AND** Dispatch an operator to open CRD AC Power Supply breakers. 2) **IF** more than one rod fails to fully insert **OR** Rx power is **not** dropping, **THEN** perform Emergency Boration (RT 12). 3) **DO NOT** continue until the reactor is shutdown.

INSTRUCTIONS	CONTINGENCY ACTIONS
② **Manually trip Turbine.** A. Verify Turbine throttle and governor valves closed.	A. Perform the following: 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. • Turbine Trip Solenoid Power Available light off. • Breaker position indications on left side of C10 off. 2) **IF** SG press is < 900 psig, **THEN** perform the following: a) Actuate MSLI for affected SG(s) **AND** actuate EFW **AND** verify proper actuation and control (RT 6). b) Advise Shift Manager to implement Emergency Action Level Classification (1903.010). c) **GO TO 1202.003, "OVERCOOLING"** procedure.

INSTRUCTIONS	CONTINGENCY ACTIONS
③ **Check adequate SCM.**	3. Check elapsed time since loss of adequate SCM **AND** perform the following: A. **IF** ≤2 minutes have elapsed, **THEN** trip all RCPs. B. **IF** >2 minutes have elapsed, **THEN** leave currently running RCPs on. C. Advise Shift Manager to implement Emergency Action Level Classification (1903.010). D. **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure.
4. **Advise Shift Manager to implement Emergency Action Level Classification (1903.010).**	
5. **Reduce Letdown by closing Orifice Bypass (CV-1223).**	
6. **Open BWST Outlet to OP HPI pump (CV-1407 or 1408).**	
7. **IF Emergency Boration is NOT in progress, THEN adjust Pressurizer Level Control setpoint to 100".**	

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

QID: 0771 **Rev:** 0 **Rev Date:** 9/03/09 **Source:** New **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine operability and/or availability safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 41.7/43.5/45.13

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

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

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

Given:

- Pressurizer Spray fails open and the ATC operator was able to close the Spray valve and stopped the Reactor Coolant system pressure decrease.

- Annunciator alarm PZR HEATER GROUND FAULT (K09-E3) comes in.

- RCS pressure response abnormally slow with Pressurizer heaters energized.

- Maintenance is requested to perform Unit 1 Emergency Powered Pressurizer Heater Checkout (1307.009) to determine operability of vital powered pressurizer heaters

Which heaters groups are the vital powered pressurizer heaters, and which KW output of the vital powered heaters will satisfy the operability requirements of Technical Specification 3.4.9?

A. Group 1 proportional heaters, Group 2 proportional heaters, Group 4 heaters, 120 KW output.

B. Group 1 proportional heaters, Group 2 proportional heaters, Group 5 heaters, 135 KW output

C. Group 1 proportional heaters, Group 3 heaters, Group 5 heaters, 124 KW output

D. Group 2 proportional heaters, Group 4 heaters, Group 5 heaters, 128 KW output

Answer:

B. Group 1 proportional heaters, Group 2 proportional heaters, Group 5 heaters, 135 KW output

Notes:

A. is incorrect wrong groups of heaters and KW output to low
B. is the correct answer correct groups of heaters and KW meets operability requirements of TS 3.4.9
C. is incorrect wrong groups of heaters and KW output to low
D. is incorrect wrong groups of heaters and KW meets operability requirements of TS 3.4.9

References:

1203.015 change 016
T.S. 3.4.9 amendment # 215

History:

New for the RO/SRO 2010 exam

SECTION 3 -- INOPERATIVE PRESSURIZER HEATER(S)

ENTRY CONDITIONS

One or more of the following:

- Pressurizer heaters do not energize in AUTO at proper setpoint:
    - Banks 1 and 2 full on: 2135 psig
    - Bank 3 on: 2135 psig
    - Bank 4 on: 2120 psig
    - Bank 5 on: 2105 psig
- RC pressure response abnormally slow with Pressurizer heaters energized
- Annunciator alarm PZR HEATER GROUND FAULT (K09-E3)

SECTION 3 -- INOPERATIVE PRESSURIZER HEATER(S)

NOTE

- In order to satisfy the requirements of TS 3.4.9, Group 1 Proportional heaters, Group 2 Proportional heaters and Group 5 vital powered heaters shall be operable. This ensures that ≥126 KW (nominal) is available in the event of a loss of offsite power concurrent with a single failure of one EDG.
- Group 5 vital powered heaters shall be capable of manual transfer via B55/56. In the event B55/56 can not be manually transferred, then one train of Pressurizer Heaters is considered inoperable.
- If Group 5 vital powered heaters are declared inoperable, then both trains of Pressurizer Heaters are considered inoperable. Per Licensing, TS 3.0.3 is NOT applicable. Entry into TS 3.4.9 Condition C is required for inoperability of both trains of Pressurizer Heaters.
- Unit 1 Emergency-Powered Pressurizer Heater Checkout (1307.009) is used to determine operability of vital powered Pressurizer Heaters.

5. **IF any Pressurizer heater is declared inoperable,**
   THEN perform the following:

   - Initiate action to repair heater.
   - Initiate a Condition Report.
   - Refer to TS 3.4.9 and TRM 3.4.9.
   - Refer to "RCS Pressure, Temperature and Flow DNB Surveillance Limits" of the ANO1 COLR (TS 3.4.1).
   - Notify Ops Manager.

END

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≥ 45 inches and ≤ 320 inches; and

b. A minimum of 126 kW of Engineered Safeguards (ES) bus powered pressurizer heaters OPERABLE.

--------------------------------------NOTE--------------------------------------
OPERABILITY requirements on pressurizer heaters do not apply in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 262°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limits.	A.1 Restore level to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 4 with RCS temperature ≤ 262°F.	24 hours
C. Capacity of ES bus powered pressurizer heaters less than limit.	C.1 Restore pressurizer heater capacity.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 4.	12 hours

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

QID: 0772 **Rev:** 0 **Rev Date:** 9/3/09 **Source:** New **Originator:** S.Pullin

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to operate and monitor the following as they apply to a small break LOCA: RB sump level

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

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 4

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

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

Given:
Small break LOCA has occurred.
The Reactor building sump is filling at a rate of 2%/minute.
Reactor Building sump level is 44%

What is the RCS leak rate and with the leak size remaining steady how long can the Reactor building sump be used for an accurate leak rate calculation?

A. RCS leak rate approximately 91 gpm, and the RB sump level can be used for 3 minutes.

B. RCS leak rate approximately 80 gpm, and the RB sump level can be used for 16 minutes.

C. RCS leak rate approximately 72 gpm, and the RB sump level can be used for 28 minutes.

D. RCS leak rate approximately 45 gpm, and the RB sump level can be used for 3 minutes.

Answer:

A. RCS leak rate approximately 91 gpm, and the RB sump level can be used for 3 minutes.

Notes:

A. is the correct answer due to sump is 45.4 gallons per/ % and the RB sump can only be used for leak rate determination up to 50% level after that level you can not get an accurate leak rate due to volume uncertainties
B. is incorrect due to the wrong leakrate and time.
C. is incorrect due to the wrong leakrate and time.
D. is incorrect due to the wrong leakrate.

References:

STM 1-08 Rev. 14

History:

New for the RO/SRO 2010 exam

2.9.2.1 RB Sump Isolation Valves

Each RB sump ECCS suction line is equipped with two isolation valves. The four ECCS isolation valves are 14-inch motor operated gate valves. The inside RB isolation valves are manufactured by Anchor Darling. They are designated as CV-1414 and CV-1415 with a design pressure and temperature of 100 psig and 150°F. Both inside isolation valves are located in the RB sump. The outside isolation valves are manufactured by Aloyco. They are designated as CV-1405 and CV-1406 with a design pressure and temperature of 75 psig and 300°F.

Isolation valves CV-1414 and CV-1405 provide the suction line from the RB Sump to P-34A/P-35A. CV-1405 is located in the "A" DH vault. They are controlled by hand-switches located on panel C-18.

Isolation valves CV-1415 and CV-1406 provide the suction line from the RB Sump to P-34B/P-35B. CV-1406 is located in the "B" DH vault. They are controlled by hand-switches located on panel C-16.

The inside containment isolation valves (CV-1414 & CV-1415) are normally left open. The outside isolation valves (CV-1405 & CV-1406) are normally closed. All four RB sump ECCS isolation valves are required to be operable whenever RB integrity is required. They can be either manually or remote-manually operable. The four RB sump isolation valves do not receive an ESAS signal to open or close since their safety function supports ECCS.

RB Sump isolation valve power supply and associated handswitch for each are provided in the following table.

RB Sump supply to P-34A & P-35A		
CV-1414 (RB Inside Isol)	HS-1414 / C-18	B-51112
CV-1405 (RB Outside Isol)	HS-1405 / C-18	B-51113
RB Sump supply to P-34B & P-35B		
CV-1415 (RB Inside Isol)	HS-1415 / C-16	B-6163
CV-1406 (RB Outside Isol)	HS-1406 / C-16	B-6166

2.9.2.2 RB Sump Level Instrumentation

(Refer to Table 8.01)

RB sump level is independently measured by level transmitter LT-1405 and level sensor LE-1405B. Each level instrument is mounted in the RB sump and has a measuring range of 0 to 56 inches, which correlates to 0 to 100%. The instruments measure sump level from 6 inches (0%) above the sump bottom to the sump maximum level of 46 inches (71.4%). Each percent from 0% to 50% of level indication equals a maximum of 45.4 gallons. Sump level indication for determining RCS leak rate is limited to 50% due to volume uncertainty above that level. Each instruments range extends 16 inches above the RB floor.

LT-1405's signal is used for indiction and an annunciator alarm function. RB sump level can be read on panel C-14 from LI-1405 or SPDS (L1405). LT-1405 has a sensitivity of ¼ inch or 0.45% of

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

QID: 0198 **Rev:** 1 **Rev Date:** 8/9/05 **Source:** Direct **Originator:** J. Haynes

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the interrelations between the Large Break LOCA and the following: Pumps.

K/A Number: EK2.02 **CFR Reference:** 41.7/45.7

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

Group: 1 **SRO Imp:** 2.7 **SRO Select:** Yes **Taxonomy:** K

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

Given:
- A large break LOCA has occurred.
- Offsite power has been lost.

Why must Reactor Building Spray flow be throttled to 1050-1200 gpm prior to transferring to Reactor Building sump suction?

A. To ensure adequate NPSH for ECCS pumps.

B. To prevent pump runout on the Spray pumps.

C. To lower load on the EDGs.

D. To limit corrosion of reactor building equipment.

Answer:

A. To ensure adequate NPSH for ECCS pumps.

Notes:

(a.) is correct.
(b.) is incorrect. The spray pumps are designed for the full flow that is achieved during ES conditions.
(c.) is incorrect. The EDGs are designed to handle the load of the spray pumps at full flow.
(d.) is incorrect. The design of RB equipment includes allowances for corrosion due to RB spray.

References:

1202.012, Chg. 008

History:

Developed for use in 98 RO Re-exam.
Selected for use in 2005 RO exam, replacement question.
Selected for the RO/SRO 2010 exam.

WARNING

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

CAUTION

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

NOTE

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

15. Shift to RB sump suction:

A. Verify both LPI pumps running (P34A and B).

1) **IF either** LPI pump is unavailable, **THEN** stop associated HPI pump.

2) Verify LPI Room Coolers running:

P-34A Room	P-34B Room
VUC1A or B	VUC1C or D

3) Verify both LPI Block valves fully open (CV-1400 and 1401).

B. Verify Letdown isolated by either:

Letdown Coolers Outlet (CV-1221)
OR
Letdown Cooler Outlets (CV-1214 and 1216).

(15. CONTINUED ON NEXT PAGE)

15. (Continued).

C. __IF__ HPI is in service, __THEN__ perform the following:

1) __IF__ **__either__** of the following __sets__ of conditions is satisfied, __THEN__ terminate HPI as follows:

__All__ of these conditions satisfied:

- CET SCM is adequate
- **__Any__** LPI flow exists
- HPI throttled to ≤ 110 gpm/pump
- RCS press and temp are **__not__** rising

__OR__

__Both__ of these conditions satisfied:

- CETs **__do not__** indicate superheated
- LPI flow meets the following criteria:

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

a) Start AUX Lube Oil pumps for running HPI pumps.

b) Stop running HPI pumps.

c) Close all HPI Block valves.

d) Close RCP Seal INJ Block (CV-1206).

2) __IF__ HPI termination criteria are **__not__** met,
__THEN__ verify __both__ Decay Heat Supply to Makeup Pump Suctions open (CV-1276 and 1277).

a) __IF__ CV-1276 or 1277 fails to open, __THEN__ stop associated HPI pump.

D. __IF__ RB Spray has actuated, __THEN__ perform the following:

1) Verify RB Spray flow throttled to maintain 1050 to 1200 gpm per train.

2) __IF__ NaOH Tank T10 level is ≤ 16',
__THEN__ close RB Spray NaOH Addition T-10 Outlets (CV-1616 and 1617).

E. Verify RB Sump Outlets open (CV-1414 and 1415).

F. Align LPI to take suction from RB sump as follows:

1) Open RB Sump Outlets (CV-1405 and 1406).

a) __IF__ CV-1405 or 1406 fails to open,
__THEN__ stop associated LPI, HPI and RB Spray pumps.

(15. CONTINUED ON NEXT PAGE)

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

QID: 0609 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** Direct **Originator:** Cork/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:** K

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

Which of the following indications would require stopping a Reactor Coolant Pump?

A. Seal cavity pressures oscillating at 600 psi peak to peak

B. Seal bleedoff temperature 160°F

C. Seal beedoff temperature 60°F above 1st stage seal temperature

D. Failure of one stage as indicated by zero stage DP

Answer:

C. Seal beedoff temperature 60°F above 1st stage seal temperature

Notes:

Answer "C" is correct, this exceeds 40°F delta-T specified in section 1 of 1203.031.
Answers "A", "B" and "D" just indicate a need for increased monitoring frequency of an RCP.

References:

1203.031, Chg. 018

History:

New for 2005 RO exam, replacement question.
Selected for the RO/SRO 2010 exam.

SECTION 1
SEAL DEGRADATION

NOTE

- RCP seal stage ΔP is determined as follows:
  - 1st stage ΔP = system pressure - lower seal cavity press.
  - 2nd stage ΔP = lower seal cavity pressure - upper seal cavity press.
  - 3rd stage ΔP = upper seal cavity pressure - RB atmospheric press.
- Third stage seal leakage by design is 0 to 0.08 gpm. Third stage leakage in excess of design will affect upper seal cavity pressure and seal bleed off flow.

4. **Determine if any of the following conditions exist:**

- RCP seal cavity pressure oscillations exceed 800 psi peak-to-peak
- ΔP across any stage exceeds 2/3 of system pressure
- A loss of seal injection
  AND ≥2.5 gpm total seal outflow, including seal bleedoff
  (excluding shaft sleeve leakage)

- 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.

A. **IF** any of the above conditions exist,
THEN reduce reactor power to within the capacity of the unaffected RCP combination, using Rapid Plant Shutdown (1203.045)

B. **WHEN** power reduction is complete,
THEN stop the affected RCP(s) per Reactor Coolant Pump Operation (1103.006).

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

(continued)

ATTACHMENT A

Page 1 of 1

RCP PARAMETERS

Seal Degradation/Seal Failure

1. **ANY of the following are criteria to SECURE the affected RCP per Section 1 Seal Degradation**
   - RCP seal cavity pressure oscillations exceed 800 psi peak-to-peak
   - ∆P across any stage exceeds 2/3 of system pressure on a running RCP **OR** exceeds 80% of system pressure on an idle RCP.
   - ≥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.
2. **ANY of the following are criteria to TRIP the affected RCP per Section 2 Seal Failure**
   - ≥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.
   - ∆P across a single stage equal to RCS press, with seal bleed off flow established.

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 or D **MOTOR** vibration; more than one channel >20 mils after startup stabilization
   - P-32A orC **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: 0773 **Rev:** 0 **Rev Date:** 9/3/2009 **Source:** New **Originator:** S. Pullin

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

Given:

- The RCS is drained to 374 feet for seal replacement.
- RCS Temperature 140 F.
- RCS pressure is 5 psig.
- RCS leakage measured at 50 gpm.
- "A" Decay Heat Pump has been stopped and CV-1050 Decay Heat Suction Valve has been closed per 1203.028, Loss of Decay Heat Removal AOP.

Per 1203.028, Loss of Decay Heat Removal AOP, what is the preferred makeup flow path for these conditions?

A. Gravity feed from the BWST.

B. Low Pressure Injection.

C. Spent Fuel Cooling Pump P-40A.

D. High Pressure Injection.

Answer:

B. Low Pressure Injection.

Notes:

A. Gravity feed from the BWST is incorrect because the RCS is pressurized
B. Low Pressure Injection is correct.
C. Spent Fuel Cooling Pump P-40A is incorrect because it is the least preferred method allowed.
D. High Pressure Injection is incorrect because HPI could overpressurize the RCS.

References:

1203.028 Change 21

History:

New for the RO/SRO 2010 exam.

ATTACHMENT H

{1 and 3}

RCS MAKEUP METHODS

1. Consider the existing plant conditions listed below:

- **RCS press**
- **RCS level**
- **RCS open or intact**
- **Leak rate**

- **DH Removal system isolated or unisolated**
- **Available MU flow rate**
- **Available time**
- **Available equipment**

NOTE

- The six RCS makeup methods are listed in order of preference.
- Each method is effective only as long as the limitations listed are met.

2. Select a makeup method below based upon existing plant conditions AND perform the applicable attachment:

METHOD	REQUIRED RCS PRESS	OTHER LIMITATIONS	APPLICABLE ATTACHMENT
Gravity Feed from BWST	0 psig	Requires BWST level >21 ft	A
LPI	<200 psig	Requires operable LPI pump	B
RB Spray Pump	<220 psig	Available flow rate 1500 gpm	C
Borated Water Recirc Pump (P-66)	<100 psig	Available flow rate 180 gpm	D
Spent Fuel Cooling Pump (P-40A)	<45 psig	Available flow rate 1000 gpm	E
HPI	N/A	Can over-pressurize RCS	F

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

QID: 0395 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Direct **Originator:** D.Slusher

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

The plant is shutdown and cooled down.
RCS pressure is 220 psig.
I&C is performing calibration checks on "A" RPS channel.

Why will I&C request the Pzr Control Pressure Selector, HS-1038, be placed in the "Y" position?

A. To allow remote indications to be checked during calibration.

B. To prevent the ERV opening, causing a rapid depressurization of the RCS.

C. To maintain pressurizer heaters off during testing.

D. To allow the ERV low setpoint to be checked.

Answer:

B. To prevent the ERV opening, causing a rapid depressurization of the RCS.

Notes:

Answer [b] is correct, testing will cause ERV to open since the LTOP setpoint is in effect.
Answer [a] is incorrect, the selector switch does not select between local and remote indications.
Answer [c] is incorrect, PZR heaters are in manual control for cooldown.
Answer [d] is incorrect, I&C verifies the setpoint, it is undesirable to operate ERV at this point.

References:

1105.006, Chg. 010
STM 1-69, Rev. 13

History:

Direct from regular exambank QID#5545 for 2001 RO/SRO Exam.
Selected for 2005 RO exam, replacement question.
Selected for the RO/SRO 2010 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).

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.

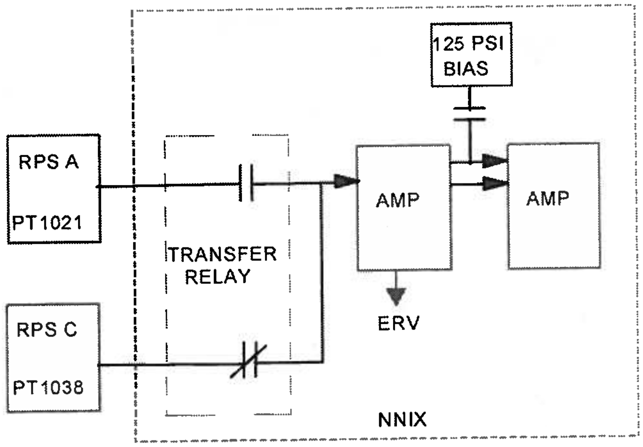
Pressure transmitters PT-1020, PT-1022, and PT-1040 provide input to A, B, and C ESAS analog channels, respectively. The pressure transmitters that supply ESAS are Rosemount differential capacitance detectors. ESAS analog channel A supplies indication on C-166 (Dasey Panel). The range of the indication 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-1-14. 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.

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

QID: 0582 **Rev:** 0 **Rev Date:** 9/3/2009 **Source:** New **Originator:** S. Pullin

TUOI: A1LP-RO-EFIC **Objective:** 26 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to determine or interpret the following as they apply to the ATWS: Reactor trip alarm.

K/A Number: EA2.02 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 4.2 **RO Select:** Yes **Difficulty:** 4

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

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

Given:
- Plant startup is in progress.
- Reactor power is 20%.
- Total Main FW flow is 1.6 x e 6 lbm/hr.
- Generator load is ~180 Mwe.

Subsequently the following indications are observed:
- Reactor power dropping rapidly,
- Regulating groups In-Limit lights ON,
- Safety groups Out-Limit lights ON.
- Turbine Generator Lockout alarm is in,
- EFW actuated on both trains.

Which of the following annunciators, and reasons for the annunciator, could cause the above indications?

A. K08-A3 "REACTOR TRIP" because the in-service MFW pump has tripped causing a reactor trip with power >9%.

B. K08-F2 "CRD MOTOR POWER FAILURE" because a loss of transformer X8 has tripped the Regulating Groups.

C. K08-A5 "AMSAC TRIP" because both Gamma Metrics NI-501 and NI-502 were not calibrated within 3% of heat balance as required.

D. K08-A3 "REACTOR TRIP" because the RPS anticipatory trip for Turbine has not been reset.

Answer:

C. K08-A5 "AMSAC TRIP" because both Gamma Metrics NI-501 and NI-502 were not calibrated within 3% of heat balance as required.

Notes:

A. Is incorrect because a reactor trip would have caused all of the control rods to insert not just the regulating groups.
B. Is incorrect because a loss of X8 would only lose one of the AC power supplies to the rods and no rods would trip.
C. Is correct, if gamma metrics indicated >45% with the given feedwater flow, an AMSAC Trip would be initiated.
D. Is incorrect because a reactor trip would have caused all of the control rods to insert not just the regulating groups.

References:

1102.002 Change 082
STM 1-59 Rev. 1

History:

New for the RO/SRO 2010 exam.

18.20 Open moisture separator-reheater 2nd Stage 1-Inch Warmup Valves on C11.

- CV-6808
- CV-6837
- CV-6814
- CV-6842

18.21 Adjust lo-load limit setpoint to 12.7% (127 MWe).

18.22 Verify that NI-501 (SPDS point NI1LP) and NI-502 (SPDS point NI2LP) Gamma Metrics Linear Power Instruments are within ±3% of heat balance power.

18.22.1 IF instruments are outside the allowable differences, THEN have I&C Dept calibrate instruments per Source Range Channels Test (1304.055).

18.23 IF available, THEN using Plant Computer UTILITY screen, PLANT MODE ASSIGNMENT (PMA), select "Power Ops" mode to enable computer alarms for the power ops mode.

18.24 Verify plant at ~25% power.

18.24.1 Adjust Low Load Limit setpoint to 20% (200 MWe).

18.24.2 WHEN gland steam condenser ΔP is >2.8 (per PDIC-2905 on C02), THEN place CV-2906 handswitch in AUTO.

Foxboro isolators NY-501G and NY-502G. NI-501 inputs reactor power to DROPS channel 1 and NI-502 inputs to channel 2.

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

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

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

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

(Refer to Figure 59.04)

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

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

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

2.4.1 DROPS Testing

Periodic testing of DROPS is required and will occur during normal plant operation. Due to DROPS interaction with other systems it is important to recognize equipment failures that can potentially cause adverse affects during testing. When performing a DROPS channel DSS or AMSAC test the "trip" function will cause the associated actuation features relays or contacts to change state. For example: When performing DROPS channel 1 DSS subsystem test the "A" or main power to CRD groups 5, 6, 7 and the Aux. bus gate drives contacts will open along with energizing one of the two relays in the turbine trip circuitry. When problems exist in either the CRD, EFIC, Gamma Metrics or the turbine trip circuitry DROPS testing should not be performed.

DROPS testing is performed by utilizing the common test jacks and test enable push-buttons located on the control module. The system test will remove the process inputs and internally simulate the inputs. The setpoints and the trip signals can be monitored at the front panel for verification during testing. Since the DROPS is a 2 out of 2 logic system the channel not being tested and the trip feature not being tested will be placed bypass. Placing the channel in bypass requires the DSS and AMSAC bypass switches to be placed in the bypass position. This will disable the associated trip function and prevent system actuation during testing.

When the associated DROPS channel being tested is placed in test the DSS / AMSAC in test annunciator will alarm alerting the operator of this condition.

For additional information refer back to section 2.1.5 Control Module and section covering the DSS and AMSAC test enable buttons.

2.5 Annunciators

This section will cover the annunciators associated with the DROPS system. For additional information refer to 1203.013G Annunciator Corrective Actions.

The annunciators associated with DROPS are systematically arranged and grouped in panel K08 located on panel C-13 in the control room.

The following alarms are provided for the DROPS system:

DSS Trip: K08-A5 will alarm when RCS pressure sensed by PT-1041 and PT-1042 indicates > 2430 psig and both DROPS channels have received an DSS trip confirm signal.

AMSAC Trip: K08-B5 will alarm when Reactor power is greater than 45% and MFW flow indicates less than 15% of total flow (.9 x 10^6 lbm/hr) and both DROPS channels have received an AMSAC trip confirm signal.

DSS / AMSAC Trouble: K08-C5 will alarm when an abnormal condition exist associated with the subsystems of DROPS. The conditions that can cause the DSS or AMSAC trouble alarm are listed below.

Figures And Diagrams/Tables Etc.

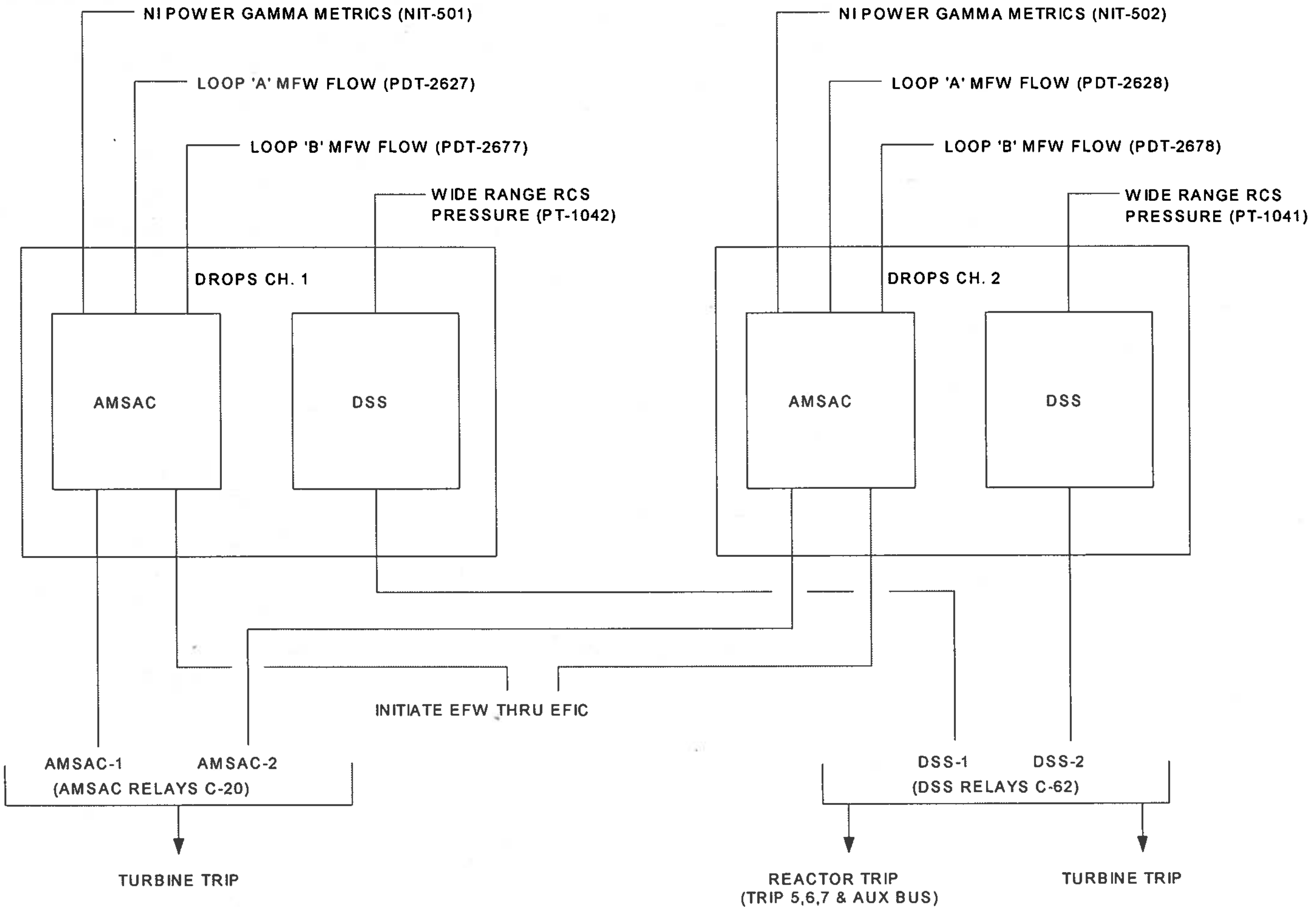


FIGURE 59.01: BLOCK DIAGRAM OF DROPS

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of EOP mitigation strategies

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

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

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

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

After a reactor trip, the following indications are observed:

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

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

A. Initiate HPI per RT-2

B. Cooldown and isolate the "B" SG

C. Cooldown and isolate the "A" SG

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

Answer:

C. Cooldown and isolate the "A" SG

Notes:

Answer [c] is correct, the SG level parameters indicate a rupture on the "A" SG and a cooldown should be commenced to reduce RCS temperature to <500 F to minimize the possibility of lifting a secondary safety on the "A" SG.
[a] is incorrect, the leak size is about 30 gpm (30.86 gal/in. x 5 in./5 min.). This is within the capacity of normal makeup.
[b] is incorrect, a cooldown and isolation is required but not on this SG.
[d] is incorrect, a rapid cooldown at this rate is not required until overfilling of ruptured SG is imminent.

References:

1202.006, Chg. 11

History:

Created for 2001 RO/SRO Exam.
Selected for 2002 RO/SRO exam.
Selected for 2005 Jon Gray RO re-exam.
Selected for 2010 RO/SRO Exam

INSTRUCTIONS

CONTINGENCY ACTIONS

5. Begin controlled plant shutdown at ≥5% per minute.

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

- OTSG N-16 monitors:

SG A	SG B
RI-2691	RI-2692

- SGTR display on SPDS.
- Plant Monitoring System Alarms.
- Steam Line High Range RAD Monitors (may be inconclusive due to insufficient shielding between MS lines):

SG A	SG B
RI-2682	RI-2681

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

7. Perform Control of Secondary System Contamination (1203.014) in conjunction with this procedure.

8. <u>WHEN</u> bad SG is known,
<u>THEN</u> place bad SG EFW pump Turbine (K3) Steam Supply valve in MANUAL <u>AND</u> close:

SG A	SG B
CV-2667	CV-2617

INSTRUCTIONS

18. IF emergency cooldown rate is not required
OR
RCS T-hot is ≤500°F,
THEN establish RCS cooldown rate of ≤100°F/hr as follows:

A. For good SG, place TURB BYP valves in HAND
AND
adjust to maintain cooldown rate ≤100°F/hr.

B. **When** RCS press is <1700 psig,
THEN bypass ESAS.

C. **IF** only one SG is bad,
THEN steam bad SG only as necessary to maintain:

- MSSVs closed
- SG press:
  - ≤990 psig if using TURB BYP valves
  - ≤1040 psig if using ATM Dump Control system
- SG level ≤410".
- SG Tube-to Shell ΔT ≤100°F (tubes colder).
- Desired cooldown rate if good SG TBV or ADV is full open.

CONTINGENCY ACTIONS

A. **IF** TURB BYP valves are **not** available,
THEN operate ATM Dump Control System for good SG in HAND to maintain cooldown rate ≤100°F/hr.

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

1) **IF** both SGs are bad,
THEN steam both SGs.

C. **IF** both SGs are bad,
THEN steam both SGs.

QID: 0551 **Rev:** 0 **Rev Date:** 3-30-05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-EOP03 **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: Consequence of PTS.

K/A Number: AK1.01 **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:** 10 **SRO:** 10

Which of the following would invoke Pressurized Thermal Shock (PTS) limits during a Steam Line Rupture?

A. HPI on with all RCPs off

B. RCS cool down rate 105°F/hr with Tcold 360°F

C. RCS cool down rate 55°F/hr with Tcold 310°F

D. SG Tube to shell DT 150°F tubes colder

Answer:

A. HPI on with all RCPs off

Notes:

Answer "A" is correct per RT-14.
Answer "B" is incorrect, cooldown rate is >100°F/hr but Tcold >355°F.
Answer "C" is incorrect, cooldown rate is >50°F/hr but Tcold >300°F.
Answer "D" is incorrect, this is a limit but not a PTS limit.

References:

1202.012, Chg. 8

History:

New for 2005 RO exam.
Selected for 2010 RO/SRO exam

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.

14. Control RCS press within limits of Figure 3.

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

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

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

1) Throttle makeup flow.

2) **IF** SCM is adequate, **THEN** throttle HPI flow by performing the following:

a.) Verify both HPI RECIRC valves (CV-1300 and 1301) open.

b.) Throttle HPI.

3) Raise Letdown flow.

a) **IF** ESAS has actuated,
THEN unless fuel damage or RCS to ICW leak is suspected,
restore Letdown flow (RT 13).

4) Verify ERV Isolation open (CV-1000) **AND** cycle ERV (PSV-1000).

(14. CONTINUED ON NEXT PAGE)

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

QID: 0774 **Rev:** 0 **Rev Date:** 9/4/2009 **Source:** Modified **Originator:** S Pullin

TUOI: A1LP-RO-EOP02 **Objective:** 8 **Point Value:** 1

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

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

Description: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): AFW adjustments needed to maintain proper T-ave and S/G level.

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

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

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

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

A reactor trip has occurred from 100% power due to a loss of both MFW Pumps.

The following conditions have existed for three minutes:

- CET temperature = 580 degrees F.
- RCS pressure = 1600 psig.

Which of the following operator actions will be performed?

A. Trip all running RCPs.

B. Verify EFW flow to each Steam Generator is ~320 gpm.

C. Verify Reflux Boiling setpoint is selected on both EFIC trains.

D. Verify EFW in hand and flow to each Steam Generator is ~570 gpm.

Answer:

C. Verify Reflux Boiling setpoint is selected on both EFIC trains.

Notes:

A. Incorrect, this would be done for loss of subcooling margin but only if <2 minutes had expired without tripping the RCPs.
B. Incorrect this is done for loss of subcooling margin but only if EFW flow is less than adequate and the value given is similar
but less than the minimum flow rate of greater than or equal to 340 gpm.
C. Correct since subcooling margin is lost and the Reflux Boiling setpoint is required to be selected in this situation.
D. Incorrect, this would be done if only one SG was available.

References:

1202.012 Change 008, RT-5

History:

Modified from QID 368.
Selected for the RO/SRO 2010 exam.

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

QID: 0368 **Rev:** 1 **Rev Date:** 8/8/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-EOP02 **Objective:** 8 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the operational implications of the following concepts as they apply to the small 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:** 4.2 **RO Select:** No **Difficulty:** 3

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

Question: RO: ☐ SRO: ☐

A reactor trip has occurred from 100% power.

The following conditions have existed for three minutes:

Parent

- RCS temperature = 590 degrees F.
- RCS pressure = 1700 psig.

Which of the following operator actions will be performed?

A. Trip all running RCPs.

B. Verify EFW flow to each Steam Generator is ~320 gpm.

C. Verify Reflux Boiling setpoint is selected on both EFIC trains.

D. Go to Overheating EOP, 1202.004.

Answer:

C. Verify Reflux Boiling setpoint is selected on both EFIC trains.

Notes:

Answer [c] is correct since subcooling margin is lost and the Reflux Boiling setpoint is required to be selected in this situation.
Answer [a] is incorrect, this would be done for loss of subcooling margin but only if <2 minutes had expired without tripping the RCPs.
Answer [b] is incorrect, this is done for loss of subcooling margin but only if EFW flow is less than adequate and the value given is similar but less than the minimum flow rate of greater than or equal to 340 gpm.
Answer [d] is incorrect, this would not be done since the entry conditions for Overheating have not been met and loss of subcooling margin .

References:

1202.012, Chg. 004-03-0, RT-5

History:

Direct from regular exambank QID 3030.
Selected for use in 2002 SRO exam.
Modified for use in 2005 RO exam, replacement question.

5. **Verify proper EFW actuation and control:**

A. Verify EFW actuation indicated on Bus 1 and 2 of both Trains A and B on C09.

B. Verify at least one EFW pump (P7A or B) 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

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

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

1) Select Reflux Boiling setpoint.

NOTE

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

2) Verify EFW CNTRL valves operate to establish and maintain SG levels 370 to 410”.

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

(1) **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”.

(2) **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”.

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

(5. CONTINUED ON NEXT PAGE)

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

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

Section: 4.1 **Type:** Generic EPEs

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

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

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

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

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

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

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

ELP-NLO-ELEC1

History:

Developed by NRC.
Used on 2004 RO/SRO Exam.
Selected for 2005 Jon Gray RO re-exam.
Selected for the RO/SRO 2010 exam.

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

Group: 1 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** K

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

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

1305.006, Chg. 030

History:

Created for 2001 RO/SRO Exam.
Selected for 2005 Jon Gray RO re-exam.
Selected for the 2010 RO/SRO exam

SUPPLEMENT 1

TEST QUANTITY	INSTRUMENT	MEASURED VALUES	NORMAL RANGE	LIMITING RANGE FOR OPERABILITY	IS DATA WITHIN LIMITING RANGE? (CIRCLE YES OR NO)
Loop II SW Control logic test	N/A	N/A	N/A	Attachment 4 satisfactory	YES NO
ES Even Channels Control logic test	N/A	N/A	N/A	Attachment 6 satisfactory	YES NO
DG2 loaded	Clock	Min	N/A	≥1 hour @2600-2750 KW AND temperatures stable	YES NO
DG1 (CH 2)	DAS Data from ESAS Actuation	Sec.	N/A	At rated speed and voltage in ≤15 sec.	YES NO
DG2 (CH 2)		Sec.	N/A		YES NO
Even channels ES load sequencing	DAS Data from Loss of Power	N/A	N/A	Shed on loss of power	YES NO
		N/A	N/A	Resequence on buses	YES NO
		HPI pump Sec.	N/A	4.7-5.3 sec	YES NO
		LPI pump Sec.	N/A	9.6-10.4 sec	YES NO
		SW pump Sec.	N/A	14.4-15.6 sec	YES NO
		RBS pump Sec.	N/A	33.6-36.4 sec	YES NO
		VSF-1C Sec.	N/A	48-52 sec	YES NO
		VSF-1D Sec.	N/A	48-52 sec	YES NO

5.2 IF "No" is circled in any space above, THEN declare the affected component inoperable, immediately notify the Shift Manager, write a Condition Report and initiate corrective action.

PERFORMED BY ______________________ OPERATOR DATE/TIME __________

______________________ OPERATOR DATE/TIME __________

______________________ OPERATOR DATE/TIME __________

______________________ OPERATOR DATE/TIME __________

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

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

What would be the effect on the SG pressure and level instruments on C03, if a loss of the RS-1 bus occurred?

A. Instrument power would automatically be transferred to YO-2 by the ABT, SG pressure and level instruments would not be effected.

B. The NNI-X S1 and S2 switches would open and SASS would transfer to NNI-Y, SG pressure and level instruments would fail to mid scale.

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

D. Instrument power would automatically be transferred to YO-1 by the ABT, SG pressure and level instruments would fail low.

Answer:

A. Instrument power would automatically be transferred to YO-2 by the ABT, SG pressure and level instruments would not be effected.

Notes:

"A" is correct, a loss of RS-1 would simply cause NNI-X to be powered from YO-2, -24vDC logic power is auctioneered and instrument power would transfer by the ABT within 0.5 seconds no effect on instruments.
"B" is incorrect, it would take a loss of both RS-1 and YO-2 to cause the S1 and S2 switches to open.
"C" is incorrect, although SG pressure does input to the BTU limit alarm, the NNI-X SG pressure indicators would not be failed due to the power transfer to YO-2.
"D" is incorrect, the alternate power to NNI-Y is from YO-1.

References:

STM 1-69, Rev. 13

History:

New for 2005 RO re-exam.
Selected for the 2010 RO/SRO exam

3.1.3 Controls and indications

The following controls and indications are associated with the SASS system.

SASS module front panel controls and indications:

- Reset switch

  Switch is located under the front cover and may be used to reset (initialize the computer and start the data gathering).

- Auto push-button

  When depressed, the associated SASS channel will return to automatic if the NNIX and NNIY signals are within the mismatch setpoint.

- Test toggle switch

  The test toggle switch inserts a +5 VDC signal into the signal conditioning board of the associated channel. The SASS computer will see the signal step change and generate a mismatch and trip indication. The SASS transfer function is blocked when the toggle switch is taken to either the X or Y position.

- Auto Indicator

  Green LED indicator shows the SASS system is capable of initiating a signal transfer. Indicator should normally be on.

- Mismatch Indicator

  Amber LED indicator lights when the computer detects a mismatch (NNIX and NNIY signals exceeds the mismatch setpoint).

- Trip X Indicator

  Red LED indicator lights when the NNIX signal has failed. The SASS channel should have initiated a transfer to the NNIY channel.

- Trip Y Indicator

  Red LED indicator lights when the NNIY signal has failed. The SASS channel should have initiated a transfer to the NNIX channel (for Tc only).

3.2 NNI Power Supplies

The NNI power supplies provide the power necessary for the NNI system operation. A 120-volt AC bus supplies power to relays, transmitters, and generally components not inside the NNI cabinets. ±24 volt DC buses supply power to the internal electronic circuits which process the signals. NNIX is supplied from RS-1 and Y-02. NNIY is supplied from RS-4 and Y-01.

The vital (RS) and instrument (Y) buses each supply one positive 24 volt DC and one negative 24 volt DC power supply through circuit breakers S-1 and S-2. S-1 and S-2 are located in the NNI cabinets. Diodes auctioneer the outputs of the two positive 24 volt DC and two negative 24 volt DC power supplies. Therefore, a loss of either of the power sources A (RS or instrument power) will not cause a loss of the ±24 volt DC power.

The power supply monitor monitors the output of the ±24-volt DC power supplies. The power supply monitor will initiate an annunciator alarm when power supply voltage is abnormal. The power supply monitor will also cause the S-1 and S-2 switches to open when either the positive or negative 24 volt DC power is lost (loss > >.5 seconds). The S-1 and S-2 switches also provide overcurrent protection for the power supply and NNI components.

An automatic bus transfer switch (ABT) is fed from the vital and instrument buses (vital is the normal source). The ABT will 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.

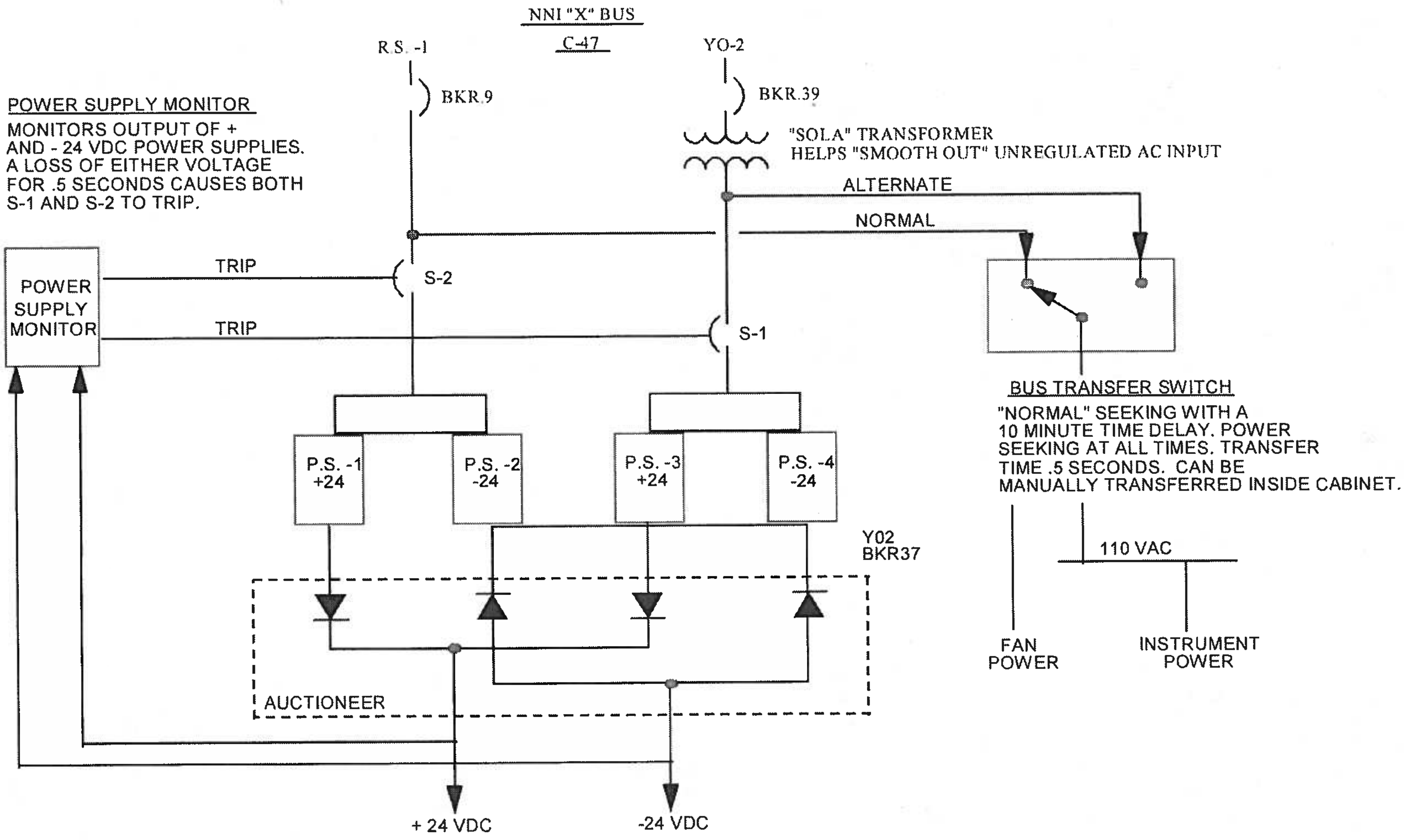
3.3 Reactor Coolant System Instrumentation

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.



<u>POWER SUPPLY MONITOR</u>

MONITORS OUTPUT OF + AND - 24 VDC POWER SUPPLIES. A LOSS OF EITHER VOLTAGE FOR .5 SECONDS CAUSES BOTH S-1 AND S-2 TO TRIP.

<u>BUS TRANSFER SWITCH</u>

"NORMAL" SEEKING WITH A 10 MINUTE TIME DELAY. POWER SEEKING AT ALL TIMES. TRANSFER TIME .5 SECONDS. CAN BE MANUALLY TRANSFERRED INSIDE CABINET.

<u>NOTE</u>: S-1 AND S-2 ALSO ACTS AS CIRCUIT BREAKERS IN THE EVENT OF AN OVERLOAD.

<u>NOTE</u>: S-1 AND S-2 MUST BE MANUALLY RESET FOLLOWING A TRIP.

FIGURE 69.01: NNI "X" BUS

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

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

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

Section: 4.2 **Type:** Generic APE

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

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

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

Tier:	1	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	Yes	**Taxonomy:**	C

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

Given the following indications at 100% power:

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

What are the actions required of the CBOT?

A. Trip the main generator output breakers.

B. Transfer D11 to emergency supply D01.

C. Trip all RCPs.

D. Transfer D21 to emergency supply D01.

Answer:

D. Transfer D21 to emergency supply D01.

Notes:

[d] is correct per 1203.036 as the conditions are indicative of a loss of D02.
[a] and [b] are incorrect due to this a loss of D02 not D01 these are actions for the loss of D01.
[c] is incorrect due to we have not loss seal injection and seal cooling, this is an action in this procedure section if both of the before mentioned system functions are lost

References:

1203.036, Chg. 08

History:

Developed for use in 98 RO Re-exam
Selected for use in 2002 RO/SRO exam, revised slightly.
Selected for 2005 Jon Gray RO re-exam.
Selected for the 2010 RO/SRO exam

SECTION 2 -- **LOSS OF D02**

1.0 SYMPTOMS

1.1 Low DC Voltage Alarms:

- D02 UNDERVOLTAGE (K01-A8)
- D21 LOSS OF VOLTAGE (K01-B8)
- RA2 LOSS OF VOLTAGE (K01-C8)
- D02 TROUBLE (K01-D8)

- H2 DC CONTROL POWER OFF (K02-B5)
- A2 DC CONTROL POWER OFF (K02-C7)
- A4 DC CONTROL POWER OFF (K02-D7)

- EOS SYSTEM TROUBLE (K04-C5)

1.2 Loss of breaker indicator lights for plant buses on right side of C10 and switchyard mimic on C10.

2.0 IMMEDIATE ACTION

NONE

3.0 FOLLOW-UP ACTIONS

3.1 IF reactor trips,
THEN perform Emergency Operating Procedures (1202.XXX) in conjunction with this procedure.

3.2 IF RCP seal injection AND seal cooling are BOTH lost,
THEN trip all running RCPs.

3.2.1 Isolate RCP Seal Bleedoff (Normal) by closing the following valves:

- CV-1270
- CV-1271
- CV-1272
- CV-1273

3.2.2 Place RCP Seal Bleedoff (Alternate Path to Quench Tank) controls in CLOSE:

- SV-1270
- SV-1271
- SV-1272
- SV-1273

SECTION 2 -- **LOSS OF D02** (continued)

3.3 Isolate letdown by closing Letdown Cooler E-29A/B Outlets:

- CV-1214
- CV-1216

3.4 At C10, transfer D21 to EMERG SUPPLY D01.

3.4.1 IF transfer of D21 is **NOT** successful,
THEN attempt local transfer of D21 to D01, while continuing.

3.5 Notify SM to implement Emergency Action Level Classification (1903.010).

3.6 IF reactor is **NOT** tripped,
THEN GO TO step 6.0.

3.7 IF transfer of D21 is successful,
THEN GO TO step 4.0.

3.8 IF transfer of D21 is **NOT** successful,
THEN perform the following:

3.8.1 Dispatch an operator to perform Attachment 2, while continuing.

3.8.2 GO TO step 5.0.

4.0 IF transfer of D21 is successful,
THEN perform the following:

4.1 Verify Condenser Vacuum Pump (C-5A OR C-5B) running.

4.2 IF OP HPI pump is tripped,
THEN restart as follows:

4.2.1 Place the following in HAND AND close:

- RC Pump Seals Total INJ Flow (CV-1207)
- PZR Level Control (CV-1235)

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

4.2.3 Start Aux Lube Oil pump for OP HPI pump.

4.2.4 Start the OP HPI pump.

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

QID: 0281 **Rev:** 0 **Rev Date:** 9-3-99 **Source:** Direct **Originator:** D. Slusher

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

Section: 4.2 **Type:** Generic AOP

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

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.

K/A Number: AK3.02 **CFR Reference:** 41.4, 41.8 / 45.7

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

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

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

Service Water Pumps P-4A, P-4B (supplied from A-4), and P-4C are running.
An ES actuation coincident with a loss of off-site power occurs.

Which service water pumps will autostart when A-3 and A-4 are re-energized?

A. P-4A, P-4B and P-4C

B. P-4A and P-4B

C. P-4B and P-4C

D. P-4A and P-4C

Answer:

D. P-4A and P-4C

Notes:

When ESAS actuates and the buses are re-energized the P-4A and P-4C handswitch position will interlock P-4B and keep P-4B from starting. Therefore, "a", "b", and "c" responses are incorrect.

References:

STM 1-42, Rev. 18, Service and Auxiliary Cooling Water, page 13, 14, 15

History:

Developed for 1999 exam.
Used in 2001 RO/SRO Exam.
Selected for the 2010 RO/SRO exam.

Each vacuum breaker returns flow back to its respective service water bay.

Each vacuum breaker is provided with a manual isolation valve and a bypass. The isolation valves, SW-118A, B & C are "Category E" valves normally locked open.

The service water pumps are driven by a 350 HP, 4160 Volt AC induction motors. The motors are located on the second floor of the Intake Structure. This location ensures pump operability in the event of a flood.

Additional information on SW pump design is contained in the table below.

Line Shaft Diameter	2-3/16"
Discharge Column	CS, Flanged
Impeller Diameter	18-1/2"

Power supplies for the motors are as follows:

- P-4A is powered from Bus A3 (4.16KV) through breaker A-302. If offsite power is unavailable and the #1 emergency diesel generator is running, A3 will be powered from DG #1 (K4A) through generator output breaker A-308.
- P-4C is powered from Bus A4 (4.16KV) through breaker A-402. If offsite power is unavailable and the #2 emergency diesel generator is running, A4 will be powered from DG #2 (K4B) through generator output breaker A-408.
- Service water pump P-4B is a swing pump. It can be powered from either A3 or A4 through motor operated disconnect (A6). P4B power can be electrically swapped using HS-3608 or by manually swapping A6 to the opposite bus. HS-3608 is located on panel C-18. To ensure system redundancy, it must be selected to the associated bus for the pump that it is in standby for. If P-4B is backup to P-4A then HS-3608 will be in the A-3 (breaker A-303) position and A-4 (breaker A-403) for P-4C backup. The MOD for P4B is located in the upper level of the Intake Structure in the electric fire pump room.

Note: Logic for auto-start is not determined by selector switch position but by breaker alignment.

The following table contains SW pump handswitch location and its associated positions.

SW Pump	Component	HS #	HS Location	Remarks
P4A	SW Loop I	3611	C-18	*
P4B	Swing SW Pump (A3)	3609	C-18	*
	Swing SW Pump (A4)	3600	C-16	*
	Bus selector switch	3608	C-18	Selects power to either A3 or A4
P4C	SW Loop II	3610	C-16	*
* Handswitch; start/stop/normal/pull-to-lock; spring return too normal.				

2.3.4.1 SW Pump Start Logic

During normal operation the A3 and A4 buses are powered from non-vital 4160-volt buses A1 and A2 respectively through bus tie breakers. A3 is fed from A1 through tiebreaker A309 and A4 is fed from A2 through tiebreaker A409. A1 or A2 can be supplied power from one of the three power supplies available. During turbine generator operation, A1 and A2 are powered from the Unit Aux transformer, which provides power to all in house loads. Following a turbine trip, electrical power is automatically transferred to the SU 1 transformer. If SU 1 becomes inoperable then power can be manually aligned to SU 2. SU2, which can provide power to either Unit 1 or Unit 2 or both is provided with a load shed feature to limit load placed on SU 2. For additional information on SU 2 load shed refer to 1107.001 Electrical System Operation.

Each SW pump is provided with 15-second time delays, which will time out prior to restarting the SW pumps previously running when power is restored. If no offsite power is available then an under voltage condition on either A3/A4 or B5/B6 will cause the bus tiebreaker to open, associated EDG to start and tie onto the bus. When A3 or A4 are re-energized, the SW pump(s) will restart after their associated time delay times out.

To prevent from exceeding EDG loading during an ESAS actuation with a loss of offsite power and three SW pumps in service, modifications to the SW pump start circuitry were incorporated. Prior to these modifications the potential existed for two SW pumps to be placed on a single EDG due to time delays for each pump are set at 15 seconds. This condition would occur if the time delay for the swing pump timed out before the lead pump. This event would start the swing SW pump and the lead SW pump overloading the EDG.

DCP-92-1016 modified the SW pump start logic to prevent this event from occurring. Modifications to the system included replacing P4A and P4C handswitches and P-4B start permissive circuitry. The new handswitches, HS-3610 for P-4C and HS-3611 for P-4A provided additional contacts that tie into the swing SW pump start logic. A contact in each handswitch is wired into the auto-start circuitry for service water pump P-4B that allows pump to auto-start when a specific condition exists.

For ease of discussion the logic explanation will cover P-4B auto-start when selected to the A3 bus. If any of the following conditions exists, then P-4B will auto-start when an ESAS actuation occurs along with or without a loss of offsite power.

- * HS-3611 (P-4A) in normal after stop (green flagged).
- * HS-3611 in "Pull to Lock".
- * HS-3611 placed in stop position.
- * Feeder breaker for P-4A trips open with HS in normal after start (red flagged).

Service water cross-connect isolation valves will automatically align to provide flow from P-4B to the affected loop. Additional information on Service Water crosstie valve logic will be discussed in section 2.3.8.

The following are automatic starting and stopping interlocks associated with the service water pumps:

- Pump motor will stop when turned to off or P-T-L.
- Pump motor will stop on a loss of voltage.
- Pump motor will stop on an electrical fault.
- Pump motor for P-4A and P-4C will restart after a loss of voltage if its handswitch is in normal after start.
- Pump motor for P-4B will restart after a loss of voltage only if the handswitch for the primary pump is in the "Stop," "Normal-After-Stop" or Pull-To-Lock" position.
- Pump motor will start when hand-switch is placed in start.

2.3.4.2 SW Pump Instrumentation and Alarms

(Refer to Figure 42.01 & Table 42.02)

P-4A, B, and C motor winding temperature are continuously monitored by temperature elements. These temperature elements send a signal for their respective pump motor windings to trend recorder TR-2808 located on panel C-19 in the control room. The SW Pump Motor winding temperatures can also be read on the plant computer.

When motor winding temperature reaches 250°F, annunciator K10-C4 "SW Pump Mtr Wdg Temp Hi" will alarm alerting the operator of this condition. TE's associated with each SW pump are listed below.

- P-4A Mtr Wdg temp (TE-3650)
- P-4B Mtr Wdg temp (TE-3613)
- P-4C Mtr Wdg temp (TE-3610)

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

QID: 0335 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-EOP04 **Objective:** 6 **Point Value:** 1

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

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

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

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

Tier:	1	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2.5
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

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

Given:

- Loss of all Feedwater
- HPI core cooling started

What indicates adequate HPI core cooling?

A. CET temperatures stable after 100 minutes.

B. T-cold tracking associated SG T-sat.

C. T-hot tracking CET temperatures.

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

Answer:

A. CET temperatures stable after 100 minutes.

Notes:

"A" is correct since the only criteria for evaluation of adequacy of core cooling via HPI is a decrease in CET temps.
"B", "C", and "D" are individual indications of adequate primary to secondary heat transfer.

References:

1202.004 Change 6

History:

Developed for 1999 exam.
Used on 2004 RO/SRO Exam.
Selected for the 2010 RO/SRO exam

INSTRUCTIONS	CONTINGENCY ACTIONS
6. (Continued).	F. Isolate Pressurizer Spray Line as follows: 1) Place Pressurizer Spray Control in HAND **AND** verify closed (CV-1008). 2) Close Pressurizer Spray Isolation (CV-1009).
7. IF MU Tank level drops below 18", THEN close Makeup Tank Outlet (CV-1275).	
8. Check Letdown in service.	8. **IF** CET SCM is adequate, **THEN** unless fuel damage or RCS to ICW leak is suspected, restore Letdown flow (RT 13).
9. Control RCS press within limits of Figure 3 (RT 14).	
10. Check CET temps stable or dropping.	10. Perform one of the following: A. **IF** HPI flow is < full flow from one HPI pump, **THEN GO TO step 18.** B. Hold at this point until one of the following conditions is met: 1) **IF** EFW becomes available, **THEN GO TO step 13**. 2) **IF** MFW or AFW pump becomes available, **THEN GO TO step 12.** 3) **IF** CET temps begin to drop, **THEN GO TO step 11.** 4) **IF** ≥120 minutes on HPI cooling elapse **AND** CET temps are still rising, **THEN GO TO step 18.** 5) **IF** CET temps are superheated **AND** moving away from the saturation line, **THEN GO TO 1202.005, "INADEQUATE CORE COOLING"** procedure.

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

QID: 0775 **Rev:** 0 **Rev Date:** 9/8/2009 **Source:** New **Originator:** S. Pullin

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: Ability to interpret reference materials, such as graphs, curves, tables, etc.

K/A Number: 2.1.25 **CFR Reference:** 41.10/43.5/45.12

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	C

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

REFERENCE PROVIDED

Given:
Plant 100% power
Electrical storm caused an grid disturbance
The Dispatcher calls Control Room and requests Unit 1 Generator power factor

With the Unit 1 Generator operating at 880 MWe gross out, what reactive load must it carry to be at a 0.98 power factor?

A. ~140 MVAR

B. ~180 MVAR

C. ~200 MVAR

D. ~260 MVAR

Answer:

B. ~180 MVAR

Notes:

Using Attachment N of Op-1102.004
B. is correct
A, C and D are associated with different power factors or generator loads.

References:

1102.004 Change 048

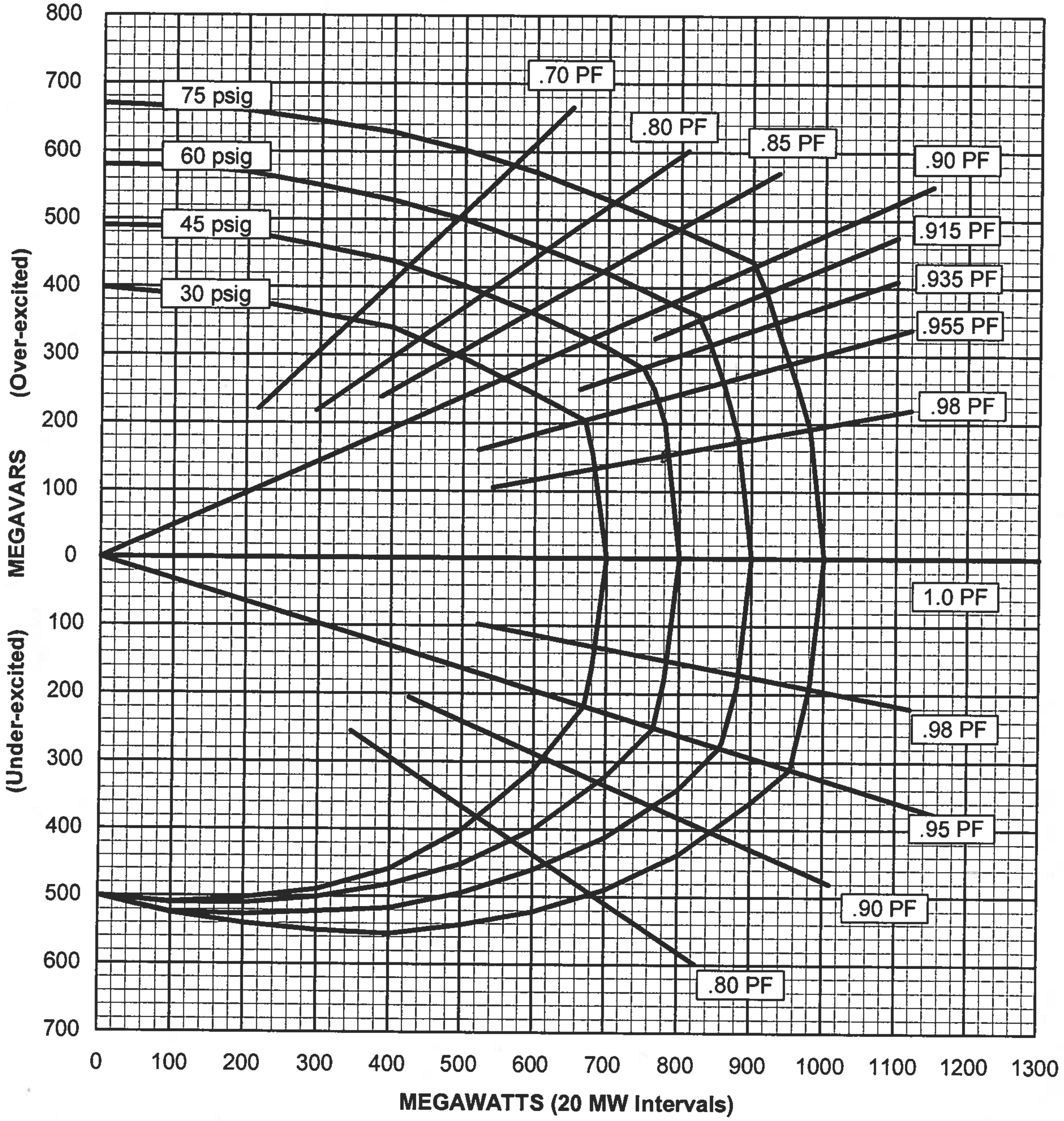
History:

Developed for the 2010 RO/SRO exam.

ATTACHMENT N

page 1 of 1

Hydrogen Inner-Cooled Turbine Generator Calculated Capability Curve at Rated Voltage



Basis Specifications:	1002.6 MVA	3 Phase
	0.90 PF	60 Hz
	22 KV	1800 RPM
	0.58 SCR	75 PSIG

RO Written Exam

Tier 1 Group 2

S-401 PWR Examination Outline Form ES-401-2

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	IR	#	QID	Type
000001 Continuous Rod Withdrawal / 1							**Not selected**	n/a			
000003 Dropped Control Rod / 1							**Not selected**	n/a			
000005 Inoperable/Stuck Control Rod / 1							**Not selected**	n/a			
000024 Emergency Boration / 1							**Not selected**	n/a			
000028 Pressurizer Level Malfunction / 2	X						**AK1.01** – PZR reference leak abnormalities.	2.8*	19	776	N
000032 Loss of Source Range NI / 7					X		**AA2.04** – Satisfactory source-range / intermediate-range overlap	3.1	20	777	N
000033 Loss of Intermediate Range NI / 7							**Not selected**	n/a			
000036 (BW/A08) Fuel Handling Accident / 8							**AK2.1 – Changed to randomly selected System 068 AK2.07**	n/a			
000037 Steam Generator Tube Leak / 3				X			**AA1.10** – CVCS makeup tank level indicator	2.9	21	778	N
000051 Loss of Condenser Vacuum / 4							**Not selected**	n/a			
000059 Accidental Liquid RadWaste Rel. / 9							**Not selected**	n/a			
000060 Accidental Gaseous Radwaste Rel. / 9							**AK1.04 – Changed to randomly selected System 028 AK1.01**	n/a			
000061 ARM System Alarms / 7			X				**AK3.02** – Guidance contained in alarm response for ARM system.	3.4	22	634	D
00067 Plant Fire On-site / 8	X						**AK1.02** – Fire Fighting	3.1	23	695	DR
0068 (BW/A06) Control Room Evac. / 8		X					**AK2.07** – ED/G	3.3	24	779	D
000069 (W/E14) Loss of CTMT Integrity / 5							**Not selected**	n/a			
000074 (W/E06&E07) Inad. Core Cooling / 4							**Not selected**	n/a			
000076 High Reactor Coolant Activity / 9							**Not selected**	n/a			
W/EO1 & E02 Rediagnosis & SI Termination / 3							**Not selected**	n/a			
W/E13 Steam Generator Over-pressure / 4							**Not selected**	n/a			
W/E15 Containment Flooding / 5							**Not selected**	n/a			
W/E16 High Containment Radiation / 9							**Not selected**	n/a			
BW/A01 Plant Runback / 1							**Not selected**	n/a			
BW/A02&A03 Loss of NNI-X/Y / 7							**Not selected**	n/a			
BW/A04 Turbine Trip / 4							**Not selected**	n/a			
BW/A05 Emergency Diesel Actuation / 6		X					**AK2.1** – Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	4.0	25	349	D
BW/A07 Flooding / 8					X		**AA2.2** - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	3.3	26	780	N
BW/E03 Inadequate Subcooling Margin / 4							**Not selected**	n/a			
BW/E08; W/E03 LOCA Cooldown - Depress. / 4							**Not selected**	n/a			
BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4							**Not selected**	n/a			
W/E13&E14 EOP Rules and Enclosures						X	**2.2.22**- Knowledge of limiting conditions for operations and safety limits.	4.0	27	595	N
CE/A11; W/E08 RCS Overcooling - PTS / 4							**Not selected**	n/a			

CE/A16 Excess RCS Leakage / 2							**Not selected**	n/a			
CE/E09 Functional Recovery							**Not selected**	n/a			
/A Category Point Totals:	2	2	1	1	2	1	Group Point Total:		9		

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

QID: 0776 **Rev:** 0 **Rev Date:** 9/8/2009 **Source:** New **Originator:** S. Pullin

TUOI: ASLP-RO-CMP02 **Objective:** 9a **Point Value:** 1

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

System Number: 028 **System Title:** Pressurizer (PZR) Level Control Malfunction

Description: Knowledge of the operational implications of the following concepts as they apply to pressurizer level control malfunctions: PZR reference leg abnormalities.

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

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

Group: 2 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** C

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

Given:

Plant at 100% power
Leak develops on the pressurizer reference leg

What effect does this have on level indication and pressurizer level control valve, CV-1235?

A. Indicated level decreases and pressurizer level control valve, CV-1235, opens to control level.

B. Indicated level decreases and pressurizer level control valve, CV-1235, closes to control level.

C. Indicated level increases and pressurizer level control valve, CV-1235, opens to control level.

D. Indicated level increases and pressurizer level control valve, CV-1235, closes to control level.

Answer:

D. Indicated level increases and pressurizer level control valve, CV-1235, closes to control level.

Notes:

D. is correct, a leak in the reference leg would cause indicated level to increase. As a result of the level rise CV-1235 will close in oreder to maintain level at setpoint.
A, B, and C are incorrect, using the different possible combinations.

References:

ASLP-RO-CMP02 Rev 2

History:

New selected for 2010 RO/SRO exam.

INSTRUCTOR GUIDE	KEY POINTS, AIDS, QUESTIONS/ANSWERS
a. An increase in ambient temperature will cause density of wet reference leg to decrease	
b. This will result in lower D/P sensed by D/P cell, and indicated level will be greater than actual level	
c. The opposite effect produces lower indicated level when ambient temperature decreases	
4. As previously discussed, radiation levels near D/P cell affect detector integrity	
a. A high radiation environment can permanently embrittle detector cell, causing cell to lose its elasticity and altering its characteristics, as well as degrade sensitive electronics	
K. Failure Indications	
1. For level detectors with wet reference leg connected to “high pressure” side of D/P cell, following failure modes exist	
a. A break in variable leg of D/P cell creates higher D/P being sensed by D/P cell, resulting in level instrument indicating low level	Objective 9b
b. Conversely, reference leg break creates lower D/P sensed across D/P cell, resulting in indicated level higher than actual level	Objective 9a

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

QID: 0777 **Rev:** 0 **Rev Date:** 9/8/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Satisfactory source-range intermediate-range overlap

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

Tier: 1 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.5 **SRO Select:** Yes **Taxonomy:** C

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

Given:

Source Range 5 X 10 4 counts
Intermediate Range 1 X 10-9 amps

During the startup, the source range instruments fail to 3 counts per second.

What is the required operator action for the given condition?

A. Immediately suspend operations involving positive reactivity changes..

B. Within 1 hour verify CRD trip breakers open.

C. Continue the startup..

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

Answer:

C. Continue the startup..

Notes:

C. is correct, procedure allows continuing with startup if intermediate range indicate >10 -10 amps.

A, B and D are incorrect due to these are the actions to take when both source range instruments fail and both intermediate range channels indicate <10 -10 amps.

References:

1203.021 Change 10

History:

New for the RO/SRO 2010 exam.

SECTION 3

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

1.0 SYMPTOMS

1.1 Source range indication reading incorrectly.

1.2 CRD WITHDRAWAL INHIBITED (K08-A2) alarm.

2.0 IMMEDIATE ACTION

NONE

3.0 FOLLOW-UP ACTIONS

NOTE

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

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

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

3.2 IF only one source range channel is operable,
OR 1 of 2 intermediate range channels indicates $>10^{-10}$ amps,
THEN continue plant operations (TS 3.3.9).

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

NOTE

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

3.3.1 Refer to TS 3.3.9 Condition A.

3.3.2 Immediately suspend operations involving positive reactivity changes.

3.3.3 Immediately initiate a shutdown and insert all control rods.

3.3.4 Within 1 hour verify CRD trip breakers open.

STM 1-67

Nuclear Instrumentation

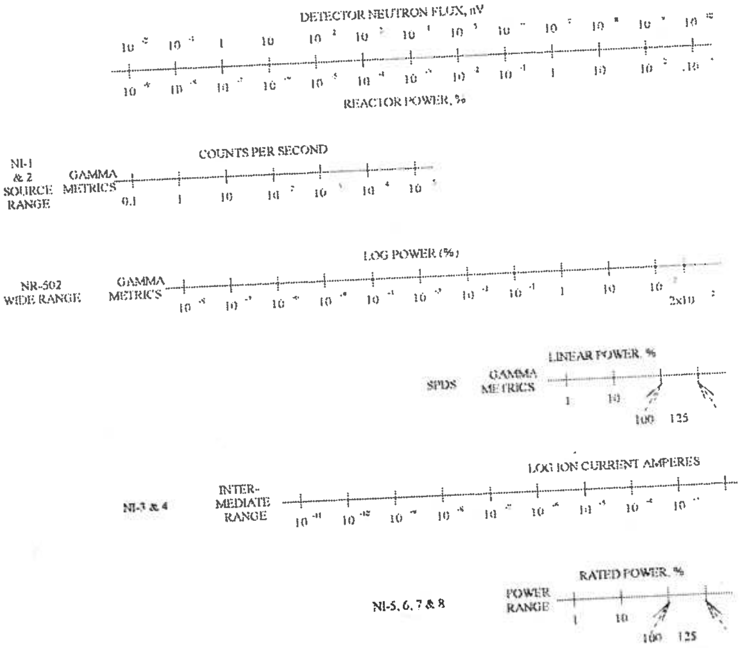
1.0 Introduction

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: 0778 **Rev:** 0 **Rev Date:** 9/8/2009 **Source:** New **Originator:** S. Pullin

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

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

System Number: 037 **System Title:** Steam Generator (S/G) Tube Leak

Description: Ability to operate and / or monitor the following as they apply to the Steam Generator Tube Leak: CVCS makeup tank level indicator.

K/A Number: AA1.10 **CFR Reference:** 41.7/45.5/45.6

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

Group: 2 **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** K

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

Given:

Plant at 100% power
Makeup Tank level dropping at 1 inch every 2 minutes.
“A” OTSG N-16 TROUBLE (K07-A5)
PROC MONITOR RADIATION HI (K10-B2)

What is the A OTSG Tube Leak rate?

A. 10 gpm

B. 15 gpm

C. 20 gpm

D. 25 gpm

Answer:

B. 15 gpm

Notes:

B. 15 gpm is correct based on makeup tank level is 30 gallons per inch, at a rate of change of 1 inch per 2 minutes equals 15 gpm leak.
A, C and D are incorrect.

References:

1203.039 Change 011

History:

New for the RO/SRO 2010 exam.

ATTACHMENT 1

Estimate of RCS Leakrate

NOTE

- The RB Sump contains 45.4 gal/percent.
- Dirty Waste Drain Tanks (T-20s) contain 52.5 gal/percent.
- Auxiliary Building Sump contains 8.98 gal/percent.
- ICW Surge Tank T-37B Level (PDIS-2229) 0.5 to 2.7 psid (1 psid = 333 gallons).
- Estimated MU Flow During RCS Cooldown is contained in Attachment 2 of this procedure.

1. **Estimate RCS leakrate using the following formula :**

   - Use the following table to perform mass balance estimate.

NOTE

- When the BWST is aligned to the Makeup Tank, Makeup Tank Level changes should generally **NOT** be used for leak rate estimation.
- Pressurizer and Makeup Tank level changes can either be added **OR** subtracted to estimate leak rate.

- IF applicable, record current cooldown rate for leak estimation: ______________
- Calculate Seal bleedoff flow for RCPs _____ + _____ + _____ + _____ = _____

Makeup Flow	F1238/C04	Gpm	**Plus**
Seal Injection Flow	F1239/C04	Gpm	**Plus**
HPI Flow	SPDS/C16 and C18	Gpm	**Plus**
Pressurizer Level Change	X 12.4 gal/in	Gpm	**Minus (IF rising)- Plus (IF lowering)**
Makeup Tank Level Change (N/A IF BWST Outlet Open)	X 30.86 gal/in	Gpm	**Plus (IF rising)- Minus (IF lowering)**
Letdown Flow	F1236/C04	Gpm	**Minus**
Seal Bleedoff flow	F1270-3/C13	Gpm	**Minus**
Makeup Flow Due to Cooldown	Attachment 2	Gpm	**Minus**
	TOTAL	Gpm	

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

QID:	0634	**Rev:**	0	**Rev Date:**	11/8/05	**Source:**	Direct	**Originator:**	J.Cork
TUOI:	A1LP-RO-RMS			**Objective:**	7			**Point Value:**	1

Section: 4.2 **Type:** Generic APEs

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

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

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

Tier:	1	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	C

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

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

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

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

A. WARNING and POWER ON lights on

B. POWER ON light off

C. HIGH ALARM light on and POWER ON light off

D. FAILURE light on

Answer:

B. POWER ON light off

Notes:

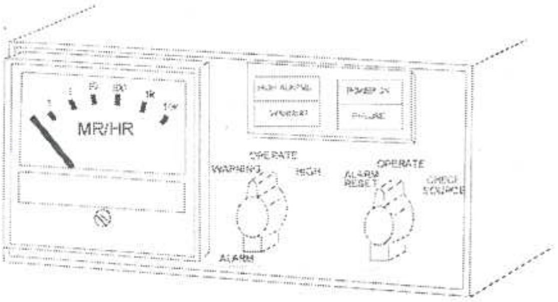
"B" is correct, a loss of power will cause both the Hi Radiation and Trouble annunciators to come in.
"A" is incorrect, this would cause the Hi Radiation but not the Trouble annunciator.
"C" is incorrect, the POWER ON light off will cause both annunicators but the HIGH ALARM light will not be on with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

References:

1203.012I, Chg. 046
STM 1-62, Rev. 11

History:

New for 2005 RO re-exam.
Selected for 2010 RO/SRO exam.



(Refer to Figure 62.09) In addition to an analog meter, station indicating units have:

- Four status lights to indicate Power On (green), Failure (white), High Alarm (red) and Warning (amber). The failure alarm occurs when the signal drops below a preset value.
- One three position switch allows for checking the warning and high alarm setpoints. Operation of the Alarm Setting switch does not bring in high alarms or initiate any automatic actuation.
- Another three position switch is provided for alarm reset and check source operation.
- When either of the three position switches is removed from the normal position of operate, a "Rad Monitor Test in Progress" Alarm will annunciate in the control room on K-10.
- Each drawer can be slid away from the panel face to gain access to potentiometers for setpoint adjustment.

High alarm of all the monitors is interlocked to give audible and visual remote alarms at the location of each monitor. The failure alarm or a loss of power to the unit will actuate the "Radiation Monitor Trouble" annunciator (K10-C1) in the Control Room. A "High" alarm or a loss of power to the unit will actuate the "Area Monitor Radiation Hi" annunciator (K10-B1).

The ARM's provide inputs to the plant computer. A listing of the ARM's and their current value can be displayed on the plant computer by going to Group Display (GD), then ARMS.

2.1.2 Control Room Radiation Monitor

The control room envelope (Unit 1 and Unit 2) is monitored for excessive radiation by five detectors. These radiation detectors are RE-8001, 2RE-8001A, 2RE-8001B, 2RE-8750-1A, and 2RE-8750-1B. The Unit 1 Control Room Area Monitor (RE-8001) is located on the east wall of the control room. In addition, radiation monitors 2RE-8001A and 2RE-8001B are mounted in the air supply and operating area ductwork for the Unit 1 Control Room. High radiation on any one of these monitors will cause Control Room isolation for both Control Rooms. The "Control Room Supply Duct Radiation Hi" annunciator (K16-D1) is the associated alarm. Refer to STM 1-12 for Control Room Ventilation.

The warning alarm on Control Room Area Monitor (RI-8001) provides the actuating signal for control room isolation. Power to this unit is from RS-4 through C-25 Bay 3.

The actuation level for high radiation is sufficiently below hazardous radiation levels to minimize operator dose during an accident and is sufficiently above normally experienced background levels to minimize spurious actuations.

Location: C16

Device and Setpoint:
See Radiation Monitoring System Check and Test (1305.001) Supplement 6, "Area Radiation Monitor Weekly Alarm Check".

AREA MONITOR
RADIATION
HI

Alarm: K10-B1

1.0 OPERATOR ACTIONS

1. Inspect C25 Bay 3 and determine alarming monitor.

   A. Determine if alarm is due to high radiation or loss of power.

2. IF alarm is due to momentary spike,
   THEN reset alarm.

3. IF loss of power,
   THEN GO TO RADIATION MONITOR TROUBLE (K10-C1).

4. IF confirmed high radiation within reactor building
   AND personnel are inside RB,
   THEN sound reactor building evacuation alarm.

   A. IF high radiation outside RB
      AND within a Radiologically Controlled Area,
      THEN GO TO step 6.

5. IF confirmed high radiation outside reactor building
   AND outside Radiologically Controlled Areas,
   THEN announce high radiation warning on plant public address system.

6. IF Control Room (RI-8001) in alarm,
   THEN refer to ACTUATION -- CONTROL ROOM ISOLATION (K16-B2).

7. Initiate action to have high radiation area surveyed.

8. IF SF Pool (RI-8009) in alarm
   AND Spent Fuel Pool is the radiation source,
   THEN maximize SF Pool purification flow per "Spent Fuel Pool Purification" section of Spent Fuel Cooling System (1104.006).

   A. IF radiation levels inside a Radiologically Controlled Area are determined to be > limits of EVACUATION (1903.030),
      THEN GO TO 1903.030.

9. IF radiation rises to ≥2.5 mrem/hour outside a Radiologically Controlled Area,
   THEN GO TO EVACUATION (1903.030).

10. IF projected summed releases exceed NUE criteria for one hour at site boundary,
    THEN notify SM to review EALs (1903.010).

11. IF it is desired to raise alarm setpoint,
THEN perform applicable sections of Area Radiation Monitor Monthly Alarm Check (1305.001 Supplement 6).

2.0 PROBABLE CAUSES

NOTE
This annunciator has multiple input without reflash.

1. Any area monitor in C25 Bay 3 senses radiation above alarm setpoint
2. Any area monitor in C25 Bay 3 de-energized
3. Any area monitor in C25 Bay 3 alarm lamp removed or burned out

3.0 REFERENCES

1. Schematic Diagram Annunciator K10 (E-460, sheets 1 - 3)

Location: C16

Device and Setpoint:
De-energization of or FAILURE ALARM on any radiation monitor in Radiation Monitoring System Panel (C25 Bays 1-3 and Bay 4 of C24). Monitors are listed on next page.

RADIATION
MONITOR
TROUBLE

Alarm: K10-C1

1.0 OPERATOR ACTIONS

1. Observe monitors at C24 and C25 for FAILURE ALARM light(s) on or POWER ON light(s) off.

2. IF power is off to all monitors in a bay,
THEN verify supply breaker closed:

   A. Rad Monitor Panel C24, Rad Monitor Panel C25, Bay 1 (RS1, bkr 8)

   B. Rad Monitor Panel C24, Rad Monitor Panel C25, Bay 2 (RS2, bkr 8)

   C. Rad Monitor Panel C25, Bay 3 (RS4, bkr 8)

3. IF either of the following monitors is inoperable (FAILURE ALARM or power loss):

   - Spent Fuel Pool (RI-8009)
   - Fuel Handling Area (RI-8017)

   AND fuel handling in progress,
   THEN stop fuel handling until radiation monitoring requirement is satisfied per Control of Unit 1 Refueling (1502.004) OR Control of Fuel and Control Rod Movement in the U-1 Spent Fuel Area (1502.010).
   (TRM 3.9.1 and TRM 3.9.2)

4. IF Control Room (RI-8001) is inoperable (FAILURE ALARM or power loss),
THEN verify control room emergency ventilation actuation. (TS 3.7.9)

5. IF Liquid Radwaste (RI-4642) is de-energized,
THEN verify CZ Disch to Flume Flow (CV-4642) is closed or auto closes.
(ODCM App.1, L2.1.1)

NOTE

The following alignment stops gaseous release and diverts flow to Waste Gas Surge Tank (T-17).

6. IF Gaseous Radwaste (RI-4830) is de-energized,
   THEN verify the following: (ODCM App.1, L2.2.1)
   - T-18s Discharge to Gaseous Radwaste Discharge Header Flow Control (CV-4820) closed
   - Gaseous Radwaste Discharge Isol (CV-4830) closed
   - ABVH Diversion to T-17 (CV-4806) open
7. IF RB Atmos Gaseous Monitor is inoperable,
   THEN refer to Reactor Building Ventilation (1104.033). (TS 3.4.15)
8. Initiate steps to survey areas for which radiation monitors are inoperable.
9. Initiate steps to have failed monitor(s) checked and repaired.
10. IF alarm was caused by FAILURE ALARM on monitors,
    THEN all monitors that are failed, must be reset using ALARM RESET switch on front of monitor to clear K10-C1.

2.0 PROBABLE CAUSES

NOTE

- This annunciator has reflash capability. If the alarm window is lit solid due to one cause and another cause actuates, the alarm will go to fast flash with an audible alarm.
- FAILURE ALARM light on monitor indicates that the monitor has had no input from the detector for one minute; detector failure.

1. Any radiation monitor FAILURE ALARM in C25 or Bay 4 of C24
2. De-energization of any radiation monitor in C25 or Bay 4 of C24
3. Any radiation monitor in C25 or Bay 4 of C24 alarm lamp removed or burned out

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

QID:	0695	**Rev:** 1	**Rev Date:** 4/1/2008	**Source:**	Repeat	**Originator:**	Steve Pullin
TUOI:	ASLP-RO-FRHAZ		**Objective:** 4B			**Point Value:**	1

Section: 4.2 **Type:** Generic APEs

System Number: 067 **System Title:** Plant Fire on Site

Description: Knowledge of the Operational implications of the following concepts as they apply to plant fire on site: fire fighting.

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

Tier:	1	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	K

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

Per 1015.007, "Fire Brigade Organization and Responsibilities," which of the following describes the Ops Manning composition of the Fire Brigade for the initial response to a fire on Unit 1?

A. Unit 1 supplies the Fire Brigade Leader,
Unit 2 supplies 3 Fire Brigade members,
Security supplies one support member.

B. Unit 1 supplies the Fire Brigade Leader and 2 Fire Brigade members,
Unit 2 supplies 1 Fire Brigade member,
Security supplies one support member.

C. Unit 2 supplies the Fire Brigade Leader,
Unit 1 supplies 3 Fire Brigade members,
Security supplies one support member.

D. Unit 2 supplies the Fire Brigade Leader and 1 Fire Brigade member,
Unit 1 supplies 2 Fire Brigade members,
Security supplies one support member.

Answer:

A. Unit 1 supplies the Fire Brigade Leader, Unit 2 supplies 3 Fire Brigade members, Security supplies one support member

Notes:

A is correct per the requirements of 1015.007
B is incorrect. This answer was previously correct for a fire on Unit 1 prior to the latest revision.
C is incorrect. This is correct for a fire on Unit 2
D is incorrect This answer was previously correct for a fire on Unit 2 prior to the latest revision.

References:

1015.007, "Fire Brigade Organization and Responsibility" Chg. 019

History:

Selected for 2008 RO Exam
Selected repeat for the 2010 RO/SRO exam

5.3 Fire Brigade Members

5.3.1 Under the direct supervision of the Fire Brigade Leader, Fire Brigade Members are responsible for primary extinguishment efforts (extinguishers, hoses, etc.).

5.3.2 The Fire Brigade Members of the unaffected unit shall respond to a fire in the affected unit.

5.3.3 Restore fire equipment after use in accordance with "Fire Equipment Restoration" section of this procedure.

5.4 Security Force

5.4.1 Shall assign a support person to the Fire Brigade Support Team to respond to a fire.

5.4.2 The support person is responsible for providing support activities under direct supervision of the Fire Brigade Leader or the three fully trained Fire Brigade Members. These activities will normally include, but are not limited to, supplying additional equipment, supplying SCBAs, hose laying, etc. Under normal circumstances the support person should not perform extinguishing activities unless directly instructed by the Fire Brigade Leader.

5.4.3 Assist with the restoration of fire equipment after use in accordance with "Fire Equipment Restoration" section of this procedure.

6.0 INSTRUCTIONS

6.1 Assignment of Fire Brigade Personnel

6.1.1 The Unit 1 Fire Brigade consists of the following:

A. Unit 1 Fire Brigade Leader

B. Three Fire Brigade Members from Unit 2

C. Fire Brigade Support Member from Security Force

6.1.2 The Unit 2 Fire Brigade consists of the following:

A. Unit 2 Fire Brigade Leader

B. Three Fire Brigade Members from Unit 1

C. Fire Brigade Support Member from Security Force

6.2 The fire is reported to the Control Room of the affected unit.

6.2.1 The SM/CRS of the affected unit dispatches the Fire Brigade to the scene of the fire. The SM/CRS of the unaffected Unit will dispatch the Fire Brigade for zones identified in 1203.049/2203.049, Fires In Areas Affecting Safe Shutdown.

6.2.2 The Fire Brigade Leader of the affected unit responds and assumes command of the fire fighting activities.

6.2.3 The Fire Brigade Members from the unaffected unit respond along with the Security Fire Brigade Support Member. This comprises the initial fire fighting force.

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

QID: 0779 **Rev:** 0 **Rev Date:** 9/8/2009 **Source:** Direct **Originator:** S. Pullin

TUOI: ANO-1-LP-RO-EDG **Objective:** 26 **Point Value:** 1

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

System Number: 068 **System Title:** Control Room Evacuation

Description: Knowledge of the interrelations between the Control Room Evacuation and the following: ED/G

K/A Number: AK2.07 **CFR Reference:** 41.7/45.7

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

Group: 2 **SRO Imp:** 3.4 **SRO Select:** Yes **Taxonomy:** K

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

Given:

Fire has occurred in the Cable Spread Room
Performing 1203.002 Alternate Shutdown
CRS follow-up actions are to place
#1 EDG and #2 EDG are running and have been placed in a "No DC" start condition.

What protection is operable to the Emergency Diesel Generators?

A. Positive crankcase pressure trip

B. Low lube oil pressure trip

C. Mechanical over speed trip

D. High jacket water temperature trip

Answer:

C. Mechanical over speed trip

Notes:

"C" will mechanically trip the fuel rack.
"A" and "B" require DC power to the emergency trip relay.
"D" does not exist.

References:

1104.036, Emergency Diesel Generator Operation, Change 049

History:

Direct Selected for 2010 RO/SRO 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 ES Electrical System Operations (1107.002), "Breaker Local Operation Without DC Control Power" section, 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)

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:**	Yes	**Difficulty:**	3
Group:	3	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

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

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?

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

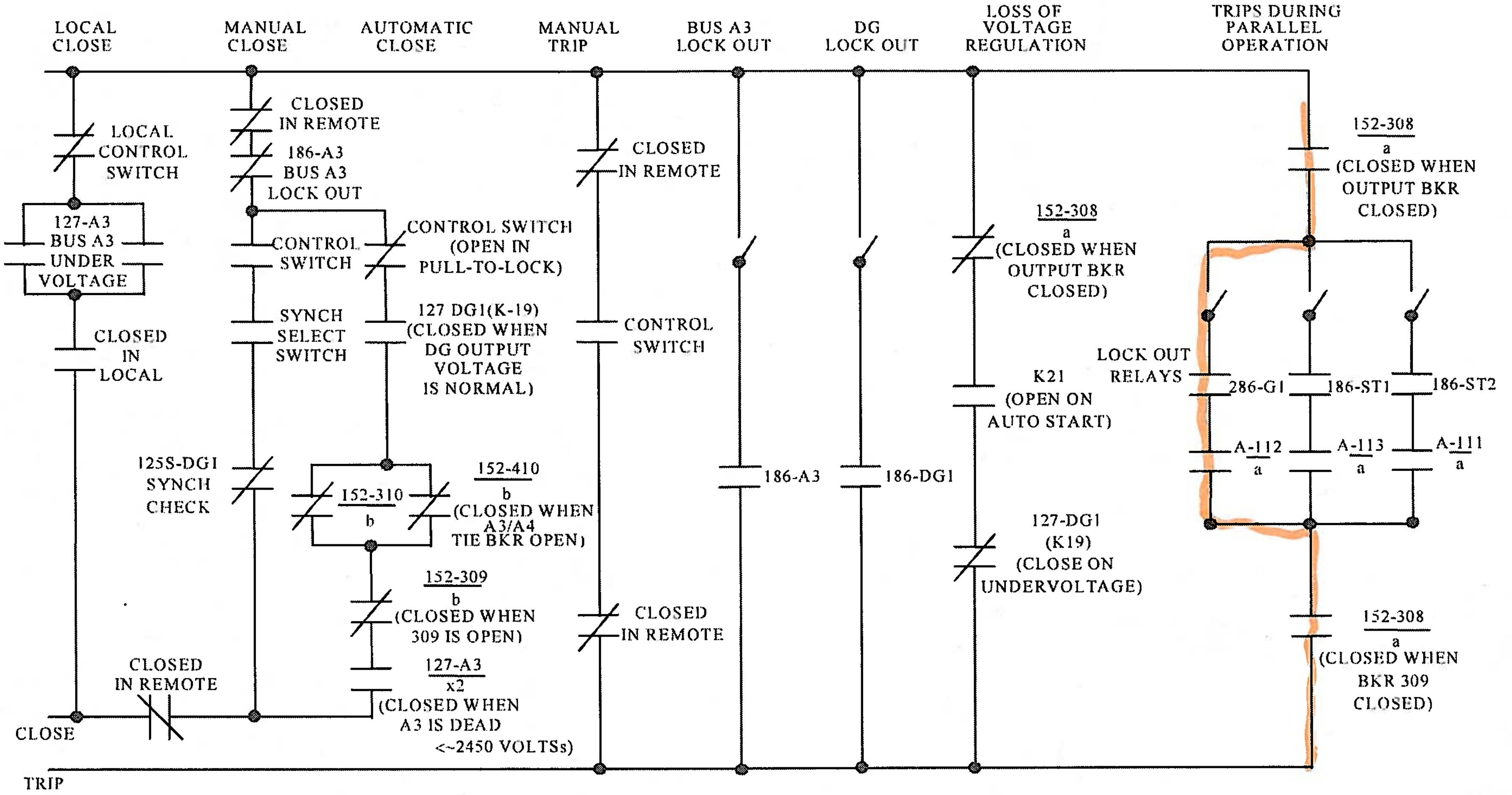


FIGURE 32.72: DIESEL GENERATOR OUTPUT BREAKER A308/A408

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

QID: 0780 **Rev:** 0 **Rev Date:** 9/09/2009 **Source:** New **Originator:** S.Pullin

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

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

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

Description: Ability to determine and interpret the following as they apply to the (flooding): adherence to appropriate procedures and operation within the limitations in the facilities license and amendments.

K/A Number: AA.2.2 **CFR Reference:** 43.5/45.13

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

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

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

Given:

Plant power 100%
"A" Decay Heat pump OOS
Dardanelle Lake Level 350 feet rising 1 ft/hr due to heavy rains
Corps of Engineers predicts peak flood levels will reach 355 feet

What action is required per Natural Emergencies procedure 1203.026 section 4 Flood?

A. Perform rapid plant shutdown per 1203.045 and align "B" Decay Heat pump for Decay Heat

B. Perform rapid plant shutdown per 1203.045 and transfer plant auxiliaries to SU 2 transformer

C. Trip Reactor and refer to 1202.001 and perform a Forced flow Cool down 1203.040

D. Trip Reactor and refer to 1202.001 and perform a Natural Circulation cool down 1203.013

Answer:

B. Perform rapid plant shutdown per 1203.045 and transfer plant auxiliaries to SU 2 transformer

Notes:

B. is correct due to 1203.025 directs you to perform a shutdown per 1203.045, and SU2 transformer is designed for flooding and should be used during a flood
A. is incorrect 1203.025 directs you to perform a shutdown per 1203.045, and align a LPI pump for DH if both pumps are operable in this case "A" DH pump is OOS
C. is incorrect the procedure does not call for a reactor trip but you should use rapid plant shut down and forced flow cool down
D. is incorrect the procedure does not call for a reactor trip but you should use rapid plant shut down and forced flow cool down not Natural Circulation CD

References:

Natural Emergencies 1203.025 change 028

History:

New for 2010 RO/SRO exam

SECTION 4
FLOOD

ENTRY CONDITIONS

- Lake level >340' and rising
- Forecasted lake level at site is >350'

SECTION 4
FLOOD

INSTRUCTIONS

1. **Notify Unit 2 Control Room.**

NOTE

Information may be obtained from the Corps of Engineers throughout the implementation of this procedure at the following numbers:

• Dardanelle Lock and Dam Project Office	479-968-5008 ext. 241
• Dardanelle Lock and Dam Powerhouse	479-229-1863 (Fri-Sun use ext. 0)
• Little Rock District Engineer	501-324-5697

2. **Establish contact with Corps of Engineers for peak flood condition forecasts and updates.**

3. **Notify Little Rock TOC Dispatcher.**

4. **Initiate lake level monitoring using one of the following methods:**

NOTE

SPDS displays level in feet. PMS/PDS displays level in inches above 324' reference level. Instructions that follow give level in feet and corresponding level from PMS/PDS in brackets, e.g., 340' [PMS 192 in.]. At flood levels >349' [PMS 300 in.], SPDS and PMS/PDS are off-scale above sensor span.

- SPDS – monitor SW pump aligned to lake (P-4A, P-4B1, P-4B2, P-4C)

- PMS/PDS – monitor SW or Circ bay aligned to lake
  (SW bays L3664, L3666, L3668, and B & C Circ Bays L3601, L3602)

- WHEN ≥ 349' [PMS 300 in.],
  THEN monitor lake level locally on an hourly basis.

5. **WHEN directed by plant management,**
   THEN begin plant shutdown per Rapid Plant Shutdown (1203.045).

   A. IF directed by plant management,
   THEN begin plant cooldown per Plant Shutdown and Cooldown (1102.010) or Forced Flow Cooldown (1203.040).

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

7. **Initiate evaluation of plant risk in accordance with COPD-024, Risk Assessment Guidelines.**

(continued)

8. **WHEN Lake Dardanelle level greater than 345 ft. (PMS 252 in.), THEN perform "Local Flooding Actions" Attachment B of this procedure.**

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

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

> **NOTE**
>
> Jumpers and associated hardware are located at Air Break Tower (B1217).

   A. Installation of jumpers from the primary side of Startup Transformer (SU-2) directly to the 161KV transmission line.

   B. Issuance of switching orders to allow work on Startup Transformer SU-2.

   C. De-energize and bypass SU-2 Voltage Regulator.

10. **IF both decay heat removal loops are available, THEN align one loop for decay heat removal as follows:**

   A. Ensure that one decay heat loop is aligned for ES standby (LPI) per Decay Heat Removal Operating Procedure (1104.004), Attachment A.

   B. Align the opposite decay heat loop for DH removal per 1104.004, "Decay Heat Removal During Cooldown" section.

      1) Align DH system AUX spray per 1104.004, "Depressurizing RCS Using DH System AUX Spray" section.

11. **IF only one decay heat removal loop is available, THEN verify loop is aligned for ES standby (LPI) per 1104.004, Attachment A.**

   A. **WHEN** plant cooldown is at the point of switching to decay heat,
   THEN align the available loop for DH removal per 1104.004, "Decay heat Removal During Cooldown" section.

   B. **IF** DH system is NOT accessible,
   THEN continue RCS cooldown utilizing steam generators
   AND ensure RC pressure is maintained as per NPSH curve for RC pump operation.

(continued)

SECTION 4 -- FLOOD (Continued)

12. Remove equipment from service AND de-energize power supplies to below-grade equipment prior to flooding.

> **NOTE**
>
> At flood levels >349' [PMS 300 in.], SPDS and PMS/PDS are off-scale above sensor span. Level must be observed locally.

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

A. Secure nonessential electrical loads.

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

C. Coordinate with Unit 2 Control Room to transfer plant auxiliaries to SU 2 using Electrical System Operation (1107.001), "Startup Transformer Operations" section.

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

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

16. Conduct further operations as directed by plant management.

END

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

QID: 0595 **Rev:** 0 **Rev Date:** 9/09/2009 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-RCS **Objective:** 26 **Point Value:** 1

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

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

Description: Knowledge of limiting conditions for operation and safety limits.

K/A Number: 2.2.22 **CFR Reference:** 41.5/43.2/45.2

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

Group: 2 **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** C

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

In accordance with Technical Specification bases, what is the purpose of the Code Safeties and what is the design bases accident that defines their minimum capacity?

A. The Code Safeties prevent exceeding the safety limit of 2500 psig during a 100% load rejection without a reactor trip.

B. The Code Safeties prevent exceeding the safety limit of 2750 psig during a 100% load rejection without reactor trip.

C. The Code Safeties prevent exceeding the safety limit of 2750 psig during a startup accident.

D. The Code Safeties prevent exceeding the safety limit of 2500 psig during a startup accident.

Answer:

C. The Code Safeties prevent exceeding the safety limit of 2750 psig during a startup accident.

Notes:

Answer "C" is correct, it lists the proper safety limit and the design basis accident.
Answer "A" is incorrect, it lists the safety setpoint (not the safety limit) and a plausible, but incorrect, accident.
Answer "B" is incorrect, it lists the proper safety limit and a plausible, but incorrect, accident.
Answer "D" is incorrect, it lists the safety setpoint (not the safety limit) and the design basis accident.

References:

Technical Specifications bases B2.1.2 amendment # 215

History:

New for 2010 RO/SRO exam

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

In SAR, Section 1.4 (Ref. 1), GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," and GDC 15, "Reactor Coolant System Design", address RCPB design and protection, respectively. The ANO-1 discussion regarding how GDC 15 is accomplished states that analysis and evaluation of all normal and abnormal operating conditions and transients are integrally related to all RCS and associated systems design. SAR Chapter 14 (Ref. 2) lists these abnormal operating conditions and transients and terms them "abnormalities". In addition, GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and abnormalities, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with the design codes (Ref. 3 and 4). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure prior to initial operation, according to the design code requirements. Inservice leak testing at not less than 2155 psig is also required, prior to MODE 2, following any opening of the reactor coolant system in accordance with ASME code, Section XI; IWA-5000. When performed at the end of refueling outages, this leak test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components (Ref. 5).

APPLICABLE SAFETY ANALYSIS

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME code for Nuclear Power Plant Components (Ref. 3). The design basis transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal event from low power.

The startup event analysis (rod withdrawal at low power) (Ref. 2) is performed using conservative assumptions relative to pressure control devices.

RO Written Exam

Tier 2 Group 1

PWR Examination Outline
Plant Systems - Tier 2/Group 1 (RO)

Form ES-401-2

	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#	QID	Type
003 Reactor Coolant Pump					X							**K5.05** – The dependency of RCS flow rates upon the number of operating RCP's	2.8*	28	781	N
										X		**A4.08** – RCP cooling water supplies	3.2	29	782	M
004 Chemical and Volume Control											X	**2.1.34 changed to 2.2.38** – Knowledge of conditions and limitations in the facility license	3.6	30	796	N
				X								**K4.03** – Protection of ion exchangers (high letdown temperatures will isolate ion exchangers)	2.8*	31	259	D
005 Residual Heat Removal		X										**K2.01** – RHR Pumps	3.0*	32	786	M
006 Emergency Core Cooling						X						**K6.10** - Valves	2.6	33	783	M
007 Pressurizer Relief/Quench Tank					X							**K5.02** – Method of forming a steam bubble in the PZR	3.1	34	561	D
008 Component Cooling Water								X				**A2.08 changed to A2.01** - Loss of CCW Pump	3.3	35	787	N
010 Pressurizer Pressure Control			X									**K3.02** - RPS	4.0	36	788	N
012 Reactor Protection						X						**K6.10** – Permissive circuits	3.3	37	784	N
											X	**2.1.32** – Ability to explain and apply system limits and precautions	3.8	38	785	N
013 Engineered Safety Features Actuation				X								**K4.10** – Safeguards equipment control reset	3.3	39	144	D
022 Containment Cooling									X			**A3.01** – Initiation of safeguards mode of operation	4.1	40	135	D
025 Ice Condenser												**Not Selected**	**N/A**			
026 Containment Spray	X											**K1.01** - ECCS	4.2	41	78	D
039 Main and Reheat Steam								X				**A2.04** – Malfunctioning steam dump	3.4	42	202	D
059 Main Feedwater									X			**A3.03** – Feed water pump suction flow pressure	2.5	43	195	D
				X								**K4.16** – Automatic trips for MFW pumps	3.1	44	789	N
061 Auxiliary/Emergency Feedwater							X					**A1.04 changed to A1.01** – S/G level	3.9	45	270	D

	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#	QID	Type
062 AC Electrical Distribution											X	**2.4.35** – Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	3.8	46	790	N
		X										**K2.01** – Major system loads	3.3	47	316	D
063 DC Electrical Distribution			X									**K3.02** – Components using DC control power	3.5	48	86	D
064 Emergency Diesel Generator		X										**K2.01** – Air compressor	2.7	49	791	N
	X											**K1.05** – Starting air system	3.4	50	792	N
073 Process Radiation Monitoring					X							**K5.01** – Radiation theory, including sources, types, units, and effects	2.5	51	672	R
076 Service Water										X		**A4.02** – SWS valves	2.6	52	793	D
							X					**A1.02** – Reactor and turbine building closed cooling water temperatures	2.6	53	794	N
078 Instrument Air	X											**K1.03 changed to K1.02** – Service air	2.7	54	535	D
103 Containment										X		**A4.06** – Operation of the containment personnel airlock	2.7	55	795	D
K/A Category Point Totals:	3	3	2	3	3	2	2	2	2	3	3	Group Point Total:			**28**	

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

QID: 0781 **Rev:** 0 **Rev Date:** 9/09/2009 **Source:** New **Originator:** S. Pullin

TUOI: A1LP-RO-ICS **Objective:** 26 **Point Value:** 1

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

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

Description: Knowledge of the operational implications of the following concept as they apply to the RCP: The dependency of RCS flow rates upon the number of operating RCP's

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

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

Group: 1 **SRO Imp:** 3.0 **SRO Select:** Yes **Taxonomy:** C

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

Given:
Plant 60% power
All RCPs are in service
"A" OTSG BTU LIMIT (K07-E2) alarm is received

What is the most likely cause of the alarm?

A. "A" Thot temperature instrument failing high

B. "A" Feed water temperature instrument failing high

C. "A" OTSG pressure instrument failing low

D. "A" RCS flow instrument failing low

Answer:

D. "A" RCS flow instrument failing low

Notes:

D. is correct RCS flow has the largest input to BTU limit
A. is incorrect although the Thot feeds this alarm the instrument is failing in the wrong direction
B. is incorrect although the Feed water instrument feeds this alarm the instrument is failing in the wrong direction
C. is incorrect although the OTSG instrument feeds this alarm the instrument is failing in the wrong direction this one is hard to figure out due to BTU limits is looking at 35 f of superheat as SG pressure goes down with RCS flow and the other instruments staying the same superheat is getting higher

---

References:

1203.012F change 028
STM 1-64 rev 10

History:

New for 2010 RO/SRO exam

Location: C13

Device and Setpoint: N/A

A OTSG
BTU
LIMIT

Alarm: K07-E2

1.0 OPERATOR ACTIONS

1. Verify ICS is not raising load.

2. Check for possible instrument failure.

   A. IF there is an ICS input signal failure,
   THEN GO TO ICS Abnormal Operation (1203.001).

   B. IF failure indicated,
   THEN select an alternate instrument.

3. IF alarm occurs during End of Cycle T-ave reduction,
   THEN determine Main Steam superheat using 1102.004 Attachment Q.

   A. IF Main Steam superheat drops to 35°F,
   THEN stop Tave reduction AND consult Rx Engineering.

NOTE

The parameter causing the BTU limit alarm may not be readily apparent. Other indications such as cross limits or feedwater-reactor limited may help determine the cause.

4. IF valid BTU limit condition exists,
   AND NOT due to Tave reduction,
   THEN either raise reactor power or lower feedwater demand (or both) as necessary to clear alarm.

5. IF necessary,
   THEN initiate steps to repair ICS or input transmitters.

2.0 PROBABLE CAUSES

NOTE

The BTU limit is derived from SG heat capacity and superheat considerations.

1. BTU Limit = $(T_{hot} + FW_{temp} + Press_{SG} - 200)$ RC flow (%)

3.0 REFERENCES

Schematic Diagram Annunciator K07 (E-457) valves and pump will return to the mode of control previously described.

2.6.3.1 Feedwater Pump Control.

With one FW Pump running. the Main FW Block valves are closed and the crossover valve is open. The 70 psid setpoint is being compared to the low auctioneered ΔP signal. The ΔP error signal is used to adjust the respective main feedwater loop demand signal to adjust pump speed to keep the lowest ΔP at setpoint.

When both FW Pumps are running with the crossover valve closed and both main block valves closed, each FW Pump is controlled by its own individual loop ΔP summed with its loop demand signal.

If both FW Pumps are running, with the crossover valve closed, and both main block valves open, each FW Pump is controlled by its respective loop flow error summed with its feedforward loop demand signal.

A characteristic of the ICS is that there are numerous tie-back schemes which enable the ICS to have "bump-less" transfers. With the main feedwater pumps the tie-back scheme works well for the "A" pump controls but potentially can cause a feedwater transient when placing the "B" pump in Auto. When both feed pumps are in HAND with the main block valves closed and the cross-tie valve open, the selected ΔP controller looks at the status of the "A" pump to determine which manual demand to track. If the "A" pump is latched, the selected ΔP controller will have been tracking the "A" pump demand signal. If the "A" pump is tripped, manual demand for the "B" pump will have been tracked. Thus, if the "B" pump is placed in AUTO first with the "A" pump just latched or latched and rolling at minimum speed, the "B" pump would be driven down toward the minimum demand of the "A" pump. To address this idiosyncrasy, a caution was placed in the Condensate, Feedwater and Steam System Operation procedure which states: "With both MFW Pumps in manual and the Feedwater Pumps Disch Crosstie (CV-2827) open, placing "B" MFW Pump in AUTO with a significant difference in demand signals between "A" and "B" MFW Pumps will cause a feedwater transient."

2.6.4 BTU Limits.

The purpose of BTU limits is to monitor for a minimum of 35°F of superheat in the steam leaving the OTSG. To insure that moisture does not carryover from the OTSG to the turbine generator, it is desirable to have a minimum of 35°F of superheat in each pound mass of steam. ICS monitors the superheat of the steam indirectly by monitoring four parameters and calculating the maximum loop feedwater flow allowable. If the loop feedwater demand is greater than the calculated limit, a BTU limit alarm is sounded to alert the operator. The four parameters used in the BTU limit calculation are:

Selected T_H

Individual OTSG steam pressure

Loop feedwater temperature

RCS flow in that loop.

Each BTU limit calculator takes these four parameters and determines what the maximum loop feedwater demand is that will not drop superheat to < 35°F. (Refer to figure 64.25)

BTU Limit = [(T_H + OTSG press + FW Temp) -200] × RCS Flow %.

RCS flow changes have the largest effect on the calculation, and note that the limit is lowered by either RCS flow, T_H, or feedwater temperature being lowered.

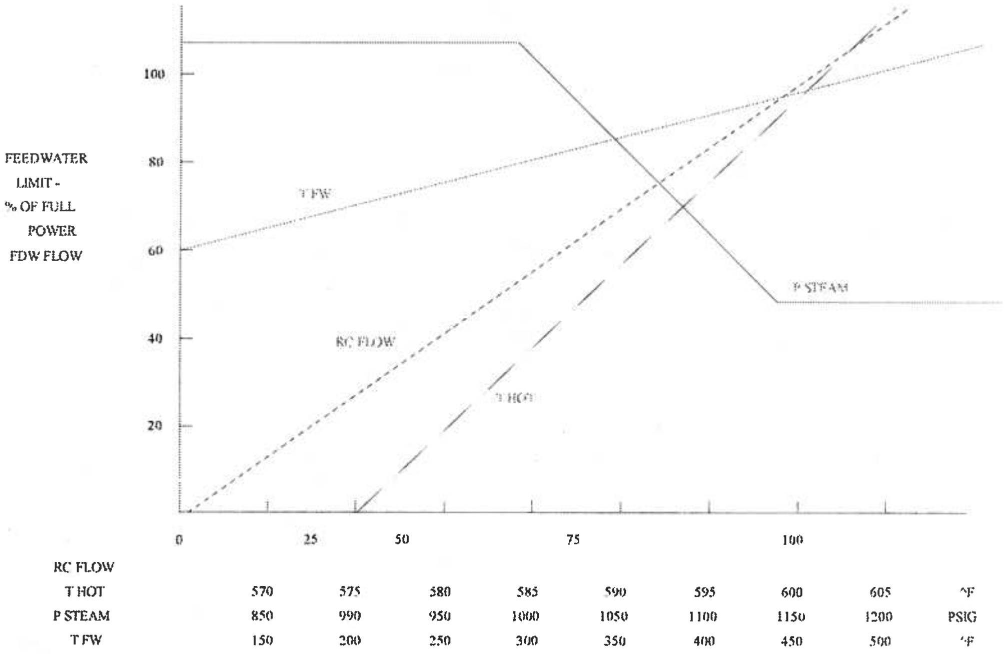


FIGURE 64.25: BTU LIMIT CURVE

Lowering feedwater temperature means that more of the primary heat is used to raise the feedwater to saturation temp. Therefore, less energy is available for superheat. The highest value that steam temperature out of an OTSG can be is to approach T_H . Therefore, if T_H decreases and OTSG saturation temperature is constant, superheat would decrease. If a constant feedwater flow to the OTSG is maintained, less RCS flow means less BTU of heat available to an OTSG and therefore superheat would decrease. If OTSG pressure increases, then saturation temperature will increase, if steam outlet temperature is constant, then superheat will decrease.

2.6.5 High Level Limit

The purpose of high level limit is to prevent flooding aspirating steam ports in the OTSG. Operate level for each OTSG is compared to the high level limit setpoint (90%) and an error signal is generated. If that signal is less than the loop feedwater flow error signal, then the low

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

QID: 0782 **Rev:** 0 **Rev Date:** 9/09/2009 **Source:** Modified **Originator:** S. Pullin

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

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

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

Description: Ability to manually operate and/or monitor in the control room: RCP cooling water supplies

K/A Number: A4.08 **CFR Reference:** 41.7/45.5 to 45.8

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	Yes	**Taxonomy:**	C

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

Given:
- Plant heat up in progress from refueling outage.
- P-32C and P-32D RCPs are running.
- Seal injection block CV-1206 is in override for testing
- Seal injection flow has been balanced and is in auto at 16 gpm total flow.
- Non-nuclear ICW to RCP motor cooling flow is 200 gpm.
- Nuclear ICW to RCP seal cooling flow is 35 gpm.
- RCS loop A & B cold leg temps are 275°F.
- RCP lift oil pressure is 1800 psig.

A start of RCP P-32A is attempted but is unsuccessful. Why?

A. Nuclear ICW to RCP seal cooling flow is low.

B. Seal injection flow is low.

C. RCP lift oil pressure is low.

D. RCP motor cooling flow is low.

Answer:

D. RCP motor cooling flow is low.

Notes:

D. is correct to satisfy the starting interlock RCP motor cooling flow needs to be >250 gpm
A is incorrect, nuclear ICW to RCPS is greater than 30 gpm.
B is incorrect, seal injection flow is greater than 3 gpm to each RCP.
C is incorrect, RCP lift oil pressure is >1750 psig

References:

1103.006 change 032

History:

Modified from QID 559
Selected for 2010 RO/SRO exam

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

QID: 0559 **Rev:** 1 **Rev Date:** 12/8/06 **Source:** Direct **Originator:** Cork/Possage

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

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

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

Description: Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: RCP bearing lift oil pump.

K/A Number: K1.13 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 2.5 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** ☐ **SRO:** ☐

Parent Question

Given:
- Plant heatup in progress from refueling outage.
- P-32A and P-32B RCPs are running.
- Seal injection flow has been balanced and is in auto at 16 gpm total flow.
- Non-nuclear ICW to RCP motor cooling flow is 275 gpm.
- Nuclear ICW to RCP seal cooling flow is 35 gpm.
- RCS loop A & B cold leg temps are 370°F.
- RCP lift oil pressure is 1600 psig.

A start of RCP P-32C is attempted but is unsuccessful. Why?

A. Nuclear ICW to RCP seal cooling flow is low.

B. Seal injection flow is low.

C. RCP lift oil pressure is low.

D. RCS cold leg temps are low.

Answer:

C. RCP lift oil pressure is low.

Notes:

"C" is correct, lift oil pressure must be > 1750 psig for pump start interlock to be met.
"D" is incorrect, RCS cold legs must be greater than 375°F to start the fourth RCP, not the third.
"A" is incorrect, nuclear ICW to RCPS is greater than 30 gpm.
"B" is incorrect, seal injection flow is greater than 3 gpm to each RCP.

References:

1103.006, Chg. 026-01-0

History:

New for 2005 RO exam by Pullin, but not used. Modified version of 615.
New for 2007 RO Exam.

5.28 During cooldown, the following RCP limits apply:

- <271°F no more than two RCPs may be operated
- <166°F no RCPs may be operated

5.29 During heatup, the following RCP limits apply:

- <241°F no more than two RCPs may be operated
- <316°F no more than three RCPs may be operated, however due to hydraulic lift of the core, no more than three RCPs may be operated until RCS temperature is >430°F
- <106°F no RCPs may be operated

5.30 RCP motor and pump vibration limits are as follows:

- P-32B or D motor vibration; more than one channel >20 mils after startup stabilization
- P-32A or C motor vibration; more than one channel >0.8 in/sec after startup stabilization
- RC pump vibration; more than one channel >25 mils after startup stabilization

5.31 Plant startup conditions could result in exceeding the Steam Generator Design Limit of 60°F Tube to Shell ΔT (tubes hotter).

5.32 Simultaneous operation of the normal and Emergency HP Oil Lift Pump (P-63 and P-80) is undesirable. Reduced oil pressure and cavitation can occur.

6.0 SETPOINTS

The following conditions must be satisfied to start an RCP from the control room.

6.1 Rx power <22%.

6.2 RCP seal injection flow >3 gpm.
If <3 gpm, alarms RCP SEAL INJ FLOW LO (K08-A7).

RCP P-32A Seal Injection Flow (FS-1280)
RCP P-32B Seal Injection Flow (FS-1281)
RCP P-32C Seal Injection Flow (FS-1282)
RCP P-32D Seal Injection Flow (FS-1283)

6.3 RCP motor cooling flow >250 gpm (non-nuclear ICW).
If <250 gpm alarms RCP MOTOR COOLING FLOW LO (K08-E6).

P-32A MTR Air LO CLR ICW RTN Flow (PDIS-2260)
P-32B MTR Air LO CLR ICW RTN Flow (PDIS-2261)
P-32C MTR Air LO CLR ICW RTN Flow (PDIS-2262)
P-32D MTR Air LO CLR ICW RTN Flow (PDIS-2263)

6.4 RCP seal cooling flow >30 gpm (nuclear ICW).
If <30 gpm alarms RCP SEAL COOLING FLOW LO (K08-E7).

P-32A Seal CLR ICW RTN Flow (PDIS-2250)
P-32B Seal CLR ICW RTN Flow (PDIS-2251)
P-32C Seal CLR ICW RTN Flow (PDIS-2252)
P-32D Seal CLR ICW RTN Flow (PDIS-2253)

6.5 RCP start interlock on low oil reservoir level

6.5.1 Upper Reservoir Oil Level Low
-2.0″ for P-32A, C, and D
-1.6″ for P-32B

RCP A Upper Lube Oil Level Lo (LS-6535)
RCP B Upper Lube Oil Level Lo (LS-6536)
RCP C Upper Lube Oil Level Lo (LS-6537)
RCP D Upper Lube Oil Level Lo (LS-6538)

6.5.2 Lower Reservoir Oil Level Low
-1.5″ for P-32A, C, and D
-1.2″ for P-32B

RCP A Lower Lube Oil Level Lo (LS-6560)
RCP B Lower Lube Oil Level Lo (LS-6561)
RCP C Lower Lube Oil Level Lo (LS-6562)
RCP D Lower Lube Oil Level Lo (LS-6563)

6.6 Computer alarms on high and low oil reservoir level

6.6.1 Upper Reservoir Oil Level High
+2.0″ for P-32A, C, and D
+1.6″ for P-32B

6.6.2 Upper Reservoir Oil Level Low
-2.0″ for P-32A, C, and D
-1.6″ for P-32B

6.6.3 Lower Reservoir Oil Level High
+1.5″ for P-32A, C, and D
+1.2″ for P-32B

6.6.4 Lower Reservoir Oil Level Low
-1.5″ for P-32A, C, and D
-1.2″ for P-32B

6.7 RCP HP oil lift pressure >1750 psig.
If <1750 psig alarms RCP LIFT OIL TROUBLE (K08-C8)
(1000 psig for P-32B)

RCP P-32A HP Lift Oil Press (PS-6530).

RCP P-32B HP Lift Oil Press (PS-6526).

RCP P-32C HP Lift Oil Press (PS-6532).

RCP P-32D HP Lift Oil Press (PS-6533).

6.8 RCP reverse rotation <12.7 gpm return oil flow/pump start permitted
If >12.7 gpm alarms plant computer (not applicable for P-32B)

RCP P32-A REVERSE ROTATION Computer Alarm (FS6510)
RCP P-32A Reverse Rotation Starting Interlock (FS-6515).

RCP P32-C REVERSE ROTATION Computer Alarm (FS6512)
RCP P-32C Reverse Rotation Starting Interlock (FS-6517).

RCP P32-D REVERSE ROTATION Computer Alarm (FS6513)
RCP P-32D Reverse Rotation Starting Interlock (FS-6518).

6.9 If starting first RCP, RCS to SG Downcomer ΔT ≤50°F.

RC Loop A Cold Leg Temp (TS-1017)
RC Loop B Cold Leg Temp (TS-1045)

A Stm Gen Downcomer Temp (TI-2665)
B Stm Gen Downcomer Temp (TI-2615)

6.10 If starting third RCP, RCS temperature >241°F.

RC Loop A Cold Leg Temp (TS-1017)
RC Loop B Cold Leg Temp (TS-1045)

6.11 If starting fourth RCP, RCS temperature >430°F.

RC Loop A Cold Leg Temp (TS-1017)
RC Loop B Cold Leg Temp (TS-1045)

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

QID: 0796 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Knowledge of conditions and limitations in the facility license.

K/A Number: 2.2.38 **CFR Reference:** 41.7/41.10/43.1/45.13

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.5	**SRO Select:**	Yes	**Taxonomy:**	C

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

REFERENCE PROVIDED

Which of the following Boric Acid Addition Tank level and concentration versus RCS Tave would require entry into TRM 3.5.1 ?

A. 8,700 ppm Boron , BAAT level 36 inches , 400 F Tave

B. 9,500 ppm Boron , BAAT level 46 inches , 450 F Tave

C. 10,000 ppm Boron , BAAT level 50 inches , 500 F Tave

D. 12,000 ppm Boron , BAAT level 56 inches , 550 F Tave

Answer:

C. 10,000 ppm Boron , BAAT level 50 inches , 500 F Tave

Notes:

C. is correct due to the values fall below and to the right of reference curve TRM figure 3.5.1-1
A, B, and D are incorrect due to the values fall above and to the left of reference curve TRM figure 3.5.1-1

REFERENCE PROVIDED FOR THIS QUESTION

References:

1104.003 change 046
TRM 3.5.1 rev 16

History:

New for 2010 RO/SRO exam

TRM 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

TRM 3.5.1 Makeup and Chemical Addition Systems

TRO 3.5.1 The Makeup and Chemical Addition System shall be OPERABLE with the following requirements:

a. Two makeup pumps shall be OPERABLE except as specified in TS 3.5.2, "Emergency Core Cooling Systems (ECCS) - Operating," and TS 3.5.3, "Emergency Core Cooling Systems (ECCS) - Shutdown,"

b. The boric acid addition tank (BAAT) shall be OPERABLE, containing at least the equivalent of the boric acid volume and concentration requirements of TRM Figure 3.5.1-1, "Boric Acid Addition Tank Volume and Concentration Vs RCS Average Temperature" as boric acid solution with a temperature of ≥ 10°F above the crystallization temperature for the concentration in the tank, and

c. One boric acid pump associated with the BAAT shall be OPERABLE.

d. System piping and valves necessary to establish a flow path from the boric acid addition tank to the makeup system shall be OPERABLE and shall have a temperature of ≥ 10°F above the crystallization temperature for the concentration in the tank.

APPLICABILITY: MODES 1, 2, 3, and 4.

---NOTE---

Condition entry is not required when the flow path from the boric acid pump(s) to the Makeup Tank is unavailable during procedurally controlled activities.

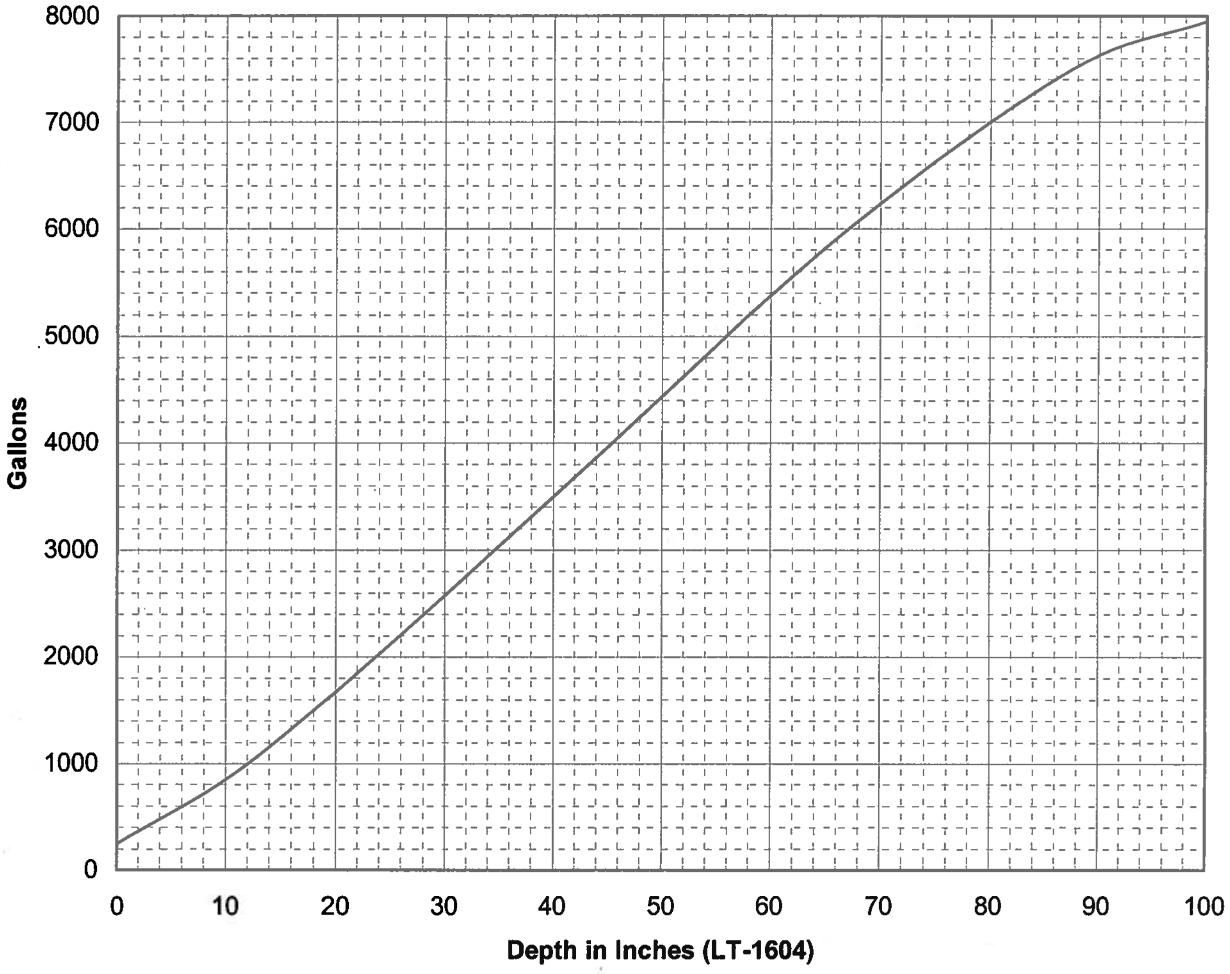
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of TRO not met.	A.1 Restore Makeup and Chemical Addition System to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Initiate a condition report to document the condition and determine any limitations for continued operation of the plant.	Immediately

ATTACHMENT H

Page 1 of 1

Volume of BAAT vs. Depth of Liquid



1.0 To calculate the BAAT (T-6) level drop corresponding to a certain feed volume:

1.1 Read initial BAAT level and determine initial volume from graph.

1.2 Subtract feed volume from initial tank level.

Example: It is desired to feed 530 gallons of boric acid.

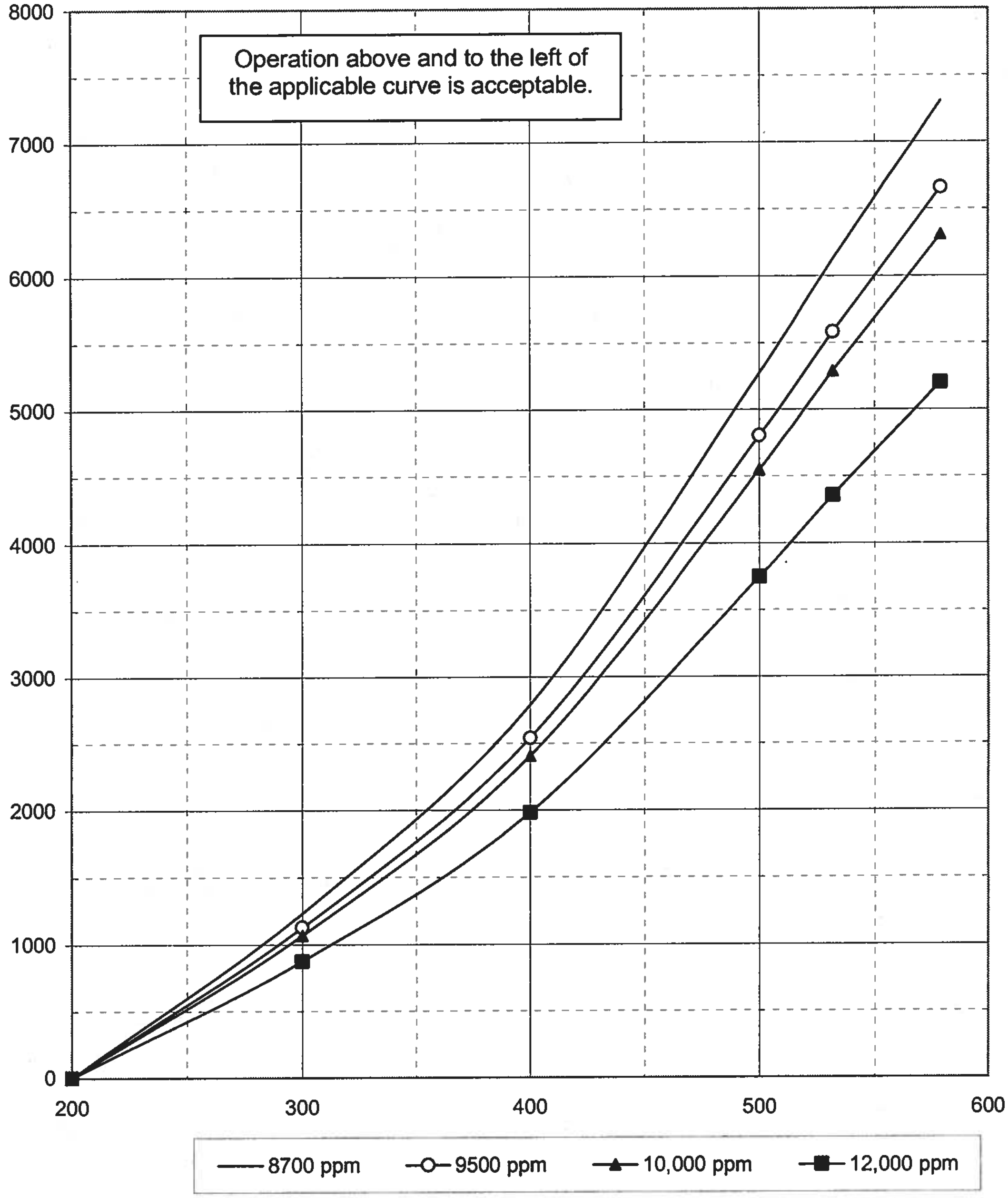
A. Initial BAAT level = 82″. (From graph, ~ 7100 gal.)

B. Initial volume - feed volume = 7100 - 530 = 6570 gal.

C. Final level, from graph, corresponding to 6570 gal. = ~ 74″.

ATTACHMENT G

BAAT Volume and Concentration Vs. RCS T-ave
(Ref. TRM Figure 3.5.1-1)



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

QID: 0259 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-MU **Objective:** 07 **Point Value:** 1

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

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

Description: Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Protection of ion exchangers (high letdown temperature will isolate ion exchangers)

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

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

Group: 1 **SRO Imp:** 2.9 **SRO Select:** Yes **Taxonomy:** K

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

What is the function of the temperature interlock associated with RCS letdown?

A. Prevents letdown fluid from flashing to steam when pressure is reduced by closing CV-1221 (letdown isolation).

B. Prevents exceeding letdown piping thermal limits by shutting CV-1213 & 1215 (letdown cooler inlet MOV).

C. Prevents degrading T36A/B resin by shutting CV-1221 (letdown isolation).

D. Prevents exceeding letdown cooler capacity by shutting CV-1213 & 1215 (letdown cooler inlet MOV).

Answer:

C. Prevents degrading T36A/B resin by shutting CV-1221 (letdown isolation).

Notes:

"A" is incorrect, this is the function of the letdown coolers.
"B" is incorrect, interlock doesn't close the inlets and piping limits will not be exceeded before the resin is damaged.
"C" is correct
"D" although the letdown cooler capacity is exceeded when temperature is exceeded the interlock doesn't close the inlet valves.

References:

1104.002 Rev 051-02-0
STM1-04 Rev 5

History:

Used in 1999 exam.
Selected for 2010 RO/SRO exam

5.0 LIMITS AND PRECAUTIONS

5.1 Do not start or continue to run a Makeup Pump (P-36A, B or C) with the RCS in a solid water condition except as directed by emergency procedures.

5.2 Maintain Makeup Tank (T-4) pressure above 10 psig.

{4.3.3}

5.3 Restricting flow through an operating Makeup Pump (P-36A, P-36B or P-36C) to < 55 gpm can cause pump damage. Accounting for instrument accuracy, HPI should be maintained such that flow through at least one HPI line is maintained ≥ 90 gpm when recirc valve is closed.

5.4 Maximum flow through a makeup pump is 500 gpm for normal operation.

5.5 Maximum flow through a makeup pump is 525 gpm for emergency operation.

5.6 Maximum flow through a Letdown Cooler (E-29A & B) is 87.5 gpm per cooler.

5.7 Allowing flow through a primary Makeup Filter (F-3A or F-3B) in excess of 80 gpm can lead to filter damage.

5.8 Allowing flow through a purification DI in excess of 123 gpm can compact the resin and restrict letdown flow.

5.9 Restricting flow through a Purification Demineralizer (T-36A or T-36B) to < 25 gpm can cause channeling of resin and reduce efficiency of demineralizer.

5.10 Maximum purification demineralizer inlet temperature is 135°F.

5.11 Placing a purification demineralizer into service that has not been borated will result in a reduction in RCS boron concentration.

5.12 Ensure clean waste system is aligned to receive waste from letdown system prior to positioning Letdown 3-Way Valve (CV-1248) to BLEED. Otherwise letdown line will over-pressurize.

5.13 When makeup pumps are subject to HPI actuation, maintain MU tank pressure/level relationship within limit of Exhibit A. Exceeding the limit reduces the time available for isolating the MU tank after HPI actuation.

5.14 When venting the makeup tank, the waste gas system shall be aligned to compress the gas for storage unless samples indicate negligible activity in the makeup tank.

5.15 MU Tank T-4 Relief Valve (PSV-1249) is not designed to relieve water. For uncontrollable high MU tank water level, open MUT Vent Valve (CV-1257).

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

QID: 0786 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** Modified **Originator:** S. Pullin

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

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

System Number: 055 **System Title:** Residual Heat Removal System

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

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

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	Yes	**Taxonomy:**	K

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

Given:

- Plant is in Mode 6
- P-34B Decay Heat pump is running

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

A. A-1 voltage of 2475 volts

B. A-2 voltage of 2475 volts

C. B-5 voltage of 428 volts

D. B-6 voltage of 428volts

Answer:

D. B-6 voltage of 428volts

Notes:

"B" Decay Heat Removal Pump is powered from A-4 via A-2. An undervoltage on the A buses or B buses will trip A-409 (A4 feeder breaker). The undervoltage setpoint for A-4 is 2450 volts. The undervoltage setpoint for B-6 is 429 volts. Therefore, "a","b", and "c" are incorrect.

References:

OP-1107.002 Change 025

History:

Modified from QID 0293
Selected for 2010 RO/SRO Exam

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

QID: 0293 **Rev:** 2 **Rev Date:** 12/14/06 **Source:** Direct **Originator:** D Slusher

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

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

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

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

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

Tier:	2	**RO Imp:**	3.0	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	Ap

Question: **RO:** ☐ **SRO:** ☐

Given:

- Plant is in Mode 5
- P-34A Decay Heat pump is running

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

Parent Question

A. A-1 voltage of 2425 volts

B. A-2 voltage of 2425 volts

C. B-5 voltage of 435 volts

D. B-6 voltage of 435 volts

Answer:

A. A-1 voltage of 2425 volts

Notes:

"A" Decay Heat Removal Pump is powered from A-3 via A-1. An undervoltage on the A buses or B buses will trip A-309 (A3 feeder breaker). The undervoltage setpoint for A-3 is 2450 volts. The undervoltage setpoint for B-5 is 429 volts. Therefore, "b", "c" and "d" are incorrect.

References:

1107.002, Chg. 023-00-0

History:

Developed for 1999 exam.
Selected for 2005 Jon Gray RO re-exam.
Modified and USED in 2007 RO Exam.

5.6 Diesel generator load limits:

- Limit diesel generator load to ≤2750KW continuous load.
- Additional loads, beyond those automatically sequenced on, may be started if needed provided that continuous diesel generator load is maintained <2750KW.

5.7 When racking out 4160V bus breakers, personnel shall use the protective equipment specified in Electrical System Operations (1107.001), Exhibit I, Electrical Safety Requirements.

5.8 Opening the DC Control Power Breaker in the following breaker cubicles results in loss of Bus-Protective Relays:

- A-309 A1 Feed to A3
- A-409 A2 Feed to A4

5.9 Load Center Transformers are NOT capable of supplying full load of two buses when crosstied. Loading must be restricted.

5.10 When racked down and disengaged from the lifting mechanism, 4160V breakers no longer meet seismic requirements.

5.11 Load Center Breaker Handling Jib Cranes do NOT meet seismic requirements when NOT secured in the stowed position.

5.12 Motor Control Centers and Load Centers require a seismic evaluation by Design Engineering to determine operability if the following conditions are exceeded:

- Two breakers removed
- One breaker removed and two breakers racked out
- Three breakers racked out

5.13 All 4160V TEST breaker operations, including racking up or down, shall be performed by Electrical/Relay Department personnel.

5.14 Except in an emergency, all 4160V breaker removal and reinstallation operations shall be performed by Electrical/Relay Department personnel.

6.0 SETPOINTS

6.1 Bus A3 and A4 undervoltage: 2450V nominal

6.2 Bus B5 and B6 undervoltage: 429.6V nominal

QID: 0783 **Rev:** 0 **Rev Date:** 9/10/2009 **Source:** Modified **Originator:** Possage

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

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

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

Description: Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Valves

K/A Number: K6.10 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 3

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

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

Given

Degraded Power condition is present with a LOCA

RCS pressure 1580 psig

Reactor Building pressure 2 psig

Diesel Generator #1 failed to start

No other failures are present

Which component would be automatically actuated to it ES position?

A. "B" Letdown cooler outlet CV-1216 would close

B. RCP Motor Air and Lube Oil Cooling Isolation valve CV-2221 would close

C. BWST outlet valve CV-1408 would open

D. HPI Pump P-36B would start

Answer:

C. BWST outlet valve CV-1408 would open

Notes:

C. Is the correct answer. CV-1408 is ES actuated open and will have power available to open..
A. Is incorrect, although it should close on ES, with the given information, it would not have power available to close.
B. Is incorrect, CV-2221 would have power to close but would not close until reactor Building pressure reached the setpoint
for Channels 5 & 6
D. Is incorrect, B HPI pump would not start unless there was a failure of P-36C and the student was given that no other failures are present.

References:

STM 1-04 Rev. 9
STM 1-43 Rev. 12

History:

Modified from exma bank ANO-OpsUnit1-06296
Selected for 2010 RO/SRO exam.

Editing ANO:OpsUnit1 :ANO-OpsUnit1-06296 You have 1 question in your editor cart.

12 pt

Question:
Given

Degraded Power condition is present with a LOCA

Diesel Generator #1 failed to start

No other failures are present

ANO-OpsUnit1-06296
Parent

Which component would be automatically actuated to it ES position/status if RCS pressure subsequently dropped below 1590 psig?

A. "B" Letdown cooler outlet CV-1216 would close
B. Penetration Room Ventilation fan VEF-38A would start
C. HPI Pump P-36B would start
D. Letdown coolers outlet CV-1221 would close

Answer: ○ A. ○ B. ○ C. ● D.

Select: ● Open Reference Question ○ Closed Reference Question

Select: ○ Handout Required with Exam ● Handout Not Required with Exam

Point Value: 1.0 points

Cognitive Level:

○ 1: Fundamental Knowledge or Memory

● 2: Comprehension or Analysis

○ 3: Synthesis or Evaluation

Question Comments:

Review Comments: *(Edited with Question Reviewer.)*

2.2.3 Leak Detection

Tube to shell leaks inside the LD Coolers is detected through radiation elements monitoring the Nuclear ICW System. Because primary pressure is higher than ICW pressure when the coolers are in service, leakage will be from the primary to the ICW side of the coolers. Small leaks will be detected by process monitor RE- 2236. Larger leaks may cause ICW Surge Tanks to overflow. Corrective actions for LD Cooler leaks are addressed in annunciator corrective actions and the Excess RCS Leakage Abnormal Operating Procedure. For additional information on the ICW System see STM 1-43.

2.3 Letdown Heat Exchanger Outlet Valves, CV-1214 and CV-1216

These motor operated valves are located in the Reactor Building and on the outlet side of the letdown coolers. They are operated from Control Room Panel C18. During an Engineering Safeguards Actuation Signal (ESAS) Channel 1 Initiation, both valves close automatically. During ES actuation, these valves provide reactor building isolation. The valves are of the split disc gate valve type (system pressure leaking past one disc aids in seating the other disc). On the bottom of each valve is a connection for local leak rate testing. The testing connection allows pressure to be placed between the valve discs to assure positive seating.

2.4 N-16 Expansion Tank

The Expansion Tank (refer to figure 4.06) is an enlargement from a 2 1/2" line to a 12" line and back to a 2 1/2" line. This expansion increases the transport time and allows for the decay of Nitrogen 16 (N16), a high energy Gamma emitter. N16 is produced in the reactor as an activation product and has a very short (~7 second) half life. Increasing the LD transport time has the positive effect of reducing radiation levels outside containment. The tank is located in the letdown cooler room.

2.5 Letdown isolation valve CV-1221

CV-1221 is the first valve outside of the containment building and is operated from Control Room Panel C16. CV-1221 is a motor operated gate valve interlocked with Temperature Switch TS-1221. At an increasing letdown temperature of 135F CV-1221 will close. This closure is provided to protect the resins of the purification demineralizers (T-36A/B) from being exhausted prematurely due to high temperature.

CV-1221 is also closed by actuation of ES Channel 2 and provides Letdown isolation in the Upper North Piping Penetration Room (UNPPR). CV-1221 is the only single line isolation in the letdown flow path.

Instructions for reopening CV-1221 after closure due to high LD temperature trip are included in OP 1104.02, "Makeup and Purification System Operation". Recovery from closure consists primarily of correcting the cause of the overheating, bypassing the LD demineralizers until flow can be reestablished and temperatures brought down to normal and then returning the proper DI to operation.

indication on panel C09 (FI-2222) and provides standby pump auto start signal through FS-2222 discussed earlier in this section.

CRD return temperature indication and high alarm functions are provided by TE-2222 located on the return line. CRD return temperature can be read on panel C09 using TI-2222 or on the plant computer (T2222).

TS-2222 will cause annunciator alarm K08-B1 "CRD Cooling Return Temp Hi" to alarm when return temperature reaches setpoint of > 160°F. This alarm also indicates inadequate SW cooling of the Non-Nuclear ICW cooler.

The CRD cooling water return line combines with the RCP Motor Air and LO Coolers return line to form a common 8-inch return line. CRD and RCP ICW return line exits the RB at penetration #60 in the USPPR. This common return line like the supply line is provided with isolation valves for system isolation during an ESAS event. Both RB isolation valves are 8-inch, motor-operated gate valves. Inside isolation valve, CV-2221 receives a closed signal from ES channel 6 through HS-2221 located on panel C16. CV-2221 is powered from vital bus B61, breaker B-6192. Outside isolation valve, CV-2220, receives a closed signal from ES channel 5 through HS-2220 located on panel C-18. CV-2220 is powered from vital bus B52, breaker B-5221.

Non-Nuclear ICW flow from the CRD and RCP's tie into the common 10-inch return header to ICW cooler E-28A. Return flow from the Isophase Bus Cooling coils also ties into this return line.

2.6.2 MFP / RCP ICW Supply Line

(Refer to Figures 43.03 and 43.04)

The first main supply line, which taps off the discharge line of P-33A, is a 10-inch line, which provides cooling water flow to the following components:

* Main Feed Water Pump L.O. Coolers.
* Reactor Coolant Pump (RCP) Motor Air Coolers.
* RCP Motor Bearing L.O. Coolers.
* RCP High Pressure Lift Oil System Coolers.
* RCP Backstop L.O. System Coolers. Note: RCP P-32B does not utilize a backstop lube oil system.

This supply line is provided with a means to isolate ICW flow to the MFP / RCP by closing isolation valve ICW-11. ICW-11 is a 10-inch butterfly valve located in the Main Chiller room. Downstream of ICW-11 the line splits into two 10-inch lines which provide cooling water to the MFP/RCP and the other line is used to divert or bypass ICW flow to the MFP's/RCP's during shutdown conditions. ICW flow is diverted back to the Non-Nuclear ICW header through bypass valve ICW-23. This valve is normally throttled during plant operation to balance ICW flow.

The 10-inch supply line to the MFP's and RCP's splits into two separate lines. The first supply line provides cooling water flow to

Design conditions on the shell side are: 150 psig, 300 °F, 110000 lbm/hr, 220 gpm

2.22 HPI Valves CV-1227, CV-1228, CV-1278, CV-1279, CV-1219, CV-1220, CV-1284, CV-1285

These eight MOV's can be operated from Panels C16 and Cl8. They are automatically opened on Engineering Safeguard Actuation. The HPI valves are actuated from the same channel as the associated HPI pump. ES channel 1 opens CV-1219, CV-1220,CV-1278, and CV-1279. ES channel 2 opens CV-1227, CV-1228, CV-1284, and CV-1285. High pressure injection enters the RCS on the discharge side of the reactor coolant pumps. Refer to figure 4.23.

Each injection line has flow instrument installed which has a readout on C16 and C18. The control room indicator has a low flow cutoff at 10 gpm to prevent indication when there is no flow. This indication is due to errors in loop flow.

There are high flow setpoints of 450 gpm total HPI flow per train and >140 gpm on each injection line. This warns the operator that insufficient flow may be going to the core due to high flow through an HPI line that has a break. A low flow alarm at 200 gpm is to warn operator of minimum flow requirements.

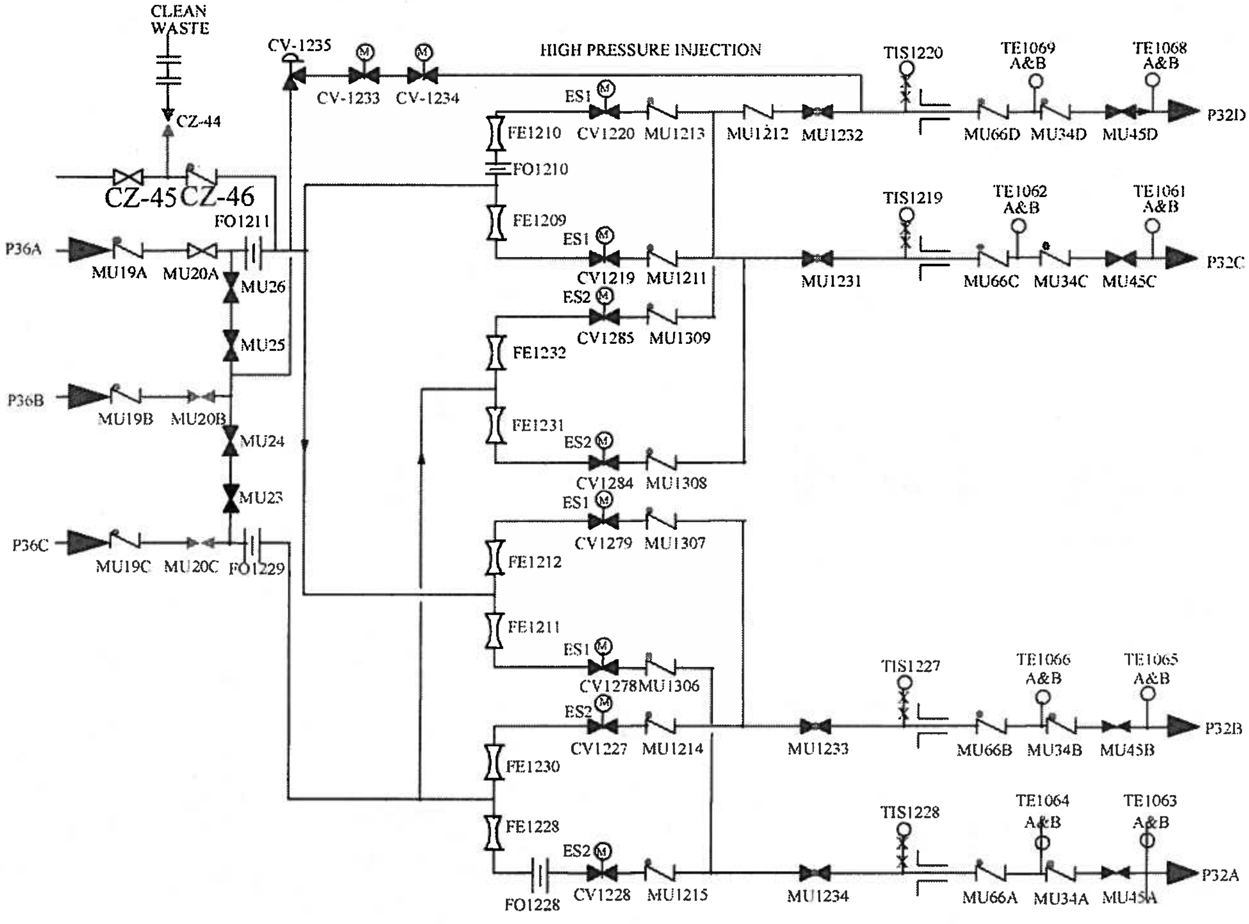


FIGURE 04.23: HPI LINE INSTRUMENTS & PIPING

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

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

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

Section: 3.5 **Type:** Containment Integrity

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

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

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

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

Group: 2 **SRO Imp:** 3.4 **SRO Select:** Yes **Taxonomy:** Ap

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

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

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

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

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

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

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

Answer:

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

Notes:

"D" is correct with Quench Tank pressure rise less than or equal to 1 psig.
All other choices contain greater than 1 psig pressure rise which indicates nitrogen is still being vented to the Quench Tank.

References:

1103.005, Chg. 036

History:

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

NOTE

Venting and bubble formation is considered complete when both of the following conditions are met:

- A three-minute blow through the ERV results in Quench Tank pressure rise of ≤ 1 psig.
- A saturation pressure/temperature relationship exists in the PZR.

7.2.9 WHEN RC pressure rises to near 70 psig, THEN repeat steps 7.2.5 through 7.2.7 as necessary until bubble forms.

7.3 System Pressurization

CAUTION

The pressurizer spray block valve shall remain closed until the ΔT between the pressurizer and the RCS is ≤ 250°F to prevent exceeding design criteria of the spray and surge lines.

7.3.1 WHEN RCS is > 200°F, THEN open Spray Block Valve (CV-1009).

7.3.2 Spray valve and heater banks may be cycled as necessary for heat-up and pressurization as outlined in Plant Startup (1102.002), "Heatup and Pressurization to ≤ 350° & ≤ 500 PSIG" section.

{4.3.7}

NOTE

ERV Isolation (CV-1000) is subject to binding if heatup continues with CV-1000 closed.

7.3.3 Verify ERV Isolation (CV-1000) remains open during heatup.

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

QID: 0787 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** **Originator:** S. Pullin

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:** CFR: 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:** 35 **SRO:** 35

Given:

- 80% power,
- P33A and P33B ICW pumps in service.
- P33C (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 all 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 all 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:

C. Loss of Nuc ICW, open all ICW cross connect valves CV-2238, CV-2239, CV-2240 and CV-2241

Notes:

"C" is correct P33C supplies the 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.
"A" is incorrect due to Non Nuc ICW loads were never lost
"B" is incorrect due to Non Nuc ICW loads were never lost
"D" is incorrect due to procedure has you open the valves and not close them

References:

OP-1104.028 Change 026

History:

New question, selected for 2010RO/SRO exam.

1.0 PURPOSE

To provide procedure for operation of the intermediate cooling water system.

2.0 SCOPE

This procedure is provided for the startup, normal operation, emergency operation, and shutdown of the ICW and CRD cooling water systems.

This procedure contains Temp Mod controls in Attachment B, Temporary Installation of a Service Water Outlet at ICW Cooler E-28C.

3.0 DESCRIPTION

The ICW system is composed of two independent closed loop cooling systems which provide an intermediate cooling water barrier between the cooled components and the Service Water system. The purpose for closed loop systems is to prevent direct contact between a radioactive system and the Service Water system.

The system uses three parallel recirculation pumps (P-33A, B, C) and three parallel heat exchangers (E-28A, B, C). The pumps circulate the ICW to various components and back through the heat exchangers which are cooled by Service Water running through the tubes. P-33A and E-28A provide cooling for the Non-Nuclear loop components and P-33C and E-28C provide cooling for the nuclear loop components. P-33B and E-28B are swing components which can be used by either loop. In normal alignment, Non-Nuc ICW is cooled by Loop II Service Water and Nuc ICW is cooled by Loop I Service Water. Both ICW loops are continuously monitored by radiation detectors to warn operators of radioactivity in the ICW system.

The Non-Nuclear loop normally has a higher activity level due to activation of ICW chemicals while over the Rx vessel head in the CRD cooling loop. There is an ICW Surge Tank (T-37A & B) associated with each loop that provides NPSH for pumps and a surge volume for the loops. This is also where makeup is added to the system from the condensate transfer system.

The CRD cooling system has two parallel recirculation pumps, CRD Pumps (P-79A and P-79B), which take a suction on the Non-Nuclear ICW loop downstream of E-28A and provide cooling water to CRD motors. It returns to the Non-Nuclear loop on the inlet to E-28A.

The RCP Seal Cooling Pumps (P-114A & B) provide added system pressure and flow for RCP seal cooling. The pumps take a suction on the Nuclear Loop inside the Reactor Building and return to the Nuclear Loop inside the Reactor Building.

3.1 Nuclear Loop cools:

- Spent Fuel Coolers (E-27A & B)
- RCP Seal Return Coolers (E-26A & B)
- Waste Gas Compressors (C-9A & B)
- Waste Gas Compressor Aftercoolers (E-40A & B)

- Vacuum Degasifier Seal Water Cooler (E-53)
- Pressurizer Sample Cooler (E-30)
- Steam Generator Sample Cooler (E-31A)
- Letdown Coolers (E-29A & B)
- RCP Seal Water Coolers (E-25A, B, C, D)

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:

- E-29A HS-2216 for Letdown Cooler Inlet Valve (CV-2216) and RC to Letdown Cooler E-29A (CV-1213)
- E-29B 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:

- E-29A HS-2216 for Letdown Cooler Inlet Valve (CV-2216) and RC to Letdown Cooler E-29A (CV-1213)
- E-29B 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 IF letdown isolated on high temperature,
THEN 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: 0788 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin

TUOI: A1LP-RO-RPS **Objective:** 5 **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 that a loss or malfunction of the PZR PCS will have on the following: RPS

K/A Number: k3.02 **CFR Reference:** 41.7 /45.6

Tier:	2	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	K

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

Given:

- 100% power,
- "A" MFW Pump trips

- PZR Spray valve (CV-1008) will not open.

What effect would this pressurizer control system malfunction have on the plant?

A Reactor trip due to AMSAC

B. Reactor trip due to anticipatory trip from RPS on loss of MFW pumps

C. Reactor trip due to High Power/Imbalance/Flow

D. Reactor trip due to High RCS Pressure

Answer:

D. Reactor trip due to High RCS Pressure

Notes:

A is incorrect because total feedwater flow will remain above trip setpoint
B is incorrect because only one MFW pump is tripped
C is incorrect because the flow in this coice refers to RCS flow
D is correct, without the spray valve opening RCS pressure will rise to the trip setpoint

---

References:

OP-1202.001 Change 31

History:

New selected for 2010 RO/SRO exam.

ENTRY CONDITIONS

- An automatic Rx trip or DSS trip.
- Failure of RPS to trip the Rx upon reaching a limit listed below:

- High power	104.9%
- High power/pumps	one pump per loop .. ≥ 55% **OR** 0 pumps in one loop .. ≥ 0%
- High power/imbalance/flow	COLR Figure
- High RCS temp	≥ 618 °F (T-hot)
- High RCS press	≥ 2355 psig
- Low RCS press	≤ 1800 psig
- Variable low RCS press	COLR Figure
- High RB press	≥ 18.7 psia
- Turbine trip	Rx power ≥ 43% **AND** Turbine is tripped
- Both MFW pumps trip	Rx power ≥ 9% **AND** both MFW pumps tripped.

- PZR level dropping < 100",
  AND
  no indication of recovery.
- PZR level > 290".
- Any MSIV closure at power.
- Either SG level < 15" or > 95%,
  AND
  no indication of recovery.
- A system degradation that requires manual Rx trip based on operator judgment.
- Abnormal Operating Procedure requirement.
- **IF** a system degradation occurs while shutdown, above DHR operation,
  THEN perform applicable steps.

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

QID: 0784 **Rev:** 0 **Rev Date:** 9/10/2009 **Source:** New **Originator:** Possage
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 the effect of a loss or malfunction of the following will have on the RPS: Permissive circuits

K/A Number: K6.10 **CFR Reference:** 41.7 / 45.7
Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 3
Group: 1 **SRO Imp:** 3.5 **SRO Select:** Yes **Taxonomy:** K

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

Given:

The plant is at 100% power

I&C is troubleshooting RPS

"B" RPS is in Manual Bypass

The Shutdown Bypass 5% bistable in Channel "A" has been pulled from the cabinet.

What would be the effect of a failure in the "B" RPS permissive circuitry that caused a short which de-energizes the "B" RPS Cabinet?

A. RPS would be in a 2 out of 3 coincidence trip logic

B. RPS would be in a 2 out of 2 coincidence trip logic

C. Reactor Trip would occur

D. High Flux trip bistable tripped in Channel "A"

Answer:

C. Reactor Trip would occur

Notes:

C. Is correct. The conditions given would result in the "A" Channel being tripped, when "B" is de-energized it would also be tripped and make up the logic to trip the reactor.
A and B are incorrect because the logic to trip the reactor has already been met.
D is incorrect, pulling the Shutdown Bypass 5% bistable would not cause a high flux trip bistable to trip in RPS.

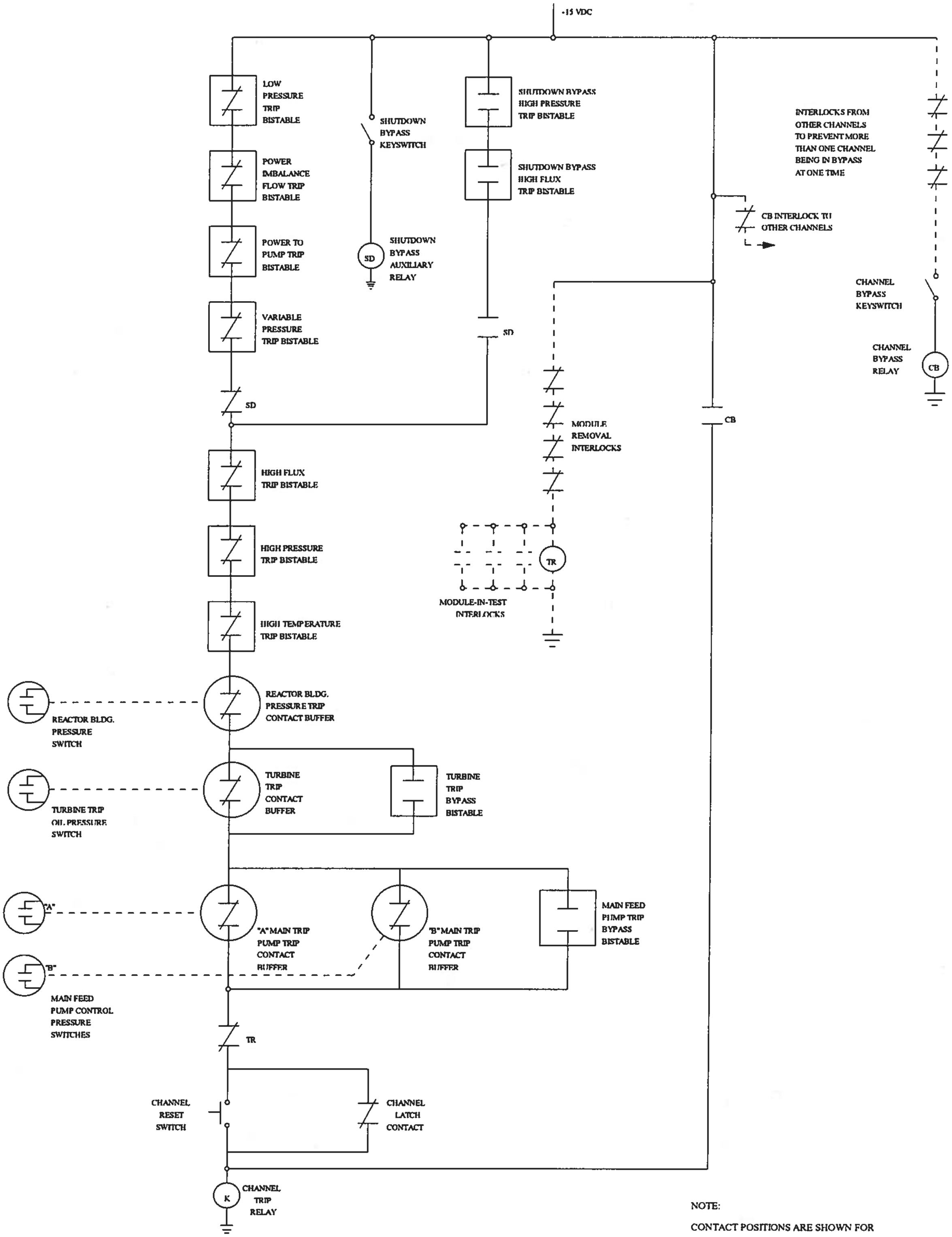
References:

STM 1-63 Rev. 7

History:

Modified from Exam Bank ANO-OPS1-1670
Selected for 2010 RO/SRO exam

FIGURE 63.01: CHANNEL TRIP CONTACT STRING



NOTE:

CONTACT POSITIONS ARE SHOWN FOR OPERATION CONDITIONS.

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

QID: 0785 **Rev:** 0 **Rev Date:** 9/10/2009 **Source:** New **Originator:** Possage

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

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

System Number: 012 **System Title:** Reactor Protection 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:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** K

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

Given:

The plant is at 100% power

"B" RPS is INOPERABLE due to a failed High Temperature Trip Bistable

All other RPS channels OPERABLE

Which of the following is NOT a required action per T.S. 3.3.1?

A. Place channel in bypass within 1 hour

B. Place channel in a trip condition within 1 hour

C. Prevent bypass of remaining channels within 1 hour

D. Open all CRD trip breakers within 1 hour

Answer:

D. Open all CRD trip breakers within 1 hour

Notes:

D is correct, with only one RPS channel inoperable T.S. does not require CRD trip breakers to be opened A, B and C are all incorrect. T.S. 3.3.1 reuires any one of listed conditions be performed for the condition given.

References:

OP-1105.001 Change 024
TS 3.3.1

History:

New Selected for 2010 RO/SRO exam

4.2.11 Reactor Protection System Channel D Test (1304.040).

4.2.12 Reactor Protection System Channel D Calibration (1304.044).

4.2.13 CRD System Operating Procedure (1105.009).

4.2.14 Emergency Operating Procedures (1202.XXX).

4.2.15 Source Range Channels Test (1304.055).

4.3 NRC COMMITMENTS

None.

5.0 LIMITS AND PRECAUTIONS

5.1 Do not place an RPS protection channel in manual bypass without first obtaining permission from the Shift Manager/CRS and notifying Control Room personnel.

5.2 When testing an RPS protection channel, only the EFIC channel associated with the RPS channel being tested may be in MAINTENANCE BYPASS. TS 3.3.1 provides guidance when an RPS channel is bypassed or contains inoperable functions.

5.3 Placing two RPS protection channels in test simultaneously will result in a reactor trip unless one is in channel bypass.

5.4 Only one RPS channel shall be key locked in the untripped state at any one time.

5.5 Only one RPS channel bypass key shall be accessible for use in the control room.

5.6 The key-operated shutdown bypass switch associated with each RPS channel shall not be used during power operation except for testing.

5.7 In the event that one of the trip devices in either of the sources supplying power to the CRDMs fails in the untripped state, perform required actions for applicable TS 3.3.4 conditions.

5.8 Do not apply power to the CRDMs without using applicable section(s) of CRD System Operating Procedure (1105.009).

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. OR 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. AND D.2 Open all control rod drive (CRD) trip breakers.	6 hours 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: 0142 **Rev:** 0 **Rev Date:** 10/28/97 **Source:** Direct **Originator:** G. Giles

TUOI: AA51002-012 **Objective:** 21 **Point Value:** 1

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

System Number: 013 **System Title:** Engineered Safety Features Actuation System(ESFAS)

Description: Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Safeguards equipment control reset.

K/A Number: K4.10 **CFR Reference:** 41.7

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

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

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

Under what conditions can the Control Board Operator bypass or defeat a component automatically actuated by ESAS?

A. Bypassing or defeating a component automatically actuated by ESAS is not allowed.

B. The Control Board Operator, after careful consideration, determines that the component is no longer required.

C. ONLY when procedurally directed by the Emergency Operating or the Abnormal Operating procedures.

D. After it is determined that the component is no longer needed and approval is obtained from the SM/CRS.

Answer:

D. After it is determined that the component is no longer needed and approval is obtained from the SM/CRS.

Notes:

[A] is incorrect, provisions are made for this action.
[B] is partially correct, the component must not be needed but the CBO cannot make this decision on his own.
[C] is only one of the directions where a component can be bypassed/reset, CRS/SS permission is the other.
[D] contains all correct elements, lack of need and supervisory (SRO) permission.

References:

OP-1202..012 Change 008

History:

Taken from Exam Bank QID # 4791
Used in A. Morris 98 RO Re-exam
Previously used under K/A: 3.2 / Reactor Coolant System Inventory Control / 013 / Engineered Safety Features Actuation System / A4.02 / Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels. / CFR: 41.7 / 45.5 to 45.8 / RO: 4.3 / SRO: 4.4
Used on 2004 RO/SRO Exam (K/A T2 G1 013 K4.06)
Selected for the 2010 RO/SRO exam

NOTE

Obtain Shift Manager/CRS permission prior to overriding ES.

10. Verify proper ESAS actuation:

A. Verify BWST Outlets open (CV-1407 and 1408).

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

B. Verify SERV WTR to DG1 and DG2 CLRs open (CV-3806 and 3807).

C. **IF** any RCP is running, **THEN** perform the following:

1) **IF** ES Channel 5 or 6 has actuated, **THEN** perform the following:

a) **IF** SCM is **adequate**,
THEN trip all running RCPs due to loss of ICW.

b) **IF** SCM is < adequate,
THEN check elapsed time since loss of adequate SCM **AND** perform the following:

(1) **IF** ≤ 2 minutes have elapsed, **THEN** trip all RCPs.

(2) **IF** >2 minutes have elapsed, **THEN** perform the following:

(a) Leave currently running RCPs on.

(b) **IF** RCS press > 150 psig,
THEN notify CRS to **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure
AND
perform contingency for failure to trip RCPs within 2 minutes.

(c) Restore RCP services (RT 8) while continuing.

2) **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.

a) **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 3820).

(10. CONTINUED ON NEXT PAGE)

QID: 0135 **Rev:** 1 **Rev Date:** 4/7/05 **Source:** Direct **Originator:** B. Short

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation.

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

Tier:	2	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.3	**SRO Select:**	Yes	**Taxonomy:**	K

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

A LOCA has occurred.
Reactor Building (RB) pressure is 47 psia.

Which ESAS channels have actuated the RB cooling units and what is the correct RB cooling alignment?

A. ES channels 3 & 4, VSF-1A, 1B, 1C, & 1D running with service water aligned to the cooling coils.

B. ES channels 3 & 4, VSF-1A, 1B, 1C, 1D, & 1E running with chilled water aligned to the cooling coils.

C. ES channels 5 & 6, VSF-1A, 1B, 1C, & 1D running with service water aligned to the cooling coils.

D. ES channels 5 & 6, VSF-1A, 1B, 1C, 1D, & 1E running with chilled water aligned to the cooling coils.

Answer:

c. ES channels 5 & 6, VSF-1A, 1B, 1C, & 1D running with service water aligned to the cooling coils.

Notes:

ESAS channels 5 & 6 actuate RB cooling fans VSF-1A through 1D and also cause the bypass dampers to drop which allows air to bypass the retum air duct and chilled water coils and flow directly to the service water coils that were aligned by ES channels 5 & 6. Thus (c) is the correct answer. (a), (b) & (d) combine other ventilation alignments with other ES channels that are incorrect.

References:

STM 1-09, Rev. 9

History:

Developed for use in 98 RO Re-exam
Selected for 2005 RO exam.
Selected for 2010 RO/SRO exam

Cooling Units VSF-1A through 1D each have an associated ES signal from either Channel 5 or 6. During normal operation, the four units are running with chilled water as the cooling medium. On an ES actuation signal, all four units receive a start signal and a bypass damper opens allowing air to bypass the return air duct and chilled water coils allowing flow directly to the service water coils. Service water valves to the coils are opened by ESAS Ch 5 or 6 and chilled water to the RB is automatically secured. The lower pressure drop caused by bypassing the chilled water coils and return plenum, permits the single speed fan to handle the quantity of air necessary for emergency cooling. This precludes the necessity of a two-speed motor with the additional controls, power source and wiring.

Unit	Control Switch	CS Location	Power Supply	ES Actuating Signal:
VSF-1A	HS-7410	C18	480v ES Bus B523	ES-5
VSF-1B	HS-7411	C18	480v ES Bus B533	ES-5
VSF-1C	HS-7412	C16	480v ES Bus B623	ES-6
VSF-1D	HS-7413	C16	480v ES Bus B633	ES-6
VSF-1E	HS-7419	C19	480v B714	None

2.1.1.2 Supply Fan Back-draft Dampers CV-7470 - 7473

Each supply fan (VSF-1A thru D) has a single blade, butterfly damper (CV-7470 thru 7473) at the discharge of the fan that opens when the fan starts. These dampers are called back-draft dampers because they prevent reverse flow through the fan when it is not running. Each damper has a Limitorque motor operator that is controlled from the same hand switch as the supply fan. They are powered from MCC B5252 for CV-7470, B5332 for CV-7471, B6212 for CV-7472 and B6332 for CV-7473. Damper position indication is provided on Control Room panels C-16 or C-18.

2.1.1.3 VCC-1A - 1E Chilled Water Cooling Coils

Refer to figure 9.01, 9.02 & 9.03

The Chilled Water Cooling Coils for the RB Cooling Units are single stage coils supplied from Main Chill Water. Isolation Valves for Main Chill Water (CV-6202 & CV-6203) are air operated outside the RB with a motor operated valve (CV-6205) for the return line inside the RB. Check valve AC-60 is used for double isolation in the supply line inside the RB. The Containment Isolation valves for Chill Water are closed by ES Channel 5 & 6 signals.

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

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

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

Section: 3.5 **Type:** Containment Integrity

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

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

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

Tier:	2	**RO Imp:**	4.2	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	K

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

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

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

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

C. To prevent overload of the EDGs.

D. To prevent water hammer of the spray headers.

Answer:

C. To prevent overload of the EDGs.

Notes:

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

References:

1107.002, Chg. 025

History:

Developed for 1998 RO/SRO Exam.
Used in A. Morris 98 RO Re-exam
Selected for 2005 Jon Gray RO re-exam.
Selected for the 2010 RO/SRO exam.

1.0 PURPOSE

To provide instructions for operating the engineered safeguard 4160V and 480V AC electrical distribution system.

2.0 SCOPE

This procedure is used for normal and infrequent operation of the 4160V and 480V ES distribution system including normal, emergency, and alternate AC power sources where those instructions differ significantly than those more generic instructions in Electrical System Operations (1107.001).

This procedure establishes operating guidelines and requirements to meet NRC Generic Letter 91-11, LCOs for Vital Instrument Buses and Tie Breakers.

3.0 DESCRIPTION

Two 4160V AC engineered safeguard buses provide power to the engineered safeguard equipment, including the 480V AC ES distribution system, through 4160V/480V transformers.

The normal power source to bus A3 and A4 is from non-ES 4160V buses A1 and A2 respectively. The emergency power source is from 4160V AC, 2750KW diesel generators, one for each bus. Emergency power is supplied automatically on loss of normal power. The Alternate AC source is a 4400KW diesel generator manually placed into service.

Normally the buses are separated and independent; however, bus tie breakers are provided for abnormal situations. Some ES loads can be powered from either bus. To maintain bus separation and independence as required by 10CFR50 Appendix R, motor operated disconnects (MODs) are provided for the B HPI pump and B service water pump.

To prevent overload due to simultaneous starting currents, ES loads are automatically sequenced onto the ES buses. This automatic sequencing occurs whether the bus is on the normal source or the emergency source.

The 480V AC engineered safeguard distribution system consists of two 480V AC load centers, B5 and B6, each containing a 1000KVA 4160V/480V step-down transformer. B5 and B6 are powered from buses A3 and A4, respectively, through the step-down transformer.

Bus B5 supplies motor control centers MCC B51, B52, B53 and B57 (MCC B53 is supplied from MCC B52).

Bus B6 supplies motor control centers MCC B61, B62, B63, B64 and B65 (MCC B63 is supplied from MCC B62. MCC B64 is supplied from MCC B65).

QID: 0202 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** R. Walters

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump.

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: 2 **SRO Imp:** 3.7 **SRO Select:** Yes **Taxonomy:** A

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

Given:

- A plant startup is in progress with the reactor critical below the point of adding heat.
- "B" OTSG Turbine Bypass Valve (CV-6688) fails full OPEN and is unable to be closed with the handjack.
- Tave 524 degrees and dropping
- Pressurizer level 205 inches and dropping
- RCS pressure 2120 psig and dropping

What is the proper course of action?

A. Initiate MSLI for the 'B' OTSG and maintain the reactor critical using 'A' OTSG Turbine Bypass Valve to control RCS temperature and pressure.

B. Continue the reactor startup maintaining startup rate <1 DPM while continuing to monitor primary and secondary plant parameters.

C. Go directly to 1203.003, OVERCOOLING for actions to mitigate the oversteaming of the 'B' OTSG.

D. Trip the reactor and follow the guidance of 1202.001 REACTOR TRIP.

Answer:

D. Trip the reactor and follow the guidance of 1202.001 REACTOR TRIP.

Notes:

(A.) is incorrect. You would not want to isolate a OTSG and maintain the reactor critical.
(B.) is incorrect. With the reactor below the point of adding heat with a stuck open TBV, this would not be possible.
(C.) is incorrect. This will be the ultimate tab that you will end up in, however, it is necessary to trip the reactor first and progress through the Reactor Trip EOP.
(D.) is correct. Taking the conservative action of tripping the reactor is appropriate due to being below the minimum temperature for criticality and the inability to maintain SUR below 1 DPM.

References:

1102.008 (Rev 023), Approach to Criticality, pages 4&5

History:

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

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

Used on 2004 RO/SRO Exam.
Selected for 2010 RO/SRO exam

QID: 0203 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** B. Short

TUOI: AA51002-008 **Objective:** 8.8 **Point Value:** 1

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

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

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:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** An

Question: **RO:** ☐ **SRO:** ☐

Red powered EFW Pump Turbine (K-3) Steam Admission Valve Bypass Valve (SV-2663) has failed to open during regularly scheduled surveillance testing. What are the required operator actions?

a. Deenergize SV-2663 closed and declare P-7A inoperable.

b. Declare SV-2663 inoperable and manually open the valve.

c. Declare SV-2663 inoperable and deenergize CV-2663 closed.

d. Deenergize SV-2617 open and declare P-7B inoperable.

Answer:

c. Declare SV-2663 inoperable and deenergize CV-2663 closed.

Notes:

(a.) is incorrect. The red powered valve being inoperable does not affect the operability of the green train P-7A.
(b.) is incorrect. SV-2663 is a solenoid operated valve and can not be manually operated.
(c.) is correct. With CV-2663 deenergized closed, the green powered valve CV-2617 is still available to operate P-7A.
(d.) is incorrect. P-7B is the electric driven pump and is not affected by the steam supply valve operability.

References:

1106.006 (Rev 58)

History:

Developed for use in 98 RO Re-exam

4.2 REFERENCES USED IN CONJUNCTION WITH THIS PROCEDURE

4.2.1 Soluble Poison Concentration Control (1103.004)

4.2.2 Reactivity Balance Calculation (1103.015)

4.2.3 CRD Operating Procedure (1105.009)

4.2.4 Plant Preheatup and Precritical Checklist (1102.001)

4.2.5 Power Operation (1102.004)

4.2.6 NI & RPS Operating Procedure (1105.001)

4.2.7 Unit 1 Technical Specifications

4.2.8 Loss of Neutron Flux Indication (1203.021)

4.2.9 Plant Startup (1102.002)

4.2.10 Infrequently Performed Tests or Evolutions EN-OP-116.

4.2.11 Reload Criticality and Low Power Physics Test (1302.020)

5.0 LIMITS AND PRECAUTIONS

5.1 Operators performing/supervising the reactor startup should not rely on the critical rod position predicted by the estimated critical position calculation, but anticipate criticality any time during rod withdrawal, boron dilution or RCS temperature changes.

5.2 Maintain at least a 1.5% shutdown margin if any condition, physical or administrative, delays approach to criticality.

5.3 Do not simultaneously change reactivity by more than one means while subcritical or prior to point of adding heat.

5.4 Operators performing/supervising the reactor startup should use all pertinent instrumentation available to monitor indication of approaching criticality. The tendency to become fixed on one indication should be avoided.

5.5 Maximum continuous SUR is ≤1 DPM. Prompt change associated with attaining this SUR shall be <1.5 DPM.

5.6 Reactor coolant temperature shall be above 525°F when the reactor is critical (TS 3.4.2).

5.7 During approach to criticality, safety rod groups shall be at upper limit and regulating rods shall be positioned as prescribed per Regulating Rod Insertion Limits curves of the COLR and (TS 3.2.1).

5.8 During startup when intermediate range instruments come on scale, flux level shall be maintained in the source range until overlap between intermediate range and source range instruments is greater than or equal to one decade (SR 3.3.10.1 Bases).

5.9 Reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable (TS 3.4.9 and TRM 3.4.9).

5.10 Reactor shall not be made critical until both Pressurizer Code Safety Valves (PSV-1001 and PSV-1002) are operable (TS 3.4.10).

5.11 The licensed Operators performing/supervising the reactor startup shall perform no other duties during reactor startup.

5.12 The licensed Control Room Operators performing/supervising the reactor startup shall not conduct shift relief until the reactor is critical at ≥1% power or shutdown by 1.5% Δk/k except during physics testing.

5.13 Prior to commencing the reactor startup, a review of activities in progress or planned shall be conducted to ensure that distractions to the startup will be minimized.

5.14 During the reactor startup, access to the control room shall be limited to ensure that a professional environment, once established, is maintained without distraction or interruption.

5.15 The Shift Manager shall oversee the reactor startup from the control room and ensure that a professional environment is maintained.

5.16 If unexpected situations/conditions arise during the reactor startup, then the Operators performing/supervising the reactor startup shall take conservative action to place the reactor in a safe condition.

5.17 During startup when withdrawing regulating groups, the overlap between two sequential groups shall be between 15% and 25% except for physics testing. (TS 3.2.1)

5.18 Reactor Engineering personnel shall be present in the Control Room to monitor the approach to criticality and to perform 1/M plots.

5.19 SR 3.2.1.3 requires verification of SDM ≥1% ΔK/K within 4 hours prior to achieving criticality.

5.20 If the reactor has been shutdown <48 hours, then contact Reactor Engineering to verify that the Fuels and Analysis calculated RHOBAL bias has been incorporated into the Estimated Critical Calculations. (CR-HQN-2009-00107) (CR-ANO-1-2009-0237)

5.21 If criticality achieved within procedural limits of ± 0.5% Δk/k but NOT within ± 0.25% Δk/k, then notify Reactor Engineering to initiate a condition report. (CR-ANO-1-2009-0237)

6.0 SETPOINTS

6.1 Observe setpoints in referenced system operating procedures.

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

QID: 0195 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** L. Kilby

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

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

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

Description: Ability to monitor automatic operation of the MFW, including: Feedwater pump suction flow pressure

K/A Number: A3.03 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	2.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	2.6	**SRO Select:**	Yes	**Taxonomy:**	K

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

Unit 1 is operating at 100% power with no abnormal conditions or alignments.
'B' MFP SUCT PRESS LO (K07-C8) annunciator is received.

Where can the Control Room Operators read the 'B' MFW pump suction pressure WITHOUT leaving the control room?

A. The 'B' MFP Lovejoy Operator Control Station (OCS).

B. 'B' MFP Suction Pressure (PI-2830) indicator.

C. 'B' MFP Suction Pressure computer point (P2830)

D. The Operator Information Touchscreen (OIT).

Answer:

C. 'B' MFP Suction Pressure computer point (P2830)

Notes:

(a.) & (d.) are incorrect. These panels are located in the control room, however, MFP suction pressure is not available on these panels.
(b.) is incorrect. This indicator is located outside the control room.
(c.) is correct. This computer point is found on the Plant Computer and the SPDS computer both of which are available in the control room.

References:

STM 1-19, Rev. 11

History:

Developed for use in 98 RO Re-exam
Selected for 2005 RO exam
Selected for 2010 RO/SRO exam

PS-2841 provides the second suction pressure trip signal used to satisfy the trip logic. Setpoint for PS-2841 is less than 200 psig.

Refer to table provided on the following page for suction pressure indications associated with the "B" MFP. Alarm and trip signal setpoints are identical to P-1A for P-1B.

PT-2830 provides suction pressure signal to plant & SPDS computers
PI-2830 provides local suction pressure indication at rack 21.
PS-2830 provides Lo & Lo-Lo alarms (K07-C8 & K07 B8). Provides Suction Pressure trip signal.
PS-2835 provides suction pressure trip signal to MFP trip logic.

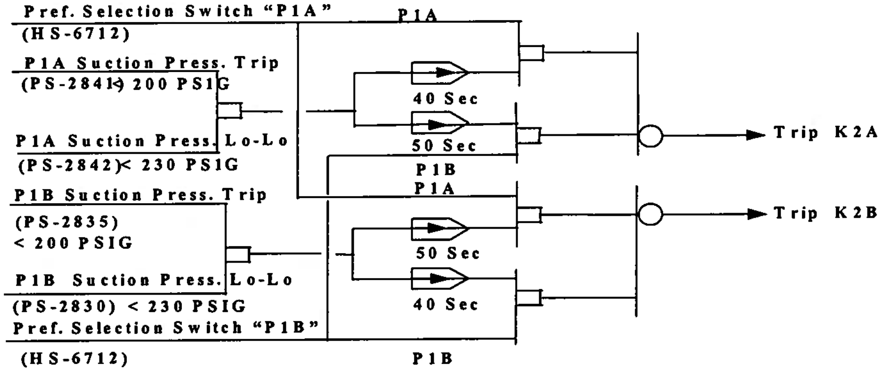
2.3.1.3 Suction Pressure Trip Logic

Operation of the MFP's with suction pressure less than 230 psig can cause pump damage. To provide MFP protection and increase plant reliability, the MFP suction pressure logic was modified requiring two separate pressure signals to trip a MFP. To increase plant reliability and inadvertent trips due to suction pressure transients, time delays were installed. During normal operation one of the MFP's will be selected for the preferred pump to trip on low suction pressure. The preferred pump is selected by handswitch HS-6712 located on panel C02. HS-6712 positions are P-1A or P-1B. Time delays associated with the preferred MFP trip are set at 40 seconds and 50 seconds for the remaining MFP.

The Lo-Lo and < 200 psig pressure switches provide the signals used to trip the preferred MFP and /or both MFP's associated with switches discussed in the above section.

If suction pressure drops to <200 psig for greater than 40 seconds the preferred or selected MFP will trip. If suction pressure remains less than 200 psig for an additional 10 seconds the remaining MFP will trip. Refer to Trip Logic String provided below.

Suction Pressure Trip Logic



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

QID: 0789 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Knowledge of MFW design feature(s) and / or interlock(s) which provide for the following: automatic trips for MFW pumps.

K/A Number: K4.16 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	Yes	**Taxonomy:**	K

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

Given:

- 100% power

Which of the following interlocks provide an automatic trip of the Main Feed Water Pump?

A. Main Feed Water Pump suction pressure reading 220 psig for 45 seconds

B. Main Feed Water Pump bearing oil pressure reading 15 psig

C. Main Feed Water Pump discharge pressure reading 1360 psig

D. Main Feed Water Pump vibration reading 14 mils

Answer:

C. Main Feed Water Pump discharge pressure reading 1360 psig

Notes:

A is incorrect, suction pressure would have to be less than 200 psig for 40 seconds.
B is incorrect, bearing oil pressure of 15 psig would cause an alarm but pressure must be less than 10 psig for a trip.
C is correct, pump discharge pressure of 1350 psig would result in a pump trip
D is incorrect, the high vibration trip is bypassed when the pump is in operation

References:

STM 1-24 Rev. 11

History:

New selected for 2010 RO/SRO exam

The pilot valve bleeds oil from the main valve disc, which allows a spring to open the main valve disc. Trip header oil pressure decreases and stop valves close as outlined above.

The overspeed trip valve will also open when auto-stop oil pressure decreases to zero (for instance, when the solenoid trip opens). This will seal in a main feedwater pump trip from the solenoid trip. To pressurize the trip header it is necessary to close the overspeed trip valve.

Closing the overspeed trip valve is accomplished through the use of the reset devices. A local reset button is used to reset (close) the overspeed trip. Depressing the reset button closes the main valve disc and pilot valve. This allows oil pressure to build up above the main valve disc and hold the valve closed. All trips must be reset to maintain the main valve closed; otherwise, the springs will open the main valve.

An overspeed trip reset solenoid valve is installed to allow the overspeed trip to be reset from a remote location. The overspeed trip reset valve will port high-pressure oil to a piston located on the shaft of the reset button. The high-pressure oil moves the piston which closes the pilot and main valve disc the same as depressing the local reset button. When the solenoid is not energized, the solenoid valve aligns the piston to drain and no pressure will be applied to the piston. The solenoid is energized when the trip-reset switch on C02 is positioned to the reset position or the reset switch at the front standard is taken to reset.

The trip lever is used to manually trip the feedwater pump. Depressing the trip lever opens a drain that bleeds pressure from the top of the main valve disc. The main valve disc opens and the pump trips as above.

3.10 Solenoid Trip Valve

The solenoid trip valves interact directly with the trip header to depressurize the trip header in response to various trip signals. The solenoid is energized to open the valve and trip the feedwater pump. For redundancy, two solenoid trip valves are used in a parallel arrangement. Trips that will trip the solenoid trip valve are:

- Electronic overspeed trip
- Low suction pressure

  Two pressure switches used (200 and 230 psig)

  40 seconds the preferred pump trips

  50 seconds the non-preferred pump trips.
- High discharge pressure (two of three pressure switches at 1350 psig)
- Low bearing oil pressure (two of three pressure switches after a 3 second time delay)

  Two pressure switches at 10 psig

  One pressure switch at 15 psig also supplies low pressure alarm
- High vibration (normally bypassed during operation)

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

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

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

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

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

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

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

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

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

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

The EFIC automatic fill rate is designed to prevent overcooling.
With the plant in a degraded power condition and given a SG pressure of 885 psig, determine the proper OTSG fill rate by EFIC
for the EFW system:

A. ~3"/min

B. ~4"/min

C. ~5"/min

D. ~6"/min

Answer:

B. ~4"/min

Notes:

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

References:

1105.005, Chg. 032

History:

Used in 1999 exam.
Direct from ExamBank, QID# 92 used in class exam
Modified for 2005 Jon Gray RO re-exam.
Selected for 2010 RO/SRO exam

6.0 SETPOINTS

6.1 Initiation Setpoints

- EFW low level initiate ~ 11", delayed 9.9 seconds
- MSLI and EFW initiate on low SG pressure ~ 600 psig.
- Loss of both MFW pumps with reactor power >7%.
- ESAS Channel 3 or Channel 4 trip.
- MFW Flow in both loops <15% with reactor power >45%. (AMSAC)
- All RCPs OFF (May be bypassed at <10% Power)

6.2 Control Setpoints

6.2.1 SG level

- Low level control ~ 31".
- Natural circulation control ~ 312".
- Reflux boiling control ~ 378".

6.2.2 Rate of SG level rise when RCPs are off is variable from 2 to 8 inches per minute depending on SG pressure. (2 inches per minute at 800 psig, 8 inches per minute at 1050 psig)

6.2.3 SG ΔP ~ 100 psi determines good (unaffected) SG to allow EFW flow and isolates bad (affected) SG on MSLI actuation.

6.2.4 Atmospheric dump control valves will control SG pressure at ~ 1020 psig at all times if not isolated.

6.3 Low condenser vacuum interlock opens atmospheric dump isolation valves at ~ 21" Hg.

6.4 MSLI actuation opens affected SG atmospheric dump isolation valve.

QID: 0790 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S Pullin

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

Section: 3.6 **Type:** Electrical

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

Description: Knowledge of local auxiliary operator tasks during emergency and the resultant operation effects.

K/A Number: 2.4.35 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

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

Given:

Unit 1 is in a Blackout condition.
Voltage has been recovered on SU#2 and is 155 kV

To restore power to A-3 and A-4, what action along with its purpose is required by the Auxiliary Operator?

A. Perform Attachment 1, Blackout Breaker Alignment and UV Relay Defeat, to defeat UV Close Permissive interlocks to allow for starting of equipment necessary to protect the core.

B. Perform Attachment 1, Blackout Breaker Alignment and UV Relay Defeat, to prevent excess current during starting of the motors.

C. Perform Attachment 2, Recovery from Blackout Breaker Alignment and UV Relay Defeat, to allow for starting of equipment necessary to protect the core.

D. Perform Attachment 2, Recovery from Blackout Breaker Alignment and UV Relay Defeat, to allow Unit 2 to tie on non-vital loads on SU#2.

Answer:

A. Perform Attachment 1, Blackout Breaker Alignment and UV Relay Defeat, to defeat UV Close Permissive interlocks to allow for starting of equipment necessary to protect the core.

Notes:

A is correct, with degraded voltage on SU#2, Att. 1 is required to defeat the UV interlocks.
B is incorrect, Att. 1 would have no effect on actual starting current for motors
C & D are incorrect, Att. 2 will only be performed when SU#2 voltage is greater than 158 kV.

References:

OP-1202.028 Change 010

History:

New selected for 2010 RO/SRO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
44. Dispatch an operator to perform Attachment 1, "Blackout Breaker Alignment and UV Relay Defeat".	

NOTE

Off-site power is considered restored to normal if either of the following conditions exists:

- SU1 ≥22KV
- SU2 ≥158KV AND all of the following conditions are met:
  - Auto X-FMR energized from 500KV
  - Auto X-FMR aligned to SU2
  - No Unit 2 buses powered from SU2
  - SU 2 V REG 3% reduction disabled

INSTRUCTIONS	CONTINGENCY ACTIONS
A. IF off-site power is restored to normal, THEN dispatch an operator to perform Attachment 2, "Recovery From Blackout Breaker Alignment and UV Relay Defeat" AND RETURN TO step 8.	
45. WHEN Attachment 1 is complete, THEN re-energize A1, A2, H1, and H2 by performing the following for each bus:	
A. Check associated bus L.O. RELAY TRIP alarm clear on K02.	A. Determine AND correct cause of L.O. RELAY TRIP before energizing bus, while continuing with this procedure (Refer to Electrical System Operation (1107.001), "Re-closing Tripped Bus or MCC Feeder Breakers" section).
B. IF buses are to be energized from SU2, THEN notify Unit 2.	
C. Turn SYNC switch for associated bus feeder breaker ON AND close breaker from handswitch.	C. Reset breaker anti-pump feature by taking handswitch to PULL-TO-LOCK AND releasing. 1) IF neither A1 nor A2 is energized, THEN **RETURN TO step 33.**

INSTRUCTIONS	CONTINGENCY ACTIONS
46. Re-energize A3 and A4 by performing the following for each bus.	
A. Check associated bus L.O. RELAY TRIP alarm clear on K02.	A. Determine **AND** correct cause of L.O. RELAY TRIP before energizing bus, while continuing with this procedure (Refer to Electrical System Operation (1107.001), "Re-closing Tripped Bus or MCC Feeder Breakers" section).
B. Turn SYNC switch for associated bus feeder breaker ON **AND** close breaker from handswitch.	B. **IF** non-vital bus voltage is <3160V, **THEN** dispatch an operator to close A3 and A4 feeder breakers in LOCAL to override Sync-check Relays (A-309 and 409).

CAUTION

- During degraded voltage conditions the following problems may occur:
  - Motors may trip on overload, overheat due to high running currents, or stall.
  - MCC starter may **not** pick up to energize loads.
  - AC auxiliary relays may **not** pick up to provide interlock or load energization features.
- Motors should be started one at a time and allowed to reach run speed to minimize further voltage degradation.
- If both Units are aligned to SU2, coordination between Units is required when starting loads.

47. Restart only equipment absolutely necessary to protect the core as follows:

A. Verify suction and discharge flow path aligned.

B. Review system operating procedure to ensure essential pump services available.

C. Consider closing centrifugal pump discharge valve before starting to reduce starting current.

(47. CONTINUED ON NEXT PAGE)

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

QID: 0316 **Rev:** 0 **Rev Date:** 9/5/99 **Source:** Direct **Originator:** J Haynes

TUOI: ANO-1-LP-RO-MU **Objective:** 3.5 **Point Value:** 1

Section: 3.6 **Type:** Electrical

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

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

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

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

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

Which of the following would explain why a loss of bus A1 will cause CV-1206 (RC Pump Seal Injection Block Valve) to close?

(Assume plant is at 100% power)

A. P36A (HPI) pump was the in-service pump.

B. Loss of instrument air to Seal Injection Control Valve, CV-1207.

C. P36C (HPI) pump was the in-service pump.

D. Loss of instrument air to Pressurizer Level Control valve CV-1235.

Answer:

A. P36A (HPI) pump was the in-service pump.

Notes:

"a" is correct, if P36A was the in-service pump, then a loss of A1 would cause a loss of A3, P-36A would cease to run, and CV-1206 would close when Seal Injection flow dropped to less than 22 gpm.
"b" is incorrect, CV-1207 fails open on a loss of Instrument Air.
"c" is incorrect, a loss of A1 would not affect P36C's power supply, bus A4.
"d" is incorrect, CV-1235 fails as-is on a loss of Instrument Air.

References:

1203.026, Change 11

History:

Used in 1999 exam.
Modified from ExamBank, QID# 3716.
Selected for 2010 RO/SRO exam.

INSTRUCTIONS

SECTION 1 -- LOSS OF HPI PUMP

NOTE

Indications of loss of HPI suction are:

- Erratic flow, and
- Erratic discharge pressure, and
- Control valves stable

1. **<u>IF</u> HPI pump has lost suction,**
   <u>THEN</u> stop the HPI pump.

2. **Isolate letdown by performing one of the following:**

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

NOTE

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

3. **Verify RC pump seals are being cooled by ICW.**

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

4. **Prepare to restart an HPI pump as follows:**

   A. **IF** OP HPI pump is unavailable
   AND STBY HPI pump is unavailable,
   THEN dispatch an operator to re-align the ES HPI pump per Attachment A of this procedure.

SECTION 1 -- LOSS OF HPI PUMP (continued)

B. Place the following valves in HAND AND close:

- RC Pumps Total INJ Flow (CV-1207)
- Pressurizer Level Control (CV-1235)

C. Verify RCP Seal Injection Block (CV-1206) closes.

D. Select Safety System Diagnostic Inst display on SPDS for OP HPI pump AND evaluate suction pressure and flow stability prior to event.

E. <u>IF</u> loss of pump suction was indicated,
<u>THEN</u> perform the following:

1) Verify Makeup Tank Outlet (CV-1275) open.

CAUTION

Indicated suction pressure could be H_2 gas pressure only and is NOT absolute assurance of adequate volume of water. HPI pump operation with inadequate water volume can damage pump.

NOTE

Addition of 600 gallons to the MU tank ensures a volume of water in the tank regardless of level indication.

2) <u>IF</u> CV-1275 was **NOT** closed,
<u>THEN</u> refill Makeup Tank (T-4) by adding ≥20" (~600 gallons) using current RCS boron concentration.

5. <u>IF</u> STBY HPI pump is available,
<u>THEN</u> perform the following:

A. Start Aux lube oil pump for STBY HPI pump.

B. **GO TO** step 8 to place STBY HPI pump into service.

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

QID: 0086 **Rev:** 0 **Rev Date:** 7/11/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-ELECD **Objective:** 37 **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: Components using dc control power.

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

Tier:	2	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	K

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

The plant is at 100% power.

Which of the following DC buses/panels, if de-energized, would cause a reactor trip?

A. Panel D41

B. Panel RA1

C. MCC D15

D. Panel D21

Answer:

B. Panel RA1

Notes:

Only "B" is capable of causing a reactor trip due to loss of two RCP contact monitors.
The others would cause a loss of vital equipment capability but as seen in Att. J of 1107.004, they would not cause a trip.

References:

1107.004, Chg. 016

History:

Developed for 1998 RO exam
Used in A. Morris 98 RO Re-exam
Selected for use in 2005 RO exam, but not used.
Selected for 2010 RO/SRO exam.

ATTACHMENT J

Consequences and Required Actions For Opening 125V DC Breakers

NOTE

- Some breaker operations render equipment inoperable and requires entry into Tech Spec LCO.
- Attachment J is not listed by priority. Locating grounds should begin with circuits of least consequences.

125V DC Bus D01 Breakers

BREAKER NUMBER	DESCRIPTION	CONSEQUENCES OF OPENING	REQUIRED ACTION
D01-21A	Supply To MCC D15	Loss of power to MCC D15 and EFW P7A valves.	None
D01-22A	DC Power Supply to Inverter Y11	Loss of Inverter Y11 DC Supply	If available, place Inverter Y15 in service.
D01-23	Supply to Panel RA1 (breaker handle not connected-fused supply)	Loss of power to RA1. Reactor trip if ≥50% power due to loss of power to RCP Contact Monitor input to RPS. MSIVs open if instrument air is not isolated	Check RA1 breakers individually first using RA1 section of this attachment. Verify reactor power <50% and not in 3 RCP operations. <u>If</u> MSIVs are closed, verify instrument air is isolated.
D01-24	Emer Supply to Panel D21 (breaker handle not connected-fused supply)	Loss of emergency supply to panel D21	Verify D21 is powered from bus D02
D01-41	Supply From Battery Charger D03A	Disconnects battery charger from bus D01	Verify battery charger D03A not in operation.
D01-42	Supply From Battery Charger D03B	Disconnects battery charger from bus D01	Verify battery charger D03B not in operation
D01-52B	DC Power Supply to Inverter Y13	Loss of Inverter Y13 DC Supply	If available, place Inverter Y15 in service.
D01-53A	DC Power Supply to Inverter Y15	Loss of Inverter Y15 DC Supply	If available, place Inverter Y11 (for RS-1) or Inverter Y13 (for RS-3) in service.

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

QID: 0791 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** **Originator:** S. Pullin

TUOI: A1LP-RO-EDG **Objective:** 19a **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators (ED/G)

Description: Knowledge of bus power supplies to the following: Air Compressor

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

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

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

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

What is the power supply to Emergency Diesel Generator Starting Air Compressors, C4A1 and C4B2?

A. B31 and B41

B. B32 and B42

C. B51 and B61.

D. B52 and B62

Answer:

A. B31 and B41

Notes:

A is correct, the other choices are alternate possibilities.

References:

OP-1107.001 Change 73

History:

New for 2010 RO/SRO exam.

ATTACHMENT D

Date _______________

MCC B41 (north electrical equipment room)

BREAKER NUMBER	DESCRIPTION	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INI-TIAL
4112	Spare	Open			
4113	Condensate to MU & Purif System CV-1251 & CA-113 position ind (E-194)	Closed			
4114	Spare	Open			
4115	Spare	Open			
4116	Degasifier Drain Pump P-43B (E-390)	Closed			
4121	Spare	Open			
4122A	Room 125 Transformer X118 P66 (E-20)	Closed			
4122B	Spare	Open			
4123A	Spare	Open			
4123B	Hot Mechanics Shop and Decon Room Utility Outlets (E-43)	Closed			
4124	Treated Waste Monitor Pump P-47B (E-392)	Closed			
4125	Clean Waste Receiver Tank Trans Pump P-49B (E-394)	Closed			
4126	Filtered Waste Pump P-53B (E-389)	Closed			
4131	Spare	Open			
4132	Dirty Waste Drain Pump P-52B (E-389)	Closed			
4133	DG-1 Starting Air Compressor C-4A2	Closed			
4134	DG-2 Starting Air Compressor C-4B2	Closed			
4135	Waste Gas Compressor C-9B (E-402)	Closed			
4136	Spare	Open			

ATTACHMENT D

Date _______________

MCC B31 (north electrical equipment room)

BREAKER NUMBER	DESCRIPTION	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INI-TIAL
3112	Vacuum Degasifier Seal Water Pump P-99 (E-397)	Closed			
3113	Primary Coolant Hydrazine Pump P-37 (E-191)	Closed			
3114	Lithium Hydroxide Pump P-38 (E-191)	Closed			
3115	DG-1 Starting Air Compressor C-4A1	Closed			
3116	Degasifier Drain Pump P-43A (E-390)	Closed			
3121	CWRT Recirc Pump P-48 (E-393)	Closed			
3122	Quench Tank Transfer Pump P-44 (E-207)	Closed			
3123A	Solid Waste Baler M-8 (E-431)	Closed			
3123B	Aux Power Receptacles (E-43)	Closed			
3124	Treated Waste Monitor Pump P-47A (E-392)	Closed			
3125	CWRT Transfer Pump P-49A (E-394	Closed			
3126	Filtered Waste Pump P-53A (E-389)	Closed			
3131	Sample Room Exhaust Fan VEF-49 (E-340)	Closed			
3132	Dirty Waste Drain Pump P-52A (E-389)	Closed			
3133	Diesel Fuel Oil Transfer Pump P-74A (E-115)	Closed			
3134	Laundry Drain Pump P-45 (E-389)	Closed			
3135	Waste Gas Compressor C-9A (E-402)	Closed			
3136	Aux Bldg Drain Transfer Pump P-46 (E-387)	Closed			
3141	Aux Bldg Sump Pump P-51A (E-383)	Closed			
3142	Core Flood Tank Recirc & MU Pump P-132 (E-385)	Closed			
3143A	Computer Room Unit Cooler VUC-5A (E-363)	Closed			
3143B	Computer Room Unit Cooler VUC-5B (E-363)	Closed			

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

QID: 0792 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin

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

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators (ED/G)

Description: Knowledge of the physical connections and / or cause-effect relationships between the ED/G system and the following systems: Starting air system.

K/A Number: K1.05 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

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

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

Given:

Plant at 100%
Performing #1 EDG monthly surveillance per 1104.036 Supplement 1

The CBOT presses the start pushbutton on C10
K01-B2, EDG 1 OVERCRANK, alarms

What is the cause of the alarm and how long did the starting air system attempt to start the engine?

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

B. #1 EDG did not exceed 300 rpm in 8 seconds and air start motors engaged for 45 seconds.

C. #1 EDG did not exceed 30 rpm in 45 seconds and air start motors engaged for 2.5 seconds.

D. #1 EDG did not exceed 30 rpm in 8 seconds and air start motors engaged for 8 seconds.

Answer:

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

Notes:

A is correct, due to meeting the annunciator logic
B, C, and D are variations of the control logic for the starting air to the engine

References:

STM-1-31 rev 10
1203.012A change 038

History:

New 2010 RO/SRO exam

Location: C10

Device and Setpoint:

EDG 1
OVERCRANK

Alarm: K01-B2

1.0 OPERATOR ACTIONS

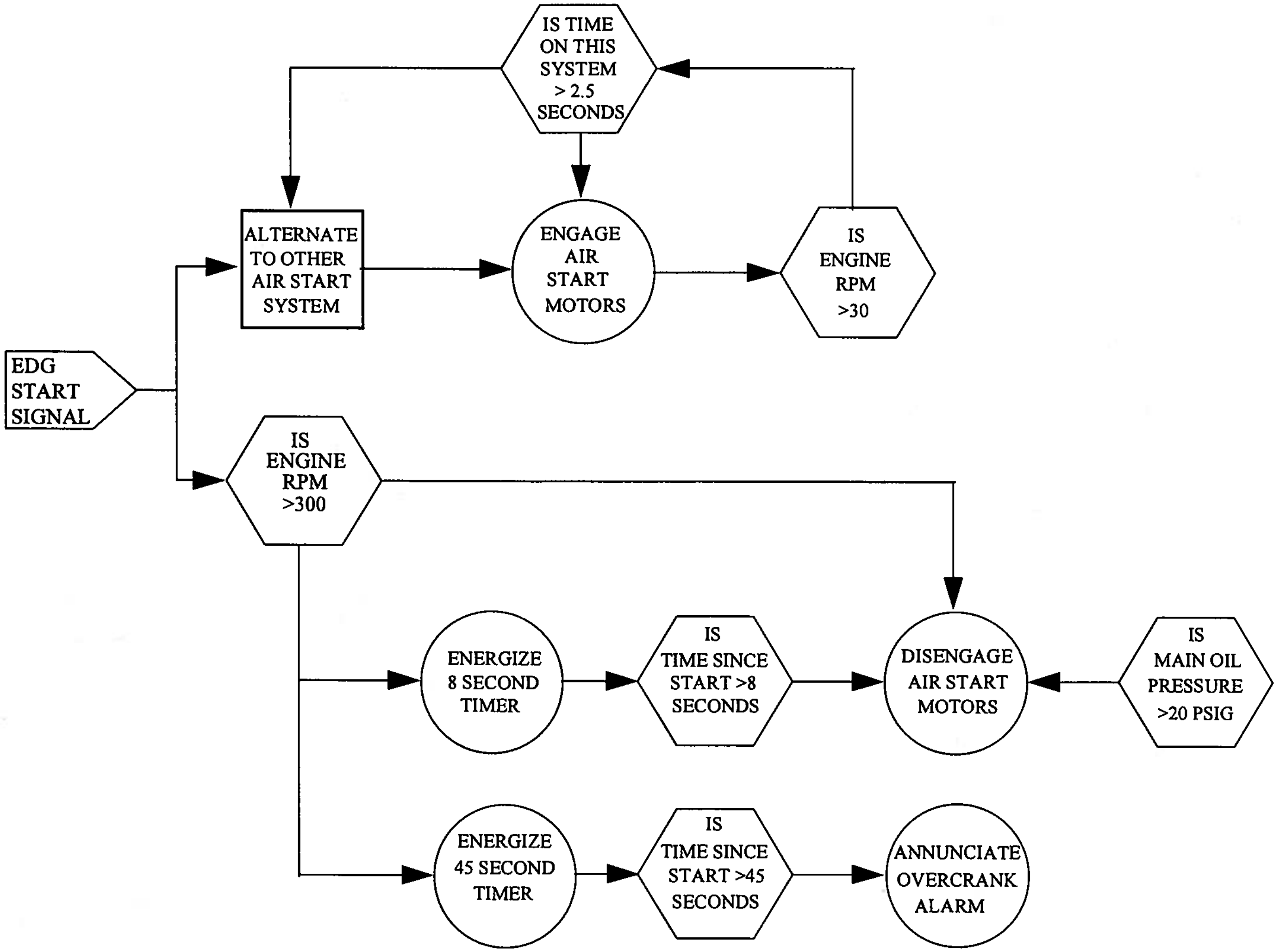
1. Place DG1 lockout switch in LOCKOUT position.
2. Reference TS 3.8.1, TS 3.8.2 and TS 3.8.3 for operability requirements.
3. Initiate action to determine cause of over-crank.
4. Operate fuel oil priming pump and verify "return fuel" sight glass is full.
5. WHEN cause of over-crank is corrected,
   THEN prove DG1 operable using Emergency Diesel Generator Operation (1104.036), Supplement 1.
6. IF DG1 inoperable,
   THEN verify proper MOD alignment for Service Water Pump (P-4B) and Makeup Pump (P-36B) per Makeup & Purification System Operation (1104.002) AND Service Water and Auxiliary Cooling System (1104.029).
7. Alarm may be cleared by ANY of the following methods:
   - Place DG1 lockout switch in LOCKOUT position
   - Depress local RESET button
   - Place Local/Maint/Remote switch in MAINT
   - Place DG1 Output (A-308) in PULL-TO-LOCK

2.0 PROBABLE CAUSES

- DG1 did not reach minimum speed within 45 seconds
- Loss of fuel oil pump prime

3.0 REFERENCES

- TS 3.8.1, TS 3.8.2 and TS 3.8.3
- Schematic Diagram Annunciator K01 (E-451)
- Schematic Diagram Diesel Generator Engine Control (E-102)



STM1-31-43 EDG STARTING SEQUENCE

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

QID: 0672 **Rev:** 0 **Rev Date:** 12/16/06 **Source:** Repeat **Originator:** Possage

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

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring System

Description: Knowledge of the operational implications of the following concepts as they apply to the PRM System: Radiation theory, including sources, types, units, and effects.

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

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

Group: 1 **SRO Imp:** 3.0 **SRO Select:** Yes **Taxonomy:** K

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

What type of detector is used by the Main Condenser Air Discharge Radiation Monitor to monitor for steam generator tube leaks?

A. Scintillation Detector

B. Geiger - Mueller Detector

C. Ion Chamber Detector

D. Beta Radiation Detector

Answer:

A. Scintillation Detector

Notes:

"A" is correct. The Main Condenser Air Dischagre Radiation Monitor is a scintillation detector.
"B" is incorrect. Area Monitors are G-M Detectors
"C" is incorrect. Ion chambers are used for RP surveys
"D" is incorrect. The Penentration Ventilation Monitors are beta sensitive monitors.

References:

STM 1-62 Rev. 11

History:

New for 2007 RO Exam.
Selected for 2010 RO/SRO exam

2.2.6 Liquid Radwaste Monitor

The Liquid Radwaste Monitor is an in-line monitor located in the liquid Radwaste common discharge line prior to its connection to the flume. The connection is between CZ-58 and CV-4642 and the monitor is physically located on the 335' elevation of the auxiliary building by the discharge flume. Liquid Radwaste passes through the pipe section of the sampler and is monitored by a gamma sensitive scintillation detector (RE-4642). The detector count rate is displayed on the digital rate meter located in the Control Room (C-25, Figure 62.14). There is an input to SPDS and the plant computer as well as a recorder readout on RR-4830.

The Liquid Radwaste Monitor is used to determine radioactive discharge activity during a release and to shut off the discharge should a pre-determined level of radioactivity be reached. On a high radiation level solenoid valve, SV-4642 operates to shut CV-4642, terminating the liquid Radwaste release. An annunciator in the Control Room will alarm on high radiation.

2.2.7 Main Condenser Air Discharge Radiation Monitor

The Main Condenser Air Discharge Radiation Monitor is an in-line monitor on the combined suction line of the condenser vacuum pumps. The detector (RE-3632) is a gamma sensitive scintillation detector and is located on a platform above and just south of the condenser vacuum pumps. The detector count rate is displayed on the digital rate meter located in the Control Room (C-25, Figure 62.14). There is an input to both SPDS and plant computer as well as recorder readout on RR-4830.

The purpose of the Main Condenser Air Discharge Radiation Monitor is to detect activity resulting from a steam generator tube leak. On a high radiation, an annunciator in the Control Room alarms.

2.2.8 Waste Gas Radiation Monitoring

The Waste Gas Radiation Monitor is an in-line monitor in the gaseous Radwaste system discharge to the vent plenum. This monitor is down stream of gaseous discharge shutoff valve CV-4830 and is located on the 404' elevation of the auxiliary building in the CRD transformer (X-8) room. The detector is a gamma sensitive scintillation detector (RE-4830). The count rate is displayed on a digital rate meter located in the Control Room (C-25, Figure 62.14) and provides an input to both SPDS and plant computer. There is also a recorder readout of radiation level on recorder RR-4830.

On a high radiation level, an annunciator in the Control Room alarms. At this alarm setpoint, solenoid valve, SV-4830, operates to shut CV-4830, isolating gaseous Radwaste discharge to the station vent plenum. Also, the following will take place: CV-4820 will be shut by solenoid valve, SV-4820, to isolate the Waste Gas Tanks discharge header; Solenoid valve, SV-4806, operates to open CV-4806, to direct miscellaneous vents from the components in the Auxiliary Building to the Waste Gas Surge Tank.

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

QID: 0793 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** Direct **Originator:** S Pullin

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

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

System Number: 076 **System Title:** Service Water System (SWS)

Description: Ability to manually operate and / or monitor in the control room SWS valves

K/A Number: A4.02 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.6	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** 52 **SRO:** 52

When starting Service Water Pump P-4A after maintenance, you observe the following symptoms.

- Pump start is indicated by normal light indication above pump control HS on.
- Annunciator K10-B3 "SW DISCH PRESS HI" alarms.
- Valve position indication in the control room indicates proper valve alignment.
- SW Bay levels are 338 feet
- No change in SW flow or discharge pressure indications on the SPDS Diagnostics screen.
- No change in SW Loop pressure indications on control room panel C09.

Which of the following is the most likely cause of these symptoms?

A. The pump discharge valve was not opened when returned to service.

B. The pump did not start when pump breaker closed.

C. P-4A cannot pump into the system because of high system pressure from the other(running) pump.

D. P-4A is running without sufficient NPSH to pump water into the SW System.

Answer:

A. The pump discharge valve was not opened when returned to service.

Notes:

A is the correct answer. With the local discharge valve closed, the SW Pump would not be able to pump water to the loop, but since the discharge pressure switch is between the pump and discharge vlave, therefore a high discharge pressure would be seen.
B is incorrect, if the pump did not start there would not be a high discharge pressure alarm.
C is incorrect, if the maintenance performed had caused low discharge pressure such that the pump was unable to pump water to the loop, there would not be a high discharge pressure alarm.
D is incorrect, with a bay level of 338 feet, suction pressure would be (356.5-338)0.433= 8 psig which is adequate.

References:

OP-1203.012I Change 046

History:

Direct ANO Exam bank QID ANO-OPS1-3284
Selected for 2010 RO/SRO exam

Location: C16

Device and Setpoint:
SW Pump P-4A running & P-4A Disch Press (PS-3611) >90 psig
SW Pump P-4B running & P-4B Disch Press (PS-3609) >90 psig
SW Pump P-4C running & P-4C Disch Press (PS-3610) >90 psig

SW PUMP
DISCH PRESS
HI

Alarm: K10-B3

1.0 OPERATOR ACTIONS

1. Determine which pump is in alarm.

2. IF experiencing a loss of Service Water
OR degraded Service Water flow,
THEN GO TO Loss of Service Water (1203.030).

3. IF lake temperature is low
OR cold weather operations with low ACW/SW demand,
THEN consider throttling open ICW Coolers Loop 1 and 2 SW Bypass (SW-4026A and SW-4026B).

4. IF T-alt is installed from ICW Cooler (E-28C) outlet,
THEN throttle open temporary valve T-1.

5. Place additional SW/ACW loads into service as needed.

2.0 PROBABLE CAUSES

NOTE
This annunciator has multiple input without reflash.

1. Improper SW Pump discharge alignment

2. Cold lake temperatures causing low ACW/SW demand

3.0 REFERENCES

1. Schematic Diagram Annunciator K10 (E-460, sheets 1 - 3)

2. NRC Commitment P 6186, Provide procedure for cause, action, and how to clear alarms of DHR.

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

QID:	0794	**Rev:**	0	**Rev Date:**	9/15/2009	**Source:** New	**Originator:** S. Pullin
TUOI:	A1LP-RO-ESAS			**Objective:**	20		**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:**	2
Group:	1	**SRO Imp:**	2.6	**SRO Select:**	Yes	**Taxonomy:**	K

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

What would be the effect to service water pressure due to an inadvertent actuation of ES Channel 5 ?

A. Service Water Pressure would drop due to SW valves to the EDG Coolers opening.

B. Service Water Pressure would drop due to SW valves to the RB Coolers opening.

C. Service Water Pressure would rise due to ACW isolation valve closing.

D. Service Water Pressure would rise due to SW to ICW isolations closing.

Answer:

B. Service Water Pressure would drop due to SW valves to the RB Coolers opening.

Notes:

A is incorrect, SW to EDG Coolers open on diesel start. EDG starts on Channels 1 or 2
B is correct, ES Channel 5 will align SW to the RB Coolers
C is incorrect, ACW isolation valve would close on ES Channel 2
D is incorrect, SW to ICW isoaltion valve will close on ES Channels 1 and 2

References:

STM 1-65 Rev. 5

History:

New, Selected for 2010 RO/SRO exam

- 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.

4.12.3 Reactor Building Cooling and Isolation

RB isolation and cooling (Channel 5 and 6 is initiated by high Reactor Building pressure of 4 psig, and as its name implies, its function is to isolate and cool the RB. The following equipment is actuated:

- CV-2234, 2235, 2220 and 2221 close to isolate Non-Nuc ICW to RC Pump Air/LO and CRD Coolers.
- CV-2214, CV2215 and CV-2233 close to isolate Nuc ICW to Letdown and RCP Seal Coolers.

- CV-6205, CV-6202 and CV-6203 close to isolate the RB Chillers.
- The RB Coolers Inlet and Outlet Valves open to VCC 2A, B, C & D (CV-3812, CV-3814 and CV-3813, CV-3815).
- RB Cooling Fan "A", "B", "C" & "D" start and SV-7410, SV-7411, SV-7412 and SV-7413 (RB Bypass Dampers open.
- VEF-38A or B, Penetration Room Fans start.
- CV-2235, CRD Cooling Coil Inlet Isolation Valve closes.
- CV-1065, Quench Tank Cond. Isolation closes.

4.12.4 Reactor Building Spray

Reactor Building Spray and Chemical Addition components are actuated when RB pressure reaches 30 psig. The components actuated are:

- P35A & B RB Spray Pumps start.
- CV-2401 and 2400 RB Spray Blocks open.
- CV-1616 and 1617 open to supply Sodium Hydroxide to the Spray Pumps.

5.0 Technical Specifications

21. Identify the Technical Specification requirements for ESAS.

The Technical Specification requirements for the Engineered Safeguards Actuation System are found in:

- 3.5 Instrumentation Systems
    - ◊ 3.5.1 Operational Safety Instrumentation
        - ⇒ 3.5.1.1 Requirements of Table 3.5.1-1
        - ⇒ 3.5.1.2 Number of channels below that required.
    - ◊ Table 3.5.1-1 Instrumentation Limiting Conditions for Operation
    - ◊ 3.5.3 Safety Features Actuation Setpoints
- 4.1 Operational Safety Items
    - ◊ Table 4.1-1 Instrument Surveillance Requirements.

The Technical Specification requirements for the systems and the components actuated by ESAS are covered in the respective systems' System Training Manual.

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

QID: 0535 **Rev:** 1 **Rev Date:** 10/13/2005 **Source:** Direct **Originator:** J.Cork
TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems
System Number: 078 **System Title:** Instrument Air System
Description: Knowledge of the physical connections and / or cause-effect relationships between the IAS and the following systems: Service Air
K/A Number: K1.02 **CFR Reference:** 41.7 / 45.5
Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3
Group: 1 **SRO Imp:** 2.8 **SRO Select:** Yes **Taxonomy:** K

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

Instrument Air pressure has dropped to 50 psig.

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

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

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

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

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

D. Close Letdown Cooler outlet to isolate Letdown.

Answer:

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

Notes:

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

References:

1104.025, Chg. 014

History:

Developed for 1998 RO exam (similar to QID 102)
Modified question for A. Morris 98 RO Re-exam
Modified for J. Gray 2005 re-exam.
Selected for 2010 RO/SRO exam.

6.2 Compressor trips on any of the following:

6.2.1 Electrical fault.

6.2.2 After cooler discharge temp high:

- C-3A After Cooler Disch Air Temp High (TS-5405) 125°F
- C-3B After Cooler Disch Air Temp High (TS-5407) 125°F

6.2.3 Lube oil pressure low: 8 psig for >10 seconds

- C-3A Low Lube Oil Press (PS-5434)
- C-3B Low Lube Oil Press (PS-5436)

6.3 Interlocks

6.3.1 Compressor start opens cooling water solenoid valve:

- E-19A After Cooler ICW Inlet (SV-2251)
- E-19B After Cooler ICW Inlet (SV-2250)

6.3.2 50 psig dropping IA pressure opens Inst Air X-over (SV-5400) and closes at ~54 psig rising IA pressure.

6.4 SA Compressor Alarms

6.4.1 Compressor cooling water outlet temp high: 125°F.

- C-3A ICW Disch Temp (TS-2261)
- C-3B ICW Disch Temp (TS-2260)

6.4.2 Compressor discharge air temp high: 340°F.

- C-3A Disch Air Temp High (TS-5404)
- C-3B Disch Air Temp High (TS-5406)

6.4.3 After cooler discharge air temp high: 110°F

- C-3A After Cooler Disch Air Temp High (TS-5405)
- C-3B After Cooler Disch Air Temp High (TS-5407)

6.4.4 Lube oil pressure low: 15 psig

- C-3A Low Lube Oil Press (PS-5434)
- C-3B Low Lube Oil Press (PS-5436)

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

QID: 0795 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** Direct **Originator:** S. Pullin

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to manually operate and / or monitor in the control room: Operation of the containment personnel airlock door

K/A Number: A4.06 **CFR Reference:** 41.7 / 45.5 to 45.8

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

Group: 1 **SRO Imp:** 2.9 **SRO Select:** Yes **Taxonomy:** C

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

Given:

Plant refueling is in progress

The Reactor Building Coordinator calls the control room and reports the following:
The inner door of the reactor building personnel hatch will not close
The outer door is operable

In accordance with Technical Specifications for Refueling Operations, how does this affect fuel movement?

A. Irradiated fuel movement in the reactor building and auxiliary building must be suspended.

B. Irradiated fuel movement in the reactor building must be suspended.

C. Irradiated fuel movement in the auxiliary building must be suspended.

D. Irradiated fuel movement may continue without restriction.

Answer:

D. Irradiated fuel movement may continue without restriction.

Notes:

D is correct, fuel movement may continue in both the Reactor Building and Aux Building provided one of the air lock doors is capable of being closed.
A, B, and C are incorrect due to the outer door being operable.

References:

T.S. 3.9.3 Amendment No. 215

History:

Direct from ANO exam bank ANO-OPS1-6622
Selected for 2010 RO/SRO exam.

3.9 REFUELING OPERATIONS

3.9.3 Reactor Building Penetrations

LCO 3.9.3 The reactor building penetrations shall be in the following status:

a. The equipment hatch is capable of being closed;

b. One door in each air lock is capable of being closed; and

c. Each penetration providing direct access from the reactor building atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
2. capable of being closed by an OPERABLE reactor building isolation valve, except reactor building purge isolation valves, or
3. capable of being closed by an OPERABLE reactor building purge isolation valve with the purge exhaust radiation monitoring channel OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more reactor building penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

RO Written Exam

Tier 2 Group 2

ES-401 PWR Examination Outline Form ES-401-2

Plant Systems - Tier 2/Group 2 (RO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#	QID	Type
001 Control Rod Drive		X										**K2.05 changed to K2.02 –** One-line diagram of power supply to trip breakers	3.6	56	429	D
002 Reactor Coolant								X				**A2.01-** Loss of coolant inventory	4.3	57	604	D
011 Pressurizer Level Control									X			**A3.03-** Charging and letdown	3.2	58	797	N
014 Rod Position Indication				X								**K4.05** – Rod hold interlocks	3.1	59	308	D
015 Nuclear Instrumentation			X									**K3.04** - ICS	3.4	60	299	D
016 Non-nuclear Instrumentation					X							**K5.01-** Separation of control and protection circuits	2.7	61	77	D
017 In-core Temperature Monitor						X						**K6.01-** Sensors and detectors.	2.7	62	240	D
027 Containment Iodine Removal												**Not selected**	**N/A**			
028 Hydrogen Recombiner and Purge Control												**Not selected**	**N/A**			
029 Containment Purge												**Not selected**	**N/A**			
033 Spent Fuel Pool Cooling												**Not selected**	**N/A**			
034 Fuel Handling Equipment												**Not selected**	**N/A**			
035 Steam Generator												**Not selected**	**N/A**			
041 Steam Dump/Turbine Bypass Control												**Not selected**	**N/A**			
045 Main Turbine Generator										X		**A4.06-** Turbine stop valves	2.8	63	138	D
055 Condenser Air Removal												**Not selected**	**N/A**			
056 Condensate												**Not selected**	**N/A**			
068 Liquid Radwaste												**K4.01-** Safety and environmental precautions for handling hot, acidic, and radioactive liquids **Rejected system to 014 Rod Position Indication**	**N/A**			
071 Waste Gas Disposal												**K3.05** – ARM and PRM systems **Rejected system to 015 Nuclear Instrumentation**	**N/A**			
072 Area Radiation Monitoring												**Not selected**	**N/A**			
075 Circulating Water											X	**2.4.11-** Knowledge of abnormal condition procedures-	4.0	64	798	N
079 Station Air												**Not selected**	**N/A**			
086 Fire Protection							X					**A1.01**- Fire header pressure	2.9	65	542	D
K/A Category Point Totals:	**0**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	Group Point Total:		**10**		

QID: 0429 **Rev:** 0 **Rev Date:** 4/30/2002 **Source:** Direct **Originator:** S.Pullin

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

Section: 3.1 **Type:** Reactivity Control

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

Description: Knowledge of bus power supplies to the following: One-line diagram of power supply to trip breakers

K/A Number: K2.02 **CFR Reference:** 41.7

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

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

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

If breaker B631 opened while operating at 100% power, the response of the Control Rod Drive system would be:

A. A ratchet trip of all regulating rods since half of the power supply has been removed.

B. No effect on regulating rods, safety rods are held by a single phase (CC) energized.

C. A ratchet trip of the safety rods due to a single phase remaining energized.

D. A trip of all safety rods since the main power has been removed.

Answer:

B. No effect on regulating rods, safety rods are held by a single phase (CC) energized.

Notes:

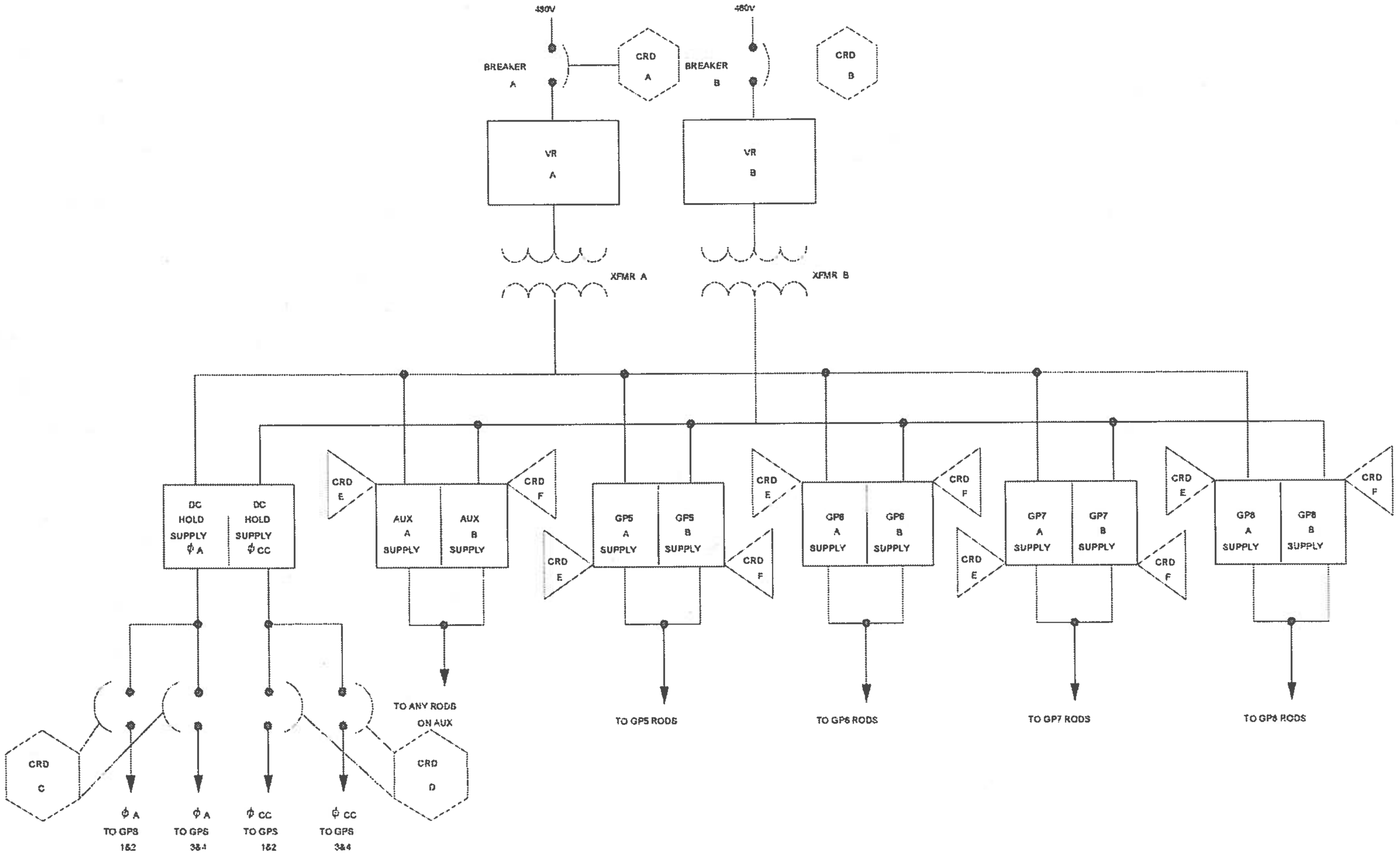
B is correct. The one-line diagram shows the power supply configuration from A-501 providing power to the CC phase on the DC hold bus which will maintain the safety rods out. Regulating rods are not effected normal movement will be supplied by the Bus 2 power supplied by A-501.

References:

STM 1-02, Control Rod Drive System, page 9, step 2.4

History:

Direct from regular exambank QID 4208.
Selected for use in 2002 RO/SRO exam.
Selected for 2010 RO/SRO exam.



SYMBOLS:

⬡ POWER TRIP SIGNALS

▷ RELAY ACTUATION OFFGATE

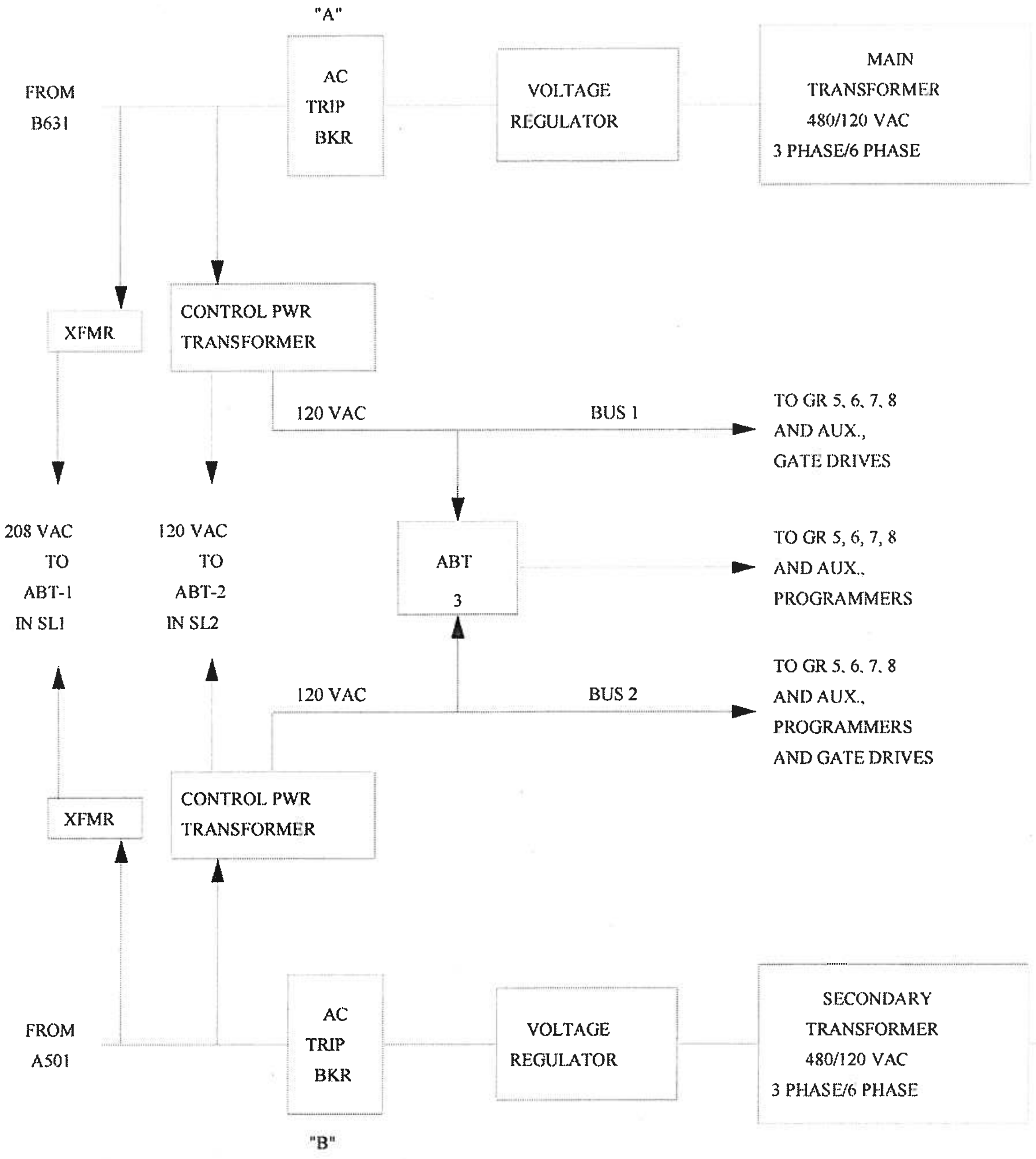
NOTES:

TO REMOVE ALL POWER FROM ALL
ONE OF THE FOLLOWING TRIP
MUST EXIST AS A

(CRD A X CRD B) + (CRD A X CRD D) + (CRD B X CRD C) + (CRD C X CRD
WHICH REDUCES TO (CRD A + CRD D) X (CRD C + CRD D) WHICH IS 1-OUT-OF-2
UNLESS A MALFUNCTION OCCURS IN THE TRIPPING
ALL BREAKERS ARE OPEN AND ALL GATES ARE TURNED
WHEN A 2-OUT-OF-4 TRIP CONDITION OCCURS IN THE

FIGURE 02.34: CRD TRIP SIGNALS

FIGURE 02.30: CRD 120 VOLT POWER SUPPLIES



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

QID: 0604 **Rev:** 0 **Rev Date:** 6/30/05 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LPR-RO-RCS **Objective:** 5 **Point Value:** 1

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

System Number: 002 **System Title:** Reactor Coolant System (RCS)

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 coolant inventory.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

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

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** C

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

A reactor trip has occurred and the CRS is directing actions per 1202.001, Reactor Trip.

Assume all actions have been performed when required by system parameters.

The CBOR reports that Pressurizer level has fallen to 30" and continuing to drop.
Pressurizer Level Control (CV-1235) is in Auto and fully open.

Which of the following is the proper response?

A Initiate HPI per RT-2.

B. Reduce Letdown by closing Orifice Bypass (CV-1223).

C. Isolate Letdown by closing Letdown Cooler Outlet (CV-1221).

D. Operate CV-1235 in HAND to control PZR level 90 to 110".

Answer:

A Initiate HPI per RT-2.

Notes:

Answer "A" is correct, this is done when level is < 30" per 1202.001.
Answer "B" is incorrect, this was done early in the procedure, shortly after immediate actions.
Answer "C" is incorrect, this was done earlier when level was < 50".
Answer "D" is incorrect, CV-1235 is operating properly in Auto, taking it to hand would not help.

References:

1202.001, Chg. 031

History:

New for 2005 RO exam, modified as a replacement question.
Selected for 2010 RO/SRO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
26. Check Pressurizer Level Control valve (CV-1235) maintains PZR level > 55".	26. Perform the following: A. **IF** CV-1235 fails to respond in AUTO, **THEN** operate CV-1235 in HAND to control PZR level 90 to 110". B. **IF** PZR level is < 55" with **no** indication of recovery, **THEN** isolate Letdown by closing either: Letdown Cooler Outlet (CV-1221), **OR** Letdown Cooler Outlets (CV-1214 and 1216). C. **IF** PZR level drops below 55", **THEN** verify Pressurizer Heaters off. D. **IF** PZR level drops below 30", **THEN** initiate HPI (RT 2).

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

QID: 0797 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** New **Originator:** S. Pullin

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

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

System Number: 011 **System Title:** Pressurizer Level Control System (PZR LCS)

Description: Ability to monitor automatic operation of the PZR LCS, including: Charging and letdown.

K/A Number: A3.03 **CFR Reference:** 41.7 /45.5

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.3	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** 58 **SRO:** 58

Given:

Plant at 100%
Letdown flow 80 gpm indicated on FI-1236
Letdown pressure 50 psig on PI-1237

CV-1244 and CV-1245 Letdown DI Inlet Isolation valves lose power.

With no operator action what would be the expected automatic response of the pressurizer level control system ?

A. FI-1236 would read 80 gpm, PI-1237 would read 50 psig and Pressurizer level control valve CV-1235 position would not change.

B. FI-1236 would read 0 gpm, PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would open.

C. FI-1236 would read 85 gpm, PI-1237 would read 45 psig and Pressurizer level control valve CV-1235 position would open.

D. FI-1236 would read 70 gpm, PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would close.

Answer:

B. FI-1236 would read 0 gpm, PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would open.

Notes:

B is correct, due to letdown DI Inlet Isolation Valves fail closed on a loss of power. Which would isolate letdown, letdown pressure would rise to the letdown relief setpoint of 150 psig, causing a LOCA. Pressurizer level would go down causing CV-1235 to open.
A,C, and D are variations of these possible combinations.

References:

STM 1-04 Rev. 9

History:

New for 2010 RO/SRO exam.

MU-4, is a backup for either the letdown orifice or the control valve. This valve, when fully open, passes as much flow as the control valve. The manual valve is opened only if the control valve is shut and secured. Thus, the maximum flow capacity is 170 gpm through the control valve or the manual valve and 45 gpm through the orifice. This yields a total of 215 gpm. The relief valve downstream of the letdown orifice is set for 150 psig and can pass up to 257 gpm. Even in the unlikely situation that all three paths are open simultaneously, which would require multiple operator error, the flow capacity of the makeup system combined with the relief valve prevents overpressuring the letdown line.

2.7.1 Letdown Orifice Isolation Valve CV-1222

This air operated, solenoid actuated pneumatic valve is used to isolate the normal letdown stream . Its hand switch is located on Control Room Panel C04 (HS-1222). Loss of Instrument Air to the valve will cause it to fail as is.

2.7.2 Letdown Orifice Bypass Valve CV-1223

CV-1223 is an air operated, electrically controlled valve, and is throttled from Panel C04 by the operator. Flow indicating controller FIC-1223 is used to electronically control flow around the letdown orifice line and give the operator final control of maximum letdown flow. CV-1223 is equipped with a voltage to pneumatic transducer, E/P-1223. Normal letdown purification flow is more than can be passed through the letdown flow orifice (FO-1222). Loss of Instrument Air to the valve will cause it to fail closed.

2.7.3 Letdown Flow Orifice FO-1222

This flow orifice was sized to limit letdown flow to approximately 45 gpm at normal RCS pressure. At this flow rate, one complete RCS volume turnover occurs each 24 hours. The orifice also causes a pressure drop from 2155 psi to about that of current M/U tank pressure.

2.7.4 Letdown Flow Orifice FO-1220

This bypass or parallel orifice can be used to obtain more flow during low pressure operations. It is also available for use if the letdown orifice bypass valve, CV-1223 is not available. It is placed in service manually by opening manual letdown valve MU4.

2.7.5 Letdown Temperature Element and Switch

Temperature element TE-1221 monitors letdown temperature and operates TIS-1221. This temperature switch sends a signal to the letdown penetration isolation valve, CV-1221, to close if letdown temperature reaches 135 ^{0}F. The interlock is designed to protect the resin of the purification demineralizers from damage due to excessively hot water.

2.8 Pressure Relief Valve PSV-1236

PSV-1236 is set for 150 psig and relieves pressure on the LD piping should the downstream piping and components be isolated. It discharges to the Auxiliary Building Equipment Drain Tank

(ABEDT). PSV-1236 can pass up to 257 GPM @ 10% above set pressure.

2.9 Letdown Flow Element, FE-1236

This Letdown Flow Element (FE-1236) provides the operator indication of letdown flow on panel C04, indicator FI-1236, SPDS, and feeds the Plant Computer.

2.10 Letdown Temperature Element TE-1237

TE-1237 is located on the letdown line on the 335 foot elevation of the Reactor Auxiliary Building. This temperature string consists of a measuring mechanism and a pneumatic transmitting mechanism. It provides pneumatic signals for letdown temperature indicator TI-1237 on Control Room panel C04. The temperature element also feeds a temperature switch and an electro-pneumatic converter, E/P-1237. The switch, TS-1237 provides electrical signals for the High Letdown Temperature annunciator alarm. The E/P converter supplies an electrical signal to the Plant Computer. Should letdown fluid temperature increase to greater than 130F, it causes annunciator K10-A8, "LETDOWN TEMP HI" to alarm.

2.11 Makeup Prefilter, F-25

The Makeup Prefilter is designed to be used in the event that extra filtering will be needed to filter out crud that would otherwise be entrained in the demineralizers. Crud could cause increasing radiation levels of the DI's and possibly shorten their useful life. RCS transient changes (pressure, temperature, pH and flow) can cause the release of crud within the RCS. F-25 is interconnected with, and normally used with, the decay heat removal system.

The Makeup Prefilter can be placed in service by verifying it isolated from the DHR system and manually opening MU-5 and MU-6 then closing Valve MU-7. Refer to figure 04.07. F-25 is used during plant start up and shut down to prevent excessive buildup of particulates in the demineralizers. It can be used anytime excessive particulate is indicated in the letdown system. The filter may be used during normal steady state operation. The filter is used when the decay heat system is in operation as part of one method of RCS drain down. This drain path is from the decay heat system, through F-25 to the letdown line and ultimately to the Clean Waste Receiver Tanks (CWRT) via the Vacuum Degasifier.

The filter is located upstream of the purification demineralizers and has a flow capacity of 140 gpm. In parallel with the filter is a differential pressure transmitter, PDIS-1400. At 25 psid across the filter "MU Sys. F-25 Filter ΔP HI" annunciator alarms on control room annunciator K10-F7.

remove ionic impurities and have some filtering capability for suspended materials. Each demineralizer contains a bed of mixed cation and anion resins. One unit is normally operating while the second unit is in standby.

The F-3A & B filters are used to keep resin fines and any particulate material that may pass through the DI's from entering the remainder of the purification system and the RCS.

2.13.1 Purification Demineralizer Inlet Valves, CV-1244 and CV-1245

These air operated gate valves are used to place either of the purification demineralizers in service. CV-1244 is for Demineralizer T-36A and CV-1245 is for Demineralizer T-36B. Both valves are operated from panel C-04 and have solenoid actuated, air operated, single acting cylinder operators. One valve is normally open, the other normally closed. These valves will fail closed on loss of air or power.

2.13.2 Demineralizer Bypass Valve, MU-9

This valve is used to bypass the purification demineralizers. It's use is directed during recovery from high temperature conditions to prevent LD DI resin depletion. During high temperature conditions in the letdown line, this valve should be opened prior to opening the letdown isolation valve (CV-1221). MU-9 is a manual valve.

2.13.3 Purification Demineralizers (DI's), T-36A/B

The HOH mixed bed demineralizers (DI's) are used to remove reactor coolant impurities from the letdown stream. Since the reactor coolant may be contaminated with dissolved fission and corrosion products, ion exchange resins are used to clean the reactor coolant. The resins remove radioactive impurities and reduce the radiation levels that might otherwise be present in the RCS piping.

Normally, the operating demineralizer is saturated with boron at a concentration equal to RCS boron concentration. The standby demineralizer may be unsaturated. This allows use of the standby demineralizer to remove boron late in core life to keep the reactor operating. A positive reactivity addition hazard may occur if the wrong DI is placed in service during power operation. The mixed bed HOH resin will remove the boron from the water passing through it. This will continue until the HOH resin comes up to an equiliabrium concentration of boron that equals RCS concentration. The RCS water passing into the demineralizer also may have an excess concentration of lithium-7, in the form of Lithium Hydroxide (LiOH). LiOH is used for pH control, thus corrosion control of the reactor coolant system. The HOH resin will also remove the Lithium from the water passing through it. This will continue until the HOH resin comes up to an equilibrium concentration of lithium that equal the RCS concentration.

Maximum and minimum flows through one demineralizer are 123 gpm (to prevent resin compacting) and 25 gpm (to avoid channeling) respectively. Table 4.2 contains design data for the purification demineralizers.

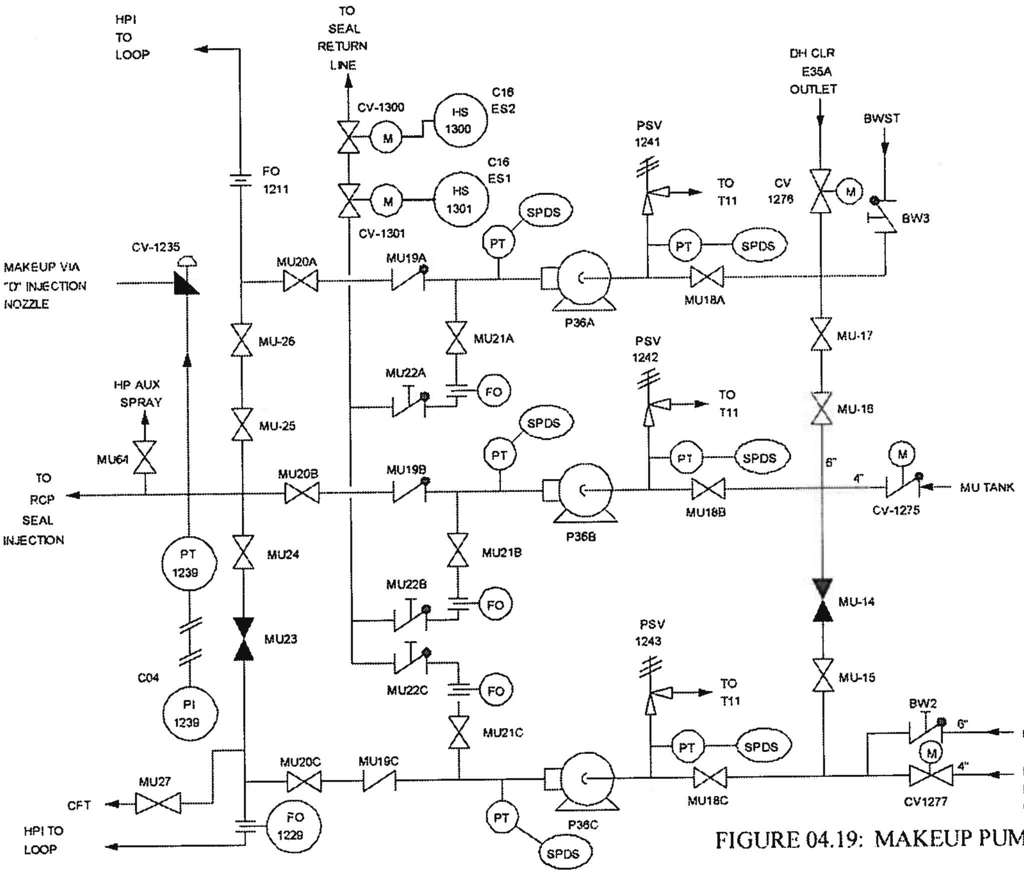


FIGURE 04.19: MAKEUP PUMPS PIPING

2.17 Pressurizer level control valve (CV-1235)

For normal makeup, flow element (FE-1238) provides indication for the operator that is displayed on C04. Makeup flow is controlled by pressurizer level control valve, CV-1235, an air operated 2 1/2 inch angled gate valve. CV-1235 is positioned by an electro-pneumatic controller which changes an electrical signal to an air signal. In auto the signal is derived from the difference between desired level (setpoint) and actual level from the Pressurizer Level Control circuit. In manual, the signal is generated from a toggle switch on the auto-manual control station. The control board operator positions the toggle as needed to increase or decrease flow. As is shown on figure 4.20 on the next page, there are two bypasses around CV-1235. CV-1235 may be isolated and makeup manually controlled through a manual bypass valve MU-1235-3. MU-1235-3 is a globe valve and can undergo significant erosin when exposed to the high differential pressure of an operating makeup pump. Therefore it is desirable to minimize the amount of time that MU-1235-3 is the only makeup flow path. A second bypass (MU-32) with a one inch angle valve is installed to ensure a continuous 10 gpm flow. (Set at 10 gpm during system startup.) This is to prevent thermal shock to HPI nozzle.

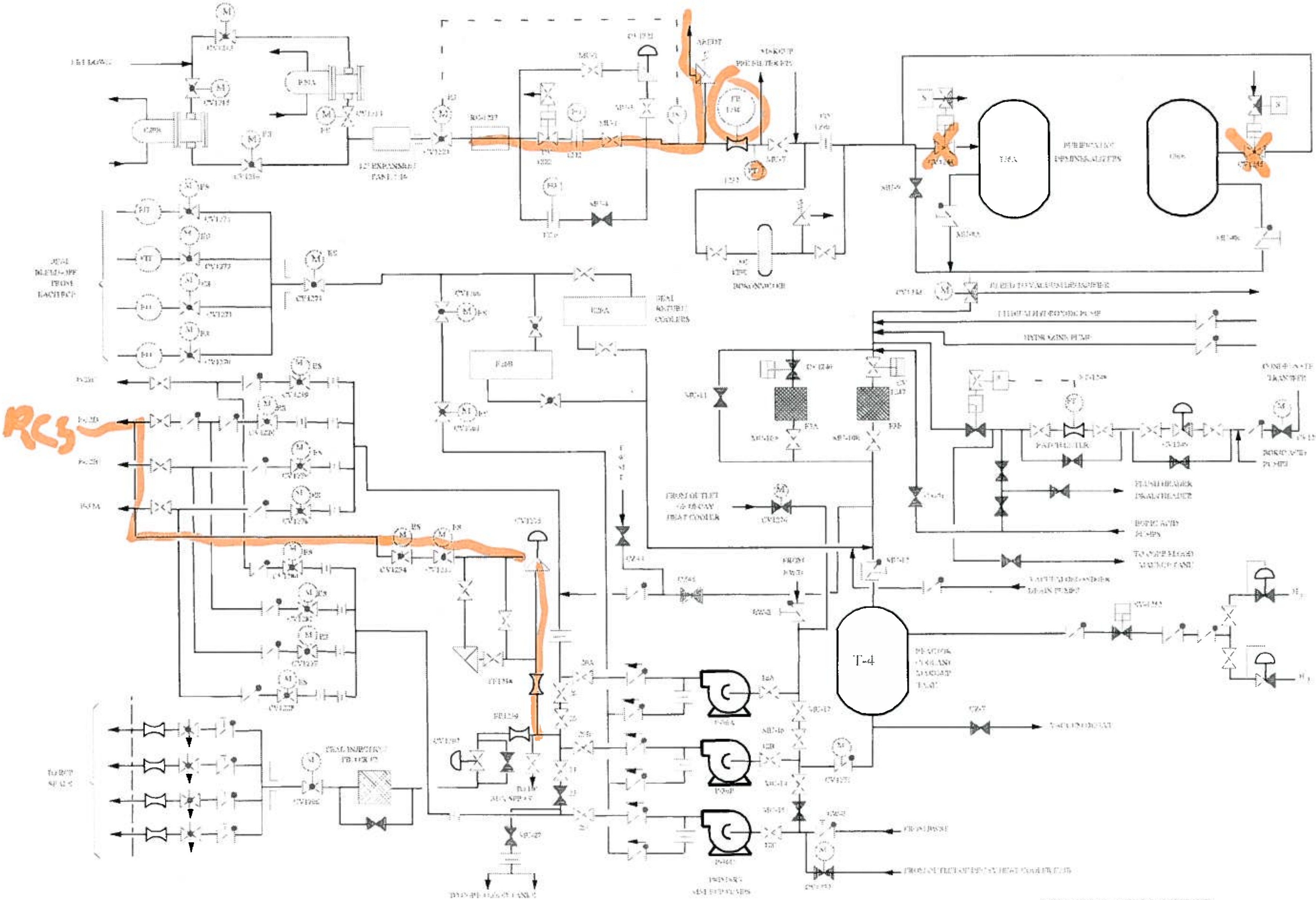


FIGURE 04 01: MU & P ONE LINE

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

QID: 0308 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J. Cork

TUOI: ANO-1-LP-RO-CRD **Objective:** 16 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 014 **System Title:** Rod Position Indication System

Description: Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Rod hold interlocks.

K/A Number: K4.05 **CFR Reference:** CFR: 41.5 / 45.7

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2.5

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

Question: **RO:** 59 **SRO:** 59

Given:

- Plant is at 100% power.
- ICS is in full automatic.

Subsequently, annunciator K07-B3 "ASYM ROD RUNBACK IN EFFECT" alarms.
A check of the PI panel shows that Rod 6 in Group 5 has dropped.

Which of the following alarms or indications would you expect to see on the diamond panel?

A. Sequence Inhibit lamp ON

B. Out Inhibit lamp ON

C. Auto Inhibit lamp ON

D. Group 5 Out Limit lamp OFF

Answer:

B. Out Inhibit lamp ON

Notes:

"a" is incorrect because the sequence inhibit is generated from relative position indications which do not use absolute position indications.
"b" is correct because the rods are interlocked so that they cannot move outward with an asymmetric rod fault with power greater than 40%.
"c" is incorrect because the rods are in auto and dropped rod is not a condition which will place the CRD system in manual.
"d" is incorrect because the group 5 out limit lamp may be from one of the other rods.

References:

1105.009 Change 32

History:

Developed for 1999 exam.
Selected for 2010 RO/SRO 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: 0299 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J Haynes

TUOI: ANO-1-LP-RO-NI **Objective:** 10 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation System

Description: Knowledge of the effect that loss or malfunction of the NIS will have on the following: ICS

K/A Number: K3.04 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	An

---

Question: **RO:** 60 **SRO:** 60

Given:

- The plant is at 80% power.
- The NI SASS mismatch alarm is bypassed due to a mismatch.

What would be the predicted plant response if NI-6 failed to 125%?

A. Control rods move inward, feedwater flows go up.

B. Control rods move inward, feedwater flows go down.

C. Control rods move outward, feedwater flows go up.

D. Control rods move outward, feedwater flow go down.

Answer:

A. Control rods move inward, feedwater flows go up.

Notes:

The mismatch alarm disables the SASS module automatic operation. When NI-6 fails to 125% power, ICS will see NI-6 as the input power. ICS will generate an error to drive rods in. At the same time a cross-limit is generated to keep feedwater balanced with reactor power. Feedwater will go up. Therefore, "B", "C", and "D" are incorrect.

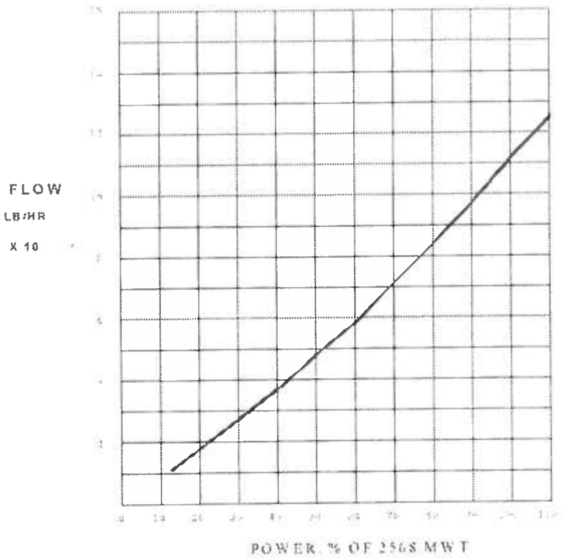
References:

STM 1-64, Integrated Control System, rev 10, page 33, step 2.6.1, page 43, step 2.7

History:

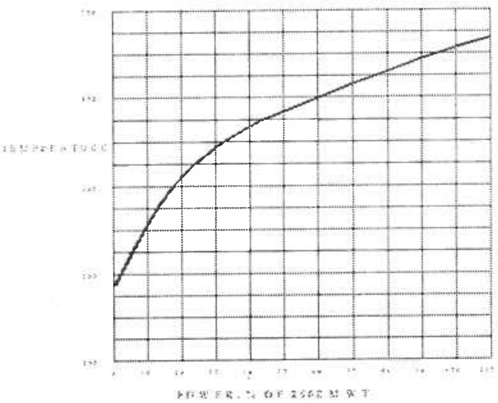
Used in 1999 exam.
Direct from ExamBank, QID# 3723
Selected for 2002 RO exam.
Used on 2004 SRO/SRO Exam.
Selected for 2010 RO/SROexam

FIGURE 64.22 TOTAL FEEDWATER FLOW vs POWER



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 64.23 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 "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 6×10^6 pounds mass per hour.

The value of the multiplier is set by the value of a control signal. This signal is the algebraic summation of two other signals. One of these signals is the RCS flow mismatch signal and will be zero when all four RCP's are properly operating. This signal will be described under "Three Pump Operations". The other signal is the ΔTc correction signal.

The control of the ratio of feedwater to each OTSG will determine the amount of heat that will be removed from the primary water in the reactor coolant system (RCS) and the relative amount of loading that each OTSG will carry. Therefore, the loading of the OTSGs can be indicated by the relative RCS return temperatures to the reactor (Tc's). If the difference in the Tc's (ΔT_c) is controlled near zero, then each OTSG will be loaded properly for the RCS flow through it. A trip of one RCP would give an immediate re-ratioing. An important benefit of keeping ΔTc low is that quadrant tilts within the reactor may be kept to a minimum.

The actual ΔTc is compared to the ΔTc setpoint. The difference (ΔTc Error) is used to generate the ΔTc correction signal. A zero ΔTc correction signal will split the signal equally between the loops.

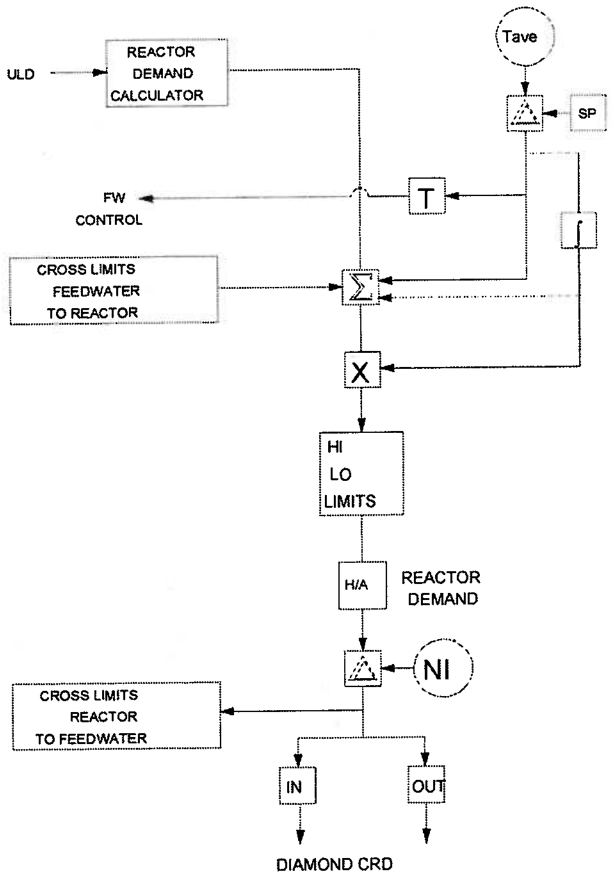
The operator may choose to manually control the ΔTc correction signal by placing the Load Ratio Hand/Automatic Station in hand. The only difference between this station and the other feedwater hand/auto stations is the additional dial and knob located under the both feedwater loop demands. When both 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 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

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

QID: 0077 **Rev:** 0 **Rev Date:** 9/29/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-1-LP-RO-NNI **Objective:** 5 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation System (NNIS)

Description: Knowledge of the operational implications of the following concepts as they apply to the NNIS: Separation of control and protection circuits.

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

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.8	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 61 **SRO:** 61

Given:

- Loop A RCS flow 70 E6 lbm/hr
- Loop B RCS flow 63 E6 lbm/hr
- Loop A Tave 578°F
- Loop B Tave 580°F
- Unit Tave 579°F

Which Tave will be selected by the SASS Auto/manual transfer switch and why?

a. Unit Tave due to Loop B flow

b. Loop A Tave due to Loop B flow

c. Loop B Tave due to Loop B flow

d. Unit Tave, flows are within tolerances

Answer:

b. Loop A Tave due to Loop B flow

Notes:

SASS will automatically select the Loop Tave for the Loop with the highest RCS flow should either flow drop below 95%. Normal RCS loop flow is ~70 E6 lbm/hr, therefore Loop B flow is <95% and SASS will select Loop A flow for Tave control, this control function protects the core from excessive heat transfer based upon flux to flow, therefore, (b) is the only correct response.

References:

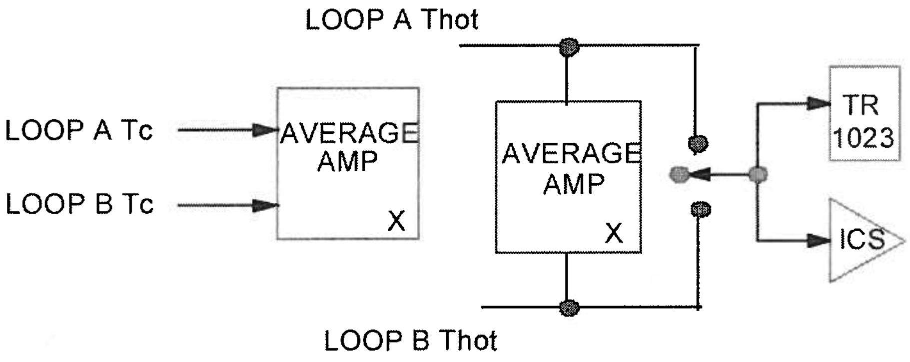
STM 1-69 (Rev 5), Non-Nuclear Instrumentation System page 12 step 3.3.5

History:

Modified QID 2517 for 1998 RO/SRO Exam.
Used in A. Morris 98 RO Re-exam
Selected for 2002 RO/SRO exam.
Selected for 2010 RO/SRO exam

3.3.4 Average Th and Tc

The SASS selected loop A and B hot leg temperatures are averaged by an average amplifier. A selector switch, located on C03, allows selecting either the loop A hot leg temperature, the loop B hot leg temperature, or the average hot leg temperature.



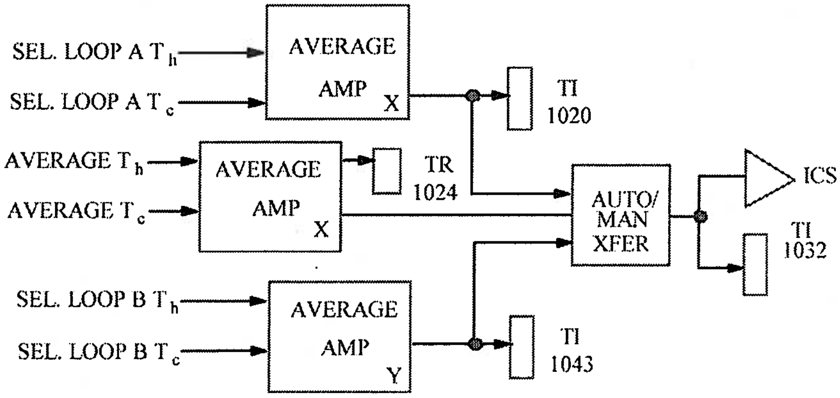
Normally the average hot leg temperature is selected. The selected temperature inputs into ICS for calculation of OTSG BTU limits. The selected temperature is also displayed on the T_{hot} temperature recorder located on C-13.

The SASS selected loop A and B cold leg temperatures are also averaged by an average amplifier. This average cold leg temperature 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.

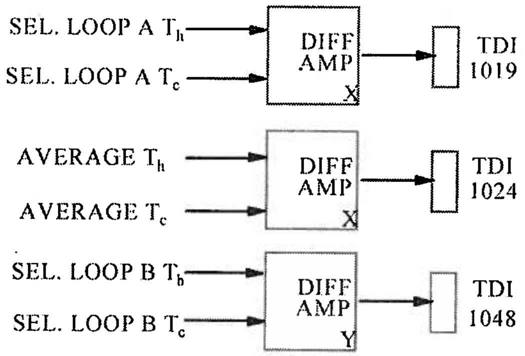
Average amplifiers average the SASS selected Th and Tc signals. RCS loop “A” average temperature is calculated by the NNIX channel. The NNIY channel calculates RCS loop “B” average temperature. Unit Tave is calculated from the average Th and Tc (loop “A” and loop “B” Th and Tc are averaged) signals.



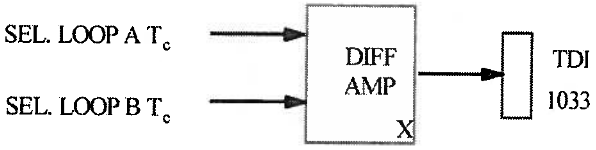
Loop A average temperature, loop B average temperature, and Unit average temperature provide input to an auto/manual selector switch. The Tave selector switch output supplies the digital Tave indicator on C03 and ICS for temperature control. Unit Tave, loop A Tave, or loop B Tave may be selected by depressing 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 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.



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

QID: 0240 **Rev:** 0 **Rev Date:** 8-17-99 **Source:** Direct **Originator:** Don Slusher

TUOI: ANO-1-LP-RO-NNI **Objective:** 25 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-Core Temperature Monitor (ITM) System

Description: Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors.

K/A Number: K6.01 **CFR Reference:** CFR: 41.7/45.7

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

Group: 2 **SRO Imp:** 3.0 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** 62 **SRO:** 62

Given:

- Plant is at 100% power
- All CETs indicate 602 °F

ICC train "B" Core Exit Thermocouple TE-1152 fails to 900 °F.

What is the effect of this failure?

A. Core Exit Thermocouple TE-1152 will be removed from the average.

B. ICC Core Exit Thermocouple indication will go to ~627 °**F**.

C. "TRAIN B SUBCLG MARG LO" annunciator will alarm.

D. "B" SPDS will switch from ATOG to the ICC display.

Answer:

A. Core Exit Thermocouple TE-1152 will be removed from the average.

Notes:

CETs are averaged together to generate alarms, indication, or action. Therefore, "b", "c", and "d" are incorrect and "a" is correct since ICCMDS will determine that TE-1152 is unreliable and remove it from the average.

References:

1105.008 Rev 17

History:

Developed for 1999 exam.
Used on 2004 RO/SRO Exam.
Selected for 2010 RO/SRO exam

- NARR Range Hotleg LVL

  LT-1189
  LT-1191
  LT-1193
  LT-1195
  LT-1190
  LT-1192
  LT-1194
  LT-1196

- RCS Pressure

  PT-1042
  PT-1041

- Reactor Coolant Pump Contacts

3.1.1 Reactor Vessel Level Sensors

Two level probes, each having nine level sensors and an absolute thermocouple near the top, are installed in the reactor vessel through the head at the center CRDM location (center CRD no longer used). Level is sensed at approximately 2' intervals from the top of the dome to near the top of the fuel assemblies.

A level sensor consists of two thermocouples connected internally to provide a signal proportional to the temperature difference. One thermocouple is heated by an internal heater element in the probe. The area around the heated thermocouple has a different heat transfer coefficient to the surrounding RCS and, therefore, has a different sensitivity to water or steam. As the water level drops below the level sensor, its ΔT changes and provides wet or dry indication.

The absolute thermocouple provides head fluid temperature indication from near the top of the head.

TS 3.3.15 includes the reactor vessel level sensors (RVLMS).

3.1.2 Core Exit Thermocouples

Twenty-four qualified core exit thermocouples provide temperature indication in a range of 50°F to 2300°F. These instruments are part of the incore detector system and are installed through the bottom of the reactor vessel through the incore instrument guide tubes. All valid CETs are averaged, and each CET is compared to the average. If a significant deviation exists, the CET is flagged SUSPECT. Failed or suspect CETs are automatically excluded from the average. TS 3.3.15 includes the core exit thermocouples.

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

QID: 0138 **Rev:** 0 **Rev Date:** 12/02/98 **Source:** Direct **Originator:** B. Short

TUOI: AA51002-013 **Objective:** 9 **Point Value:** 1

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

System Number: 045 **System Title:** Main Turbine Generator System

Description: Ability to manually operate and/or monitor in the control room: Turbine stop valves.

K/A Number: A4.06 **CFR Reference:** 41.7 / 45.5 to 45.8

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

Group: 2 **SRO Imp:** 2.7 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 63 **SRO:** 63

During the performance of Main Turbine Governor Valve testing, while governor valve #1 was closed in the test position governor valve #3 fails closed. What turbine problems does this impose?

A. Moisture impingement on the turbine blading.

B. Thermal shock to the turbine rotor.

C. Turbine will trip due to low load.

D. Turbine overspeed condition.

Answer:

B. Thermal shock to the turbine rotor.

Notes:

(A) is incorrect. The closure of both valves does not change the quality of the steam.
(B) is correct. Closure of GV1 and GV3 with GV2 & GV4 open or closure of GV2 & GV4 with GV1 & GV3 open causes thermal shock on the turbine rotor.
(C) is incorrect. The load shifts through the two valves that remain open.
(D) is incorrect. The load will stay essentially the same so that an overspeed condition should not occur.

References:

1106.009 (Change 37)

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2010 RO/SRO exam

CAUTION

- Thermal shock can damage turbine rotor if either of the following occurs:
  - Simultaneous closing of GV-1 and GV-3 with GV-2 and GV-4 both open.
  - Simultaneous closing of GV-2 and GV-4 with GV-1 and GV-3 both open.
- Under no circumstance should more than one Governor Valve be operated at the same time.

12.3.3 Continuously monitor Governor Valve positions.

- Do NOT allow GV-1 and GV-3 to be closed simultaneously with GV-2 and GV-4 both open.
- Do NOT allow GV-2 and GV-4 to be closed simultaneously with GV-1 and GV-3 both open.

12.3.4 IF plant response becomes erratic when closing or opening Governor Valves,
THEN release the pushbutton, allow plant to stabilize and then continue.

CAUTION

Governor Valve operation with turbine controls not in ICS auto may cause large load changes.

NOTE

- Both GV CLOSE and GV OPEN pushbuttons will be backlit after GV CLOSE pushbutton is depressed.
- Pushbuttons will be backlit even if governor valve is already closed.

12.4 Slowly close the Governor Valve associated with the servo being repaired/replaced by depressing the GV CLOSE pushbutton.

- GV-1 "A" Governor Valve
- GV-2 "B" Governor Valve
- GV-3 "C" Governor Valve
- GV-4 "D" Governor Valve

12.4.1 WHEN Governor Valve is closed
AND associated pushbutton backlights are on,
THEN release GV CLOSE pushbutton.

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

QID: 0798 **Rev:** 0 **Rev Date:** 9/16/2009 **Source:** New **Originator:** S. Pullin

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

Section: 3.8 **Type:** Plant Services System

System Number: 075 **System Title:** Circulating Water System

Description: Knowledge of abnormal condition procedures.

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

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

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

Question: **RO:** 64 **SRO:** 64

Given:

- Plant at 100% power
- Lake Temperature is 65 F
- P-3A, P-3B, and P-3C Circulating Water Pumps are running
- P-3A Circulating Water Pump trips.
- P-3D Circulating Water Pump standby pump was started.
- It is noticed that the condenser waterbox discharge temperature is 10 degrees higher and condenser vacuum is dropping.
- AOP 1203.016, Loss of Condenser Vacuum, has been entered.

Which of the following is the cause for these conditions?

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. Lake temperature is too high for 3 circulating water pump operation per 1104.008, Circulating Water and Water Box Vacuum System Operation.

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. 1104.008 states that 4 CW Pumps are needed when lake temperature is above 67 F

References:

1104.008, Circulating Water System, change 27, page13, Caution

History:

New for 2010 RO/SRO exam

CAUTION

- Stopping CW pump during radwaste release could result in Tech Spec violation for exceeding MPC requirements at site boundary.
- Stopping CW pump during chemical release (NT dump or biocide injection) could result in violation of NPDES requirements.
- 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.
- Debris in circ water bay will become stirred when Circ Water Pump is stopped. A greater potential of Service Water pump strainer fouling exists when stopping P-3B OR P-3C.

8.4 Circ Water Pump Stop

8.4.1 IF normal pump rotation,
THEN verify no radioactive releases in progress on either unit.

A. IF a radioactive release in progress,
THEN verify either of the following is performed:

- Do NOT continue until the release is complete, or
- Terminate the release.

8.4.2 IF normal pump rotation
AND trench release in progress per Turbine Building Draining System (1104.044), "Turbine Building Trench Continuous Release",
THEN verify Chemistry notified of change in Circ Water flow configuration.

8.4.3 IF normal pump rotation,
THEN verify no radioactive release permits have been submitted to Chemistry from either unit.

A. IF radioactive release permit has been submitted,
THEN verify either of the following is performed:

- Release permit is cancelled, or
- Release calculations are re-performed for new estimated dilution flow rate.

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

QID: 0542 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Direct **Originator:** NRC

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

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

System Number: 086 **System Title:** Fire Protection System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fire Protection System controls including: Fire header pressure.

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

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

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

Question: **RO:** 65 **SRO:** 65

You are on watch in the Control Room when the following annunciator alarms:

- K12-A1, "FIRE"

As Fire Water Header pressure drops from 110 psig to 80 psig select the order that fire pumps would start.

A. Jockey FWP P-11; Diesel Fire Pump P-6B starts second; Electric Fire Pump P-6A starts last.

B. Electric Fire Pump P-6A; Diesel Fire Pump P-6B starts second; Jockey FWP P-11 starts last.

C. Electric Fire Pump P-6A; Jockey FWP P-11 starts second; Diesel Fire Pump P-6B starts last.

D. Jockey FWP P-11; Electric Fire Pump P-6A starts second; Diesel Fire Pump P-6B starts last.

Answer:

D. Jockey FWP P-11; Electric Fire Pump P-6A starts second; Diesel Fire Pump P-6B starts last.

Notes:

"D" is correct, Jockey FWP P-11; Electric Fire Pump P-6A starts second; Diesel Fire Pump P-6B starts last. The other choices are incorrect based on pressure to start for each one.

References:

STM 1-60, Fire Protection System, rev 8, page 2.

History:

Developed by NRC.
Used on 2004 RO Exam.
Selected for 2010 RO/SRO exam

The Pre-Fire Plans are used by operations and the fire brigade. They contain detailed information for each fire zone. Copies are located in both control rooms and the fire locker on 386 elevation (for fire brigade use).

The Fire Brigade provides the trained on-site personnel required to attack and control a fire. Three fire brigade members and two support personnel are on-site at all times. The fire brigade has been trained and drilled in fire fighting strategy. Abnormal operating procedures and the Emergency Plan provide for contacting the Russellville Fire Department and other agencies should the magnitude of the fire exceed on-site fire fighting capabilities.

2.0 COMPONENT DESCRIPTION

2.1 Fire Water System

(Refer to Figure 60.1)

Three fire water pumps in the intake structure take their suction from the Service Water bays. The motor-driven jockey pump (P11) is a small displacement pump whose function is to makeup for small leaks and thereby maintain fire main pressure. This keeps the high capacity pumps P6A&B from starting inadvertently, reducing wear on the larger pumps and thus increasing their reliability. An actuation of a sprinkler system will cause header pressure to drop significantly, starting the electric motor driven fire pump (P6A) first and, if pressure drops further, the diesel-driven fire pump (P6B) next.

The fire water pumps discharge to the fire water loop, a 12-inch main that surrounds both units and is buried below the frost line for freeze protection. A loop configuration with isolation valves is used so a broken pipe in the loop can be isolated without isolating all of the fire header downstream of the break.

There are other fire water loops tapping off the yard fire main. These loops supply water to the Turbine, Auxiliary, Administration, and Reactor Buildings, and to the Switchyard and Transformer areas.

All fire water loops have hose reels spaced at a maximum of 100 ft. apart. Each hose reel has fifty feet of 1½ inch hose available for use. Hydrants are also provided at various locations and are spaced approximately 250 ft apart.

2.1.1 Fire Water Pumps

The fire water pumps, P6A&B, are vertical mounted, 3 stage, centrifugal pumps rated at 2500 gpm at a discharge pressure of 150 psig. The pumps are identical while the drivers are diverse to ensure a high volume fire water source will be available for all plant conditions. Both pumps use identical relief type recirc valves which are set at 150 psig. Flow through the relief can be checked by a sight glass. The pumps are located in the intake structure and take suction on the Service Water bays.

RO Written Exam

Tier 3

Facility: **Arkansas Nuclear One – Unit 1** Date of Exam: **3/5/2010**

Category	K/A #	Topic	RO			
			IR	#	QID	Type#
1. Conduct of Operations	**2.1.23**	Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.3	66	482	D
	2.1.31	Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	4.6	67	800	N
	2.1.32	Ability to explain and apply system limits and precautions.	3.8	68	799	N
	Subtotal			3		
2. Equipment Control	**2.2.1**	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could effect reactivity.	4.5	69	160	D
	2.2.37	Ability to determine operability and / or availability of safety related equipment	3.6	70	801	N
	Subtotal			2		
3. Radiation Control	**2.3.4**	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	71	802	N
	2.3.11	Ability to control radiation releases.	3.8	72	436	R
	Subtotal			2		
4. Emergency Procedures / Plan	**2.4.6**	Knowledge of EOP mitigation strategies.	3.7	73	803	N
	2.4.11	Knowledge of abnormal condition procedures.	4.0	74	161	D
	2.4.50	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.2	75	804	M
	Subtotal			3		
Tier 3 Point Total				10		

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

QID: 0482 **Rev:** 0 **Rev Date:** 10/7/2003 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-WCO-CZ **Objective:** 13 **Point Value:** 1

Section: 3.9 **Type:** Radioactivity Release

System Number: 068 **System Title:** Liquid Radwaste System (LRS)

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

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

Tier: 3 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 66 **SRO:** 66

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

A. Estimate radiation level every four hours during the release.

B. Have an independent sample obtained and analyzed prior to release.

C. Estimate flow rate at least once every three hours during release.

D. T-16A can NOT be released if RI-4642 is inoperable.

Answer:

B. Have an independent sample obtained and analyzed prior to release.

Notes:

Answer "B" contains the requirement from Att. B1 of 1104.020. The other answers are incorrect.

2004 Exam Development Note: Randomly selected alternate K/A 2.1.23 to replace 2.1.31 due to lack of CR controls at ANO for the Liquid Radwaste system.

References:

1104.020, Change 49, Att. B1, section 2

History:

Modified regular exambank QID #2765.
Used on 2004 RO/SRO Exam.
Selected for 2010 RO/SRO

1.5 Record the following:

1.5.1 Number of CW Pumps running ____

<u>AND</u> CW pump Disch Press _______ psig

1.6 <u>IF</u> adjustments are made to CW flow,
<u>THEN</u> terminate release.

1.7 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 <u>IF</u> Liquid Radwaste Process Monitor (RI-4642) is inoperable
<u>OR</u> unavailable as identified in either "Request", or "Verification of Pre-Release Requirements" sections of this permit,
<u>THEN</u> 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 <u>IF</u> release is radioactive
<u>AND</u> release desired,
<u>THEN</u> 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 <u>IF</u> Liquid Radwaste Process Monitor (RI-4642) is inoperable
<u>OR</u> unavailable as identified in either "Request", or "Verification of Pre-Release Requirements" section of this permit,
<u>THEN</u> perform independent analysis of computer data input.

Date/Time___________/__________

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

QID: 0800 **Rev:** 0 **Rev Date:** 9/16/2009 **Source:** New **Originator:** S Pullin

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

Section: 2 **Type:** Generic K&A

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to locate control room switches, controls, and indications, and to determine they correctly reflect the desired plant lineup.

K/A Number: 2.1.31 **CFR Reference:** 41.10 / 45.12

Tier: 3 **RO Imp:** 4.6 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 67 **SRO:** 67

Given:

LOCA in progress has caused ESAS actuation of Channel 1-4

Which of the following combinations of indications and locations
are correct for the given condition?

A. CV-3820, "SW TO ICW," green light, on C16;
CV-1270, "RCP SEAL BLEEDOFF FROM D RCP," red light, on C18;
CV-1053, "QUENCH TANK DRAIN," green light, on C16

B. CV-1233, "RCS MAKEUP," red light, on C16;
CV-1441, "BWST PURIF RECIRC ISOL," green light, on C13;
CV-5612 ,"FIRE WATER TO RB," green light, on C18.

C. CV-1285, "HIGH PRESSURE INJECTION," red light, on C16;
CV-1407, "BWST OUTLET," red light, on C18;
CV-3841, "LPI PUMP BRG CLR E-50 INLET," red light, on C16

D. CV-1408, "BWST OUTLET," red light, on C18;
CV-7402, "RB PURGE INLET," green light, on C18;
CV-4804, "RB VENT," red light, on C16

Answer:

C. CV-1285, "HIGH PRESSURE INJECTION," red light, on C16;
CV-1407, "BWST OUTLET," red light, on C18;
CV-3841, "LPI PUMP BRG CLR E-50 INLET," red light, on C16

Notes:

C is correct in that it has the correct indications and panel locations.
A is incorrect in that it has the incorrect indications and correct panel locations.
B is incorrect in that it has the correct indications and incorrect panel locations.
D is incorrect in that it has the incorrect indications and incorrect panel locations.

References:

STM 1-65 Rev 5 ESAS
STM 1-05 Rev 16 DHR

History:

New selected for 2010 RO/SRO exam

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).

20. Identify all devices actuated by ESAS to include post actuation condition or condition

- 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.

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).

20. Identify all devices actuated by ESAS to include post actuation condition or condition

- 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.

The recirc flow path ensures a flow of >80 gpm for pump protection during periods of low flow.

If Service Water flow is maintained to the cooler, run time in the recirculation mode is not restricted due to the high volume recirculation flow. No discharge isolation valves are provided. Each decay heat pump has a discharge check valve (DH-2A & 2B) on the discharge line to the DHR cooler. A motor current monitor with a variable alarm setpoint has been provided to detect vortex formation when the system is operating with reduced RCS levels in the DHR mode of operation. There are no interlocks on the DHR Pumps that would prevent a pump start with the suction or discharge valves closed. Engineered Safeguards (ES) signals will start the pumps based on RCS or Reactor Building conditions regardless of valve alignments. Pump operation with the suction lines closed can cause pump damage in a very short time. For this reason, it is very important to check valve alignments when DHR/LPI Pumps are prepared for immediate or standby operation.

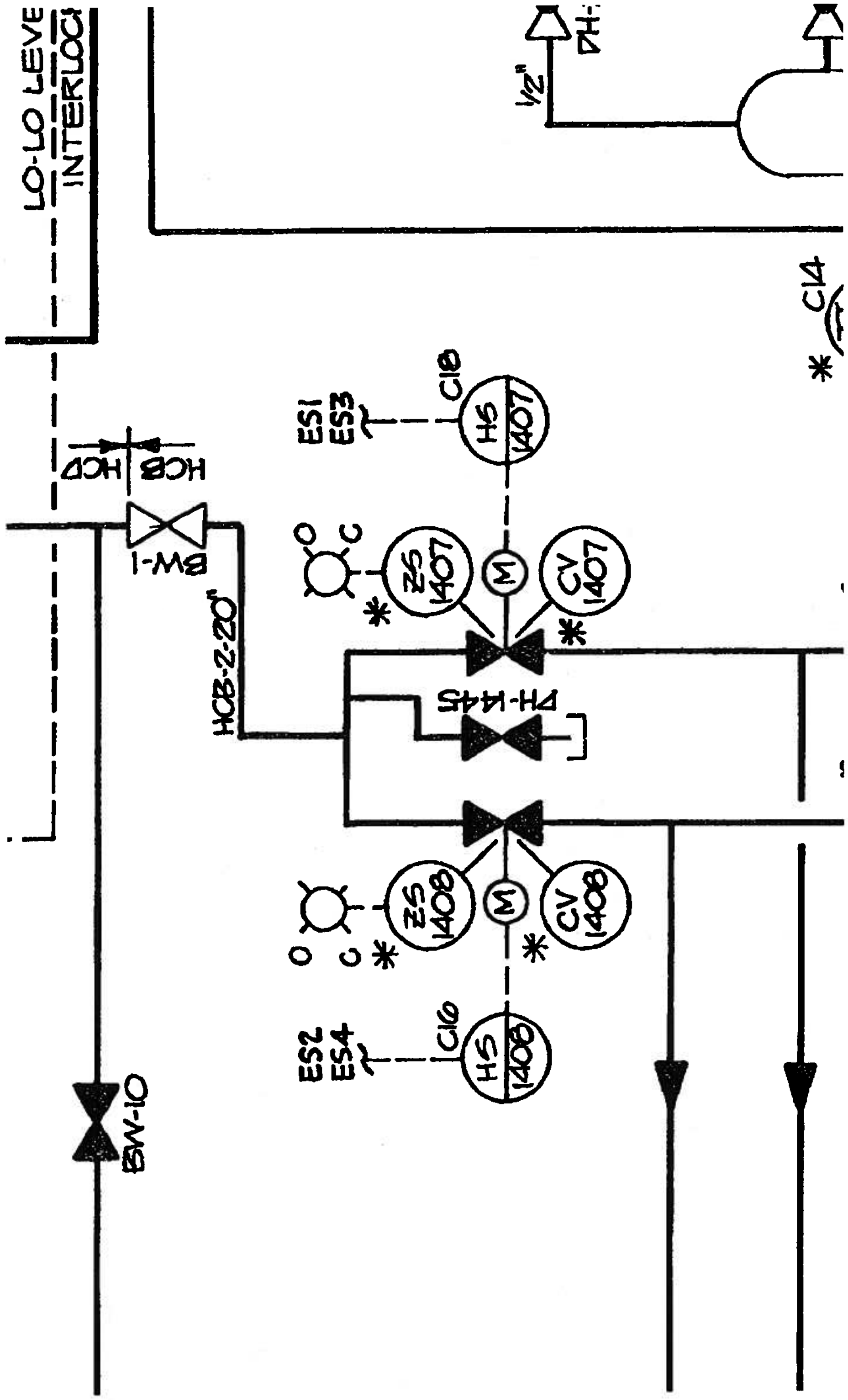
2.1.3.1 Pump Bearing Lubrication

DHR /LPI pump and motor bearings are lubricated by a slinger ring. A loose collar is hung on the pump shaft at each bearing location. The collar or "slinger ring" hangs down into a bearing oil sump and slings oil onto the rotating parts in the bearing housing. Oil level is checked using "Bulls eye" sight glasses on the motor and vertical sight glasses at the pump bearings. "Trico" automatics oilier are provided at the bearings to maintain oil level in the bearing sumps. The Automatic oilers are clear plastic reservoirs inverted on a supply pipe. Oil level is visible at all times. Experience has shown that the Trico oilers are susceptible to vapor binding and close checks of oil levels are required after extended runs of the associated pump. Level gauges at the bearing sump should be checked to verify proper sump oil levels.

To determine if slinger rings are operating properly observation ports are provided. The observation ports are located on the opposite side of the level indicator. By using a flashlight during pump operation, the slinger ring can be viewed to determine if the slinger rings are operating properly. During pump operation the slinger ring should be rotating on the shaft. If the slinger rings are not rotating on the shaft notify control room personnel at once.

2.1.3.2 Pump Cooling

Pump bearing and stuffing box cooling is supplied by the Service Water System. Cooling water flows through the pump oil and stuffing box jacket coolers (E-50A & B) and then flows to the Service Water return line. Service water flow through the cooler is normally isolated when the pump is secured by an air operated control valve. CV-3840 provides isolation for P-34A and CV-3841 for P-34B. The associated control valve receives an open signal when the pump is started. The control valves can be manually opened when maintenance is performed or when valve fails required stroke time to maintain pump operability. Indication is provided on C-16 & C-18.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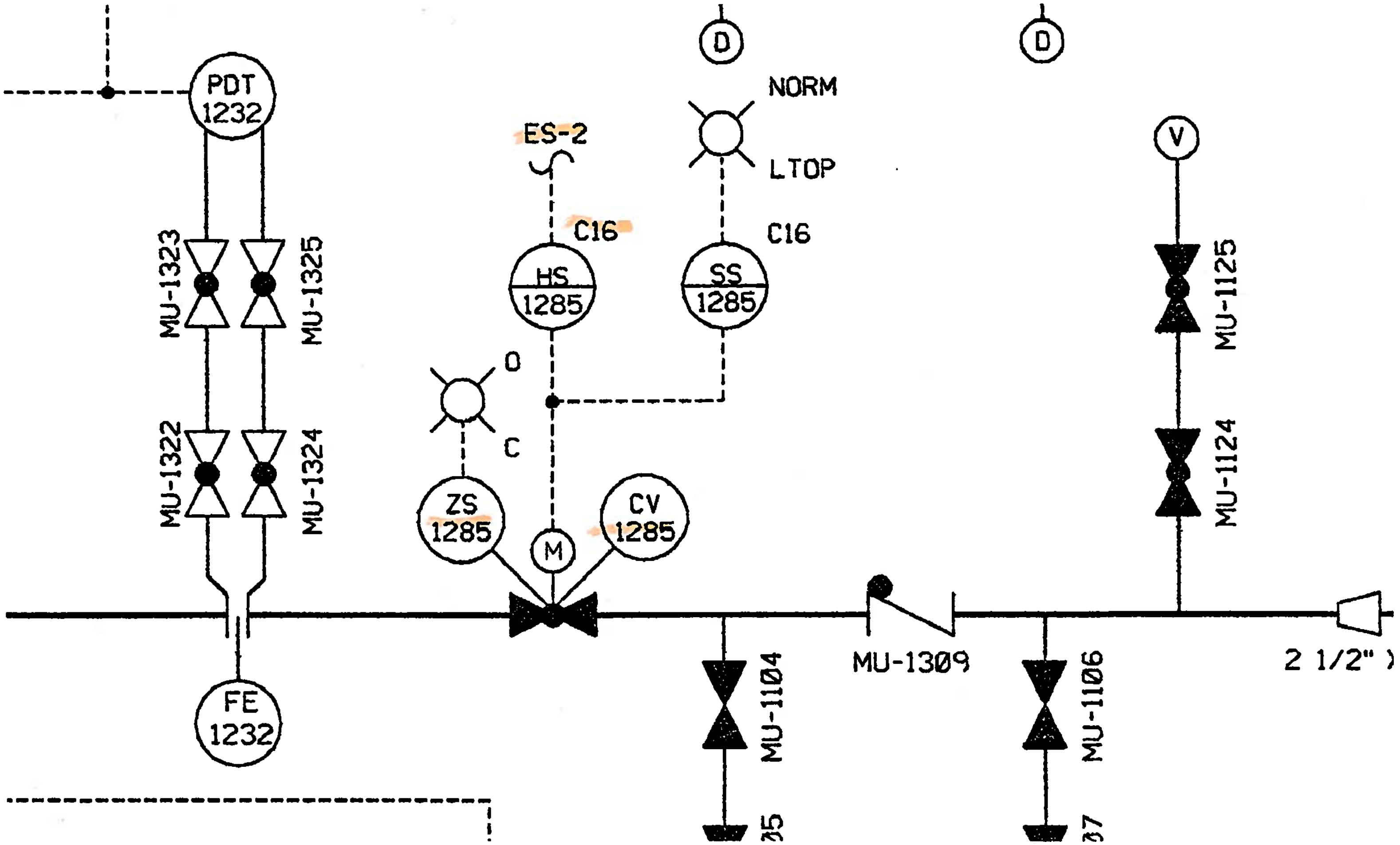
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History:

New for 2010 RO/SRO exam

QID: 0799 **Rev:** 0 **Rev Date:** 9/16/2009 **Source:** New **Originator:** S Pullin

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

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

System Number: 2.1 **System Title:** Conduct of Operations

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:	3	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	4
Group:		**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** 68 **SRO:** 68

Procedure 1105.004, "Integrated Control system" limit and precaution states do not operate Reactor Demand H/A station in Auto with both S/Gs on low level limits.

What is the reason for this precaution and does any exception apply?

A. Due T-ave reduction as power lowers rods will pull to maintain T-ave at setpoint, you can operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits if you adjust T-ave setpoint to match reactor power

B. Due T-ave reduction as power lowers rods will not move due to T-ave error, you can not operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits

C. When S/Gs are on Low Level Limits, the Tave calibrating integral is blocked,. you can operate with Reactor Demand H/A station in Auto with both STGs on low level limits providing you verify calibrating integral is blocked on PDS.

D. When S/Gs are on Low Level Limits, the Tave calibrating integral is released, you can not operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits.

Answer:

A. Due T-ave reduction as power lowers rods will pull to maintain T-ave at setpoint, you can operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits if you adjust T-ave setpoint to match reactor power

Notes:

A is correct, due to lowering power with S/G on LLL will cause Tave to ramp down. The Rx Demand station will try to pull rods to maintain 579 F. Limit & Precaution allows this mode of operation only if you reduce Tave setpoint to match Rx power.
B, C and D are incorrect

References:

OP-1105.004 Change 20

History:

New selected for 2010 RO/SRO exam.

5.6 If it is necessary to operate either the startup or low load valve in manual, both startup and low load valves in that Loop should be placed in manual and valves operated in normal sequence.

5.6.1 Do not modulate startup valve unless low load valve is closed.

5.6.2 Do not modulate low load valve unless startup valve is full open.

5.7 Do not operate Reactor Demand H/A station in AUTO with both OTSGs on low level limits unless T-ave setpoint is adjusted to match desired reactor power.

5.8 Operation of either Startup Valve in HAND when SG pressure is ≥750 psig requires entry into TS 3.7.3 Condition D.

5.9 Operation of either Low Load Valve in HAND when SG pressure is ≥750 psig requires entry into TS 3.7.3 Condition C.

5.10 Operation of both Startup Valve and Low Load Valve in HAND when SG pressure is ≥750 psig requires entry into TS 3.7.3 Condition E.

5.11 Due to offset of the prongs on the light bulbs used in the ICS H/A stations, it is necessary to ensure proper alignment prior to replacing bulbs to avoid damage to the H/A station or shorting of ICS circuitry. (CR-ANO-1-2004-2382, CR-ANO-1-2004-2384)

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

QID: 0116 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork
TUOI: A1LP-RO-NOP **Objective:** 7 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As
System Number: 2.2 **System Title:** Equipment Control
Description: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

K/A Number: 2.2.1 **CFR Reference:** 45.1
Tier: 3 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2
Group: **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 69 **SRO:** 69

During an INITIAL approach to criticality, if criticality is NOT achieved within _________ of the ECC, then insert ________ and ________ .

A. Plus or minus 1.0% delta k/k
control rods to achieve 1.5% SD margin
establish hot shutdown conditions

B. Plus or minus 1.0% delta k/k
regulating groups to achieve 1.0% SD margin
notify Reactor Engineering

C. Plus or minus 0.5% delta k/k
control rods to achieve 1.5% SD margin
verify calculation

D. plus or minus 0.5% delta k/k
regulating groups to achieve 1.0% SD margin
verify calculation

Answer:

C. plus or minus 0.5% delta k/k
control rods to achieve 1.5% SD margin
verify calculation

Notes:

Answer "C" is correct per 1102.008.

References:

1102.008, Chg. 023

History:

Used in 1998 RO exam
Used in NRC developed RO exam 8/24/92, no. 88
Used in A. Morris 98 RO Re-exam
Used in 2001 RO Exam
Selected for 2010 RO/SRO exam

9.7 Sequentially withdraw regulating groups in ≤30% increments, per CRD System Operating Procedure (1105.009), "Regulating Group Sequential Withdrawal" section. Perform the following during rod withdrawal:

- IF unexpected situations/conditions arise,
  THEN take conservative actions to place the reactor in a safe condition.
- Continuously monitor available instrumentation for doubling count rate and unplanned criticality.
- IF unexpected count rate/power rise is observed
  THEN immediately insert control rods to stop rise or if required trip the reactor.
- At ≤30% rod position increments stop rod withdrawal, allow count rate to stabilize, and collect data for 1/m plot.

9.8 IF criticality is achieved within
procedural limits of ± 0.5% Δk/k
AND NOT within ± 0.25% Δk/k,
THEN notify Reactor Engineering to initiate a condition report AND continue this procedure. (CR-ANO-1-2009-0237)

9.9 IF this is a startup immediately following refueling
AND a rod index of 300% is within the ECC band
AND criticality is NOT achieved by a rod index of 300%,
THEN inform Reactor Engineering and refer to 1302.020 for completion of the approach to criticality.

9.10 IF criticality is NOT achieved within ±0.5% Δk/k of the ECC,
THEN insert control rods to obtain ≥1.5% subcritical conditions, and perform the following:

9.10.1 Inform Reactor Engineering.

9.10.2 Verify boron concentrations.

9.10.3 Verify ECC calculation.

9.10.4 Verify position of all control rods by comparing API to zone or limit position switches.

9.10.5 WHEN cause of ECC error is determined,
AND cause corrected,
THEN re-perform AND re-initial applicable steps of this procedure.

9.11 Record time reactor is made critical _________.

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

QID: 0801 **Rev:** 0 **Rev Date:** 9/17/2009 **Source:** New **Originator:** S Pullin

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

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

System Number: 2.2 **System Title:** Equipment Control

Description: Ability to determine operability and / or availability of safety related equipment.

K/A Number: 2.2.37 **CFR Reference:** 41.7 / 43.5 / 45.12

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

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

Question: **RO:** 70 **SRO:** 70

REFERENCE PROVIDED

Which of the following plant conditions would require entry into LCO 3.2.1 due to exceeding Regulation Rod Insertion Limits per the COLR?

A. 80% Power, 4 RCP's in service, 150 EFPD, Rod Index of 250 %

B. 70% Power, 4 RCP's in service, 300 EFPD, Rod Index of 220 %

C. 60% Power, 3 RCP's in service, 100 EFPD, Rod Index of 265 %

D. 50% Power, 3 RCP's in service, 350 EFPD, Rod Index of 255 %

Answer:

B. 70% Power, 4 RCP's in service, 300 EFPD, Rod Index of 220 %

Notes:

Per the graphs in the COLR answer (B) falls within the Operation Restricted area of the figure and would require entry into LCO 3.2.1.
A, C, and D do not require entry into LCO

References:

ANO-1 Cycle 22 COLR Figures 3-A through 4-B

History:

New for 2010 RO/SRO exam

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Regulating Rod Insertion Limits

LCO 3.2.1 Regulating rod groups shall be within the physical insertion, sequence, and overlap limits specified in the COLR.

---NOTE---
Not required for any regulating rod repositioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

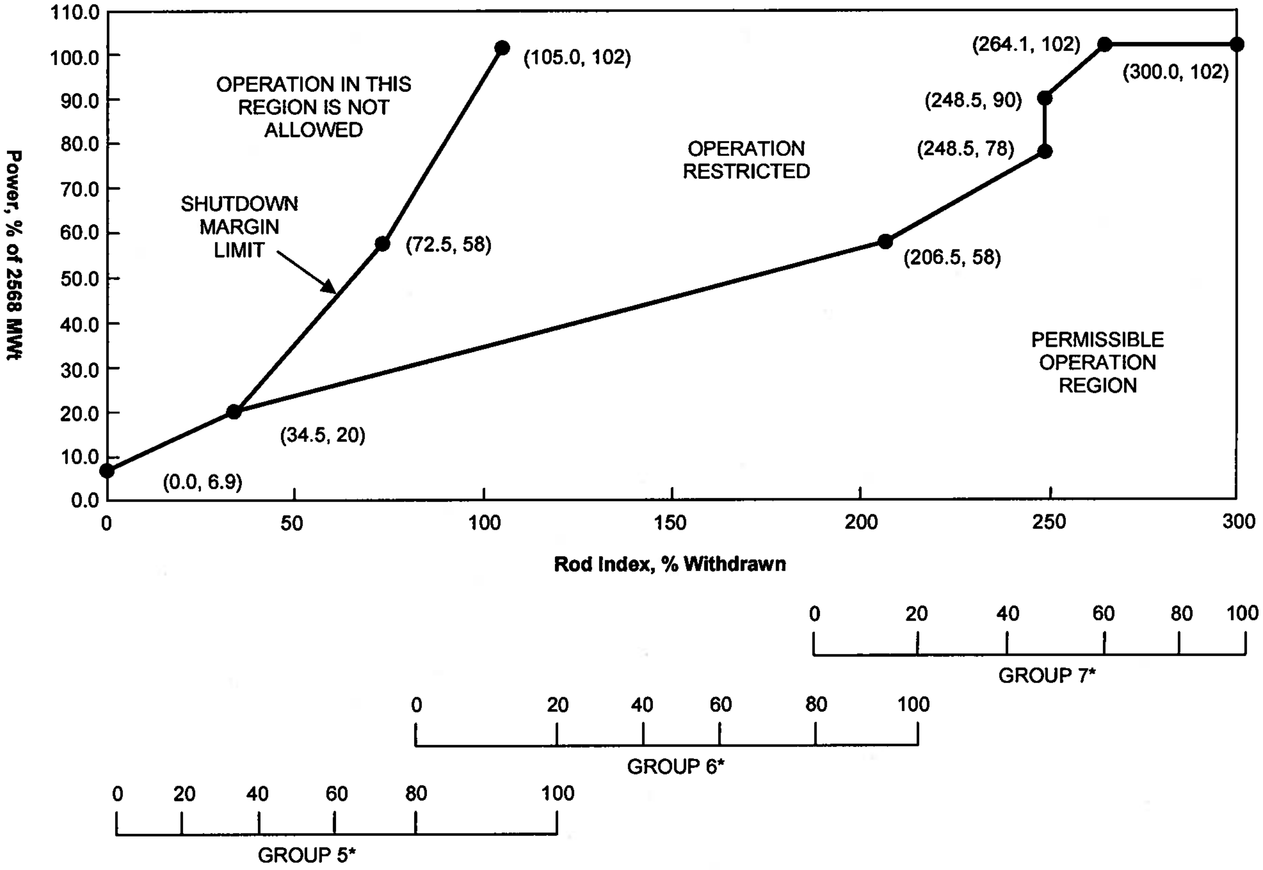
ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Regulating rod groups inserted in restricted operation region.	A.1	-------------NOTE------------- Only required when THERMAL POWER is > 20% RTP. ------------------------------------	
		Perform SR 3.2.5.1.	Once per 2 hours
	<u>AND</u>		
	A.2	Restore regulating rod groups to within acceptable region.	24 hours from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group insertion limits.	2 hours
C. Regulating rod groups sequence or overlap requirements not met.	C.1	Restore regulating rod groups to within limits.	4 hours

Figure 3-A

Regulating Rod Insertion Limits for Four-Pump Operation From 0 to 200 ± 10 EFPD

(Figure is referred to by Technical Specification 3.2.1)

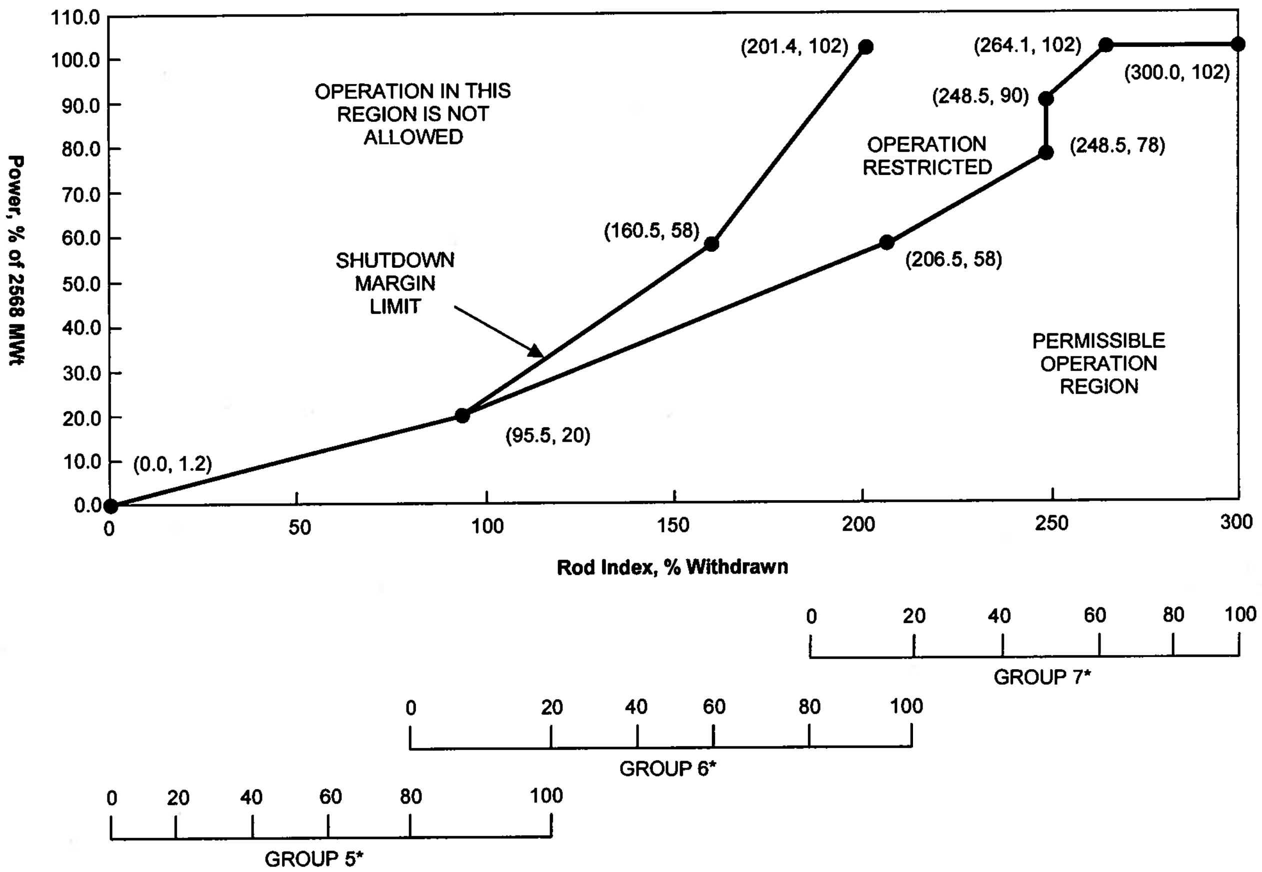


* Technical Specification 3.5.2.5(2) requires that operating rod group overlap be 20% ± 5% between two sequential groups, except for physics tests.

Figure 3-B

Regulating Rod Insertion Limits for Four-Pump Operation From 200 ± 10 EFPD to EOC

(Figure is referred to by Technical Specification 3.2.1)

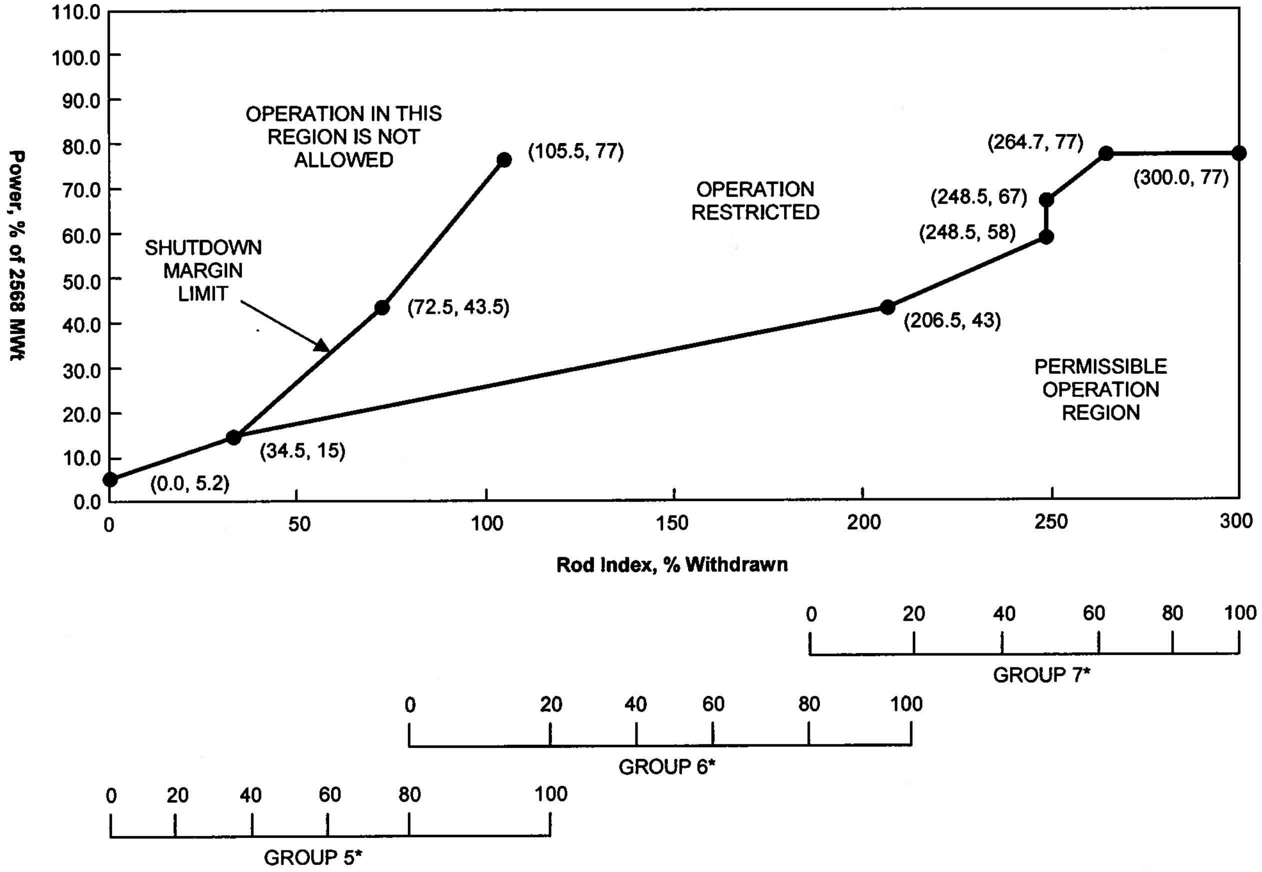


* Technical Specification 3.5.2.5(2) requires that operating rod group overlap be 20% ± 5% between two sequential groups, except for physics tests.

Figure 4-A

Regulating Rod Insertion Limits for Three-Pump Operation From 0 to 200 ± 10 EFPD

(Figure is referred to by Technical Specification 3.2.1)

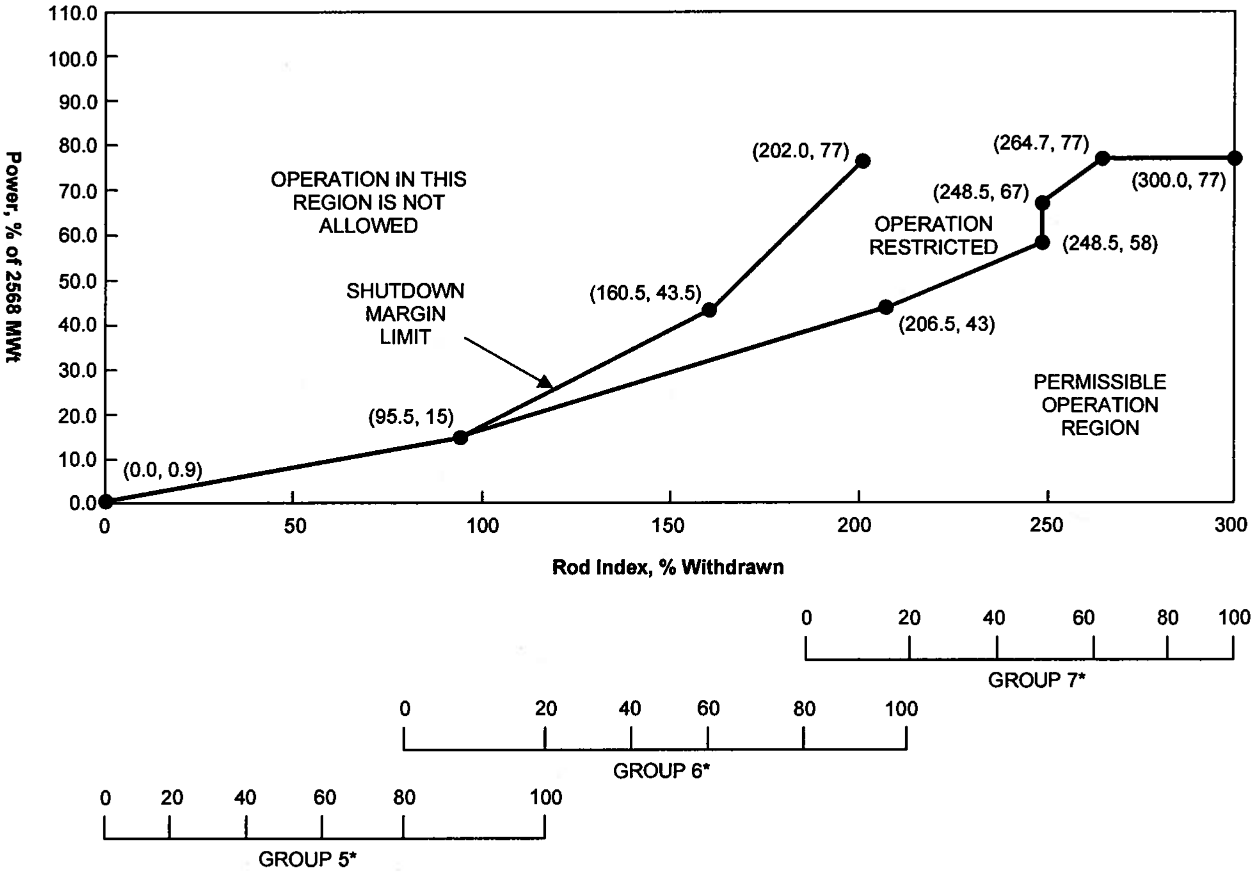


* Technical Specification 3.5.2.5(2) requires that operating rod group overlap be 20% ± 5% between two sequential groups, except for physics tests.

Figure 4-B

Regulating Rod Insertion Limits for Three-Pump Operation From 200 ± 10 EFPD to EOC

(Figure is referred to by Technical Specification 3.2.1)



* Technical Specification 3.5.2.5(2) requires that operating rod group overlap be 20% ± 5% between two sequential groups, except for physics tests.

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

QID: 0802 **Rev:** 0 **Rev Date:** 9/17/2009 **Source:** New **Originator:** S. Pullin

TUOI: ASLP-RO-RADP **Objective:** 15 **Point Value:** 1

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

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier:	3	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** 71 **SRO:** 71

Given:

- A General Emergency has been declared on Unit 1.
- A Maintenance crew must enter a radiological area with a dose rate of 150 Rem/Hr to protect valuable property.

Which of the following is the MAXIMUM time an individual team member can stay in this area?

A. 4 minutes

B. 6 minutes

C. 8 minutes

D. 10 minutes

Answer:

A. 4 minutes

Notes:

A is correct, for protecting valuable property 10 Rem is the does limit.
B, C and D exceed 10 Rem limit.

References:

OP-1903.033 Change 019-01-0

History:

New for 2010 RO/SRO exam

Dose limit* (rem TEDE)	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Life saving or protection of large populations	Lower dose not practicable
>25	Life saving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved (refer to Attachment 1 of this procedure for health risks).

* Workers performing services during emergencies should limit dose to the lens of the eye to <u>three times</u> the listed value and doses to any other organ (including skin and body extremities) to <u>ten times</u> the listed value.

6.1.4 Rescue/repair and damage control personnel shall perform their duties in the most safe and efficient manner possible. Once their operations have been completed, they shall follow self-monitoring and personnel decontamination procedures as specified by the Health Physics Supervisor.

6.2 ACTIONS

NOTE

[During a "Personnel Emergency" the Emergency Medical Team may enter Radiologically Controlled Areas without SRDs or Alarming Dosimeters as long as an HP Technician is providing radiological instructions and is monitoring dose rates and time in the area. Prompt medical attention shall take precedence over HP procedures for a seriously injured individual.]

6.2.1 Personnel selected for the rescue/repair and damage control teams should report to the OSC (unless otherwise instructed) for their briefing.

6.2.2 The rescue/repair and damage control team leader shall function under the direction of the Shift Manager/OSC Director.

6.2.3 <u>Immediate Actions</u>

A. **<u>IF</u>** exposure to significant radioiodine concentrations is possible,
<u>THEN</u> refer to procedure 1903.035, "Administration of Potassium Iodide" for guidance.

B. Rescue/repair and damage control teams shall be briefed using Form 1903.033B, "OSC Team Briefing Form". This form serves as an emergency RWP and Work Order. Instructions for conducting re-entry team briefings are contained in Attachment 3.

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

QID: 0436 **Rev:** 0 **Rev Date:** 4/30/2002 **Source:** Repeat **Originator:** J.Cork

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:	G	**SRO Imp:**	4.3	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 72 **SRO:** 72

The WCO is preparing to commence a liquid release on TWMT T-16A when he notices that there is no tag hanging on T-16A inlet valve CZ-47A (tank was sampled several hours ago).

What action should be taken?

A. Document discrepancy via CR, install tag on CZ-47A, and continue with the release.

B. Terminate the release, install tag on CZ-47A and submit new release permit to nuclear chemistry.

C. Install tag on CZ-47A and continue with the release.

D. Install tag on CZ-47A, inform nuclear chemistry and resample with current release permit.

Answer:

B. Terminate the release, install tag on CZ-47A and submit new release permit to nuclear chemistry.

Notes:

Per 1104.020 Chg 043-05-0 if tag is missing when preparing to perform release, then the operator shall: Terminate release, Install tag on CZ-47A, and Submit new relesase permit to Chemistry. Therefore: "B" is correct, all other answers do not contain the correct information.

References:

1104.020, Chg 043-05-0

History:

Modified regular exambank QID 2761 for 2002 RO exam. Was KA 2.1.32 for Liquid Radwaste System
Selected for use on 2007 RO Exam.
Selected for 2010 RO/SRO exam

4.0 Release (Operations)

> **CAUTION**
>
> Unauthorized discharge to Lake Dardanelle via the flume shall be avoided.

4.1 Verify CZ Disch to Flume Flow (CV-4642) closed.

4.2 Verify T-16A X-fer PP (P-47A) stopped.

> **NOTE**
>
> Tag contains information to remind personnel that tank is isolated for chemistry sample.

4.3 Verify Treated Waste Monitor Tank T-16A Inlet (CZ-47A) closed AND tagged.

4.3.1 <u>IF</u> tag is missing <u>or</u> has been removed since tank was last sampled,
<u>THEN</u> perform the following:

A. Terminate this release.

B. Install tag on CZ-47A.

C. Submit new release permit to Chemistry.

4.4 Verify Treated Waste Monitor Tank T-16A Outlet (CZ-48A) open.

4.5 Verify F-560 in-service by performing the following:

4.5.1 Verify the following valves open:

- CZ-74 (LRW Disch Filter F-560 Inlet)
- CZ-77 (LRW Disch Filter F-560 Outlet)

4.5.2 Verify CZ-83 (LRW Disch Filter F-560 Bypass) closed.

4.6 Verify Treated Waste Discharge Valve to Header from P-47B (CZ-55B) closed.

4.7 Verify Treated Waste Monitor Tank T-16A Recirc Inlet (CZ-54A) closed.

4.8 Open Treated Waste Discharge Valve to Header from P-47A (CZ-55A).

4.9 Open Treated Waste Discharge to Circ. Water Flume (CZ-58).

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

QID: 0803 **Rev:** 0 **Rev Date:** 9/17/2009 **Source:** **Originator:** S Pullin

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

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

System Number: 2.4. **System Title:** Emergency procedure / plan

Description: Knowledge of EOP mitigation strategies.

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

Tier:	3	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 73 **SRO:** 73

General rules of the Generic Emergency Operating Guidelines are that symptoms are treated whenever they occur based on priorities.

Which of the following transients has top priority per the GEOG?

A. Overheating

B. Overcooling

C. Loss of Subcooling Margin

D. Steam Generator Tube Rupture

Answer:

C. Loss of Subcooling Margin

Notes:

C is correct per the GEOG LOSM has top priority.

References:

Volume 1 GEOG Part 1, Introduction

History:

New for 2010 RO/SRO exam.

Part I

Introduction

The Generic Emergency Operating Guideline (GEOG) Bases is a guideline developed from the technical bases contained in Volume 3. The GEOG is intended to demonstrate how the individual sections of the Technical Bases document (TBD) can be assembled into one overall transient mitigation guideline. It represents the vendor-preferred path relative to options included in the TBD.

The GEOG is not a procedure nor should it be used as a direct model for a procedure. The development of this document did not rigorously adhere to any set of human factors principles other than to achieve consistency in the use of terms, such as IF-THEN statements (the users of this document have their own plant specific procedure writer's guides to control procedure format and content). The GEOG should also not be used as a stand-alone document. All of the TBD volumes must be read and understood before implementing TBD guidance.

GEOG Structure

Seven parts comprise the GEOG:

- Introduction: basic information on use.
- List of acronyms and abbreviations.
- Diagnosis and mitigation: covers entry, diagnosis of abnormal conditions, mitigation of transients and plant stabilization.
- Cooldown: covers cooldown under abnormal conditions of LOCA, HPI cooling, or degraded SGs.
- Repetitive tasks: covers guidance for tasks that may apply in several mitigation or cooldown sections.
- Rules: covers important guidance that always applies after the reactor is shutdown when the stated conditions exist.
- Figures: provides any figures used in the GEOG other than the section flowcharts.

- Symptoms are treated whenever they occur, and are treated in order of priority. This precludes the need for repeated steps in the guidelines to require symptom status checks. Symptom checks are specified where their occurrence is more likely or as a transfer check at the completion of a section.

- Symptom priorities are, in descending order:

  - Loss of SCM

  - Upsets in heat transfer (lack of or excessive)

  - Steam generator tube rupture

  ICC is not a symptom, and can only occur following a loss of SCM. The possibility of ICC conditions developing is always monitored when SCM does not exist.

- Rules (Part VI) are used for specific guidance that always applies when the stated conditions exist. This also reduces the need for repeated steps, but more importantly fosters the better response and consistency that is achievable using rule-based behavior.

- The intent of the guidelines is to proceed through the appropriate actions without undue delay and to primarily mitigate transients from the control room when possible. Except for specific hold points or loops, it is not expected that delays will be encountered due to either prolonged attempts to achieve satisfactory results from a lesser impact action or due to attempting significantly time-consuming actions from outside the control room. For example, it is expected that a feedwater pump will be tripped to terminate overfeeding a SG if initial attempts to control flow were unsuccessful, rather than repeated attempts at local valve manipulation.

Transient Mitigation Sections

The transient mitigation sections are intended to provide the necessary guidance to bring the plant to a safe and stable condition following the occurrence of a symptom (i.e., abnormal transient).

Once the plant is in a safe stable configuration, the guidance routes to either an appropriate cooldown section, or back to Section III.A for completion of VSSV checks, or provides the option to remain in the stable configuration and await station management's decision relative to continued operation or shutdown.

The transient mitigation sections are:

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

QID: 0161 **Rev:** 1 **Rev Date:** 4/24/2002 **Source:** Direct **Originator:** J. Cork

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

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

System Number: 2.4 **System Title:** Emergency procedure / plan

Description: Knowledge of abnormal condition procedures.

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

Tier:	3	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	G	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 74 **SRO:** 74

Given:

- Power escalation is in progress following a shutdown.
- Reactor power is 35%.
- Rod 6 of Group 7 drops.

Which of the following actions should be taken?

A. Insert all regulating rods in sequential mode.

B. Trip the reactor and go to Reactor Trip, 1202.001.

C. Verify plant stabilizes at 320 MWe after ICS runback.

D. Verify SDM within COLR limit within one hour.

Answer:

D. Verify SDM within COLR limit within one hour.

Notes:

[a] would only be performed if power was <2%.
[b] would not be done because only one rod dropped.
[c] power is <360 MWe so there wouldn't be any runback, the value given would require a power increase.
[d] is the correct answer per TS.

References:

1203.003, Control Rod Drive Malfunction Action, change 023, page 12, step 4

History:

Developed for use in 98 RO Re-exam.
Used in 2001 RO/SRO Exam.
Selected for 2002 RO/SRO exam. Revised to agree with ITS.
Selected for 2010 RO/SRO exam

SECTION 2
DROPPED ROD – REACTOR CRITICAL

NOTE

- Technical Specifications defines an inoperable rod as follows:
  - Safety Rod that is NOT fully withdrawn within one hour, except during performance of rod exercise surveillance (TS 3.1.5). If the Safety Rod is declared inoperable in TS 3.1.5, then TS 3.1.4 must also be entered.
  - Inability to move control rod (SR 3.1.4.2) or APSR (TS 3.1.6).
  - Rod can not be located with API, RPI or limit lights (TS 3.1.7). Not meeting TS 3.1.7 results in not meeting either TS 3.1.4 or 3.1.6.
- The misaligned (>6.5%) rod's position is NOT to be used in the calculation of the rod group average position.

4. IF rod is declared inoperable
OR is misaligned >6.5%,
THEN perform the following:

NOTE

If the inoperable control rod is fully inserted, then it is not necessary to consider it inoperable for the purposes of shutdown margin calculations because it has inserted its negative reactivity. A control rod is considered to be inoperable if it is not free to insert into the core within the required insertion time, or does not have at least one position indicator channel operable, i.e., cannot be located. (Ref. TS 3.1.4 Bases)

- Within 1 hour **AND** once every 12 hours thereafter, verify 1.5% available shutdown margin per Reactivity Balance Calculation (1103.015) OR initiate boration to restore SDM to be within COLR limit within 1 hour.

A. IF control rod is NOT fully inserted,
OR the control rod can NOT be located,
THEN use worksheet 4 and use the inoperable rod option (does NOT apply to APSRs).

B. IF rod is fully inserted,
THEN use worksheet 4 and do NOT use the inoperable rod option.

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

QID: 0804 **Rev:** 0 **Rev Date:** 9/17/2009 **Source:** Modified **Originator:** S. Pullin

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

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

System Number: 2.4 **System Title:** Emergency procedure / plan

Description: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

K/A Number: 2.4.50 **CFR Reference:** 41.10 / 43.5 / 45.3

Tier:	3	**RO Imp:**	4.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	G	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 75 **SRO:** 75

Given:

- Plant is in cold shutdown.
- All necessary components have been aligned per 1305.006, Integrated ES System Test.
- All ES EVEN Digital Channels actuated per procedure using RB pressure transmitters.
- Annunciator "RB SPRAY P35B ES FAILURE" K11-C7 is in alarm.

What caused the alarm and what is the proper response to this alarm (K11-C7)?

A. Flow is < 1050 gpm, no response required, the Spray pump breaker is racked down for this test.

B. Flow is < 1500 gpm, raise RB Spray flow using CV-2400, RB Spray Block valve.

C. Flow is < 1050 gpm, raise RB Spray flow using DH-9, DH-10 Bypass valve.

D. Flow is < 1500 gpm, No response needed, expected alarm due to no flow through the flow transmitter.

Answer:

C. Flow is < 1050 gpm, raise RB Spray flow using DH-9, DH-10 Bypass valve.

Notes:

"C" is correct for the ES test since the RB Spray pump is recircing on the BWST. Low flow alarm setpoint is 1050 gpm.
"A" is incorrect, the Spray pumps are operated while the HPI pumps' breakers are racked down for this test
"B" is incorrect, although this would be done for an actual ES actuation, this would spray the RB down during this test, hence the valve is closed and tagged.
"D" is incorrect, the flow transmitter is in service for this test.

References:

OP-1203.012J Change 37
OP-1305.006 Change 30

History:

Modified from QID 564
Selected for 2010 RO/SRO exam

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

QID: 0564 **Rev:** 0 **Rev Date:** 4/7/05 **Source:** Direct **Originator:** S.Pullin

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

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

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

Description: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

K/A Number: 2.4.50 **CFR Reference:** 45.3

Tier: 2 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** ☐ **SRO:** ☐

Given:

- Plant is in cold shutdown.
- All necessary components have been aligned per 1305.006, Integrated ES System Test.
- All ES EVEN Digital Channels actuated per procedure using RB pressure transmitters.
- Annunciator "RB SPRAY P35B ES FAILURE" K11-C7 is in alarm.

Which of the following is a proper response to this alarm (K11-C7)?

PARENT

Question

A. No response required, the Spray pump breaker is racked down for this test.

B. Raise RB Spray flow using CV-2400, RB Spray Block valve.

C. Raise RB Spray flow using DH-9, DH-10 Bypass valve.

D. No response needed, expected alarm due to no flow through the flow transmitter.

Answer:

C. Raise RB Spray flow using DH-9, DH-10 Bypass valve.

Notes:

"C" is correct for the ES test since the RB Spray pump is recircing on the BWST.
"A" is incorrect, the Spray pumps are operated while the HPI pumps' breakers are racked down for this test
"B" is incorrect, although this would be done for an actual ES actuation, this would spray the RB down during this test, hence the valve is closed and tagged.
"D" is incorrect, the flow transmitter is in service for this test.

References:

1203.012J, Chg. 035-00-0
1305.006, Chg. 020-04-0

History:

New for 2005 RO exam

Location: C18

Page 1 of 2

Device and Setpoint (either of the following):

A. P-35B breaker (A-404) is open 55 seconds after ES CH 8 actuation

B. RB spray flow <1050 gpm 55 seconds after ES CH 8 actuation

RB SPRAY
P35B ES
FAILURE

Alarm: K11-C7

1.0 OPERATOR ACTIONS

CAUTION

Attempting to reclose breaker with protective relay tripped may damage motor and circuit components.

1. IF breaker A-404 open,
THEN perform the following:

NOTE

Indications such as PI-2408, ESAS actuation alarms on K11, ES Cabinet pressure indicators, wide range pressure indicators and recorders PI-2412, PI-2413, PR-2413 and SPDS/PMS may be relied upon.

A. IF RB Press >30 psig,
THEN perform the following:

1) Verify that ES CH 7 actuated.

2) Check that RB Spray Pump (P-35A) has started.

B. Determine cause of P-35B failure.

2. IF RB Spray P-35B Flow on C16 is low,
THEN perform the following:

NOTE

Indications such as PI-2408, ESAS actuation alarms on K11, ES Cabinet pressure indicators, wide range pressure indicators and recorders PI-2412, PI-2413, PR-2413 and SPDS/PMS may be relied upon.

A. IF RB Press >30 psig,
THEN perform the following:

1) Verify that ES CH 7 actuated.

2) Check that RB Spray Pump (P-35A) has started.

B. Determine and correct cause of low flow.

3. Refer to TS 3.6.5 for RB Spray Pump operability requirements.

4. IF desired to clear alarm,
   THEN perform either of the following:

   - Clear ES condition

     – Reset ES CH 8 per Engineered Safeguards Actuation System (1105.003)

   - Close breaker A-404

     – Raise RB spray flow to >1050 gpm

2.0 PROBABLE CAUSES

Pump P-35B failure to auto start

3.0 REFERENCES

Schematic Diagram Annunciator K11 (E-461, sheets 1-3)

3.7.12 Perform the following to vent P-35B discharge piping:

A. Connect hose to Pressure Point (PP-2400) and run end of hose to floor drain. ______

B. Slowly open PP-2400 Isol Before CV-2400 (BS-2400C) until solid stream of water flows out of hose. ______

C. Close BS-2400C. ______

3.7.13 Start RB Spray Pump (P-35B). ______

3.7.14 Adjust DH-9 to obtain ~1500 GPM spray flow. ______

A. IF necessary to obtain ~1500 GPM
AND DH-9 is fully open,
THEN throttle open DH Test & Recirc Isol (DH-10). ______

3.7.15 Stop P-35B AND leave handswitch in NORMAL-AFTER-STOP. ______

3.7.16 Close CV-1408. ______

3.8 Align DH Pump (P-34B) for start in DH mode as follows:

3.8.1 Close P-34B Suction From BWST (CV-1437). ______

3.8.2 Open P-34B Suction From RCS (CV-1435). ______

3.8.3 Unlock and close B DH Cooler SW Outlet Isol (SW-22B). ______

3.8.4 Verify LPI Block Valve (CV-1400) closed. ______

3.8.5 Verify Decay Heat Cooler Outlet (CV-1429) open. ______

3.8.6 Verify Decay Heat Cooler Bypass (CV-1432) closed. ______

3.8.7 Start DH Pump (P-34B). ______

3.8.8 Open LPI Block Valve (CV-1400). ______

3.8.9 Verify proper alignment by observing LPI P-34B Flow ~3500 gpm. ______

3.8.10 Stop P-34B AND leave handswitch in NORMAL-AFTER-STOP. ______

3.8.11 Close CV-1400. ______

Facility: **Arkansas Nuclear One – Unit 1** Date of Exam: **3/5/2010**

Tier	Group	RO K/A Category Points												SRO-Only Points				
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2		G*		Total
1. Emergency & Abnormal Plant Evolutions	1	0	0	0				0	0			0	0	3		3		6
	2	0	0	0				0	0			0	0	3		1		4
	Tier Totals	0	0	0	N/A			0	0	N/A		0	0	6		4		10
2. Plant Systems	1	0	0	0	0	0	0	0	0	0	0	0	0	3		2		5
	2	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2		3
	Tier Totals	0	0	0	0	0	0	0	0	0	0	0	0	4		4		8
3. Generic Knowledge and Abilities Categories					0		0		0		0		0	1	2	3	4	7
					0		0		0		0			2	1	2	2	

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

SRO Written Exam

Tier 1 Group 1

ES-401

PWR Examination Outline
Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)

Form ES-401-2

EAPE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#	QID	Type
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1					X		**EA2.1**- Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.0	76	588	D
000008 Pressurizer Vapor Space Accident / 3							**Not selected**	**N/A**			
000009 Small Break LOCA / 3							**Not selected**	**N/A**			
000011 Large Break LOCA / 3							**Not selected**	**N/A**			
000015/17 RCP Malfunctions / 4							**Not selected**	**N/A**			
000022 Loss of Rx Coolant Makeup / 2					X		**AA2.04**- How long PZR level can be maintained within limits	3.8	77	805	N
000025 Loss of RHR System / 4						X	**2.4.31** Knowledge of annunciator alarms, indications, or response procedures.	4.1	78	806	N
000026 Loss of Component Cooling Water / 8					X		**AA2.01**- Location of a leak in the CCWS	3.5	79	807	N
000027 Pressurizer Pressure Control System Malfunction / 3							**Not selected**	**N/A**			
000029 ATWS / 1							**Not selected**	**N/A**			
000038 Steam Gen. Tube Rupture / 3						X	**2.4.18** – Knowledge of the specific bases for EOPs.	4.0	80	585	N
0040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4						X	**2.4.6**- Knowledge of symptom based EOP mitigation strategies	4.7	81	584	D
000054 (CE/E06) Loss of Main Feedwater / 4							**Not selected**	**N/A**			
000055 Station Blackout / 6							**Not selected**	**N/A**			
000056 Loss of Off-site Power / 6							**Not selected**	**N/A**			
000057 Loss of Vital AC Inst. Bus / 6							**Not selected**	**N/A**			
000058 Loss of DC Power / 6							**Not selected**	**N/A**			
000062 Loss of Nuclear Svc Water / 4							**Not selected**	**N/A**			
000065 Loss of Instrument Air / 8							**2.4.18** – Knowledge of the specific bases for EOPs **Rejected system to 038 Steam Gen Tube Rupture**	**N/A**			
W/E04 LOCA Outside Containment / 3							**Not selected**	**N/A**			

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)			Form ES-401-2	
APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#	QID	Type
W/E11 Loss of Emergency Coolant Recirc. / 4							**Not selected**	**N/A**			
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4							**Not selected**	**N/A**			
000077 Generator Voltage and Electric Grid Disturbances / 6							**Not selected**	**N/A**			
K/A Category Totals:					3	3	Group Point Total:		**6**		

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

QID: 0588 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-EOP04 **Objective:** 11 **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): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A Number: EA2.1 **CFR Reference:** 43.5 /45.13

Tier: 1 **RO Imp:** 2.5 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 76

Given:

- Reactor tripped due to a loss of both MFWPs approximately 15 minutes ago.
- Annunciator K02-B6 "A3 L.O. RELAY TRIP" is in alarm.
- AFW pump, P-75, is tagged out for maintenance.
- Steam Driven EFW Pump, P-7A, has tripped on overspeed.
- RCS pressure is 2000 psig.
- CETs are 612°F.
- Both OTSG levels are 30".

Which of the following procedures should be in use for the above conditions?

A. 1202.002, Loss of Subcooling Margin

B. 1202.004, Overheating

C. 1202.011, HPI Cooldown

D. 1203.037, Abnormal ES Bus Voltage

Answer:

B. 1202.004, Overheating

Notes:

Answer "B" is correct, the Overheating EOP should be entered with CETs > 610°F and all MFW and EFW lost during loss of adequate Subcooling Margin.
Answer "A" is incorrect, this procedure would have been in use up to the point where CETs became > 610°F.
Answer "C" is incorrect, this procedure is entered from Loss of Subcooling Margin.
Answer "D" is incorrect, this procedure is used when ES bus voltage is low but not de-energized.

References:

1202.004, Chg. 006

History:

New for 2005 SRO exam.
Selected for 2010 SRO 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.

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

QID: 0805 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Ability to determine and interpret the following as they apply to Reactor Coolant Makeup: How long PZR level can be maintained within limits.

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

Tier:	1	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** ☐ **SRO:** 77

Given:

- RCS Cooldown in progress
- Tave is 295 F
- RCS Pressure is 440 psig.
- Pressurizer level is 85 inches
- All makeup has been lost
- Pressurizer level is dropping at 5 inches per minute
- Assuming pressurizer level rate of change remains the same

When will LCO 3.4.9 Pressurizer, be entered due to low Pressurizer level and what is the bases per Technical Specification for the low level?

A. 2 minutes and to prevent violating NDTT Curve.

B. 4 minutes and to prevent violating LTOP Curve.

C. 6 minutes and to maintain the minimum ES bus powered pressurizer heaters OPERABLE.

D. 8 minutes and to maintain on scale pressurizer level indication.

Answer:

D. 8 minutes and to maintain on scale pressurizer level indication.

Notes:

D is correct, the limit per LCO 3.4.9 is less than or equal to 45 inches and the minimal water level limit has been established to ensure that water level is above the minimum detectable level.
A is incorrect, due to PZR level would be 75 inches which is below the administrative limit per OP-1102.010 for PZR level, but does not violate the NDTT curve.
B is incorrect, due to PZR level would be 65 inches which is below the administrative limit per OP-1102.010 for PZR level, but does not violate the LTOP curve.
C is incorrect, due to PZR level would be 55 inches which is at the pressurizer heater cutoff level which would deenergize the ES powered heaters.

References:

T.S. 3.4.9 Amendment 215

History:

New selected for 2010 SRO exam.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≥ 45 inches and ≤ 320 inches; and

b. A minimum of 126 kW of Engineered Safeguards (ES) bus powered pressurizer heaters OPERABLE.

---NOTE---
OPERABILITY requirements on pressurizer heaters do not apply in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 262°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limits.	A.1 Restore level to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 4 with RCS temperature ≤ 262°F.	24 hours
C. Capacity of ES bus powered pressurizer heaters less than limit.	C.1 Restore pressurizer heater capacity.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 4.	12 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls. Pressurizer safety valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves."

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for abnormalities. The water level limit thus serves two purposes:

a. Provides pressure control during normal operation; and

b. Prevents the peak RCS pressure from exceeding the safety limit of 2750 psig during an abnormality.

The maximum water level limit thus permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, so that both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) during abnormalities, thus ensuring that pressure relief devices (electromatic relief valve (ERV) or code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to an abnormality that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2750 psig or damage may occur to the ERV or pressurizer code safety valves.

The minimum water level limit has been established to ensure that water level is above the minimum detectable level.

The pressurizer heaters are used to maintain a pressure in the RCS so reactor coolant in the loops is subcooled and thus in the preferred state for heat transport to the steam generators (SGs). This function must be maintained with a loss of offsite power. Consequently, the emphasis of this LCO is to ensure that the Engineered Safeguards (ES) bus powered heaters are adequate to maintain pressure for RCS loop subcooling with an extended loss of offsite power.

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

QID: 0806 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** S. Pullin

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

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

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

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

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

Tier:	1	**RO Imp:**	4.2	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** **SRO:** 78

Given:

- Mode 5
- RCS temperature 170 F
- RCS pressure 0 psig
- "A" RCP seal removed for maintenance
- "A" Decay Heat in service
- Following alarms are received
  - DECAY HEAT FLOW HI/LO (K09-A8)
  - DECAY HEAT VORTEX WARNING (K09-D8)
  - ISOL VLV OPEN RC PRESS LO (K10-E5)

Which section of OP-1203.028, Loss of Decay Heat Removal, will be entered for the given conditions?

A. Section 6, Decay Heat Pump Trip

B. Section 7, Suction Valve Closure

C. Section 9, Loss of Both DH Systems - RCS Pressure Boundary Intact

D. Section 10, Loss of Both DH Systems - RCS Pressure Boundary Open

Answer:

B. Section 7, Suction Valve Closure

Notes:

B is correct, with the given alarms K10-E5 would automatically cause the DHR Suction valve to close.
A is incorrect, the DHR Pump would still be running for the given condition. The pump does not automatically stop on valve closure.
C is incorrect, although the RCS is still intact with an RCP seal removed, the transition to loss of both DHR Pumps does not occur until RCS temperature is greater than 280 F
D incorrect, the RCS is not open and the transition to loss of both DHR Pumps does not occur until RCS temperature is greater than 280 F

References:

OP-1203.028 Change 021
op-1203.012I Change 046

History:

New selected for 2010 SRO exam.

SECTION 7 – SUCTION VALVE CLOSURE

ENTRY CONDITIONS

One or more of the following:

- DECAY HEAT FLOW HI/LO (K09-A8) alarm
- RCS temp rise

- Train A CET TEMP HI (K09-D6) alarm
- Train B CET TEMP HI (K09-E6) alarm
- CV-1050 AUTO CLOSE (K09-B7) or CV-1410 AUTO CLOSE (K09-B8) alarm

SECTION 7 – SUCTION VALVE CLOSURE

INSTRUCTIONS

1. **Stop the running DH pump.**

2. **Notify Shift Manager/CRS to implement Emergency Action Level Classification (1903.010).**

3. **Terminate any operation causing pressure rise.**

4. **IF maintenance activities in the Reactor Building could be affected by RCS level rise,**
   THEN perform local evacuation of the affected areas.

5. **IF RCS temp exceeds 280°F,**
   THEN GO TO applicable "Loss of Both DH Systems" section of this procedure.

NOTE

- Containment closure must be established prior to steam release.
- Decay Heat Removal and LTOP System Control (1015.002), Form 1015.002B provides estimate of time to 200°F, time to steam release, time to core uncovery, heatup rate, and required makeup rate.

6. **IF any of the following conditions occur:**
   - Time remaining to steam release is, or becomes <1 hour
     AND DH removal can **NOT** be immediately restored
   - RCS press >150 psig
     AND RCS loops **NOT** filled
   - RCS press >Decay Heat Sys Max Pressure limit of Plant Shutdown and Cooldown (1102.010), Attachment A

   THEN initiate containment closure per Attachment G of this procedure, while continuing with this section.

(continued)

SECTION 7 – SUCTION VALVE CLOSURE

K10 - ES

NOTE

- CV-1050 will close automatically if Core Flood Tank T-2A Outlet (CV-2415) comes off its closed seat or if RCS press exceeds 320 psig.
- CV-1410 will close automatically if Core Flood Tank T-2B Outlet (CV-2419) comes off its closed seat or if RCS press exceeds 385 psig.
- The auto close interlock is automatically reset when RCS press is <290 psig.

7. Determine and correct cause of valve closure.

8. IF RCS press is greater than the applicable limit listed below,
THEN perform the following:

- RCS loops not filled -- 150 psig
- RCS loops filled -- Decay Heat Sys Max Pressure limit of Plant Shutdown and Cooldown (1102.010), Attachment A.

A. Initiate containment closure per Attachment G of this procedure.

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

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

1) Stop the running DH pump.

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

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

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

(continued)

Location: C16

ISOL VLV OPEN
RC PRESS
LO

Alarm: K10-E5

Device and Setpoint:
RC Loop "A" Unit Press (PS-1021) <700 psig along with
valve open signal from either of the following:
CFT T-2A Outlet Limit Switch (ZS-2415A)
CFT T-2B Outlet Limit Switch (ZS-2419A)

1.0 OPERATOR ACTIONS

1. Unless a loss-of-coolant accident, or valve testing is in progress, secure depressurization, and verify CFT outlets closed:

    A. Core Flood Tank T-2A Outlet (CV-2415)

    B. Core Flood Tank T-2B Outlet (CV-2419)

2. IF loss-of-coolant accident is indicated,
THEN GO TO Emergency Operating Procedure series (1202.XXX).

2.0 PROBABLE CAUSES

1. Valve testing during plant shutdown.

2. CFT outlet open and RC pressure <700 psig

3.0 REFERENCES

1. Schematic Diagram Annunciator K10 (E-460, sheets 1 - 3)

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

QID: 0807 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** S Pullin

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

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

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

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

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

Tier:	1	**RO Imp:**	2.9	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** ☐ **SRO:** 79

Given:

- Plant at 100%
- The following alarms are received
- ICW COOLER OUTLET TEMP HI (K12-E4)
- RCP BEEDOFF TEMP HI (K08-C7)
- "A" RCP seal temperature rising
- Skewed RCP Seal Injection flows indicated on CO4
- RCS leak rate is 50 gpm

Which of the following procedures provide the actions necessary to mitigate the abnormal operating condition?

A. OP-1203.039, Excess RCS Leakage

B. OP-1203.026, Loss of Reactor Coolant Makeup

C. OP-1203.031, Reactor Coolant Pump and Motor Emergency

D. OP-1102.016, Power Reduction and Plant Shutdown

Answer:

A. OP-1203.039, Excess RCS Leakage

Notes:

A is correct, because Excess RCS Leakage procedure is the only procedure that combats an intersystem LOCA.
B is incorrect, OP-1203.026 has a section to address makeup & purification system leaks, but with the indications given this is not considered a makeup & purification system leak.
C is incorrect, with the given indications the student could misdiagnose this as a seal failure issue,
D is incorrect, with the given leak rate, a rapid plant shutdown would be necessary.

References:

OP-1203.039 Change 11

History:

New selected for 2010 SRO exam

7. Check for primary to secondary leak indicated as follows:

A. Alarm or rising count rate on any of the following:

- A OTSG N-16 Detector (RI-2691)
- B OTSG N-16 Detector (RI-2692)
- Main Condenser Radiation Process Monitor (RI-3632)
- Steam Line A High Range Rad Monitor (RI-2682)

- Steam Line B High Range Rad Monitor (RI-2681)
- SPING 2 Unit 1 Radwaste Area (RX-9825)
- SG-A N-16 AVG Leakrate GPM (SGALRGPM)
- SG-A N-16 Leakrate ROC (Rate of Change) GPM/HR (SGAROC1)

- SG-B N-16 AVG Leakrate GPM (SGBLRGPM)
- SG-B N-16 Leakrate ROC (Rate of Change) GPM/HR (SGBROC1)

B. Chemistry samples indicate rising secondary activity.

C. A leaking SG may exhibit the following at low feedwater flow rates:

- Higher SG level
- Lower FW flow rate
- Lower MFW pump speed

D. **IF** primary to secondary leakage is indicated,
THEN GO TO Small Steam Generator Tube Leaks (1203.023).

8. Check RCP seals for proper staging.

A. **IF** seal degradation **OR** seal failure is indicated,
THEN GO TO Reactor Coolant Pump and Motor Emergencies (1203.031).

9. Check indications of Makeup and Purification System leakage.

- AUX BLDG Sump level rising
- Dirty Waste Drain Tank (T20A and T20B) level rising
- Equipment Drain Tank (T11) level rising
- AREA MONITOR RADIATION HI alarm (K10-B1) for AUX BLDG area

A. **IF** Makeup and Purification System leakage is indicated,
THEN GO TO Loss of Reactor Coolant Makeup (1203.026), "Large Makeup and Purification System Leak" section.

10. Monitor RB Sump level.

A. <u>**IF**</u> leakage into RB Sump is indicated,
<u>**THEN**</u> continue with efforts to locate and isolate the leak using RCS Leak Detection (1103.013)
<u>**AND**</u> continue with this procedure.

11. Monitor Quench Tank (T42) pressure, level, and temperature.

A. <u>**IF**</u> leakage is indicated into Quench Tank,
<u>**THEN**</u> **GO TO** Pzr Systems Failure (1203.015).

12. Check indications of RCS leakage into ICW system.

NOTE

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

- Nuclear Loop ICW Surge Tank (T37B) level rising
- Nuclear Loop ICW activity rising
- Indication of Letdown Cooler RCS leak into ICW:
  - Letdown Cooler ICW Outlet temp rising on PMS:
    - 8P ICW trend
    - T2214 for E29A
    - T2215 for E29B
- Indication of RCP Seal Cooler RCS leak into ICW:
  - RCP Seal Temp rising
  - RCP Seal Bleedoff Temp rising
  - Skewed RCP Seal Injection Flows

NOTE

With small leak rates, sufficient time should be available to isolate one cooler at a time.

A. <u>**IF**</u> RCS leak into **LETDOWN COOLER** is indicated,
<u>**THEN**</u> perform the following:

1) Isolate one or both Letdown Cooler (s) (E29A/B) by closing associated valves:

- RC to Letdown Coolers E29A (C04) (CV-1213)
  <u>**AND**</u>
  Letdown Coolers Outlet (RCS) E29A (C18) (CV-1214)
- RC to Letdown Coolers E29B (C04) (CV-1215)
  <u>**AND**</u>
  Letdown Coolers Outlet (RCS) E29B (C18) (CV-1216)

(12.A CONTINUED ON NEXT PAGE)

NOTE

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

14. IF total RCS leakage is in excess of that allowed by Tech Spec 3.4.13
AND
poses an immediate threat to plant operations,
THEN perform the following:

A. IF reactor is Critical,
THEN commence plant shutdown per Rapid Plant Shutdown (1203.045).

B. IF reactor is shutdown,
THEN perform RCS cooldown by one of the following:

1) IF RCS is cooling down due to HPI/break flow, independent of SG cooling,
THEN perform Small Break LOCA Cooldown (1203.041), while continuing with this procedure.

2) IF any RCP is running,
THEN perform Forced Flow Cooldown (1203.040), while continuing with this procedure.

3) IF all RCPs are off,
THEN perform Natural Circulation Cooldown (1203.013), while continuing with this procedure.

15. IF total RCS leakage is in excess of that allowed by Tech Spec 3.4.13
AND
poses no immediate threat to plant operations,
THEN perform the following:

A. Bring reactor to cold shutdown per Power Reduction and Plant Shutdown (1102.016) and Plant Shutdown and Cooldown (1102.010).

16. Advise Shift Manager to implement Emergency Action Level Classification (1903.010).

17. IF leakage is within Tech Spec 3.4.13 limits,
THEN continue with efforts to locate and isolate the leak using RCS Leak Detection (1103.013) and proceed as directed by Operations Manager.

18. IF leak is isolated,
THEN proceed as directed by Operations Manager.

END

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

QID: 0585 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** B. Possage
TUOI: A1LP-RO-EOP **Objective:** 9 **Point Value:** 1

Section: 4.1 **Type:** Generic EPE's
System Number: 038 **System Title:** Steam Generator Tube Rupture
Description: Knowledge of the specific bases for EOPs.

K/A Number: 2.4.18 **CFR Reference:** 41.10 / 43.1 / 45.13
Tier: 1 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 3
Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** ☐ **SRO:** 80

Given:
- SGTR in progress
- Rx is tripped
- RCS pressure 1350 psig
- RCS Thot 540°F
- Projected dose rate at site boundary at NUE criteria
- "B" SG level at 395" and rising rapidly
- "A" SG level stable at 40"

Considering the above conditions, which of the following procedural actions will cause higher tube stresses than normal limitations but is acceptable during a SGTR per the EOP technical bases document?

A. Perform a cool down to less than 500°F at 100°F/hr and isolate bad SG.

B. Steam bad SG to maintain bad SG Tube-to-Shell DT <150°F (tubes colder).

C. Steam bad SG to maintain bad SG Tube-to-Shell DT <100°F (tubes hotter).

D. Establish a cool down rate of 250°F/hr to 500°F Thot.

Answer:

B. Steam bad SG to maintain bad SG Tube-to-Shell DT <150°F (tubes colder).

Notes:

B is correct, per Technical Bases during emergency cool downs the tube to shell delta T limits are relaxed. With the given information an emergency cool down is required at the rate of </= 240 F/hr.
A is incorrect, this rate is the normal cool down rate.
C is incorrect, this is the normal tube to shell delta T limit.
D is incorrect, this exceeds the allowed emergency cool down limit.

References:

OP-1202.006 Change 11
B&W EOP Technical Bases Document

History:

New selected for 2010 SRO exam

INSTRUCTIONS

17. **__IF__ bad SG level is approaching 410" due to leakage**

    __OR__

    dose rate ≥ Alert criteria is projected at Site boundary,
    __THEN__ establish emergency cooldown rate of ≤240°F/hr (≤4°F/min) to 500°F T-hot as follows:

    A. For good SG, place TURB BYP valves in HAND

    __AND__

    adjust to maintain cooldown rate ≤240°F/hr.

    B. **__WHEN__** RCS press is <1700 psig,
    __THEN__ bypass ESAS.

    C. __IF__ only one SG is bad,
    __THEN__ steam bad SG only as necessary to maintain:

    - MSSVs closed
    - SG press:
      - ≤990 psig if using TURB BYP valves
      - ≤1040 psig if using ATM Dump Control system
    - SG level ≤410".
    - SG Tube-to Shell ΔT ≤150°F (tubes colder).
    - Desired cooldown rate if good SG TBV or ADV is full open.

CONTINGENCY ACTIONS

17. **GO TO step 18.**

    A. **__IF__** TURB BYP valves are **__not__** available,
    __THEN__ operate ATM Dump Control System for good SG in HAND to maintain cooldown rate ≤240°F/hr.

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

    1) **__IF__** both SGs are bad,
    __THEN__ steam both SGs.

    C. **__IF__** both SGs are bad,
    __THEN__ steam both SGs.

3.3.1.2 Tube-to-Shell ΔT

The normal tube-to-shell ΔT limit for cooldowns is 100°F (tubes colder) and, during an emergency cooldown (3.3.1.1) this limit may be increased to 150°F. Methods to control tube-to-shell ΔT are discussed in Chapter III.G.

This relaxation is allowed to facilitate an emergency cooldown should it be required. However, two important points should be considered:

a. Whenever tube-to-shell ΔT exceeds 100°F a post-transient stress evaluation will be required.

b. Higher tube-to-shell ΔTs will increase the tensile stresses on the tubes and may lead to higher leak flows. Indications of this occurring have been observed during actual tube leak transients.

Therefore, some judgment is required before a decision is made to increase tube-to-shell ΔT. Normally, it is recommended that tube-to-shell ΔT be kept much lower than the normal cooldown ΔT limit if at all possible. However, there may be cases where an increase in ΔT is necessary to accommodate an expeditious cooldown which may be accomplished with little or no risk (e.g., decision has already been made to isolate the affected SG and allow it to fill, thus increases in leak flow rate may not significantly impact the transient). As noted in section 3.3.1.1, the use of the emergency cooldown rate to 500°F should not result in excessive tube-to-shell ΔTs.

3.3.1.3 Cooldown Limits

The normal cooldown limit is the Technical Specification limit. With the exception of section 3.3.1.1, this limit should not be exceeded during a plant cooldown when the RCS is subcooled. If the RCS is not subcooled, then this limit does not apply as discussed in Chapter III.B.

3.3.1.4 Summary of Limits During Cooldown

The following limits should be observed, if at all possible:

a. If section 3.3.1.1 applies, then above 500°F the cooldown rate limit is 240°F/hr

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

QID: 0584 **Rev:** 0 **Rev Date:** 5/20/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-EOP03 **Objective:** 10 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of symptom based EOP mitigation strategies.

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

Tier:	1	**RO Imp:**	3.7	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 81

A steam line rupture has occurred in the Reactor Building with the following conditions now present:
- ESAS actuated on channels 1 thru 6.
- All RCPs secured per RT-10.
- RB pressure 19 psig and dropping.
- HPI throttled due to existence of adequate SCM.
- RCS pressure is 1050 psig.
- T-hot is 490°F.
- EOP actions have terminated the overcooling.

The SE recommends to the CRS to restore normal operating pressure per RT-14 in order to reset ESAS and re-start RCPs.
As CRS, does this recommendation follow the EOP mitigation strategies?

A. Yes, overcooling event has been terminated.

B. No, this could overstress reactor vessel.

C. Yes, adequate SCM has been restored.

D. No, RB pressure is not within normal limits.

Answer:

B. No, this could overstress reactor vessel.

Notes:

"B" is correct, trainee must recognize that with RCPs secured and HPI having been initiated that PTS limits apply until an evaluation is performed prior to returning to normal pressure. PTS limits prevent overstressing reactor vessel.
"A" is incorrect, yes the overcooling has been terminated but normal operating pressure would violate procedure.
"C" is incorrect, subcooling margin was never lost but normal operating pressure would violate procedure.
"D" is incorrect, although RB pressure is a concern the overriding concern is with PTS concerns.

THIS QUESTION IS TIED to 43.1

References:

1202.012, chg. 004-03-0, RT-14

History:

New for 2005 SRO exam.
Selected for the 2010 SRO exam

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.

14. Control RCS press within limits of Figure 3.

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

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

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

1) Throttle makeup flow.

2) **IF** SCM is adequate, **THEN** throttle HPI flow by performing the following:

a.) Verify both HPI RECIRC valves (CV-1300 and 1301) open.

b.) Throttle HPI.

3) Raise Letdown flow.

a) **IF** ESAS has actuated,
THEN unless fuel damage or RCS to ICW leak is suspected,
restore Letdown flow (RT 13).

4) Verify ERV Isolation open (CV-1000) **AND** cycle ERV (PSV-1000).

(14. CONTINUED ON NEXT PAGE)

SRO Written Exam

Tier 1 Group 2

S-401 PWR Examination Outline Form ES-401-2

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	IR	#	QID	Type
000001 Continuous Rod Withdrawal / 1							**AA2.05-** Uncontrolled rod withdrawal from available indications **Rejected system to 005 Inoperable/Stuck Control Rod**	**N/A**			
000003 Dropped Control Rod / 1							**Not selected**	**N/A**			
000005 Inoperable/Stuck Control Rod / 1					X		**AA2.03** – Required actions if more than one rod is stuck or inoperable	4.4	82	589	D
000024 Emergency Boration / 1					X		**AA2.05** – Amount of boron to add to achieve the required SDM	3.9	83	808	M
000028 Pressurizer Level Malfunction / 2							**Not selected**	**N/A**			
000032 Loss of Source Range NI / 7							**Not selected**	**N/A**			
000033 Loss of Intermediate Range NI / 7							**Not selected**	**N/A**			
000036 (BW/A08) Fuel Handling Accident / 8							**Not selected**	**N/A**			
000037 Steam Generator Tube Leak / 3							**Not selected**	**N/A**			
000051 Loss of Condenser Vacuum / 4							**Not selected**	**N/A**			
000059 Accidental Liquid RadWaste Rel. / 9							**Not selected**	**N/A**			
000060 Accidental Gaseous Radwaste Rel. / 9							**Not selected**	**N/A**			
00061 ARM System Alarms / 7							**Not selected**	**N/A**			
000067 Plant Fire On-site / 8							**Not selected**	**N/A**			
000068 (BW/A06) Control Room Evac. / 8							**Not selected**	**N/A**			
000069 (W/E14) Loss of CTMT Integrity / 5							**Not selected**	**N/A**			
000074 (W/E06&E07) Inad. Core Cooling / 4							**Not selected**	**N/A**			
000076 High Reactor Coolant Activity / 9							**Not selected**	**N/A**			
W/EO1 & E02 Rediagnosis & SI Termination / 3							**Not selected**	**N/A**			
W/E13 Steam Generator Over-pressure / 4							**Not selected**	**N/A**			
W/E15 Containment Flooding / 5							**Not selected**	**N/A**			
W/E16 High Containment Radiation / 9							**Not selected**	**N/A**			
BW/A01 Plant Runback / 1							**Not selected**	**N/A**			
BW/A02&A03 Loss of NNI-X/Y / 7					X		**AA2.1** – Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.0	84	591	D
BW/A04 Turbine Trip / 4							**Not selected**	**N/A**			
BW/A05 Emergency Diesel Actuation / 6							**Not selected**	**N/A**			
BW/A07 Flooding / 8							**Not selected**	**N/A**			
BW/E03 Inadequate Subcooling Margin / 4							**Not selected**	**N/A**			

BW/E08; W/E03 LOCA Cooldown - Depress. / 4						X	**2.4.47**- Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe management guidelines	4.0	85	592	D
BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4							**Not selected**	**N/A**			
BW/E13&E14 EOP Rules and Enclosures							**Not selected**	**N/A**			
CE/A11; W/E08 RCS Overcooling - PTS / 4							**Not selected**	**N/A**			
CE/A16 Excess RCS Leakage / 2							**Not selected**	**N/A**			
CE/E09 Functional Recovery							**Not selected**	**N/A**			
K/A Category Point Totals:					**3**	**1**	Group Point Total:		**4**		

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

QID: 0589 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** Direct **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Inoperable/Stuck Control Rod: Required actions if more than one rod is stuck or inoperable.

K/A Number: AA2.03 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	3.5	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** ☐ **SRO:** 82

Given:
- Plant is at 40% power.
- Group 4, Rod 4 is stuck and is mis-aligned from the group by 7.5%.
- The rod can not be re-aligned with the group.

Subsequently Group 7 Rod 6 drops to 0% withdrawn.

What are the required action(s) per Technical Specifications for the above conditions?

A. Immediately trip the reactor.

B. Borate to restore SDM within 1 hour and perform Linear Heat Rate surveilllance, SR 3.2.5.1, within 6 hours.

C. Borate to restore SDM within 1 hour and verify the potential ejected rod worth is within the assumptions of the rod ejection analysis within 6 hours.

D. Borate to restore SDM within 1 hour and place the plant in Mode 3 within 6 hours.

Answer:

D. Borate to restore SDM within 1 hour and place the plant in Mode 3 within 6 hours.

Notes:

Answer "D" is correct per TS 3.1.4 action "C" for two inoperable rods.
Answer "A" is incorrect, this action is performed for two dropped rods.
Answer "B" is incorrect, this action is performed for one inoperable rod and the time given for the stated condition is incorrect.
Answer "C" is incorrect, this action is performed for one inoperable rod and the time given for the stated condition is incorrect.

References:

T.S. 3.1.4 amendment 215

Do not include this spec in the student handout!!!

History:

New for 2005 SRO exam.
Selected for 2010 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 AND 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	

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2.3 --------------NOTE------------ Only required when THERMAL POWER is > 20% RTP. ---------------------------------- Perform SR 3.2.5.1.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours
C. More than one CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both.	C.1.1 Verify SDM to be within the limit provided in the COLR.	1 hour
	OR	
	C.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	C.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual CONTROL ROD positions are within 6.5% of their group average height.	12 hours
SR 3.1.4.2	Verify CONTROL ROD freedom of movement for each individual CONTROL ROD that is not fully inserted.	92 days

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

QID: 0808 **Rev:** 0 **Rev Date:** 9/22/2009 **Source:** Modified **Originator:** S.Pullin

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

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

System Number: 0024 **System Title:** Emergency Boration

Description: Ability to determine and interpret the following as they apply to the Emergency Boration: Amount of boron to add to achieve the required SDM.

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

Tier:	1	**RO Imp:**	3.3	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** ☐ **SRO:** 83

REFERENCE PROVIDED

- Rx has tripped with three CRDs stuck full out.
- Core lifetime = 150 EFPD
- RCS initial Boron concentration = 810 ppm
- Chemistry reports that the RCS boron concentration is 2200 ppm.

Which of the following contains guidance that must be used, for the above conditions?

A. No action required, SDM is adequate

B. 1202.012, RT-12 Emergency Boration

C. 1203.017, Moderator Dilution

D. 1103.015, Reactivity Balance Calculation

Answer:

B. 1202.012, RT-12 Emergency Boration

Notes:

Answer "B" is correct, using Att. B-16 from the plant data book, the examinee should determine that adequate SDM has not been established and Emergency Boration must be performed until adequate SDM is established.
Answer "A" is incorrect, SDM is not adequate.
Answer "C" is incorrect, although this might seem like a logical choice, this procedure should not be used for these conditions.
Answer "D" is incorrect, although this might seem like a logical choice, use of the Reactivity Balance Calculation procedure does not have any plant actions in it.

References:

1202.012 RT-12, Chg. 008
CALC-ANO1-NE-08-00007
NOTE: CALC Att. B-16 must be in SRO handout!!!!

History:

Modified from QID 678
Selected for 2010 SRO exam

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

QID: 0678 **Rev:** 0 **Rev Date:** 2/2//07 **Source:** Direct **Originator:** Cork/Possage

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

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

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

Description: Ability to determine and interpret the following as they apply to the Emergency Boration: Amount of boron to add to achieve the required SDM.

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

Tier: 1 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** An

---

Question: **RO:** ☐ **SRO:** ☐

PARENT Question

Given:

- Rx has tripped with two CRDs stuck full out.
- Core lifetime = 250 EFPD
- All immediate actions have been performed
- RCS initial Boron concentration = 638 ppm

Chemistry reports that the RCS boron concentration is 1656 ppm.

(Reference Provided)

Which of the following contains guidance that must be used, if any, for the above conditions?

A. No action required, SDM is adequate.

B. 1203.017, Moderator Dilution

C. 1202.012, RT-12 Emergency Boration

D. 1202.010, ESAS

Answer:

C. 1202.012, RT-12 Emergency Boration

Notes:

Answer "C" is correct, using Att. B-16 from the plant data book, the examinee should determine that adequate SDM has not been established and Emergency Boration must be performed until adequate SDM is established.
Answer "A" is incorrect, SDM is not adequate.
Answer "B" is incorrect, although this might seem like a logical choice, this procedure should not be used for these conditins.
Answer "D" is incorrect, use of the ESAS procedure is not required for inadequate SDM.

References:

1202.012 RT-12, Chg. 004-06-0
CALC-A1-NE-2005-003, Rev. 0
NOTE: CALC Att. B-16 must be in SRO handout!!!!

History:

New for 2007 SRO exam.

12. (Continued).

10) <u>**IF**</u> Batch Controller output rate <5 gpm
<u>**THEN**</u> perform the following:

  a) Stop running Boric Acid pump(s) (P-39A, P-39B).

  b) Close CV-1250.

  c) Stop Batch Controller by depressing stop key.

  d) **GO TO** step B.

11) Adjust Pressurizer Level Control Setpoint to 220".

12) Open BWST Outlet to OP HPI Pump (CV-1407 or 1408).

13) <u>**WHEN**</u> PZR level is ≥ 100", <u>**THEN**</u> establish maximum Letdown flow.

14) Perform the following as necessary to maintain MU Tank level 55 to 86":

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

  b) Stop running Boric Acid Pump(s) (P-39A, P-39B).

  c) Place 3-Way valve in BLEED.

  d) <u>**WHEN**</u> MU Tank level is lowered to desired level, <u>**THEN**</u> perform the following:

    (1) Return 3-Way valve to LETDOWN.

    (2) Start available Boric Acid Pump(s) (P-39A or B or both).

    (3) Open Batch Controller Outlet (CV-1250).

15) As time permits, determine actual required boration as follows:

  a) Obtain required boron concentration from the Plant Data Book ________ ppmB.

  b) Calculate batch add required using Plant Computer
  <u>**OR**</u>
  Soluble Poison Concentration Control (1103.004), Attachment A.3, "Calculation of Feed Volume For Batch Boration or Dilution". ________ gal.

  c) Use 1103.004, Attachment D, "Volume of BAAT vs. Depth of Liquid" to determine desired final BAAT level. ________ in.

(12. CONTINUED ON NEXT PAGE)

12. (Continued).

16) **WHEN** required amount of boric acid has been added per **step 15)**
OR
as determined by Reactor Engineering,
THEN perform the following:

a) Stop Boric Acid pump (P39A and B).

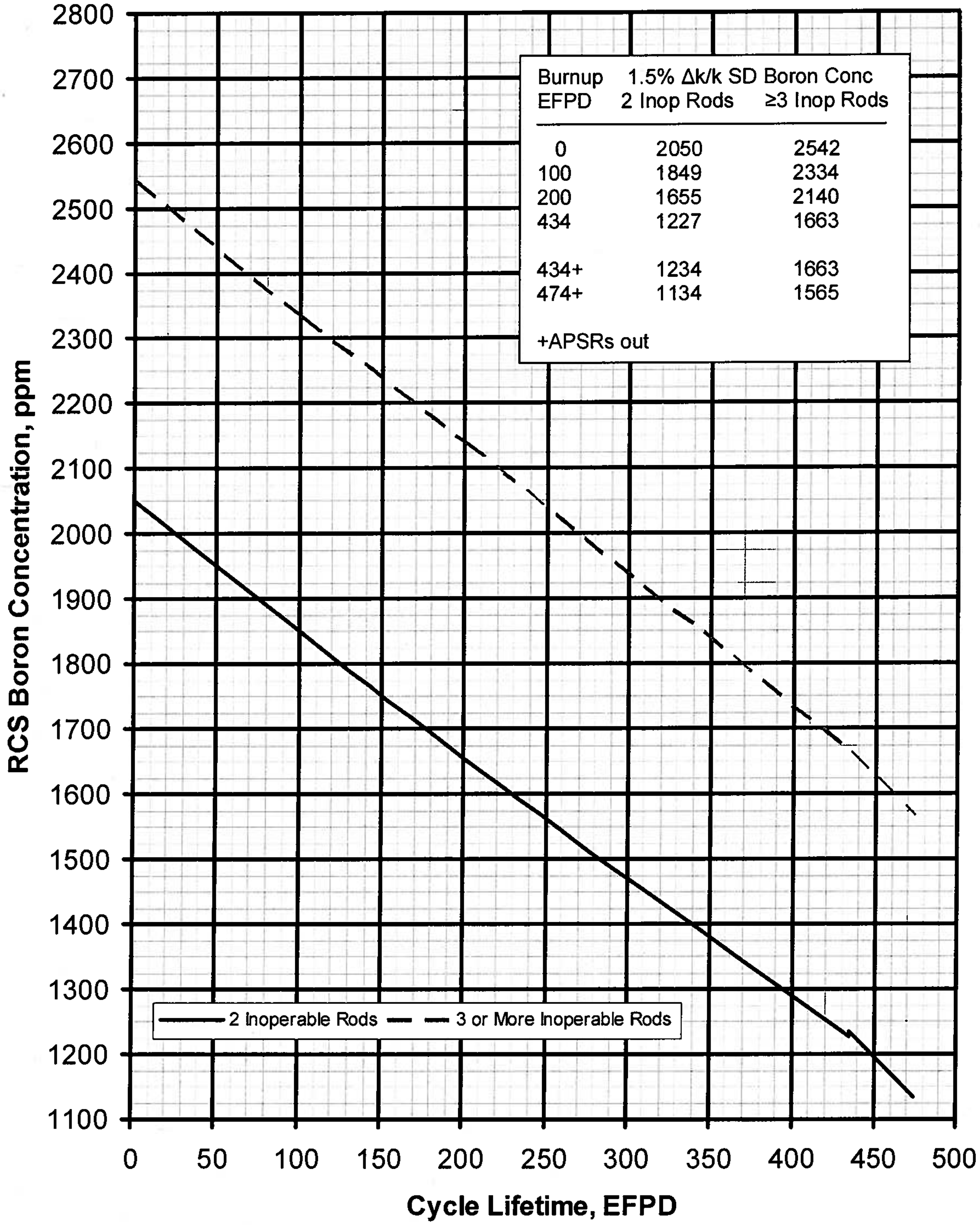
b) Close Batch Controller Outlet (CV-1250).

c) Verify MU Tank level 55 to 86" **AND** close BWST Outlet to OP HPI pump (CV-1407 or 1408).

d) Adjust Letdown flow to desired rate.

(12. CONTINUED ON NEXT PAGE)

Attachment B-16: Boron Concentration for 1.5% Shutdown Margin During Emergency Boration



Burnup	1.5% Δk/k SD Boron Conc	
EFPD	2 Inop Rods	≥3 Inop Rods
0	2050	2542
100	1849	2334
200	1655	2140
434	1227	1663
434+	1234	1663
474+	1134	1565

+APSRs out

QID: 0591 **Rev:** 0 **Rev Date:** 6/6/05 **Source:** Direct **Originator:** S.Pullin

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

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

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

Description: Ability to determine and interpret the following as they apply to the (NNI-X): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A Number: AA2.1 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** ☐ **SRO:** 84

Given:

- Pressurizer Level Control Valve CV-1235 indicates 50% open.
- RC Pump Seals Total Inj Flow valve CV-1207 indicates 50% open.
- Letdown flow indication is zero.
- Letdown pressure indication is zero.
- Letdown Orifice Bypass valve CV-1223 indicates 50% open.
- RCS pressure is 2210 psig and slowly rising.
- Pressurizer Spray valve CV-1008 indicates closed.

What procedure should be in use due to the above conditions?

A. 1203.015, Pressurizer Systems Failure

B. 1203.024, Loss of Instrument Air

C. 1203.047, Loss of NNI Power

D. 1203.012B, ACA for K10-A8 "LETDOWN TEMP HI"

Answer:

C. 1203.047, Loss of NNI Power

Notes:

Answer "C" is correct since the conditions given are representative of a loss of NNI X and Y power.
Answer "A" is incorrect, this would be in use if Spray valve was failed due to something other than a loss of NNI power.
Answer "B" is incorrect, this would be in use for failed valves due to loss of IA, but the positions given are different than for loss of air alone.
Answer "D" is incorrect, this is chosen for hi letdown temp but letdown flow would still be indicated while the question states there is none.

References:

1203.047, Chg. 000-01-0

History:

New for 2005 SRO exam.
Selected for 2010 SRO exam

NOTE

- MU Tank level recorder is inoperable.
- 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.
- Letdown Flow indication is lost.
- If NNI Y AC power is lost, the following occurs:
  - * Letdown Orifice Bypass (CV-1223) fails to 50%
  - * Letdown Pressure indication is lost

<u>INSTRUCTIONS</u>	<u>CONTINGENCY ACTIONS</u>
2. <u>IF</u> any combination of both NNI X and NNI Y power is lost, <u>THEN</u> perform the following:	2. **RETURN TO step 1.**
A. Trip the Rx **<u>AND</u>** perform **1202.001, "REACTOR TRIP"** in conjunction with this procedure. B. Manually actuate EFW **<u>AND</u>** verify proper actuation and control (1202.012, RT 5). C. Trip both MFW pumps. D. Open BWST Outlet to OP HPI pump (CV-1407 or 1408).	
E. Operate TURB BYP valves in HAND to control SG press 970 to 1020 psig.	E. **<u>IF</u>** TURB BYP valves are **<u>not</u>** available, **<u>THEN</u>** verify ATM Dump Control System operates to maintain SG press 1000 to 1040 psig.
F. Close RCS Makeup Block (CV-1233 or 1234) G. Operate HPI Block (CV-1220 or 1285) associated with OP HPI pump to maintain PZR level 90 to 110".	

(2. CONTINUED ON NEXT PAGE)

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

QID: 0592 **Rev:** 0 **Rev Date:** 6/6/05 **Source:** Direct **Originator:** S.Pullin

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

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

System Number: E08 **System Title:** LOCA Cool down

Description: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe management guidelines

K/A Number: 2.4.16 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** ☐ **SRO:** 85

Given:
- Rx was shutdown using 1203.045 Rapid Plant Shutdown,
- Due to a RCS leak
- RCS pressure 1720 psig and lowering slowly
- HPI flow 150 gpm
- A & B SG pressure 910 psig
- RCS cool down rate 35°F per hour
- All Turbine bypass valves closed

Which procedure should be in use?

A. 1202.001, Overcooling

B. 1203.041, Small Break LOCA cool down

C. 1203.040, Forced Flow cool down

D. 1202.010, ESAS

Answer:

B. 1203.041, Small Break LOCA cool down

Notes:

Answer "B" is correct with an uncontrolled cool down continuing due to break/HPI flow, regardless of SG status.
Answer "A" is incorrect, Overcooling entry conditions have not yet been met
Answer "C" is incorrect, although RCPs are running, there is no control of the cool down.
Answer "D" is incorrect, although parameters are close to ES actuation setpoints, the ESAS procedure would eventually transition to 1203.041.

References:

1203.039, Chg. 011

History:

New for 2005 SRO exam.
Selected for 2010 SRO exam

NOTE

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

14. IF total RCS leakage is in excess of that allowed by Tech Spec 3.4.13
AND
poses an immediate threat to plant operations,
THEN perform the following:

A. IF reactor is Critical,
THEN commence plant shutdown per Rapid Plant Shutdown (1203.045).

B. IF reactor is shutdown,
THEN perform RCS cooldown by one of the following:

1) IF RCS is cooling down due to HPI/break flow, independent of SG cooling,
THEN perform Small Break LOCA Cooldown (1203.041), while continuing with this procedure.

2) IF any RCP is running,
THEN perform Forced Flow Cooldown (1203.040), while continuing with this procedure.

3) IF all RCPs are off,
THEN perform Natural Circulation Cooldown (1203.013), while continuing with this procedure.

15. IF total RCS leakage is in excess of that allowed by Tech Spec 3.4.13
AND
poses no immediate threat to plant operations,
THEN perform the following:

A. Bring reactor to cold shutdown per Power Reduction and Plant Shutdown (1102.016) and Plant Shutdown and Cooldown (1102.010).

16. Advise Shift Manager to implement Emergency Action Level Classification (1903.010).

17. IF leakage is within Tech Spec 3.4.13 limits,
THEN continue with efforts to locate and isolate the leak using RCS Leak Detection (1103.013) and proceed as directed by Operations Manager.

18. IF leak is isolated,
THEN proceed as directed by Operations Manager.

END

SRO Written Exam

Tier 2 Group 1

PWR Examination Outline
Plant Systems - Tier 2/Group 1 (SRO)

Form ES-401-2

	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#	QID	Type
003 Reactor Coolant Pump								X				**A2.02** – Conditions which exist for an abnormal shutdown of a RCP in comparison to a normal shutdown of RCP	3.9	86	809	N
004 Chemical and Volume Control												**Not Selected**	**N/A**			
005 Residual Heat Removal												**Not Selected**	**N/A**			
006 Emergency Core Cooling												**Not Selected**	**N/A**			
007 Pressurizer Relief/Quench Tank												**Not Selected**	**N/A**			
008 Component Cooling Water												**Not Selected**	**N/A**			
010 Pressurizer Pressure Control								X				**A2.02** – Spray failures	3.9	87	762	R
012 Reactor Protection												**Not Selected**	**N/A**			
013 Engineered Safety Features Actuation								X				**A2.06** – Inadvertent ESFAS actuation	4.0	88	812	N
022 Containment Cooling												**Not Selected**	**N/A**			
025 Ice Condenser												**Not Selected**	**N/A**			
026 Containment Spray												**Not Selected**	**N/A**			
039 Main and Reheat Steam												**Not Selected**	**N/A**			
059 Main Feedwater												**Not Selected**	**N/A**			
061 Auxiliary/Emergency Feedwater											X	**2.2.22** – Knowledge of limiting conditions for operations and safety limits	4.7	89	811	N
062 AC Electrical Distribution												**Not Selected**	**N/A**			
063 DC Electrical Distribution											X	**2.2.42** – Ability to recognize system parameters that are entry-level conditions for Technical Specifications	4.6	90	810	N
064 Emergency Diesel Generator												**Not Selected**	**N/A**			
073 Process Radiation Monitoring												**Not Selected**	**N/A**			
076 Service Water												**Not Selected**	**N/A**			
078 Instrument Air												**Not Selected**	**N/A**			
103 Containment												**Not Selected**	**N/A**			
K/A Category Point Totals:										3	2	Group Point Total:	5			

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

QID: 0809 **Rev:** 0 **Rev Date:** 9/23/2009 **Source:** New **Originator:** S Pullin

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPs; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP

K/A Number: A2.02 **CFR Reference:** 41.5/43.5/45.3/45/13

Tier: 2 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** ☐ **SRO:** 86

Given:

- 100% Power,
- "C" RCP seal bleed off temperature 210 F.
- "C" RCP motor bearing temperature 185 F and stable,
- "C" RCP motor inboard vibration alert alarm,
- "C" RCP seal cavity pressure oscillating from 650 to 1250 psig.

What is the appropriate section and action of 1203.031, "Reactor Coolant Pump and Motor Emergency" which will mitigate the consequences of these malfunctions?

A. Section 2, "Seal Failure", Reduce reactor power to within the capacity of unaffected RCP combination and stop the affected RCP per Reactor Coolant Pump Operation, OP1103.006.

B. Section 2, "Seal Failure", Trip the Reactor and trip the affected RCP.

C. Section 5, "Motor / Bearing Trouble", Reduce reactor power to within the capacity of unaffected RCP combination and stop the affected RCP per Reactor Coolant Pump Operation, OP1103.006.

D. Section 5, "Motor / Bearing Trouble", Trip the Reactor and trip the affected RCP.

Answer:

B. Section 2, "Seal Failure", Trip the Reactor and trip the affected RCP.

Notes:

B is correct, a seal bleedoff temperature of greater than 200 F with no change in cooling (seal injection or ICW flow) meets the requirements to trip the RCP due to seal failure section.
A is incorrect. The given conditions require an abnormal shutdown of an RCP instead of a normal shutdown of an RCP.
C is incorrect. The given conditions require an abnormal shutdown of an RCP instead of a normal shutdown of an RCP.
D is incorrect. The given conditions do not indicate a bearing problem that warrents stopping the RCP.

References:

OP-1203.031 Change 018

History:

New selected for 2010 SRO exam

ATTACHMENT A

Page 1 of 1

RCP PARAMETERS

Seal Degradation/Seal Failure

1. **ANY of the following are criteria to SECURE the affected RCP per Section 1 Seal Degradation**

   - RCP seal cavity pressure oscillations exceed 800 psi peak-to-peak
   - ΔP across any stage exceeds 2/3 of system pressure on a running RCP **OR** exceeds 80% of system pressure on an idle RCP.
   - ≥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.

2. **ANY of the following are criteria to TRIP the affected RCP per Section 2 Seal Failure**

   - ≥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.
   - ΔP across a single stage equal to RCS press, with seal bleed off flow established.

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 or D **MOTOR** vibration; more than one channel >20 mils after startup stabilization
   - P-32A orC **MOTOR** vibration; more than one channel >0.8 in/sec after startup stabilization
   - ANY RC PUMP SHAFT vibration ≥29 mils after startup stabilization

SECTION 2
SEAL FAILURE

ENTRY CONDITIONS

One or more of the following:

- ≥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.
- ΔP across a single stage equal to RCS press, with seal bleed off flow established.

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)

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

QID: 0762 **Rev:** 0 **Rev Date:** 11/11/2000 **Source:** Repeat **Originator:** Steve Pullin

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures

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

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

Question: **RO:** ☐ **SRO:** 87

Given:

-Unit 1 is operating at 40% power.
-The Unit is in three pump ops due to the failure of P-32B.
-The Pressurizer Spray Control valve (CV-1008) fails open.

The ATC attempts to close the Pressurizer Spray Isolation valve (CV-1009) and it will NOT close

-Reactor Coolant Pressure is at 2100 psig and slowly lowering with all Pzr Heaters on.

What is the correct procedure and correct action for this condition?

A. 1202.001 Reactor Trip, and trip the Reactor.

B. 1202.001 Reactor Trip, and stop P-32C.

C. 1203.015 PZR System Failure, and trip the Reactor.

D. 1203.015 PZR System Failure, and stop P-32C.

Answer:

D. 1203.015 PZR System Failure, and stop P-32C.

Notes:

A is incorrect. Since the Power to Pump trip entry conditions are not met.
B is incorret with the correct action but with the incorrect procedure since the Power to Pump trip entry conditions are not met.
C is incorrect with the correct procedure but incorrect action.
D is correct.

References:

1203.015 Pzr System Failure Chg 16

History:

New for the 2009 Retake SRO Exam
Selected for 2010 SRO exam REPEAT

SECTION 6 -- PRESSURIZER SPRAY VALVE (CV-1008) FAILURE

INSTRUCTIONS

1. **IF failed open,**
   THEN place Pressurizer Spray Control switch in HAND AND attempt to close CV-1008 (modulating valve).

> **NOTE**
>
> CV-1009 torque switch can be overridden in the OPEN or CLOSE direction by holding the hand switch in the respective position.

A. IF CV-1008 will NOT close,
THEN close Pressurizer Spray Isolation Valve (CV-1009).

B. Verify Pressurizer heaters return RCS pressure to normal.

> **CAUTION**
>
> Pressurizer spray shall not be used if the temperature difference between the Pressurizer and the spray fluid is >430°F (TRM 3.4.3). Closing CV-1009 isolates the CV-1008 bypass spray flow.

C. IF necessary,
THEN control spray flow by cycling Pressurizer Spray Isolation Valve (CV-1009) open and closed.

D. IF both CV-1008 and CV-1009 do NOT close
AND RCS pressure is dropping,
THEN perform the following:

1) Verify all PZR heaters ON.

2) Immediately begin reducing load to 40% at 10%/min per Rapid Plant Shutdown (1203.045).

3) IF 4 RCPs are running
AND BOTH of the following conditions are met:

- Load is reduced to ≤675 MWe (≤75% load)
- Reactor power is ≤75%,

THEN perform the following:

a. Start "C" RCP HP Oil Lift Pump (P-63C) and "C" RCP Backstop Lube Oil Pump (P-81C).

b. Stop "C" RCP (P-32C).

c. WHEN zero speed is indicated,
THEN stop P-63C and P-81C.

(continued)

SECTION 6 – PRESSURIZER SPRAY VALVE (CV-1008) FAILURE

NOTE

In Modes 1 and 2, operation with only one RCP in each loop causes entry into TS 3.4.4 Condition A.

4) **IF** 3 RCPs running
AND all of the following conditions are met:

- Load is reduced to ≤360 MWe (≤40% load)
- Reactor power is ≤55%,
- "C" and "D" RCPs in-service

THEN perform the following:

a) Start "C" RCP HP Oil Lift Pump (P-63C) and "C" RCP Backstop Lube Oil Pump (P-81C).

b) Stop "C" RCP (P-32C).

c) **WHEN** zero speed is indicated,
THEN stop P-63C and P-81C.

d) Enter TS 3.4.4 Condition A.

5) **IF** 3 RCPs running,
AND "D" RCP is secured,
THEN perform the following:

a) Trip Reactor.

b) Secure P-32C as follows:

(1) Start "C" RCP HP Oil Lift Pump (P-63C) and "C" RCP Backstop Lube Oil Pump (P-81C).

(2) Stop "C" RCP (P-32C).

(3) **WHEN** zero speed is indicated,
THEN stop P-63C and P-81C.

c) Perform Reactor Trip (1202.001) while continuing with this procedure.

d) Enter TS 3.4.5 Condition A.

6) **WHEN** conditions permit a reactor building entry,
THEN attempt to manually close either CV-1008 or CV-1009.

E. Contact Ops Manager.

(continued)

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

QID: 0812 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based ability on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertant ESFAS actuation.

K/A Number: A2.06 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** ☐ **SRO:** 88

Given

- Plant at 100% power
- P-2B Condensate Pump OOS
- Inadvertent actuation of ES Channel #1
- S/U #1 OOS for maintenance LCO 3.8.1.A 72 hour Time Clock in effect

What would be the impact to the plant due to this malfunction and what procedure would be used to mitigate the effects?

A. #1 Emergency Diesel Generator would start and use OP-1105.003, Engineered Safeguards Actuation System
to reset the tripped channel.

B. Red Train High Pressure Injection would occur and use 1202.010, ESAS EOP to override HPI

C. Loss of power to A-1 bus and use 1202.001, Reactor Trip EOP

D. All Seal Return isolates and use OP1203.031, Reactor Coolant Pump and Motor Emergencies to realign seal bleed off.

Answer:

C. Loss of power to A-1 bus and use 1202.001, Reactor Trip EOP

Notes:

C is correct, the Unit Aux supply breaker to A-1 would open on ES Channel #1 actuation and would result in a reactor trip due to a loss of all Main Feedwater.
A is incorrect, although the EDG would start with a reactor trip the EOP would have priority over securing the EDG
B is incorrect, although HPI would occur the ESAS EOP would not be utilized to secure HPI for an inadvertant actuation.
D is incorrect, seal return would be realigned to the Quench Tank rather than isolate.

References:

STM 1-32 Rev 33
OP-1107.001 Change 073

History:

New selected for 2010 SRO exam

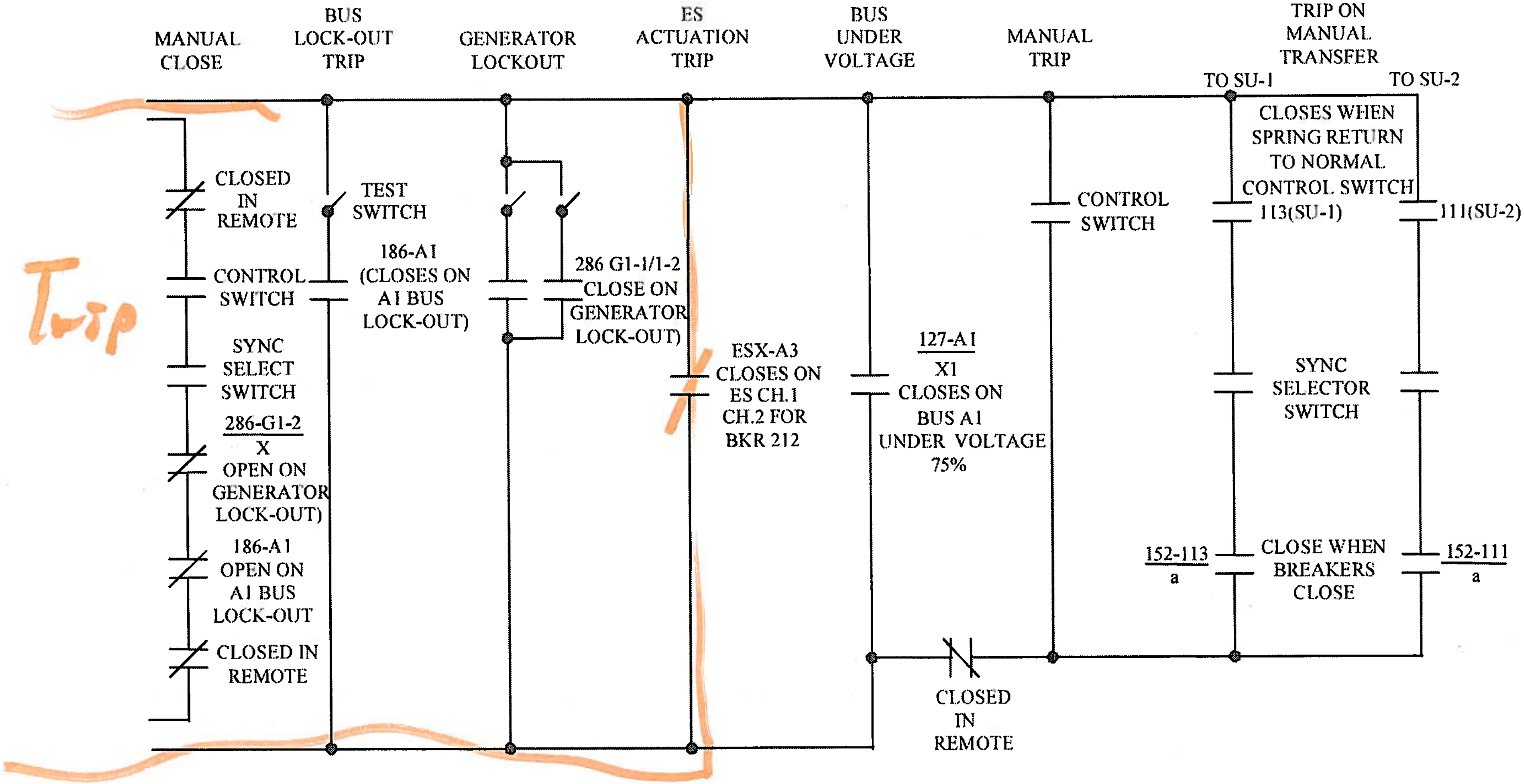


FIGURE 32.62: UNIT AUXILIARY TRANSFORMER FEEDER BRK 112(212)

ATTACHMENT B

Page 1 of 3

Date ______________

4160V SWITCHGEAR (NON-ES) CHECKLIST

WARNING

Operating a breaker without knowing the consequences could lead to injury or equipment damage.

NOTE

- This attachment assumes Unit 1 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 (except where specified Racked Down)
- Breaker control selector in REMOTE. (A-116 and A-206 may be in either local or remote as desired)
- Breaker labeled properly.

1.1 Log any breaker that is danger tagged <u>or</u> not in desired position on Lineup Exception Sheet (E-doc 1015.001F).

1.2 Notify plant labeling of any label discrepancies.

4160V Bus A1					
BREAKER NUMBER	DESCRIPTION	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INI-TIAL
A-113	Startup Xfmr #1 Feed to A1 (E-91)	--			
A-112	Unit Auxiliary Xfmr Feed to A1 (E-90)	Open			
A-111	Startup Xfmr #2 Feed to A1 (E-92)	--			
A-110	Circ Water Pump P-3A (E-271)	--			
A-109	Circ Water Pump P-3C (E-271)	--			
A-108	Main Chiller VCH-1A (E-372)	Closed			
A-107	Heater Drain Pump P8A (E-304)	Open			
A-106	Condensate Pump P2C (E-306)	--			

ATTACHMENT B

Date _______________

Page 2 of 3

BREAKER NUMBER	DESCRIPTION	DESIRED POSITION	ACTUAL POSITION	TAG (✓)	INI-TIAL
A-105	Condensate Pump P2A (E-306)	-			
A-104	A1 Feed to X14 (E-104)	Closed			
A-103	A1 Feed to X3 (E-104)	Closed			
A-102	A1 Feed to X1 (E-104)	Closed			
A-114	Electric Fire Pump P-6A (E-346)	Closed			
A-115	A1 Feed to X7 (E-104)	Closed			
A-116	Admin Bldg Unit Sub (E-106)	Closed			

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

QID: 0811 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

TUOI: A1LP-RO-EFIC **Objective:** 43 **Point Value:** 1

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

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

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: 2 **RO Imp:** 4.0 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 89

Given

- 'A' SG Low level transmitter feeding the 'D' EFIC Channel failed Lo
- 'B' SG Pressure transmitter feeding the 'C' EFIC Channel failed Hi

What operator actions are required per Technical Specifications?

A. Place 'D' channel in bypass per 3.3.11.A

B. Place 'C' channel in bypass per 3.3.11.B

C. Trip 'D' channel per 3.3.11.B

D. Trip 'C' channel per 3.3.11.A

Answer:

B. Place 'C' channel in bypass per 3.3.11.B

Notes:

B is correct, the Low Level transmitter failing low will result in a trip of the D Channel, 3.3.11.B requirements for two inoperable channels requires one to be placed in bypass and the other one tripped.
A is incorrect, "D" Channel is already tripped and placing in bypass would have no effect. TS 3.3.11.A is only applicable to one inoperable channel. The question asks what to do for two inoperable channels
C is incorrect, because it is only half of the action required by 3.3.11.B
D is incorrect because tripping "C" Channel would result in an EFIC actuation.

References:

TS 3.3.11 Amendment 215

History:

New selected for 2010 SRO exam

3.3 INSTRUMENTATION

3.3.11 Emergency Feedwater Initiation and Control (EFIC) System Instrumentation

LCO 3.3.11 The EFIC System instrumentation channels for each Function in Table 3.3.11-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.11-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Emergency Feedwater (EFW) Initiation or Main Steam Line Isolation Functions listed in Table 3.3.11-1 with one channel inoperable.	A.1 Place channel(s) in bypass or trip.	1 hour
B. One or more EFW Initiation or Main Steam Line Isolation Functions listed in Table 3.3.11-1 with two channels inoperable.	B.1 Place one channel in bypass.	1 hour
	<u>AND</u>	
	B.2 Place second channel in trip.	1 hour
C. One EFW Vector Valve Control channel inoperable.	C.1 Restore channel to OPERABLE status.	72 hours
D. Required Action and associated Completion Time not met for Function 1.b.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 4.	12 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met for Function 1.a or 1.d.	E.1 Reduce THERMAL POWER to ≤ 10% RTP.	6 hours
F. Required Action and associated Completion Time not met for Functions 1.c, 2, or 3.	F.1 Be in MODE 3. <u>AND</u> F.2 Reduce steam generator pressure to < 750 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.

SURVEILLANCE		FREQUENCY
SR 3.3.11.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.11.2	Perform CHANNEL FUNCTIONAL TEST.(Notes 1 & 2)	31 days
SR 3.3.11.3	Perform CHANNEL CALIBRATION.(Notes 1 & 2)	18 months

The following notes apply only to the SG Level – Low function:

Note 1: If the as-found channel setpoints are conservative with respect to the Allowable Value but outside their predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoints are not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Note 2: The instrument channel setpoint(s) shall be reset to a value that is equal to or more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint and the predefined as-found acceptance criteria band are specified in the Bases.

Table 3.3.11-1
Emergency Feedwater Initiation and Control System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUES
1. EFW Initiation				
a. Loss of MFW Pumps (Control Oil Pressure)	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 55.5 psig
b. SG Level - Low	1,2,3	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 9.34 inches[c,d]
c. SG Pressure - Low	1,2,3[a]	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
d. RCP Status	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2	NA
2. EFW Vector Valve Control				
a. SG Pressure – Low	1,2,3[a]	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
b. SG Differential Pressure – High	1,2,3[a]	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≤ 150 psid
3. Main Steam Line Isolation				
a. SG Pressure – Low	1,2,3[a][b]	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig

(a) When SG pressure ≥ 750 psig.

(b) Except when all associated valves are closed and deactivated.

(c) The SG Level – Low "Limiting Trip Setpoint" in accordance with NRC letter dated September 7, 2005, *Technical Specification For Addressing Issues Related To Setpoint Allowable Values*, is ≥ 10.42 inches.

(d) Includes an actuation time delay of ≤ 10.4 seconds.

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

QID: 0810 **Rev:** 0 **Rev Date:** 9/23/2009 **Source:** New **Originator:** S. Pullin

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

Section: 2 **Type:** Generic Knowledge and Abilities

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

Description: Ability to recognize system parameters that are entry-level conditions for Technical Specifications

K/A Number: 2.2.42 **CFR Reference:** 41.7/41.10/43.2/43.3/45.3

Tier: 2 **RO Imp:** 3.9 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** ☐ **SRO:** 90

Which of the following conditions requires entry into Technical Specification 3.8.4, "DC Sources, Operating" and what is the bases for Technical Specification 3.8.4?

A. D04A, "Battery Charger" inoperable and D06, "Battery" operable.
Bases is to insure reactor coolant pressure boundary limits are
not exceeded as a result of abnormalities

B. D04B, "Battery Charger" inoperable and D03B, "Battery Charger" inoperable.
Bases is to insure reactor coolant pressure boundary limits are
not exceeded as a result of abnormalities

C. D04A, "Battery Charger" inoperable and D04B, "Battery Charger" inoperable.
Bases is to insure adequate core cooling is provided, and reactor building operability
and other functions are maintained in the event of a postulated DBA

D. D03B, "Battery Charger" inoperable and D07, "Battery" operable.
Bases is to insure adequate core cooling is provided, and reactor building operability
and other functions are maintained in the event of a postulated DBA

Answer:

C. D04A, "Battery Charger" inoperable and D04B, "Battery Charger" inoperable.
Bases is to insure adequate core cooling is provided, and reactor building operability
and other functions are maintained in the event of a postulated DBA

Notes:

C is correct, with both battery chargers on the same train being inoperable, the subsystem is inoperable requiring entry into TS 3.8.4. The bases for TS 3.8.4 is to insure adequate core cooling is provided, and reactor building operability and other functions are maintained in the event of a postulated DBA
A is incorrect, Only one of the two charges being inoperable does not affect the operability of the subsystem. The bases used for this option is partially correct.
B is incorrect, two battery chargers are inoperable but since they are on different trains they do not affect the operability of either subsystem. The bases used for this option is partially correct.
D is incorrect, Only one of the two charges being inoperable does not affect the operability of the subsystem. The bases used for this option is partially correct.

References:

T.S. 3.8.4 Amendment 215

History:

New selected for 2010 SRO exam

LCO

The DC electrical power subsystems, each subsystem consisting of one battery, one of two battery chargers and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an abnormality or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires the associated battery to be OPERABLE and connected to the associated DC bus and one of its respective chargers to be operating and connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of abnormalities; and

b. Adequate core cooling is provided, and reactor building OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed by the definition of OPERABILITY for each required supported load.

ACTIONS

<u>A.1</u>

Condition A represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 8 hour limit is consistent with the allowed time for an inoperable DC distribution system train.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery chargers, or inoperable battery chargers and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power subsystems with attendant

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Both DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	8 hours
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 124.7 V on float charge.	7 days
SR 3.8.4.2	Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	18 months

SRO Written Exam

Tier 2 Group 2

ES-401 | PWR Examination Outline | Form ES-401-2

Plant Systems - Tier 2/Group 2 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#	QID	Type
001 Control Rod Drive												**Not selected**	**N/A**			
002 Reactor Coolant												**Not selected**	**N/A**			
011 Pressurizer Level Control												**Not selected**	**N/A**			
014 Rod Position Indication												**Not selected**	**N/A**			
015 Nuclear Instrumentation												**Not selected**	**N/A**			
016 Non-nuclear Instrumentation											X	**2.2.40** – Ability to apply technical specifications for a system	4.7	91	599	D
017 In-core Temperature Monitor												**Not selected**	**N/A**			
027 Containment Iodine Removal												**Not selected**	**N/A**			
028 Hydrogen Recombiner and Purge Control												**2.4.23** – Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations. **Rejected system replaced with 016 Non-Nuclear Instrumentation**	**N/A**			
029 Containment Purge												**Not selected**	**N/A**			
033 Spent Fuel Pool Cooling												**Not selected**	**N/A**			
034 Fuel Handling Equipment											X	**2.1.40** –Knowledge of refueling administrative requirements.	3.9	92	600	D
035 Steam Generator								X				**A2.01** – Faulted or ruptured S/Gs.	4.6	93	813	N
041 Steam Dump/Turbine Bypass Control												**Not selected**	**N/A**			
045 Main Turbine Generator												**Not selected**	**N/A**			
055 Condenser Air Removal												**Not selected**	**N/A**			
056 Condensate												**Not selected**	**N/A**			
068 Liquid Radwaste												**Not selected**	**N/A**			
071 Waste Gas Disposal												**Not selected**	**N/A**			
072 Area Radiation Monitoring												**Not selected**	**N/A**			
075 Circulating Water												**Not selected**	**N/A**			
079 Station Air												**Not selected**	**N/A**			
086 Fire Protection												**Not selected**	**N/A**			
K/A Category Point Totals:								1			2	Group Point Total:		3		

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

QID: 0599 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-NNI **Objective:** 35 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation

Description: Ability to apply technical specifications 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:**	4
Group:	2	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** ☐ **SRO:** 91

REFERENCE PROVIDED

The plant is operating at 100% power.
Both PZR level transmitters LT-1001 and LT-1002 have failed LOW.

Which of the following actions are required by Technical Specification 3.3.15 and Table 3.3.15-1?

A. Be in Mode 3 within 6 hours.

B. Both channels must be restored within 7 days.

C. Restore one channel to operable status within 30 days or be in Mode 3 within 6 hours.

D. Restore one channel to operable status within 7 days or be in Mode 3 within 6 hours.

Answer:

D. Restore one channel to operable status within 7 days or be in Mode 3 within 6 hours.

Notes:

Answer "D" is correct in accordance with Table 3.3.15-1 and actions C and E.
Answer "A" is incorrect, there is still an allowance of 7 days per action C.
Answer "B" is incorrect, only one channel must be restored.
Answer "C" is inocrrect, this is a combination of A and E.

References:

T.S. 3.3.15 Amendment 232

Note: T.S. 3.3.15 must be in students' handout.

History:

Direct from regular exam bank QID#ANO-OPS1-6623
Selected for 2005 SRO exam.
Selected for 2010 SRO exam.

3.3 INSTRUMENTATION

3.3.15 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.15 The PAM instrumentation for each Function in Table 3.3.15-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to prepare and submit a Special Report.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.15-1 for the channel.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3.	6 hours
	AND	
	E.2 Be in MODE 4.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---

These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.15.2	----------------NOTE---------------- Neutron detectors are excluded from CHANNEL CALIBRATION. ------------------------------------ Perform CHANNEL CALIBRATION.	18 months

Table 3.3.15-1
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Wide Range Neutron Flux	2	E
2. RCS Hot Leg Temperature	2	E
3. RCS Hot Leg Level	2	F
4. RCS Pressure (Wide Range)	2	E
5. Reactor Vessel Water Level	2	F
6. Reactor Building Water Level (Wide Range)	2	E
7. Reactor Building Pressure (Wide Range)	2	E
8. Penetration Flow Path Automatic Reactor Building Isolation Valve Position	2 per penetration flow path(a)(b)	E
9. Reactor Building Area Radiation (High Range)	2	F
10. Deleted		
11. Pressurizer Level	2	E
12. a. SG “A” Water Level – Low Range	2	E
b. SG “B” Water Level – Low Range	2	E
c. SG “A” Water Level – High Range	2	E
d. SG “B” Water Level – High Range	2	E
13. a. SG “A” Pressure	2	E
b. SG “B” Pressure	2	E
14. Condensate Storage Tank Level	2	E
15. Borated Water Storage Tank Level	2	E
16. Core Exit Temperature (CETs per quadrant)	2	E
17. a. Emergency Feedwater Flow to SG “A”	2	E
b. Emergency Feedwater Flow to SG “B”	2	E
18. High Pressure Injection Flow	2	E
19. Low Pressure Injection Flow	2	E
20. Reactor Building Spray Flow	2	E

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

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

QID: 0600 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** J.Cork

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

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

System Number: 034 **System Title:** Fuel Handling Equipment

Description: Knowledge of refueling administrative requirements.

K/A Number: 2.1.40 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 2 **RO Imp:** 2.8 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** ☐ **SRO:** 92

Given:

- Plant is in a Refueling outage.
- Core re-load is in progress.
- Approximately 90% of the core is in the Reactor vessel.

The Main Fuel Handling Bridge has a once-burned fuel assembly and is in the process of indexing over the specified core location when NI-502 fails to 0.1 cps.

What action should be taken?

A. No action necessary because with NI-501 operating, Tech Spec NI requirements for operablility are met.

B. Contact the Main Fuel Bridge operator and place the assembly in a core location without any adjacent fuel assemblies.

C. Halt operations on the Main Fuel Bridge. Core geometry cannot be changed unless two neutron flux monitors are operable.

D. Verify boron concentration in the Refueling Canal is greater than 2326 ppm and then continue fuel load.

Answer:

C. Halt operations on the Main Fuel Bridge. Core geometry cannot be changed unless two neutron flux monitors are operable.

Notes:

Answer "C" is correct per 1502.004, 5.3, and T.S. 3.9.2
Answer "A" is incorrect, although only one is required in Mode 6, two NI's are required during core alterations.
Answer "B" is incorrect, this is still a core alteration.
Answer "D" is incorrect, this is simply a requirement for refueling.

References:

1502.004, Chg. 041
T.S. 3.9.2 Amendment 215

History:

Direct from regular exam bank QID#3178
Selected for 2005 SRO exam.
Selected for 2010 SRO exam

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.

5.0 LIMITS AND PRECAUTIONS

5.1 During movement of any fuel assemblies within the reactor building, radiation levels shall either be monitored by RE-8017 or applicable TRM 3.9.1 Condition has been entered and the Required Action to place a portable survey instrument of appropriate range and sensitivity in-service have been performed. (TRM 3.9.1).

5.2 During movement of any fuel assemblies within the auxiliary building, radiation levels shall either be monitored by RE-8009 or applicable TRM 3.9.2 Condition has been entered and the Required Action to place a portable survey instrument of appropriate range and sensitivity in-service have been performed. (TRM 3.9.2).

5.3 One source range neutron flux monitor shall be operable in Mode 6. Two source range neutron flux monitors shall be operable during core alterations (TS 3.9.2).

5.4 One decay heat removal loop shall be operable and in operation in Mode 6 with water level ≥23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.4 for contingencies and exceptions.

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2

a. One source range neutron flux monitor shall be OPERABLE, and

b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable during CORE ALTERATIONS.	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately Immediately
B. No OPERABLE source range neutron flux monitor.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status. AND B.2 Perform SR 3.9.1.1.	Immediately Once per 12 hours

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

QID: 0813 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

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

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

System Number: 035 **System Title:** Steam Generator System (S/GS)

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the S/G and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulted or ruptured S/Gs.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier: 2 **RO Imp:** 4.5 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.6 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 93

Given:

- Plant at 100% power

Simultaneously the following occurs:
- Reactor trips on low RCS Pressure
- N-16 alarm on "A" Steam Generator
- Steam Line High Range Radiation monitor RI-2681 in alarm.
- RCS pressure drops to 1300 psig
- CET's indicate 550°F
- Reactor Building and Aux Building sump levels are stable.

Starting with 1202.001, Reactor Trip EOP, which of the following lists the order of EOP's to mitigate this event?

A. 1202.002 Loss of Subcooling Margin and 1202.006 Tube Rupture

B. 1202.002 Loss of Subcooling Margin and 1202.010 ESAS

C. 1202.006 Tube Rupture and 1202.010 ESAS

D. 1202.006 Tube Rupture and 1202.012 RT-10

Answer:

A. 1202.002 Loss of Subcooling Margin and 1202.006 Tube Rupture

Notes:

A is correct, The Reactor Trip EOP immediate actions will send the operator to Loss of Subcooling margin, with the only LOCA being a tube rupture the Loss of Subcooling Margin procedure will send the operator to Tube Rupture.
B is incorrect, ESAS would only be entered if RCS pressure dropped below 150 psig.
C and D are incorrect, Reactor Trip would send the operator to Loss of Subcooling Margin EOP first.

References:

OP-1202.001 Change 031
OP-1202.002 Change 006

History:

New selected for 2010 SRO exam

INSTRUCTIONS	CONTINGENCY ACTIONS
③ **Check adequate SCM.**	3. Check elapsed time since loss of adequate SCM **AND** perform the following: A. **IF** ≤2 minutes have elapsed, **THEN** trip all RCPs. B. **IF** >2 minutes have elapsed, **THEN** leave currently running RCPs on. C. Advise Shift Manager to implement Emergency Action Level Classification (1903.010). D. **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure.
4. **Advise Shift Manager to implement Emergency Action Level Classification (1903.010).**	
5. **Reduce Letdown by closing Orifice Bypass (CV-1223).**	
6. **Open BWST Outlet to OP HPI pump (CV-1407 or 1408).**	
7. **IF Emergency Boration is NOT in progress, THEN adjust Pressurizer Level Control setpoint to 100".**	

INSTRUCTIONS

8. **Check SG tube integrity:**

   A. None of the following rad monitor indications rising OR in alarm:

   - Main Condenser (RI-3632)
   - OTSG N-16 Gross (RI-2691 and 2692)
   - Steam Line High Range (RI-2681 and 2682).

   B. No report from Nuclear Chemistry that SG tube leak exists.

   C. No rise in unidentified RCS leakage accompanied by:

   - Higher than expected SG level
   - Lower than expected FW flow rate

9. **IF CET SCM is adequate, THEN control RCS press low within limits of Figure 3 (RT 14).**

10. **Check RCS press remains ≥150 psig.**

11. **Check SG levels at or approaching one of the following:**

SCM adequate	SCM < adequate
300 to 340"	370 to 410"

CONTINGENCY ACTIONS

8. IF CET SCM is adequate,
   OR
   no other LOCA indications exist (RB and Aux Bldg sump levels are stable),
   THEN GO TO 1202.006, "TUBE RUPTURE" procedure.

10. IF RCS press is <150 psig,
    THEN GO TO 1202.010, "ESAS" procedure.

11. Re-verify proper EFW actuation and control (RT 5).

    A. IF all MFW and EFW is lost
       AND
       either of the following conditions is met,
       THEN GO TO 1202.004, "OVERHEATING" procedure.

    - CET SCM adequate
    - CET temps ≥610°F

SRO Written Exam

Tier 3

Facility: **Arkansas Nuclear One – Unit 1** Date of Exam: **3/5/2010**

Category	K/A #	Topic	RO			
			IR	#	QID	Type#
1. Conduct of Operations	**2.1.7**	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.7	94	492	D
	2.1.35	Knowledge of the fuel-handling responsibilities of SROs	3.9	95	814	N
	Subtotal		2			
2. Equipment Control	**2.2.25**	Knowledge of bases and technical specifications for limiting conditions of operations and safety limits.	4.2	96	646	D
	2.2.19	Knowledge of maintenance work order requirements.	3.4	97	815	N
	Subtotal		2			
3. Radiation Control	**2.3.14**	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.8	98	816	N
	Subtotal		1			
4. Emergency Procedures / Plan	**2.4.30**	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.	4.1	99	411	D
	2.4.35	Knowledge of local auxiliary operator tasks during an emergency and the operational resultant effects.	4.0	100	750	D
	Subtotal		2			
Tier 3 Point Total			7			

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

QID: 0492 **Rev:** 1 **Rev Date:** 12/4/06 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-EOP08 **Objective:** 7 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

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: 3 **RO Imp:** 4.4 **RO Select:** No **Difficulty:** 4

Group: **SRO Imp:** 4.7 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** ☐ **SRO:** 94

Given the plant conditions following a reactor trip:

- RCS temperature: 605 degrees stable
- RCS pressure: 2300 psig slowly dropping
- ERV: open in AUTO
- OTSG shell temperature: 558 degrees
- OTSG levels 20 inches, steady
- PZR level 180 inches, rising

Which of the following actions are required?

A. Trip the running RCP per 1202.002, Loss of Subcooling Margin.

B. Isolate the ERV per 1202.001, Reactor Trip.

C. Select the reflux boiling setpoint per RT-5.

D. Initiate Full HPI per RT 3.

Answer:

B. Isolate the ERV per 1202.001, Reactor Trip.

Notes:

Answer "B" is correct. A pressurizer steam space leak is indicated by PZR level rising with RCS pressure dropping and no rise in RCS temperature. ERV is open and should have closed at 2395 psig.
Answer "A" is incorrect, Tube to Shell delta T of 60 degrees tubes hotter would require this action however the delta T is only 47 degrees in the question.
Answer "C" is incorrect, although RCS temperature/pressure conditions are close to a loss of subcooling margin which would require selection of Reflux Boiling but SCM is still adequate.
Answer "D" is incorrect, Full HPI would be required if the ERV opened in Auto with the Overheating EOP in effect but the Overheating entry conditions are not met.

References:

1202.001, Chg. 031

History:

Modified from regular exambank QID#3314.
Used on 2004 SRO Exam.
Modified for use on 2007 SRO Exam.
Selected for 2010 SRO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
28. Verify ERV, Pressurizer Spray, and Pressurizer Heaters operate to control RCS press 2050 to 2250 psig.	28. Perform the following: A. **IF** ERV is open in AUTO **AND** RCS press < 2395 psig, **THEN** verify ERV Isolation closed (CV-1000). B. **IF** Pressurizer Spray valve is open in AUTO **AND** RCS press < 2155 psig, **THEN** close Pressurizer Spray Isolation (CV-1009). C. **IF** Pressurizer Heaters fail to operate in AUTO, **THEN** operate Heaters manually to control RCS press 2050 to 2250 psig.
29. Check at least one RCP running.	29. Perform the following: A. Verify proper EFW actuation and control (RT 5). B. **IF** H1 or H2 is energized with normal voltage (≥6900V) **AND** CET SCM is adequate **AND** RCPs are available, **THEN** perform the following: 1) Start one RCP in each loop (RT 11).
30. Check RCS T-cold remains ≥ 540°F.	30. **IF** RCS T-cold is < 540°F **AND** dropping, **THEN** **GO TO 1202.003, "OVERCOOLING"** procedure.
31. Check adequate SCM.	31. **GO TO 1202.002, "LOSS OF SUBCOOLING MARGIN"** procedure.

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

QID: 0814 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

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

Section: 2 **Type:** Generic Knowledge and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of the fuel-handling responsibilities of SROs

K/A Number: 2.1.35 **CFR Reference:** 41.10 / 43.7

Tier:	3	**RO Imp:**	2.2	**RO Select:**	No	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** **SRO:** 95

Which of the following conditions would require the SRO in charge of fuel handling to order a stop to fuel movement in the Reactor Building?

A. Outage Control Center reports that the reactor has been subcritical for 90 hours.

B. National Weather Service declares a Tornado Watch in effect for Conway County.

C. One Control Room Emergency Air Conditioning System (CREACS) inoperable for the past 5 days.

D. Reactor Building Radiation monitor RE-8017 inoperable, and portable survey instrument is being monitored on the fuel handling bridge.

Answer:

A. Outage Control Center reports that the reactor has been subcritical for 90 hours.

Notes:

A is correct, the reactor must be subcritical for greater than 100 hours prior to fuel movement.
B is incorrect, Pope, Johnson, Yell and Logan counties in a tornado watch would require stopping fuel movement. Conway county is immediately east of Pope county.
C is incorrect, with one CREACS channel inoperable we have 30 days to repair prior to stopping fuel movement.
D is incorrect, RE-8017 is desired to be operable for monitoring radiation levels on the bridge, however if it becomes inoperable any portable survey instrument is allowed for monitoring rad levels and continue fuel movement.

References:

OP-1502.004 Change 041

History:

New selected for 2010 SRO exam.

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.

5.0 LIMITS AND PRECAUTIONS

5.1 During movement of any fuel assemblies within the reactor building, radiation levels shall either be monitored by RE-8017 or applicable TRM 3.9.1 Condition has been entered and the Required Action to place a portable survey instrument of appropriate range and sensitivity in-service have been performed. (TRM 3.9.1).

5.2 During movement of any fuel assemblies within the auxiliary building, radiation levels shall either be monitored by RE-8009 or applicable TRM 3.9.2 Condition has been entered and the Required Action to place a portable survey instrument of appropriate range and sensitivity in-service have been performed. (TRM 3.9.2).

5.3 One source range neutron flux monitor shall be operable in Mode 6. Two source range neutron flux monitors shall be operable during core alterations (TS 3.9.2).

5.4 One decay heat removal loop shall be operable and in operation in Mode 6 with water level ≥23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.4 for contingencies and exceptions.

5.9 Refueling canal water level shall be maintained ≥23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel during movement of irradiated fuel assemblies within the reactor building (TS 3.9.6).

5.10 A minimum of 10 feet separation shall be maintained between fuel assemblies when two assemblies are moved simultaneously in the transfer canal (TRM 3.9.3).

5.11 Each required reactor building penetration shall be verified in the required status within 7 days prior to refueling operations and at least every 7 days thereafter (SR 3.9.3.1).

5.12 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 100 hours (TRM 3.9.3).

5.13 In the event of a complete core offload, the decay heat load to be transferred to the Spent Fuel Pool shall be verified to be within the limits of the spent Fuel Pool Cooling System (TRM 3.7.3).

5.14 No tornado watches shall be in effect for Pope, Yell, Johnson, or Logan counties in Arkansas during movement of any fuel assemblies within the auxiliary building (TRM 3.9.2) or the Reactor Building (ER-ANO-2002-1078-007 Rev. 0).

5.14.1 Upon issue of a tornado watch for any of these counties, enter TRM 3.9.2 Condition B, cease all fuel handling in the auxiliary building and Reactor Building. Fuel handling in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position.

5.15 Loads in excess of 2000 pounds (such as the cask loading pit gate and tilt pit gate, ~4000 lbs. each) shall not travel over fuel assemblies in the storage pool (TRM 3.7.2).

5.16 During movement of irradiated fuel assemblies, either two Control Room Emergency Ventilation System (CREVS) trains shall be operable, and one CREVS train shall be capable of automatic operation, or the applicable TS 3.7.9 Condition has been entered and the Required Actions of TS 3.7.9 have been performed. The control room boundary may be opened intermittently under admin controls (TS 3.7.9).

5.17 During movement of irradiated fuel assemblies, either two Control Room Emergency Air Conditioning System (CREACS) shall be operable or applicable TS 3.7.10 Condition has been entered and the Required Actions of TS 3.7.10 have been performed.

5.18 During movement of irradiated fuel assemblies, either two channels of Control Room Isolation - High Radiation shall be operable or the applicable TS 3.3.16 Condition and the Required Actions of TS 3.3.16 (immediately place one Operable CREVS train in emergency recirculation mode) have been performed.

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

QID: 0646 **Rev:** 0 **Rev Date:** 10/23/2006 **Source:** Direct **Originator:** Cork/Possage
TUOI: A1LP-RO-EDG **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic Knowledge and Abilities
System Number: 2.2 **System Title:** Equipment Control
Description: Knowledge of 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: 3 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3
Group: G **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 96

REFERENCE PROVIDED

Given:

- #1 EDG has one Air Start Compressor and it's associated Air Receiver Tanks tagged out.
- The remaining Air Start Compressor on #1 EDG trips while EDG is running for a surveillance.
- The Air Receiver Tanks' pressure is 145 psig.

In accordance with Technical Specifications, what is the required action for the above conditions?

A. No actions are necessary since the EDG is running and an air start system is not needed.

B. Restore required starting air receiver pressure to within limits in 48 hours.

C. Declare #1EDG inoperable immediately.

D. Be in Mode 3 within 12 hours.

Answer:

C. Declare #1EDG inoperable immediately.

Notes:

Answer "C" is correct, with only one receiver bank and pressure <158 psig the EDG must be declared inoperable per 3.8.3.E.1.
Answer "A" is incorrrect, although the EDG is running, if it tripped there would not be enough air for a re-start.
Answer "B" is incorrect, this is the action from 3.8.3.D and would be applicable if pressure was between 158 and 175 psig.
Answer "D" is incorrect, this action is from 3.8.1.F and would be applicable if the EDG was not made operable within 7 days.

References:

3.8.3 and Bases Amendment 215

History:

Uses QID 447 stem with some modifications, all answers are different, therefore it is a new question.
New question for 2007 SRO exam.
Selected for the 2010 SRO exam

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil and Starting Air

LCO 3.8.3 The stored diesel fuel oil and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DG fuel oil storage tank(s) with fuel volume < 20,000 gallons and > 17,140 gallons.	A.1 Restore fuel oil volume to within limits.	48 hours
B. One or more DGs with stored fuel oil total particulates not within limit.	B.1 Restore fuel oil total particulates to within limits.	7 days
C. One or more DGs with new fuel oil properties not within limits.	C.1 Restore stored fuel oil properties to within limits.	30 days
D. One or more DGs with required starting air receiver pressure < 175 psig and ≥ 158 psig.	D.1 Restore required starting air receiver pressure to within limits.	48 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met. OR One or more DGs with diesel fuel oil or required starting air subsystem not within limits for reasons other than Condition A, B, C, or D.	E.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 20,000 gallons of fuel.	31 days
SR 3.8.3.2	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.3	Verify each DG required air start receiver pressure is ≥ 175 psig.	31 days
SR 3.8.3.4	Check for and remove accumulated water from each fuel oil storage tank.	31 days

ACTIONS (continued)

B.1

This Condition is entered as a result of a failure to meet the acceptance criterion of Specification 5.5.13. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

C.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.2 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

D.1

With starting air receiver pressure < 175 psig in the required receivers, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is ≥ 158 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that the credited DG start is accomplished on the first attempt, and the low probability of an event during this brief period.

E.1

With a Required Action and associated Completion Time not met, or one or more DGs with fuel oil or required starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

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

QID: 0815 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

TUOI: ASLP-SRO-MNTC **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic K&A

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of maintenance work order requirements

K/A Number: 2.2.19 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	3	**RO Imp:**	2.3	**RO Select:**	No	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.4	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** ☐ **SRO:** 97

Given:

- Annunciator K12-B5, P-7A Turbine Trip alarms
- WCO reports that the linkage for the trip throttle valve has broken.

You are the Shift Manager,

Per EN-WM-100, "Work Request (WR) Generation, Screening and Classification," which work order process should be used to correct this condition.

A. Priority One Work Order

B. Priority Two Work Order

C. Tool Pouch Maintenance / No work order required

D. FIN Team / No work order required

Answer:

A. Priority One Work Order

Notes:

A is correct, since P-7A inoperability is a 72 hour Time Clock, a Priority 1 work order would be initiated to begin maintenance and work around the clock to completion.
B is incorrect, Priority 2 work orders are entered into the T-3 week schedule and would not be urgent enough to meet the needs of the plant.
C is incorrect, Tool pouch maintenance is not allowed on safety related equipment even though the repairs are skill of the craft.
D is incorrect, FIN Team can not work on safety related equipment with out a work order even though the repairs are skill of the craft.

References:

EN-WM-100 Rev 3

History:

New selected for 2010 SRO exam

significance of the condition identified. For on line work, the priority is determined in accordance with Attachment 9.1. Each priority state is shown below along with guidelines for starting the work.

- Priority 1: Begin immediately following planning of the work order and work around the clock.
- Priority 2: Schedule at earliest opportunity within T-3.
- Priority 3: Schedule at next available system week within the 12 week process or next available system window.
- Priority 4: Schedule as resources allow within the normal process.
- Priority 5: Work only when time allows (fill in activity).
- Priority 8: Outage work where performance is mandatory (required / de-rate)
- Priority 9: Outage work where performance is discretionary (potential)

[10] Power Block Equipment - All SSC's required for the safe and reliable operation of the station. It will include all safety-related and balance-of-plant system and components required for the operation of the station, including radioactive waste processing and storage, and switchyard equipment maintained by the station. Systems, structures, or components required to maintain federal or state regulatory compliance should be included in this grouping. This classification does not include buildings or structures that support station staff, such as offices or storage structures, or the HVAC and support systems focused only on habitability of those structures.

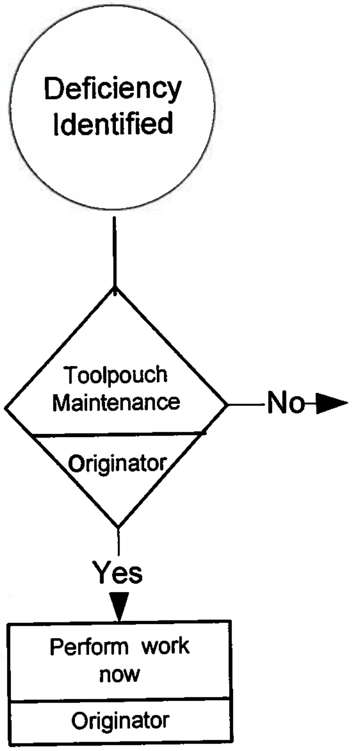
[11] Skill Of The Craft – A task that workers are familiar with and experienced in performing, which are not complex in the actions required and are common to their craft. Familiarity may have been gained through training or on the job performance. To perform the task safely and successfully, the worker would not require further instruction or oversight.

[12] Work Instructions - A set of work steps included in a work package provided to direct how work is to be accomplished.

[13] Work Request Screening Committee – The Work Request Screening Committee, chaired by Scheduling, meets each normal workday and reviews WR's contained on the work screening report (Attachment 9.5). The standard report is WEB based and includes work requests that have not been previously converted to work orders or approved as toolpouch, and includes all work requests generated since the previous meeting.

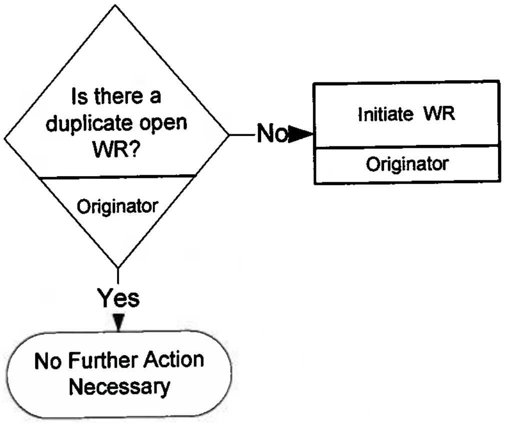
, Scheduling will review the work screening report prior to the meeting and provide

5.2[2] cont.



The individual discovering the deficiency, or another person, can repair it, if qualified to do so, utilizing Toolpouch Maintenance, and if:

- the activity does not affect a safety related function
- there is no risk of a plant transient
- the activity does not require either a procedure, work instructions or material other than consumable
- the activity does not involve work on ASME Code or EQ equipment
- the activity does not alter plant configuration
- the activity is not complex and is within the skill of the personnel
- the activity does not affect a Maintenance Rule function
- the activity requires no additional support beyond that for normal plant access



If the requirements for Toolpouch Maintenance are not met, the identifier should generate a WR in IAS. IAS required actions for Work Request Initiation, Operability Screening and Classification are contained in Attachment 9.6.

The identifier of the activity provides the following information:

- Originator name (Defaults in IAS)
- The date identified (Defaults in IAS)
- identification of the component including Equipment Number
- A description of the deficiency (Attach. 9.3)
- Recommended solution, if known

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

QID: 0816 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

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

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

K/A Number: 2.3.14 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier: 3 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 3

Group: G **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** ☐ **SRO:** 98

During a fuel handling accident Krypton-85 is the major source of gaseous activity released from a damaged Fuel assembly.

Which portion of the body will receive the highest dose after a fuel handling accident?

A. Skin dose from Beta

B. Whole body dose from Gamma

C. Extremities dose from Beta

D. Internal Organ dose from Gamma

Answer:

A. Skin dose from Beta

Notes:

A is correct, skin dose rates from K-85 are 100 times higher than the whole body , gamma dose rates.
B, C, and D are all incorrect.

References:

OP-1203.042 Change 005-03-0

History:

New selected for 2010 SRO exam

SECTION 1 -- FUEL HANDLING ACCIDENT

2. IF damage to a spent fuel assembly is suspected,
THEN perform the following:

WARNING

Krypton-85, a beta emitter, is the major source of gaseous activity released from a damaged spent fuel assembly that has decayed >190 days. Skin dose rates from Kr-85 are 100 times higher than the whole body, gamma dose rate. Instruments not sensitive to beta, such as self-reading dosimeters and survey meters with their beta windows closed, will read less than the actual values.

A. Direct RP personnel to proceed to the area and inform them of the beta hazard associated with a damaged spent fuel assembly.

B. Inspect the spent fuel assembly with all available means to determine if damage has occurred.

C. **IF** even slight spent fuel assembly damage is detected,
THEN take actions for confirmed damage per this procedure.

D. **IF** fuel assembly is not damaged,
THEN proceed as directed by Plant Management.

END

DISCUSSION

A fuel handling accident has occurred when fuel handling equipment malfunctions or other occurrences result in damage to a spent fuel assembly. Until proven otherwise, it is assumed that one or more fuel pins are ruptured. This assumption is made regardless of how slight the damage to the fuel assembly(ies) appears.

Ruptured spent fuel cladding releases fission product gases in undetermined quantities to the pool water or possibly to atmosphere in case of dry fuel storage handling accident. A mechanical damage type accident is considered the maximum potential source of activity release during refueling operations.

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

QID: 0411 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** Direct **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.

What is the Emergency Plan 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 Change 036

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

ARKANSAS NUCLEAR ONE		
E-DOC TITLE: **EMERGENCY CLASS INITIAL NOTIFICATION MESSAGE**	**E-DOC NO.** **1903.011-Y**	**CHANGE NO.** **036**

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

[redacted]

QID: 0750 **Rev:** 2 **Rev Date:** 6/23/08 **Source:** Direct **Originator:** Spullin

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

Section: 2.0 **Type:** Generic K/As

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

Description: Knowledge of local auxiliary operator tasks during an emergency and the operational resultant effects

K/A Number: 2.4.35 **CFR Reference:** 41.10/43.5/45.13

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

Question: **RO:** ☐ **SRO:** 100

Given:
- Severe Fire on 335 Auxiliary Building on Unit 1
- Reactor has been tripped

Which of the following actions would the CRS direct the Outside AO to perform and what procedural guidance would be used?

A. Fire fighting tasks per "Fire or Explosion" procedure 2203.034.

B. Securing Polishers per "Reactor Trip/Outage Recovery" procedure 1102.006.

C. Placing the Startup Boiler in service per "Startup Boiler Operation" procedure 1106.022.

D. Throttle CV-2627 EFW Supply to "A" SG per "Fires in Areas Affecting Safe Shutdown" procedure 1203.049

Answer:

D. Throttle CV-2627 EFW Supply to "A" SG per "Fires in Areas Affecting Safe Shutdown" procedure 1203.049

Notes:

"A." is incorrect; due to recent procedures changes have the opposite Unit AO's fighting the fire, but the WCO non licensed operator has fire fighting duties
"B." is incorrect; under normal Reactor trip conditions this would be an Outside AO action promptly following Rx trip, but it is not in the Reactor procedure
"C." is incorrect; under normal Reactor trip conditions this would be an Outside AO would perform following Rx trip
"D." is correct; this is a new procedure action for the non licensed operators

---

References:

1203.049 Fires in Areas affecting Safe Shutdown Change 005

History:

Selected for 2010 SRO exam

Fire Area C (335' Aux Building)

Outside AO Required Actions

5. **IF directed by CBOR,**
THEN close MSIVs by manually opening the following valves:

- IA Vent to MSIV "A" (IA-2691B)
- IA Vent to MSIV "A" (IA-2691C)
- IA Vent to MSIV "A" (IA-2691D)
- IA Vent to MSIV "A" (IA-2691E)

- IA Vent to MSIV "B" (IA-2692B)
- IA Vent to MSIV "B" (IA-2692C)
- IA Vent to MSIV "B" (IA-2692D)
- IA Vent to MSIV "B" (IA-2692E)

6. **WHEN notified by CBOT that CV-1405 is de-energized,**
THEN verify RB Sump Line A Outlet (CV-1405) closed (A Decay Heat Vault).

    A. Notify CBOR that CV-1405 is de-energized and closed.

7. **WHEN directed by CBOR**
AND notified by CBOT that CV-2627 is de-energized,
THEN locally close EFW P-7A to SG-A Isol (CV-2627) in UNPPR.

8. **WHEN notified by CBOT that CV-1220 is de-energized,**
THEN locally verify HPI to P-32D Discharge (CV-1220) is open (UNPPR).

    A. Notify CBOR that CV-1220 is de-energized and open.

9. **WHEN directed by CRS,**
THEN throttle CV-2627 as directed (UNPPR).

10. **WHEN notified by CBOT that CV-1407 is de-energized,**
THEN verify BWST T-3 Outlet (CV-1407) open (behind Waste Gas Panel on 354' EL).

    A. Notify CBOR that CV-1407 is de-energized and open.