NOTE REGARDING THE TWO VERSIONS OF THE DRAFT RO EXAM

The following file includes the originally submitted Draft RO and SRO exams. In addition, there is a second Draft RO exam. The reason for the second Draft RO exam is that portions of the original Draft RO exam were compromised late in the development process. This compromise necessitated revision of the original draft by replacing and/or modifying approximately half of the originally submitted questions. Both versions of the draft are included here for completeness. The second version of the Draft RO exam includes resolution of comments from the original Draft RO submittal, so that the second Draft RO exam closely resembles the Final RO exam (and it provides the necessary reference material and KA information for the questions since the Final RO exam file does not include this extra material).

FIRST (ORIGINAL) VERSION
OF THE
DRAFT RO and SRO WRITTEN EXAMS

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

QID: 0023 **Rev:** 1 **Rev Date:** 3/16/05 **Source:** Modified **Originator:** S.Pullin

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the reasons for the following as they apply to a reactor trip: Actions contained in EOP for reactor trip.

K/A Number: EK3.01 **CFR Reference:** CFR: 41.5 / 41.10 / 45.6 / 45.13

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

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

Question: RO: 1 SRO:

The following conditions exist immediately after a reactor trip:

- Group 2, Rod 4 failed to fully insert into the core
- RCS pressure is at 1750 psig and trending down
- Pressurizer level is at 50 inches and trending down
- A OTSG pressure is at 880 psig and trending down
- B OTSG pressure is at 885 psig and trending down
- Subcooling margin is at 55 °F and trending up slowly
- Turbine Trip Solenoid Power Available light is OFF

Choose the appropriate operator response:

a. Manually actuate MSLI for affected SG(s) and EFW.

b. Commence emergency boration per RT-12.

c. Trip all Reactor Coolant Pumps.

d. Initiate High Pressure Injection per RT-2.

Answer:

a. Manually actuate MSLI for affected SG(s) and EFW.

Notes:

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

References:

1202.001 (Rev 028-03-0), Reactor Trip, Immediate Actions

History:

Developed for 1998 RO/SRO Exam.
Modified for use in 2005 RO Exam.

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.

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

The following plant conditions exist:

- Pressurizer temperature is 645 °F
- Pressurizer level is 245 inches and rising
- RCS Pressure is 1950 psig and lowering
- Quench Tank pressure is 50 psig and rising
- The ERV acoustic monitor indicates flow noise

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

A. Approximately 258 °F

b. Approximately 281 °F

c. Approximately 298 °F

d. Approximately 645 °F

Answer:

c. Approximately 298 °F

Notes:

Candidates should be provided with steam tables. The temperature elements on the ERV tailpipe would indicate the saturation temperature for the Quench Tank pressure, therefore, answer (c) is correct. The disclaimers are incorrect because: (a) is the saturation temperature for 35 psia, (b) is saturation temperature for 50 psia and (d) is the saturation temperature of the pressurizer.

References:

ASME Steam Tables

History:

Developed for 1998 RO Exam.
Selected for 2002 RO exam under 008 AK1.01.
Modified for 2005 RO exam.

Table 2: Saturated Steam: Pressure Table

Abs Press. Lb/Sq In. p	Temp Fahr t	Specific Volume: Sat. Liquid v_f	Specific Volume: Evap v_{fg}	Specific Volume: Sat. Vapor v_g	Enthalpy: Sat. Liquid h_f	Enthalpy: Evap h_{fg}	Enthalpy: Sat. Vapor h_g	Entropy: Sat. Liquid s_f	Entropy: Evap s_{fg}	Entropy: Sat. Vapor s_g	Abs Press. Lb/Sq In. p
0.08865	32.018	0.016022	3302.4	3302.4	0.0003	1075.5	1075.5	0.0000	2.1872	2.1872	0.08865
0.25	59.323	0.016032	1235.5	1235.5	27.382	1060.1	1087.4	0.0542	2.0425	2.0967	0.25
0.50	79.586	0.016071	641.5	641.5	47.623	1048.6	1096.3	0.0925	1.9446	2.0370	0.50
1.0	101.74	0.016136	333.59	333.60	69.73	1036.1	1105.8	0.1326	1.8455	1.9781	1.0
5.0	162.24	0.016407	73.515	73.532	130.20	1000.9	1131.1	0.2349	1.6094	1.8443	5.0
10.0	193.21	0.016592	38.404	38.420	161.26	982.1	1143.3	0.2836	1.5043	1.7879	10.0
14.696	212.00	0.016719	26.782	26.799	180.17	970.3	1150.5	0.3121	1.4447	1.7568	14.696
15.0	213.03	0.016726	26.274	26.290	181.21	969.7	1150.9	0.3137	1.4415	1.7552	15.0
20.0	227.96	0.016834	20.070	20.087	196.27	960.1	1156.3	0.3358	1.3962	1.7320	20.0
30.0	250.34	0.017009	13.7266	13.7436	218.9	945.2	1164.1	0.3682	1.3313	1.6995	30.0
40.0	267.25	0.017151	10.4794	10.4965	236.1	933.6	1169.8	0.3921	1.2844	1.6765	40.0
50.0	281.02	0.017274	8.4967	8.5140	250.2	923.9	1174.1	0.4112	1.2474	1.6586	50.0
60.0	292.71	0.017383	7.1562	7.1736	262.2	915.4	1177.6	0.4273	1.2167	1.6440	60.0
70.0	302.93	0.017482	6.1875	6.2050	272.7	907.8	1180.6	0.4411	1.1905	1.6316	70.0
80.0	312.04	0.017573	5.4536	5.4711	282.1	900.9	1183.1	0.4534	1.1675	1.6208	80.0
90.0	320.28	0.017659	4.8779	4.8953	290.7	894.6	1185.3	0.4643	1.1470	1.6113	90.0
100.0	327.82	0.017740	4.4133	4.4310	298.5	888.6	1187.2	0.4743	1.1284	1.6027	100.0
110.0	334.79	0.01782	4.0306	4.0484	305.8	883.1	1188.9	0.4834	1.1115	1.5950	110.0
120.0	341.27	0.01789	3.7097	3.7275	312.6	877.8	1190.4	0.4919	1.0960	1.5879	120.0
130.0	347.33	0.01796	3.4364	3.4544	319.0	872.8	1191.7	0.4998	1.0815	1.5813	130.0
140.0	353.04	0.01803	3.2010	3.2190	325.0	868.0	1193.0	0.5071	1.0681	1.5752	140.0
150.0	358.43	0.01809	2.9958	3.0139	330.6	863.4	1194.1	0.5141	1.0554	1.5695	150.0
160.0	363.55	0.01815	2.8155	2.8336	336.1	859.0	1195.1	0.5206	1.0435	1.5641	160.0
170.0	368.42	0.01821	2.6556	2.6738	341.2	854.8	1196.0	0.5269	1.0322	1.5591	170.0
180.0	373.08	0.01827	2.5129	2.5312	346.2	850.7	1196.9	0.5328	1.0215	1.5543	180.0
190.0	377.53	0.01833	2.3847	2.4030	350.9	846.7	1197.6	0.5384	1.0113	1.5498	190.0
200.0	381.80	0.01839	2.2689	2.2873	355.5	842.8	1198.3	0.5438	1.0016	1.5454	200.0
210.0	385.91	0.01844	2.16373	2.18217	359.9	839.1	1199.0	0.5490	0.9923	1.5413	210.0
220.0	389.88	0.01850	2.06779	2.08629	364.2	835.4	1199.6	0.5540	0.9834	1.5374	220.0
230.0	393.70	0.01855	1.97991	1.99846	368.3	831.8	1200.1	0.5588	0.9748	1.5336	230.0
240.0	397.39	0.01860	1.89909	1.91769	372.3	828.4	1200.6	0.5634	0.9665	1.5299	240.0
250.0	400.97	0.01865	1.82452	1.84317	376.1	825.0	1201.1	0.5679	0.9585	1.5264	250.0
260.0	404.44	0.01870	1.75548	1.77418	379.9	821.6	1201.5	0.5722	0.9508	1.5230	260.0
270.0	407.80	0.01875	1.69137	1.71013	383.6	818.3	1201.9	0.5764	0.9433	1.5197	270.0
280.0	411.07	0.01880	1.63169	1.65049	387.1	815.1	1202.3	0.5805	0.9361	1.5166	280.0
290.0	414.25	0.01885	1.57597	1.59482	390.6	812.0	1202.6	0.5844	0.9291	1.5135	290.0
300.0	417.35	0.01889	1.52384	1.54274	394.0	808.9	1202.9	0.5882	0.9223	1.5105	300.0
350.0	431.73	0.01912	1.30642	1.32554	409.8	794.2	1204.0	0.6059	0.8909	1.4968	350.0
400.0	444.60	0.01934	1.14162	1.16095	424.2	780.4	1204.6	0.6217	0.8630	1.4847	400.0
450.0	456.28	0.01954	1.01224	1.03179	437.3	767.5	1204.8	0.6360	0.8378	1.4738	450.0
500.0	467.01	0.01975	0.90787	0.92762	449.5	755.1	1204.7	0.6490	0.8148	1.4639	500.0
550.0	476.94	0.01994	0.82183	0.84177	460.9	743.3	1204.3	0.6611	0.7936	1.4547	550.0
600.0	486.20	0.02013	0.74962	0.76975	471.7	732.0	1203.7	0.6723	0.7738	1.4461	600.0
650.0	494.89	0.02032	0.68811	0.70843	481.9	720.9	1202.8	0.6828	0.7552	1.4381	650.0
700.0	503.08	0.02050	0.63505	0.65556	491.6	710.2	1201.8	0.6928	0.7377	1.4304	700.0
750.0	510.84	0.02069	0.58880	0.60949	500.9	699.8	1200.7	0.7022	0.7210	1.4232	750.0
800.0	518.21	0.02087	0.54809	0.56896	509.8	689.6	1199.4	0.7111	0.7051	1.4163	800.0
850.0	525.24	0.02105	0.51197	0.53302	518.4	679.5	1198.0	0.7197	0.6899	1.4096	850.0
900.0	531.95	0.02123	0.47968	0.50091	526.7	669.7	1196.4	0.7279	0.6753	1.4032	900.0
950.0	538.39	0.02141	0.45064	0.47205	534.7	660.0	1194.7	0.7358	0.6612	1.3970	950.0
1000.0	544.58	0.02159	0.42436	0.44596	542.6	650.4	1192.9	0.7434	0.6476	1.3910	1000.0
1050.0	550.53	0.02177	0.40047	0.42224	550.1	640.9	1191.0	0.7507	0.6344	1.3851	1050.0
1100.0	556.28	0.02195	0.37863	0.40058	557.5	631.5	1189.1	0.7578	0.6216	1.3794	1100.0
1150.0	561.82	0.02214	0.35859	0.38073	564.8	622.2	1187.0	0.7647	0.6091	1.3738	1150.0
1200.0	567.19	0.02232	0.34013	0.36245	571.9	613.0	1184.8	0.7714	0.5969	1.3683	1200.0
1250.0	572.38	0.02250	0.32306	0.34556	578.8	603.8	1182.6	0.7780	0.5850	1.3630	1250.0
1300.0	577.42	0.02269	0.30722	0.32991	585.6	594.6	1180.2	0.7843	0.5733	1.3577	1300.0
1350.0	582.32	0.02288	0.29250	0.31537	592.3	585.4	1177.8	0.7906	0.5620	1.3525	1350.0
1400.0	587.07	0.02307	0.27871	0.30178	598.8	576.5	1175.3	0.7966	0.5507	1.3474	1400.0
1450.0	591.70	0.02327	0.26584	0.28911	605.3	567.4	1172.8	0.8026	0.5397	1.3423	1450.0
1500.0	596.20	0.02346	0.25372	0.27719	611.7	558.4	1170.1	0.8085	0.5288	1.3373	1500.0
1550.0	600.59	0.02366	0.24235	0.26601	618.0	549.4	1167.4	0.8142	0.5182	1.3324	1550.0
1600.0	604.87	0.02387	0.23159	0.25545	624.2	540.3	1164.5	0.8199	0.5076	1.3274	1600.0
1650.0	609.05	0.02407	0.22143	0.24551	630.4	531.3	1161.6	0.8254	0.4971	1.3225	1650.0
1700.0	613.13	0.02428	0.21178	0.23607	636.5	522.2	1158.6	0.8309	0.4867	1.3176	1700.0
1750.0	617.12	0.02450	0.20263	0.22713	642.5	513.1	1155.6	0.8363	0.4765	1.3128	1750.0
1800.0	621.02	0.02472	0.19390	0.21861	648.5	503.8	1152.3	0.8417	0.4662	1.3079	1800.0
1850.0	624.83	0.02495	0.18558	0.21052	654.5	494.6	1149.0	0.8470	0.4561	1.3030	1850.0
1900.0	628.56	0.02517	0.17761	0.20278	660.4	485.2	1145.6	0.8522	0.4459	1.2981	1900.0
1950.0	632.22	0.02541	0.16999	0.19540	666.3	475.8	1142.0	0.8574	0.4358	1.2931	1950.0
2000.0	635.80	0.02565	0.16266	0.18831	672.1	466.2	1138.3	0.8625	0.4256	1.2881	2000.0
2100.0	642.76	0.02615	0.14885	0.17501	683.8	446.7	1130.5	0.8727	0.4053	1.2780	2100.0
2200.0	649.45	0.02669	0.13603	0.16272	695.5	426.7	1122.2	0.8828	0.3848	1.2676	2200.0
2300.0	655.89	0.02727	0.12406	0.15133	707.2	406.0	1113.2	0.8929	0.3640	1.2569	2300.0
2400.0	662.11	0.02790	0.11287	0.14076	719.0	384.8	1103.7	0.9031	0.3430	1.2460	2400.0
2500.0	668.11	0.02859	0.10209	0.13068	731.7	361.6	1093.3	0.9139	0.3206	1.2345	2500.0
2600.0	673.91	0.02938	0.09172	0.12110	744.5	337.6	1082.0	0.9247	0.2977	1.2225	2600.0
2700.0	679.53	0.03029	0.08165	0.11194	757.3	312.3	1069.7	0.9356	0.2741	1.2097	2700.0
2800.0	684.96	0.03134	0.07171	0.10305	770.7	285.1	1055.8	0.9468	0.2491	1.1958	2800.0
2900.0	690.22	0.03262	0.06158	0.09420	785.1	254.7	1039.8	0.9588	0.2215	1.1803	2900.0
3000.0	695.33	0.03428	0.05073	0.08500	801.8	218.4	1020.3	0.9728	0.1891	1.1619	3000.0
3100.0	700.28	0.03681	0.03771	0.07452	824.0	169.3	993.3	0.9914	0.1460	1.1373	3100.0
3200.0	705.08	0.04472	0.01191	0.05663	875.5	56.1	931.6	1.0351	0.0482	1.0832	3200.0
3208.2*	705.47	0.05078	0.00000	0.05078	906.0	0.0	906.0	1.0612	0.0000	1.0612	3208.2*

*Critical pressure

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

QID: 0028 **Rev:** 1 **Rev Date:** 3/21/05 **Source:** Direct **Originator:** J.Cork

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 determine or interpret the following as they apply to a small break LOCA: Actions to be taken, based on RCS temperature and pressure, saturated and superheated.

K/A Number: EA2.01 **CFR Reference:** CFR: 43.5 / 45.13

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

Group: 2 **SRO Imp:** 4.8 **SRO Select:** No **Taxonomy:** C

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

A small break LOCA cooldown is in progress with the following:
* Subcooling margin less than adequate.
* HPI flow is less than full flow from a single HPI pump.
* RV head voiding is indicated.
* CETs indicate 350°F

Which of the following cooldown limits apply?

a. 100 °F/hr

b. 50 °F/hr

c. 25 °F/hr

d. No cooldown limits apply

Answer:

d. No cooldown limits apply

Notes:

To regain subcooling margin the operators must either re-pressurize the RCS or cooldown. With subcooling margin less than adequate the Small Break LOCA cooldown procedure specifies that no cooldown limits apply, therefore, answer (d) is correct.
Answers (a), (b) and (c) are cooldown limits if subcooling margin is adequate.

References:

1202.002 (Rev 004-02-0), Loss of Subcooling Margin

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

- This section is used for rapid RCS cooldown if adequate SCM is lost and either of the following is true:
  - HPI flow is < full flow from one HPI pump **AND** RV Head void is indicated
    OR
  - RCPs were not tripped within 2 minutes of LOSCM
- Cooldown rate limits do **not** apply.
- During this cooldown, primary to secondary heat transfer will be temporarily lost as the primary level drops from the bottom of the hot leg bend to below the secondary level or below the EFW nozzles if EFW flow exists.

19. IF conditions requiring rapid cooldown are met, THEN perform rapid cooldown as follows:

A. **WHEN** SG press is <720 psig, **THEN** bypass MSLI as follows:

1) On Initiate module in each EFIC cabinet, place each SG Bypass toggle switch in BYPASS and release.

B. Begin rapid cooldown at maximum attainable rate by fully opening TURB BYP valves or ATM Dump valves.

C. Place **all** EFW CNTRL valves in VECTOR OVERRIDE:

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

D. Place **all** EFW ISOL valves in MANUAL:

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

(19. CONTINUED ON NEXT PAGE)

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

QID: 0507 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Repeat **Originator:** NRC

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

Section: 4.1 **Type:** Generic EPE

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:** No **Taxonomy:** C

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

Given:

- Unit 1 is at 100% Power.
- A LOCA occurs with RCS pressure stabilizing at 200 psig
- BWST level is 8 feet.

The purpose of the PIGGYBACK mode during recovery from a LOCA is to:

A. Provide a minimum recirculation flowpath for the LPI pumps.

B. Provide Net Positive Suction Head to the HPI pumps.

C. Provide a low pressure discharge to the RCS when RCS pressure is <200psig.

D. Provide a means of transferring to long-term core cooling with the LPI pumps and the RB sump when RCS Pressure is <200psig.

Answer:

B. Provide Net Positive Suction Head to the HPI pumps.

Notes:

"B" is correct, ensures NPSH to HPI pumps prior to transfer to RB sump recirculation.
"A" is incorrect, although this will provide some flow for the LPI pumps, this is not the purpose.
"C" is incorrect, although the LPI pumps are used, the HPI pumps provide a high pressure discharge.
"D"is incorrect, although this does provide long term cooling, this is done in case RCS pressure stablilizes above LPI discharge pressure.

References:

STM 1-05, Rev 13

History:

Developed by NRC for 2004 RO/SRO Exam.
Repeated for use on 2005 RO Exam.

When operating in the LPI mode the LPI screens on SPDS will provide the following information when selected. LPI pumps motor amps, LPI flow, pump suction and discharge pressures.

2.6 Piggyback Mode

(Refer to Figure 05.16)

Briefly described in section 2.4 was the piggyback mode of operation used to supply suction to the HPI pumps from the RB sump via the LPI pump discharge.

During accident conditions such as a LOCA, the BWST water is used to inject into the core. Depending on the size of the leak and the ability to perform a plant cooldown will determine the time HPI will be required to inject water into the core. If RCS pressure is less than ~150 psig the LPI system suction will be aligned to the RB sump and will supply long term cooling of the core. If RCS pressure is greater than the discharge head of the LPI pump and the BWST is emptied due to coolant leakage, the piggyback mode of operation allows for the HPI pumps to provide water from the RB sump back to the core.

The two piggyback valves are 4-inch gate valves manufactured by Anchor Darling. The valves are designated as CV-1276 and CV-1277 with a design pressure and temperature of 560 psig and 300°F. The LPI supply to the HPI pump comes off the main line prior to the injection block valves and ties in on the suction header of the "A" & "C" HPI pumps. P-34A discharge is directed to the suction of P-36A through CV-1276. CV-1276 is located in the "A" makeup pump room. P-34B discharge is directed to the suction of the P-36C through CV-1277. CV-1277 is located in the "C" makeup pump room.

The piggyback valves are controlled and aligned by the operator from hand-switches located on panels C-16 & C-18.

Piggyback mode of operation will be aligned when directed by Small Break LOCA Cooldown procedure 1203.041. The LPI suction is aligned to the RB sump when BWST level reaches 6 feet. Steps necessary to align LPI suction to the RB sump and piggyback mode are contained in Repetitive Task 15 of the Emergency Operating Procedure (EOP).

Repetitive Task 15 provides guidance on actions required for the following conditions.

- Only 1 LPI pump is available for piggyback.
- Throttling of the RB Spray system if actuated to ensure adequate NPSH for the LPI pumps is available.
- Isolation of the NaOH tank when level is less than 25 feet.
- Actions when RB sump isolation valve fails to open.
- Steps for a complete loss of piggyback flow during swap to RB sump.
- Steps to cross connect LPI pump suctions when only 1 LPI pump available and power to RB sump isolation valves is not available for the operable side.

- Actions to take if unable to establish suction from the RB sump.
- Steps to isolate back leakage into the BWST through BW-1 or the test & recirc header.
- Steps to maintain Makeup tank (T-4) level less than 86 inches.

2.7 Reg. Guide 1.97 Instruments

As a follow up to NUREG 0737, Reg. Guide 1.97 identifies the minimum instrumentation required to provide control room indications during accidents.

Instruments associated with the Low Pressure Injection / Decay Heat system that are classified as Reg. Guide 1.97 instruments are listed in Table 5.4. For explanation of category and type refer to Non-nuclear Instrumentation STM 1-69 or the SAR.

2.8 Industry Events

Over a number of years, Loss of Decay Heat Removal events have impacted the Nuclear industry. This section will discuss the conditions that can occur to cause a loss of Decay Heat Removal. A number of diverse industry events have taken place that can directly or indirectly affect operation of DHR and LPI Systems.

These events are:

- Mis-positioning of valves.
- thermal binding of various types of gate valves.
- check valve failure or degradation.
- spurious closure of DHR suction valves.
- DHR Pump trips.
- flooding of power plant buildings.
- loss of DHR Pump suction.
- INPO document SEN 212

This list does not cover all of the conditions that can cause a loss of DHR. Due to the number of maintenance activities that occur during a refueling outage, any activity that is not controlled properly can lead to a loss of DHR.

2.8.1 SOER's

Significant Operating Experience Reports are written by INPO when a trend in the industry is indicated and if significant for safe operation a SOER is written.

SOER 85-4 was written in August of 1985. This SOER covered the "Loss or Degradation of Residual Heat Removal Capability in PWRs". It discussed events involving loss or degradation of DHR capability at pressurized water reactors. SOER 85-4 focused on the three most common ways of losing DHR:

- low reactor vessel level resulting in loss of DHR pump suction
- closure of the DHR pump suction valve

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

QID: 0452 **Rev:** 1 **Rev Date:** 3/17/2005 **Source:** Modified **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APE

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 seals.

K/A Number: AK2.07 **CFR Reference:** 41.7 / 45.7 / 43.5

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 4

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

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

The following conditions exist:

- Plant is currently at 100% power.
- The following annunciators alarm:

RCP SEAL INJ FLOW LO (K08-A7)
RCP SEAL COOLING FLOW LO (K08-E7)
RCP BLEED OFF TEMP HI (K08-C7)

- CBOT reports that all above annunciators are caused by one RCP, P-32C.

Which of the following actions is procedurally directed for the above conditions?

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

B. Trip P-32C RCP, trip reactor, and go to 1202.001, Reactor Trip.

C. Trip P-32C RCP and isolate seal bleedoff to all RCPs.

D. Trip reactor, trip P-32C RCP, and go to 1202.001, Reactor Trip.

Answer:

D. Trip reactor, trip P-32C RCP, and go to 1202.001, Reactor Trip.

Notes:

"D" is correct since Rx Power is >92% and tripping an RCP would trip the reactor.
"A" is incorrect, this would only be done if Rx power were less than 92%.
"B" is incorrect, the reactor is tripped before the RCP.
"C" is incorrect, tripping RCP is correct but seal bleedoff should not be isolated to all RCPs and Rx should be tripped.

References:

1203.031, Reactor Coolant Pump and Motor Emergency, change 016-02-0, page 14, section 3, step 3.

History:

Created for 2002 SRO exam.
Used on 2004 RO/SRO exam.
Modified for 2005 RO exam.

SECTION 3
SIMULTANEOUS LOSS OF SEAL INJECTION AND SEAL COOLING FLOW

INSTRUCTIONS

CAUTION

Continued RCP operation without seal injection or seal cooling will cause seal damage due to overheating

1. **Attempt to restore seal injection OR seal cooling within 2 minutes, while continuing with this procedure.**

2. **IF at least 1 RCP in each loop is unaffected,**
THEN begin reducing reactor power at maximum rate using Rapid Plant Shutdown (1203.045) to within limits of unaffected RCP combination.

3. **IF either seal injection OR seal cooling cannot be restored within 2 minutes,**
THEN perform the following:

NOTE

- Flux/ΔFlux/Flow reactor trip setpoints are shown on COLR Figure 10.
- High power/pumps reactor trip setpoints are:
  - One pump per loop ≥55%
  - Zero pumps in one loop ≥0%
- Tripping 1 RCP with reactor power >92% may result in reactor trip on high power/imbalance/flow.

A. IF tripping the affected RCP(s) will result in reactor trip on high power/pumps,
THEN perform the following:

1) Trip reactor.

2) Trip affected RCP(s).

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

B. IF tripping the affected RCP(s) will NOT cause a reactor trip on high power/pumps,
THEN perform the following:

CAUTION

Operation with only 1 RCP in each loop is limited to 18 hours, after which the plant shall be placed in Mode 3 within an additional 6 hours per TS 3.4.4.

1) Trip affected RCP(s).

2) Verify proper ICS response.

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

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

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

Section: 4.2 **Type:** Generic APE

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

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

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

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

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

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

Given the following:

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

Which of the following actions should be perfomed in response to these indications?

A. Trip HPI pump and isolate Letdown by closing Letdown Isolation, CV-1221.

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

C. Take manual control of Pressurizer Level Control, CV-1235, and stabilize Pressurizer level.

D. Trip HPI pump, trip reactor, and go to EOP 1202.001, Reactor Trip.

Answer:

A. Trip HPI pump and isolate Letdown by closing Letdown Isolation, CV-1221.

Notes:

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

References:

1203.026, Rev. 009-04-0

History:

New for 2005 RO exam.

SECTION 2 -- LARGE MAKEUP AND PURIFICATION SYSTEM LEAK

1.0 SYMPTOMS

1.1 Dropping MU tank level or pressurizer level in conjunction with rising level in AUX building sump or Dirty Waste Drain Tanks (T-20A and T-20B).

1.2 Annunciator alarms

- AREA MONITOR RADIATION HI (K10-B1)
- HPI PUMP TRIP (K10-A6)
- RCP SEAL INJ FLOW LO (K08-A7)

- MU TANK LEVEL HI/LO (K10-B7)
- MU TANK PRESS HI/LO (K10-B8)

1.3 Loss of or erratic makeup flow and seal injection flow.

1.4 Loss of or erratic makeup (HPI) pump discharge header pressure.

2.0 IMMEDIATE ACTION

2.1 None.

3.0 FOLLOW-UP ACTIONS

NOTE

Indications of loss of HPI suction are erratic flow, <u>and</u> erratic discharge pressure <u>and</u> control valves stable.

3.1 <u>If</u> HPI pump has lost suction, stop the HPI pump.

3.2 Isolate letdown by closing either:

- Letdown Coolers Outlet (CV-1221),

  <u>or</u>
- Letdown Cooler Outlets (RCS) (CV-1214 <u>and</u> CV-1216).

NOTE

With HPI pump off, ICW cooling of RCP seals should provide adequate time to isolate and correct leaks, providing no pre-condition exists, such as excessive RCP shaft sleeve leakage. HPI can provide necessary makeup for normal operations or plant shutdown.

3.3 <u>If</u> HPI pump is stopped, verify RC pump seals are being cooled by ICW.

3.3.1 <u>If</u> ICW to RCP seals is <u>not</u> available, perform Reactor Coolant Pump and Motor Emergency (1203.031), "Simultaneous Loss of Seal Injection and Seal Cooling Flow" section.

SECTION 2 -- LARGE MAKEUP AND PURIFICATION SYSTEM LEAK (continued)

NOTE

- With HPI pump off, dropping MU tank level indicates leak is in MU tank or downstream of MU tank.
- Until leak is identified, discretion should be used in placing MU and purification system back in service.

3.4 If leak is suspected downstream of HPI pump
or
MU tank level cannot be maintained >55",
stop the HPI pump.

3.4.1 Verify RC pump seals are being cooled by ICW.

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

3.4.2 If MU tank continues to drop, close Makeup Tank Outlet (CV-1275).

3.5 Attempt to locate leak as follows:

WARNING

Airborne radioactivity hazards may exist in vicinity of leakage site.

3.5.1 Notify HP and dispatch personnel with respiratory protection to visually inspect system for leakage in auxiliary building.

3.5.2 Review area monitors and process monitors for indication of leakage location.

3.5.3 Monitor AUX building sump level and Dirty Waste Drain Tank (T-20A and T-20B) levels.

3.6 If HPI pump is stopped, perform the following:

3.6.1 If both the OP and STBY HPI pumps are unavailable,
and
leak is isolated,
dispatch and operator to re-align the ES HPI pump per Attachment A of this procedure.

3.6.2 Place the following valves in HAND and close:

- RC Pumps Total INJ Flow (CV-1207)
- Pressurizer Level Control (CV-1235)

3.6.3 Verify RCP Seal Injection Block (CV-1206) closes.

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

QID: 0033 **Rev:** 1 **Rev Date:** 3/17/2005 **Source:** Direct **Originator:** JCork

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

Section: 4.2 **Type:** Generic APEs

System Number: 025 **System Title:** Loss of Residual Heat Removal System (RHRS)

Description: Knowledge of annunciator response procedures.

K/A Number: 2.4.10 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2

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

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

Given:
- RCS is in reduced inventory for nozzle dam installation.
- Annunciator K09-D8, DECAY HEAT VORTEX WARNING, is in alarm.
- DH flow is oscillating from 1000 gpm to 3000 gpm.

Which of the following is one of the initial follow-up actions performed for loss of DH Removal due to vortexing?

a. Close at least one DH suction valve from the RCS.

b. Start the other DH pump to makeup to RCS from BWST.

c. Initiate containment closure per Att. G of 1203.028.

d. Stabilize flow by throttling one of the discharge flowpath valves.

Answer:

d. Stabilize flow by throttling one of the discharge flowpath valves.

Notes:

"D" is correct per 1203.028, section 3.
"A" is an immediate action for sections 1 and 2.
"B" and "C" are follow-up actions only.

References:

1203.028, Loss of Decay Heat Removal, Rev. 016-03-0, page 20

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam.

Location: C14

Device and Setpoint: Decay Heat Pump motor current outside the operator-adjustable setpoint for SPDS point KP34ALO, KP34AHI, KP34BLO, or KP34BHI.

DECAY HEAT
VORTEX
WARNING

Alarm: K09-D8

1.0 OPERATOR ACTIONS

1. Verify adequate decay heat removal system performance.

2. IF decay heat removal system performance is erratic,
THEN refer to Loss of Decay Heat Removal (1203.028).

NOTE

- The Decay Heat Pump motor current alarms should be set so that the alarm will alert the operator to an unexpected change in motor current.
- The low current alarm (KP34ALO and KP34BLO) for the idle decay heat pump must be set at 0 to clear the alarm.
- SPDS points I1A305 and I1A405 provide P-34A and P-34B motor currents respectively.

3. IF decay heat removal system performance is satisfactory,
THEN adjust setpoint of KP34ALO, KP34AHI, KP34BLO, KP34BHI using SPDS computer U5 function to values just below and above the current value of Decay Heat Pump motor current.

   A. Make adjustments to both SPDS computers.

4. IF Decay Heat Removal pump is idle,
THEN set the appropriate low current alarm (KP34ALO or KP34BLO) at 0 using the SPDS U5 function.

   A. Make adjustments to both SPDS computers.

2.0 PROBABLE CAUSES

NOTE

This annunciator has multiple input without reflash.

1. Change in decay heat system flow.

3.0 REFERENCES

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

SECTION 3 -- LOSS OF DH FLOW DUE TO VORTEXING

INSTRUCTIONS

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

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

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

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

4. **IF loss of inventory is indicated by rising level in any of the following,**
   THEN GO TO "Loss of Inventory" section of this procedure:

   - RB sump
   - Aux. building sump
   - Aux. Building Equipment Drain Tank (T-11)
   - Dirty Waste Drain Tank (T-20A/B)

NOTE

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

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

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

7. **IF maintenance activities in the Reactor Building could be affected by RCS level rise,**
   THEN perform local evacuation of the affected areas.

8. **IF RCS temp exceeds 280°F,**
   THEN GO TO applicable "Loss of Both DH Systems" section of this procedure.

(continued)

SECTION 3 -- LOSS OF DH FLOW DUE TO VORTEXING

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.

9. <u>IF</u> either of the following conditions occurs, <u>THEN</u> initiate containment closure per Attachment G of this procedure, while continuing with this section.

- Time remaining to steam release is or becomes <1 hour <u>AND</u> DH removal cannot be immediately restored.
- RCS press >150 psig.

10. Using one of the RCS makeup methods listed on Attachment H of this procedure, establish AND maintain RCS hot leg level per one of the following limits:

- ≥375'
- Within "Operating Region" of 1104.004, Attachment B

CAUTION

- If RCS loops are not filled and DH pump is started with high flow, severe water hammer can damage DH system pipe.
- If RCS press is >150 psig and DH pump is dead-headed, DH system design press can be exceeded.
- Venting can cause airborne activity or an oxygen deficient atmosphere.

11. <u>WHEN</u> RCS level is ≥375' <u>OR</u> within the "Operating Region" of 1104.004, Attachment B, <u>THEN</u> restore DH Removal flow as follows:

A. <u>IF</u> RCS press >150 psig, <u>THEN</u> perform the following:

1) Verify containment closure initiated per Attachment G of this procedure

2) Cycle the ERV as necessary to maintain RCS press ≤150 psig.

3) <u>IF</u> RCS press can <u>NOT</u> be reduced to ≤150 psig, <u>THEN</u> GO TO applicable "Loss of Both DH Systems" section of this procedure.

(continued)

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

QID: 0282 **Rev:** 0 **Rev Date:** 9-3-99 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-AO-MSSS **Objective:** 8 **Point Value:** 1

Section: 4.2 **Type:** Generic APE

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

Description: Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: Valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition.

K/A Number: AA2.03 **CFR Reference:** 41.5, 41.10 / 45.6 / 45.13

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

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

Given:

- Unit 1 is at 100% power.
- ICW pumps P-33A and P-33B are in service.

Subsequently the ICW pump supplying the Non-Nuclear ICW loop trips. Which of the following actions should you verify as the proper system response to the above conditions?

a. P-33B and P-33C running
P-33A to P-33B suction and discharge crosstie valves open
P-33B to P-33C suction and discharge crosstie valves closed

b. P-33A and P-33C running
P-33A to P-33B suction and discharge crosstie valves closed
P-33B to P-33C suction and discharge crosstie valves open

c. P-33B and P-33C running
P-33A to P-33B suction and discharge crosstie valves open
P-33B to P-33C suction and discharge crosstie valves open

d. P-33A and P-33C running
P-33A to P-33B suction and discharge crosstie valves closed
P-33B to P-33C suction and discharge crosstie valves closed

Answer:

a. P-33B and P-33C running
P-33A to P-33B suction and discharge crosstie valves open
P-33B to P-33C suction and discharge crosstie valves closed

Notes:

P-33A supplies the non-nuclear ICW loop. P-33C will start when P-33A trips and when it is running the suction and discharge valves will align so that P-33B is supplying the non-nuclear ICW loop and P-33C is supplying the nuclear ICW loop. Therefore, "b", "c", and "d" are incorrect.

References:

1203.012K, Rev. 032-06-0

History
Developed for 1999 exam
Selected for 2005 exam

Location: C19

Device and Setpoint: see page 2 of 2.

ICW PUMP AUTO START

Alarm: K12-A4

1.0 OPERATOR ACTIONS

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

2. If P-33A or P-33C auto started on low P-33B discharge pressure, perform the following:

   A. Verify ICW Pumps Discharge Crossconnects (CV-2238 and CV-2239) are closed.

   B. Verify ICW Pumps Suction Crossconnects (CV-2240 and CV-2241) are closed.

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

3. If P-33A or P-33B auto started on low P-33C discharge pressure, perform the following:

   A. Verify ICW Pumps Discharge Crossconnect (CV-2238) is closed.

   B. Verify ICW Pumps Discharge Crossconnect (CV-2239) is open.

   C. Verify ICW Pumps Suction Crossconnect (CV-2240) is closed.

   D. Verify ICW Pumps Suction Crossconnect (CV-2241) is open.

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

4. If P-33B or P-33C auto started on low P-33A discharge pressure, perform the following:

   A. Verify ICW Pumps Discharge Crossconnect (CV-2238) is open.

   B. Verify ICW Pumps Discharge Crossconnect (CV-2239) is closed.

   C. Verify ICW Pumps Suction Crossconnect (CV-2240) is open.

   D. Verify ICW Pumps Suction Crossconnect (CV-2241) is closed.

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

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

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

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

QID: 0155 **Rev:** 4 **Rev Date:** 05/13/93 **Source:** Direct **Originator:** K. Canitz

TUOI: ANO-1-LP-RO-NNI **Objective:** 13 **Point Value:** 1

Section: 4.2 **Type:** Generic AOPs

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

Description: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Sensors and detectors.

K/A Number: AK2.02 **CFR Reference:** 41.7 / 45.7

Tier: 1 **RO Imp:** 2.4 **RO Select:** Yes **Difficulty:** 4

Group: 1 **SRO Imp:** 2.6 **SRO Select:** No **Taxonomy:** An

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

The plant is steady at 100% power.
Loop 'A' RCS narrow range pressure instrument, PT-1021, fails high instantaneously.

What effect will this failure have on plant equipment?

a. The ERV will open.

b. RPS channel A will trip.

c. Pressurizer heaters will go off.

d. PZR spray valve will open.

Answer:

b. RPS channel A will trip.

Notes:

If PT-1021 fails high instantaneously, then SASS will select the other instrument. Therefore the only effect will be to trip the A RPS channel which is processed "upstream" of SASS. All the other choices would occur if PT-1021 was "hard" selected via the handswitch on C04.

References:

STM 1-63
STM 1-69

History:

Modified from Exam Bank QID # 3623
Used in A. Morris 98 RO Re-exam
Selected for use in 2005 RO exam.

1.5 Non-Nuclear Instrumentation

The non-nuclear instrumentation used by the reactor protection system utilizes differential pressure detectors for reactor coolant flow, resistance temperature detectors (RTD) to measure RCS hot leg temperature (Th), pressure transmitters for reactor coolant system pressure, reactor building pressure, and pressure switches for main feedwater pump and main turbine status. The reactor coolant pump breakers are also monitored as discussed in above section which covered flux to flow and flux to pump trips. For additional information on non-nuclear instrumentation refer to STM 1-69.

1.5.1 NNI RPS Trip & Setpoints

The protective functions described below apply to each of the four protection channels. The trip settings for protective system instrumentation are listed in the Unit One Technical Specifications table 2.3-1 and in the current operating cycle COLR. The safety analysis has been based upon these protection system instrumentation trip setpoints plus calibration and instrumentation errors. To ensure Tech Spec limits are not exceeded, the allowable band for setpoint adjustment is on the conservative side of setpoint and margin.

Refer to Technical Specifications and the COLR for additional information. Trip setpoints are also contained in the Emergency Operating Procedure 1202.01 Entry Conditions.

Reactor Coolant Pressure High & Low

(Refer to Figure 63.04B and 63.06)

Each of the four protective channels receive a signal from separate narrow range reactor coolant system pressure transmitters. Each pressure transmitter is supplied with its own power supply, mounted in the associated protective channel cabinet. The output of each pressure transmitter is applied to the input of a isolation module. This buffer amplifier acts as a signal conditioner where the signal is converted to a 0 to 10 volts dc output which corresponds to the pressure range of 1700 to 2500 psig. The 0 to 10 volt dc analog pressure signal is in turn applied to the high pressure, low pressure, and pressure-temperature trip bistables. In addition, it is applied to the high pressure trip bistable associated with the shutdown by-pass circuitry. When RCS pressure exceeds the trip setpoint for the high or low pressure bistable it trips de-energizing the channel trip relay.

During a slow reactivity addition insertion rate startup accident or a slow reactivity addition insertion rate from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 63.05 for high reactor coolant system pressure of **2355 psig** has been established to maintain the system pressure below the safety limit of **2750 psig** for any design transient. During plant shutdown conditions the RPS channels are placed in a shutdown bypass condition which allows for CRD testing, zero power physics testing and to allow for plant startup. When placed in shutdown bypass a high pressure trip setpoint of **1720 psig** is imposed to provide overpressure protection to the RCS.

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.

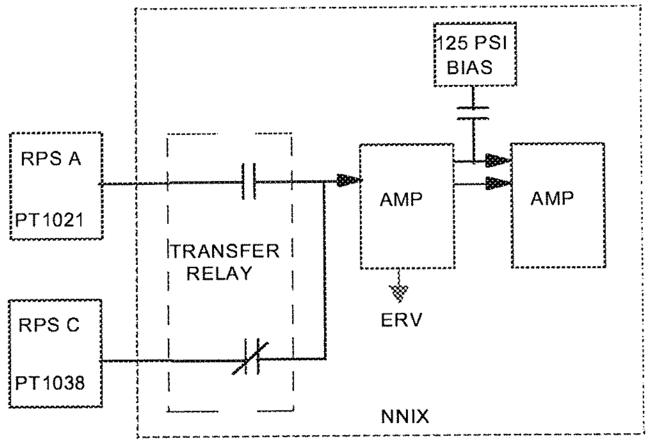
Pressure transmitters PT-1020, PT-1022, and PT-1040 provide input to A, B, and C ESAS analog channels, respectively. The pressure transmitters that supply ESAS are Rosemount differential capacitance detectors. ESAS analog channel A supplies indication on C-166 (Dasey Panel). The range of the indication 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. With the handswitch in the AUTO position, the signal monitor opens and closes the ERV. When either handswitch is placed in the OPEN position, the ERV solenoid is energized and ERV is opened.

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

QID: 0550 **Rev:** 0 **Rev Date:** 3-30-05 **Source:** New **Originator:** S.Pullin

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to operate and monitor the following as they apply to a SGTR: Isolation of a ruptured S/G.

K/A Number: EA1.32 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 4.6 **RO Select:** Yes **Difficulty:** 3

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

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

As CBOR you are monitoring parameters following a Reactor Trip with a Steam Generator Tube Rupture.

Of the criteria listed below, which requires isolation of the bad SG?

A. T-hot 490°F, BWST level 24 ft.

B. T-hot 480°F, offsite dose projection meets NUE criteria

C. T-hot 500°F, BWST level 23 ft.

D. T-hot 485°F, offsite dose projection meets Alert criteria

Answer:

D. T-hot 485°F, offsite dose projection meets Alert criteria

Notes:

"D" is correct since T-hot is less than 490°F and meets one of the three criteria listed in step 53.
All others do not meet the criteria of step 53.

References:

1202.006, Chg. 007-04-0, step 53

History:

New for 2005 RO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS

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

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

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

Contingency actions:

A. Perform the following:

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

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

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

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

(3) **GO TO step 53.A.2).**

b) **IF** ERV **cannot** be opened,
THEN adjust HPI as necessary to maintain RCS press/temp within limits of Figure 3.

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

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

(53. CONTINUED ON NEXT PAGE)

INSTRUCTIONS | **CONTINGENCY ACTIONS**

53. (Continued).

2) Verify bad SG Main Feedwater Isolation valve closed:

SG A	SG B
CV-2680	CV-2630

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

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

4) **IF** RCS press is >950 psig, **THEN** reduce RCS press to ≤950 psig, while maintaining adequate SCM by any or all of the following:

a) Maintain emergency cooldown rate of ≤240°F/hr to 500°F.

b) Raise AUX Pressurizer Spray flow.

c) Maximize Letdown flow.

d) Throttle HPI.

e) Open High Point Vents:

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

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

(53. CONTINUED ON NEXT PAGE)

INSTRUCTIONS	CONTINGENCY ACTIONS

53. (Continued).

5) Verify bad SG ATM Dump ISOL open
AND
ATM Dump CNTRL valve in AUTO **AND** closed:

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

a) IF ATM Dump CNTRL is open
AND
SG press is <1020 psig,
THEN verify associated ATM Dump ISOL in MANUAL **AND** close.

6) Close bad SG MSIV:

SG A	SG B
CV-2691	CV-2692

a) IF both SGs are isolated,
THEN GO TO 1202.011, "HPI COOLDOWN" procedure.

54. Before RCS temp drops to 390°F, verify no more than 3 RCPs running.

55. WHEN RCS temp drops to 300°F, THEN reduce RCS cooldown rate to ≤50°F/hr.

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

QID: 0551 **Rev:** 0 **Rev Date:** 3-30-05 **Source:** New **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:** No **Taxonomy:** C

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

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 cooldown rate 110°F/hr with Tcold 360°F

C. RCS cooldown rate 75°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. 004-03-0

History:

New for 2005 RO exam.

Page 1 of 2

NOTE

- PTS limits apply if <u>any</u> 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 <u>low</u> 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.

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).

D. **IF** RCS press is high, **THEN** limit press using one or more of the following:

1) Throttle makeup flow.

2) **IF** HPI is in service **AND** SCM is adequate **AND** any of the following conditions is met,
THEN throttle HPI flow:

- HPI Cooling (RT 4) **not** in progress
- CET temps dropping
- RCS press rising with ERV open

a) **IF** ESAS has actuated **AND** HPI must be throttled,
THEN override **AND** throttle HPI.

3) **IF** RCP is running, **THEN** operate Pressurizer Spray (CV-1008) in HAND.

4) **IF** PZR AUX Spray is in service, **THEN** adjust Pressurizer AUX Spray (CV-1416).

5) Place Pressurizer Heaters in OFF.

(14. CONTINUED ON NEXT PAGE)

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

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

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

Section: 4.2 **Type:** Generic EPEs

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

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

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

Tier:	1	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	K

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

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

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

B. All 4160V busses de-energized

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

D. All 6900V busses de-energized

Answer:

B. All 4160V busses de-energized

Notes:

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

References:

1202.008, chg. 007-01-0

History:

New for 2005 RO exam

ENTRY CONDITIONS

NOTE

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

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

- All 4160V buses de-energized

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

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

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the operational implications of the following concepts as they apply to the Loss of Off-site Power: Principles of cooling by natural convection.

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

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

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

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

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

1202.012, chg. 004-03-0, RT-6 step J
1202.013, rev. 4, Fig. 2

History:

New for 2005 RO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
2. Verify SW to DG1 and DG2 CLRs open to operating EDGs (CV-3806 and 3807).	
3. Verify <u>OR</u> start a Service Water pump on <u>each</u> operating DG, after 15-second time delay (P4A, B, C).	3. <u>IF</u> both EDGs are operating **<u>AND</u>** only one Service Water pump can be started **<u>AND</u>** ESAS has <u>not</u> actuated, **<u>THEN</u>** perform the following: A. Close ACW Isolation (CV-3643). B. Verify both Service Water to ICW Coolers Supply valves open (CV-3811 and 3820).
4. Actuate MSLI for both SGs <u>AND</u> verify proper actuation and control of EFW and MSLI (RT 6): A. Operate ATM Dump CNTRL valves in HAND to minimize cycling and conserve Instrument Air. B. <u>IF</u> Instrument Air to ATM Dump CNTRL valves is lost, **<u>THEN</u>** perform the following: 1) Dispatch an operator with a radio to place ATM Dump CNTRL valves on hand jack **<u>AND</u>** fully open (Refer to Alternate Shutdown (1203.002), Exhibit A) 2) Establish SG press control using ATM Dump ISOL valves in MANUAL from the Control Room.	4. <u>IF</u> all EFW is lost, **<u>THEN</u> GO TO step 53.**
5. Check RCS press remains ≥1700 psig <u>AND</u> PZR level remains ≥30".	5. Initiate HPI (RT 2).

6. (Continued).

J. <u>**IF**</u> all RCPs are off,
<u>**THEN**</u> check primary to secondary heat transfer in progress indicated by all of the following:

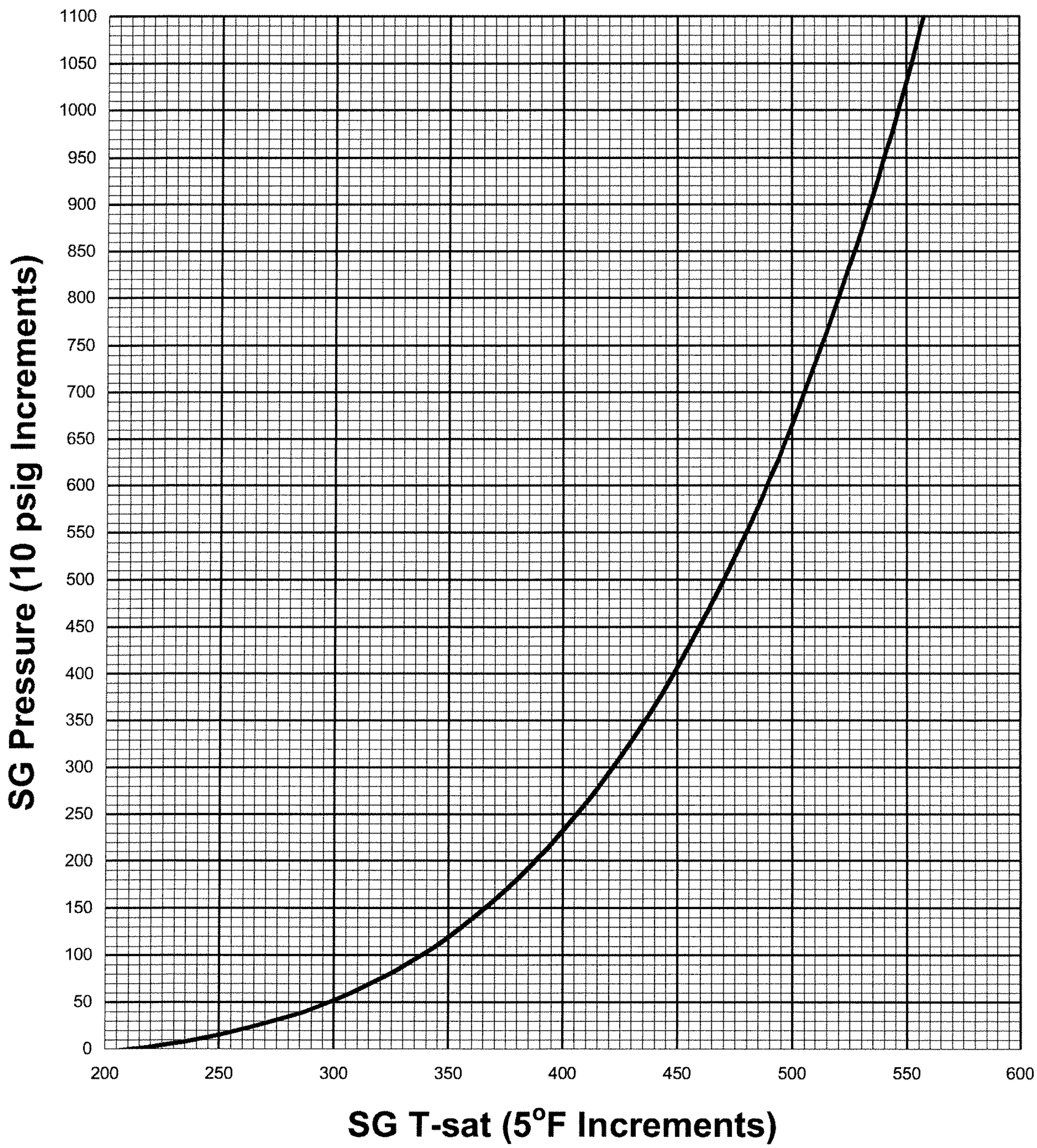
- T-cold tracking associated SG T-sat (Fig. 2)
- T-hot tracking CET temps
- T-hot/T-cold ΔT stable or dropping

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

<u>**Table 2**</u>
Examples of Less Than Adequate EFW Flow Indications
• SG level < 20" <u>**AND**</u> no EFW flow indicated • All RCPs off <u>**AND**</u> SG level <u>**not**</u> tracking EFIC calculated setpoint • All RCPs off <u>**AND**</u> EFIC level setpoint <u>**not**</u> trending toward applicable level band
Examples of Excessive EFW Flow Indications
• SG press drops ≥ 100 psig due to EFW flow induced overcooling • SCM approaching minimum adequate due to EFW flow induced overcooling • EFW CNTRL valve open with associated SG level > applicable setpoint level band

END

FIGURE 2
SG Pressure vs T-sat



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

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

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

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

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

Description: Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual inverter swapping.

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

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

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

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

a. power RS-2 from Y-25.

b. de-energize RS-4.

c. parallel RS-2 and RS-4.

d. damage the Y-25 inverter.

Answer:

b. de-energize RS-4.

Notes:

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

References:

STM 1-32, Rev. 26

History:

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

Power distribution to and from the inverters is as follows:

INVERTER	AC INPUT	DC INPUT	OUTPUT TO
Y-11	51	D01	RS-1
Y-13	53	D01	RS-3
Y-15	57	D01	RS1/RS3
Y-22	65	D02	RS-2
Y-24	61	D02	RS-4
Y-25	63	D01	RS2/RS4
Y-41	56	D41	RC-1
Y-28	61	D02	C-540

4.3.1.2 Simplified Inverter Operation

(SEE FIGURE 32.54)

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

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

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

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

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

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

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

QID: 0336 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** J Simmons

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

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

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

Description: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of DC power.

K/A Number: AK3.02 **CFR Reference:** 41.5,41.10 / 45.6 / 45.1

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

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

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

Given:

- Turbine Lockout Relay DC Failure (K04-B5),
- D01 Undervoltage (K01-A7),
- D01 Trouble (K01-D7),
- Loss of breaker position indicator lights for plant buses on left side of C10.

Which action should be performed?

a. Start both Diesel Generators from C-10.

b. Trip the Generator Output Breakers.

c. Transfer D11 to its Emergency Power Supply.

d. Line up Battery Charger D03A or D03B to the D01 Bus.

Answer:

c. Transfer D11 to its Emergency Power Supply.

Notes:

The most expedient method of restoring power to D11 and the action prescribed in 1203.036 for Loss of D01 is found in answer "c".
"a" is incorrect, although some might choose this since control power could be construed to be lost to DG undervoltage relays.
"b" is incorrect, the output breakers may or may not trip, although some might confuse this with the logic sequence within 1203.036 if transfer of D11 to D02 is UNsuccessful.
"d" is incorrect, this would take too much time and may not work if all of "A" train power is lost .

References:

1203.036, chg. 005-04-0

History:

Used in 1999 exam
Direct from ExamBank, QID# 1891
Selected for 2005 RO exam

DISCUSSION (In back of procedure)

SECTION 1 -- **Loss of D01**

1.0 SYMPTOMS

1.1 Low DC voltage alarms:
(alarms inoperable if generator output breakers open).

- D01 UNDERVOLTAGE (K01-A7)
- D11 LOSS OF VOLTAGE (K01-B7)
- RA1 LOSS OF VOLTAGE (K01-C7)
- D01 TROUBLE (K01-D7)

- H1 DC CONTROL POWER OFF (K02-B4)
- A1 DC CONTROL POWER OFF (K02-C6)
- A3 DC CONTROL POWER OFF (K02-D6)
- SU 1 L.O. RELAY DC FAILURE (K02-D1)

- SU 2 L.O. RELAY DC FAILURE (K02-E3)
- GENERATOR L.O. RELAY DC FAILURE (K04-D8)
- TURBINE L.O. RELAY DC FAILURE (K04-B5)
- EOS SYSTEM TROUBLE (K04-C5)

1.2 Loss of breaker position indicator lights for plant buses on left side of C10.

1.3 "Trip Solenoid Power Available" light on C01 **not** lit.

2.0 IMMEDIATE ACTION

NONE

3.0 FOLLOW-UP ACTIONS

3.1 At C10, transfer D11 to EMERG SUPPLY D02.

3.2 **IF** reactor trips, **THEN** perform the following:

3.2.1 **IF** SG pressure is <900 psig,
THEN actuate MSLI and EFW for both SGs.

3.2.2 Perform Emergency Operating Procedures (1202.XXX) in conjunction with this procedure.

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

3.4 **IF** transfer of D11 is **not** successful,
THEN attempt local transfer of D11 to D02, while continuing.

3.5 **IF** reactor is **not** tripped, **THEN GO TO step 7.0.**

3.6 **IF** transfer of D11 is successful, **THEN GO TO step 4.0.**

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

QID: 0554 **Rev:** 0 **Rev Date:** 3/30/05 **Source:** Direct **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.

K/A Number: 2.4.9 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	K

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

Plant heat up is in progress with RCS temperature at 215 degrees F.
Service water is lost to the in-service DH cooler.

What action is required during the re-establishment of SW flow through the DH cooler and why?

A. Establish SW slowly to prevent DH cooler water hammer.

B. Establish SW slowly to prevent SW pump runout.

C. Establish SW quickly to prevent DH cooler thermal shock.

D. Establish SW quickly to prevent RCS heat up.

Answer:

A. Establish SW slowly to prevent DH cooler water hammer.

Notes:

"A" is the correct answer per 1203.028 step 10, the others are concerns but are shown incorrectly.

References:

1203.028, chg. 016-03-0

History:

Direct from regular exam bank QID #ANO-OPS1-2788
Selected for 2005 RO exam

SECTION 4 -- LOSS OF SERVICE WATER FLOW

INSTRUCTIONS

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

2. **IF maintenance activities in the Reactor Building could be affected by RCS level rise, THEN perform local evacuation of the affected areas.**

{2}

3. **IF fuel loading is in progress, THEN terminate movement into core until DH cooling is returned to normal.**

4. **IF SW is available to idle DH cooler, THEN place respective DH loop in service for DH removal per applicable steps of Decay Heat Removal Operating Procedure (1104.004), "Decay Heat Removal During Cooldown" section.**

5. **Continue DH flow to provide mixing and minimize heatup rate of RCS.**

   A. IF RCS temp exceeds 280°F, 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.

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 Time remaining to steam release is OR becomes <1 hour AND DH removal can NOT be immediately restored, THEN initiate containment closure per Attachment G of this procedure, while continuing with this section.**

(continued)

SECTION 4 -- LOSS OF SERVICE WATER FLOW

7. **IF RCS press approaches Decay Heat Sys. Max Pressure limit of Plant Shutdown and Cooldown (1102.010), THEN perform the following:**

   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.

8. **Investigate cause of loss of heat sink.**

CAUTION

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

9. **IF RCS >200°F AND there is NO SW flow through DH cooler, THEN perform the following:**

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

Cooler	Valve	Breaker
E-35A	CV-3822	B-5182
E-35B	CV-3821	B-6183

(continued)

SECTION 4 -- LOSS OF SERVICE WATER FLOW

10. WHEN SW is regained,
THEN restore SW flow to DH cooler as follows:

A. Station operator at cooler to listen for evidence of water hammer during next step.

> **NOTE**
> If SW side of DH cooler has reached saturation temp, slowly cutting in SW will minimize thermal shock and water hammer.

B. Slowly open applicable SW Inlet to E-35A or E-35B DH Cooler manually:

E-35A	E-35B
CV-3822	CV-3821

1) **IF** water hammer is observed,
THEN open SW Inlet more slowly.

C. **WHEN** SW valve is fully open,
THEN close associated supply breaker:

CV-3822	CV-3821
B-5182	B-6183

END

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

QID: 0555 **Rev:** 0 **Rev Date:** 4/4/05 **Source:** New **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing.

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

Tier:	1	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	No	**Taxonomy:**	K

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

A Loss of Instrument Air has been in progress when Instrument Air pressure degrades to 58 psig.

Which of the following actions should be taken?

A. Trip the reactor and go to 1202.001, Reactor Trip.

B. Shutdown at greater than or equal to 10%/min. per 1203.045, Rapid Plant Shutdown.

C. Shutdown at greater than or equal to 3%/min. per 1203.045, Rapid Plant Shutdown.

D. Shutdown at approximately 30%/hr. per 1102.016, Power Reduction and Plant Shutdown.

Answer:

B. Shutdown at greater than or equal to 10%/min. per 1203.045, Rapid Plant Shutdown.

Notes:

"B" contains the proper rate and procedure per 1203.024.
"A" would be correct if IA pressure dropped to 35 psig but is incorrect for the given pressure.
"C" is incorrect, the procedure is the proper procedure but the rate is too low.
"D" is incorrect, a plant shutdown is warranted but not using the normal plant shutdown procedure.

References:

1203.024, chg. 010-08-0, section 2, step 3.1

History:

New for 2005 RO exam

SECTION 2 -- LOW-LOW INSTRUMENT AIR PRESSURE (≤60 PSIG)

DISCUSSION

The air pressure in the IA line(s) nearest the leak will be lower than the indicated air header pressure. For this reason, components nearest the leak will exhibit abnormal indication and abnormal control characteristics first. The longer air pressure remains degraded, the greater the number of malfunctions.

If IA pressure continues to degrade, power reduction at the maximum rate is specified to minimize plant impact. Close monitoring of air operated systems and components is required to minimize transients resulting from failures. For additional discussion, see Attachment D.

1.0 SYMPTOMS

1.1 IA header pressure ≤60 psig.

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

3.1 Commence plant shutdown at ≥10% per minute per Rapid Plant Shutdown (1203.045) while continuing with this procedure.

3.1.1 IF instrument air pressure recovers (>60 psig), THEN stop plant shutdown.

3.2 IF IA header pressure drops below 35 psig,
OR any system degrades to the extent that in the judgment of the operator requires reactor trip,
THEN **GO TO** Section 3, "Loss of Instrument Air Pressure (≤35 PSIG)".

3.3 Place RCP Seal INJ Block (CV-1206) pushbutton in OVRD (OVRD light on).

NOTE

- Using HPI Block Valves CV-1220 or CV-1285 will minimize nozzle stress cycles because of the normal makeup path.
- Pressurizer Makeup Flow Control Valve (CV-1235) will fail as is, when IA pressure drops to ~45 psig.

3.4 IF necessary to maintain pressurizer level ≥100",
THEN use HPI block valves to provide makeup to RCS, or take manual control of CV-1235 as follows:

3.4.1 Align stem and gear holes.

3.4.2 Install lock pin.

3.4.3 Open equalizing valve.

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

QID: 0556 **Rev:** 0 **Rev Date:** 8/2/00 **Source:** Direct **Originator:** E.Jacks

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

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

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

Description: Ability to determine and interpret the following as they apply to the (Inadequate Heat Transfer): Operating behavior characteristics of the facility.

K/A Number: EA1.2 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

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

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

Given:
- All MFW and EFW has been lost.
- HPI cooling is in progress per RT-4.

Why is the ERV manually cycled vs. allowing the ERV to cycle automatically?

A. To reduce RCS pressure and reduce flow out the ERV.

B. To maximize the number of times the ERV is cycled.

C. To increase HPI flow while preventing ESAS and maintaining SCM.

D. To prevent the code safeties from lifting.

Answer:

C. To increase HPI flow while preventing ESAS and maintaining SCM.

Notes:

"C" is correct per the BWOG Bases document and ANO's deviation document.
"A" is incorrect, while this will reduce RCS pressure, flow out the ERV will not be reduced.
"B" is incorrect, this will minimize the number of times ERV is cycled.
"D" is incorrect, although the ERV's auto setpoint will normally accomplish this.

References:

1202.012, chg. 004-03-0, RT-4
Tech Document, Vol. 3, III.C-12

History:

Direct from regular exam bank QID #2790
Selected for 2005 RO exam

4. Initiate HPI cooling:

A. **IF** RCP Seal Injection is in service,
THEN place RCP Seal INJ Block (CV-1206) in OVRD. Otherwise, verify CV-1206 closed.

B. Open both BWST Outlets (CV-1407 and 1408).

C. Verify ERV Isolation (CV-1000) open.

D. **IF** OP or STBY HPI pump is running, **THEN** perform the following:

1) **WHEN** associated BWST Outlet is open,
THEN fully open all associated HPI Block valves.

E. Verify **one** of the following:

- **Both** HPI RECIRC valves open (CV-1300 and 1301)
- Open HPI Block valve(s) as follows to prevent dead-heading pump
  - **Fully** open **one** HPI Block valve associated with ES HPI pump (CV-1220 or 1285) to prevent dead-heading pump
  - **IF** OP and STBY HPI pumps are both off,
    THEN **fully** open **one** HPI Block valve associated with OP or STBY HPI pump (CV-1220 or 1285).

F. Place ES HPI pump in service as follows:

1) Start AUX Lube Oil pump for ES HPI pump.

2) **WHEN** BWST Outlet is open, **THEN** start ES HPI pump.

3) Stop AUX Lube Oil pump.

4) Fully open all associated HPI Block valves.

5) **IF** ERV opens in auto, **THEN** perform **step 4.I** while continuing.

(4. CONTINUED ON NEXT PAGE)

4. (Continued).

G. **<u>IF</u>** OP and STBY HPI pumps are **<u>both</u>** **<u>off</u>**,
<u>THEN</u> place OP or STBY HPI pump in service as follows:

1) **<u>IF</u>** P36B will be used, **<u>THEN</u>** verify the following selected to energized bus:

- P36B Bus Select MOD Control
- P64B Transfer Switch

2) Start AUX Lube Oil pump for OP or STBY HPI pump.

3) **<u>WHEN</u>** BWST Outlet is open, **<u>THEN</u>** start OP or STBY HPI pump.

4) Stop AUX Lube Oil pump.

5) **<u>Fully</u>** open **<u>all</u>** associated HPI Block valves.

H. **<u>IF</u>** **<u>no</u>** HPI pumps are available, **<u>THEN</u>** notify CRS **<u>AND</u>** **GO TO step 4.L.**

I. Manually cycle ERV (PSV-1000) as follows, while continuing:

1) Open ERV.

2) **<u>WHEN</u>** **<u>either</u>** of the following criteria is met, **<u>THEN</u>** place ERV in AUTO:

- RCS press drops to 1650 psig if ES is armed
- SCM approaches minimum adequate

<u>CAUTION</u>

ERV Isolation (CV-1000) must be left open until HPI cooling is no longer required.

a) **<u>IF</u>** ERV fails open, **<u>THEN</u>** **<u>DO NOT</u>** close ERV Isolation.

3) **<u>WHEN</u>** RCS press reaches 2400 psig
<u>OR</u>
approaches NDTT Limit on Figure 3,
<u>THEN</u> repeat **step 4.I** until ERV can remain open with the following criteria met:

- RCS press >1650 psig if ES is armed
- SCM adequate

4) **<u>IF</u>** ERV is closed
<u>AND</u>
CET temp rise causes adequate CET SCM to be lost, without a drop in RCS press,
<u>THEN</u> open ERV **<u>AND</u>** leave open to maximize cooling.

J. Close HPI RECIRC (CV-1300 or 1301).

(4. CONTINUED ON NEXT PAGE)

4. (Continued).

K. **IF** only one train of HPI is available
AND
RCS press is >600 psig,
THEN throttle the HPI Block valve with the highest flow to within 20 gpm of the next highest flow.

L. Turn off all Pressurizer Heaters.

M. Trip all but one RCP.

N. Maximize RB cooling:

1) Verify all four RB Cooling Fans running:

VSF1A	VSF1C
VSF1B	VSF1D

2) Open RB Cooling Coils Service Water Inlet and Outlet valves:

CV-3812	CV-3813
CV-3814	CV-3815

3) Unlatch key-locked Chiller Bypass Dampers:

SV-7410	SV-7412
SV-7411	SV-7413

O. Isolate possible RB leak paths as follows:

1) **IF** RB Sump draining is in progress,
THEN close RB Sump Drain to AUX Sump valves (CV-4400 and 4446).

2) Stop RB Leak Detector RX-7460 Sample pump on C25.

3) Close RB Leak Detector Isolations (SV-7454 and 7456) on C26.

END

If HPI is not available at this point, actions must be taken to control RC pressure as discussed in Section 3.5.C of this Chapter.

B. Feedwater not available

For a total LOFW, HPI cooling will be required to keep the core covered and adequately cooled. There are two important aspects of HPI cooling. First, operator initiation is required. There are no automatic systems to initiate HPI cooling. Consequently, the operator needs to continually monitor conditions for when HPI cooling should be started.

Second, the HPI cooling heat removal rate will probably not initially match the decay heat rate. Therefore, it must be started early enough to slow RC inventory depletion enough so that HPI cooling will match decay heat before the core is uncovered. Consequently, for all plants except Davis Besse, the HPI addition must be started as soon as RC inventory starts being lost, i.e., when the PORV open setpoint is reached or the first automatic PORV lift, whichever occurs first.

For Davis Besse, MU/HPI cooling must be considered as soon as feedwater becomes unavailable. If only one MU pump is operable then MU/HPI cooling must be initiated immediately. If two MU pumps are available then MU/HPI cooling initiation can wait until the core outlet temperature reaches 600°F provided the RCS RV PT limit is not exceeded.

3.3.A Establish HPI Cooling - except Davis Besse

Establishment of HPI cooling is accomplished as follows:

1. Two HPI pumps are started and verified to be operating at full flow for the existing RC pressure. HPI flow must be verified before opening the PORV.

2. If HPI flow can not be established, the PORV must not be maintained open. Because the only source of decay heat removal is the inventory that exists in the RCS, that inventory must be preserved as much as possible. The PORV should be closed and manually cycled only as necessary to prevent automatic lifting of the PORV. If the PORV fails open, then the PORV block valve is closed to isolate the PORV. This will minimize the RCS inventory loss as well as maximize its energy removal capability while reducing the number of PORV cycles that would occur if it remained in automatic. Any running RCP(s) should be tripped to eliminate their energy input to the RCS and to reduce the SG tube heatup rate. In this case, HPI and/or FW must be restored to preclude core damage.

3. If flow can be achieved from only one HPI pump, the PORV must be opened and left open. This will reduce RC pressure and maximize the available flow from the degraded HPI system.

<u>NOTE</u>: Analyses have been performed that indicate the core will not uncover, and thus be adequately cooled, with only one HPI pump and without opening the PORV. However, considering the collapsed liquid level can approach the top of the core and that the HPI system is already degraded, the PORV <u>must</u> be opened to retain the maximum RCS inventory for assurance of adequate core cooling.

4. If full flow can be achieved from two HPI pumps, the PORV may be opened or allowed to open automatically when RC pressure increases to the PORV set-point, or it may be manually cycled. The preferred method is to maintain the PORV open which will reduce RC pressure and allow greater HPI flow thus providing increased core cooling. However, adequate core cooling can be achieved with two HPI pumps if the PORV is allowed to automatically open. In the event the PORV fails open during cycling, DO NOT close the PORV block valve.

   If the preferred method of maintaining the PORV open is used, then it is acceptable to use full pressurizer spray flow, if available, to reduce the rate of RCS pressurization in order to delay reaching the PORV setpoint. At some plants, manual action is required to fully open the pressurizer spray valve. If such actions are taken, then actions to close the spray valve or return control of the valve back to automatic should be ensured. The magnitude of the time delay depends on the decay heat level; the lower the decay heat level, the longer the time delay, including the possibility of preventing the PORV setpoint from being reached with low decay heat levels. Conversely, with high decay levels, use of pressurizer spray may have almost no noticeable effect in delaying reaching the PORV setpoint.

   Delaying reaching the PORV setpoint allows the pressurizer to fill more due to insurges. This may prevent loss of SCM after the PORV is maintained open.

   The reason for this is that the insurge to the pressurizer will decrease the size of the steam bubble so that when the PORV is opened, it will relieve steam for a shorter period of time and begin relieving two-phase liquid sooner. Once two-phase liquid is being relieved, the magnitude and rate of RCS depressurization is greatly reduced compared to single-phase (steam only) being relieved. By relieving two-phase liquid sooner, the magnitude of the RCS depressurization may be small enough to prevent loss of SCM.

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

QID: 0557 **Rev:** 0 **Rev Date:** 10/23/98 **Source:** Direct **Originator:** C.Alden

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Dropped Control Rod: Signal inputs to rod control system.

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

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

Group: 2 **SRO Imp:** 2.8 **SRO Select:** No **Taxonomy:** C

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

Given:
- The CRD system is in automatic and the plant is at 100% power.
- A dropped rod in Group 6 results in an asymmetric rod runback.

Group 6 rods will drive "in" when a Group 6 "in limit" is ON because:

A. In automatic, a Group 6 "out limit" bypasses the "in limit".

B. In automatic, an asymmetric rod runback bypasses the group "in limit".

C. The dropped rod's relative position is aligned with the group.

D. The "in limit" is only functional when the Diamond is in manual.

Answer:

B. In automatic, an asymmetric rod runback bypasses the group "in limit".

Notes:

"B" is correct per the CRD control logic.
The other choices represent actual CRD logic conditions but do not cause runbacks.

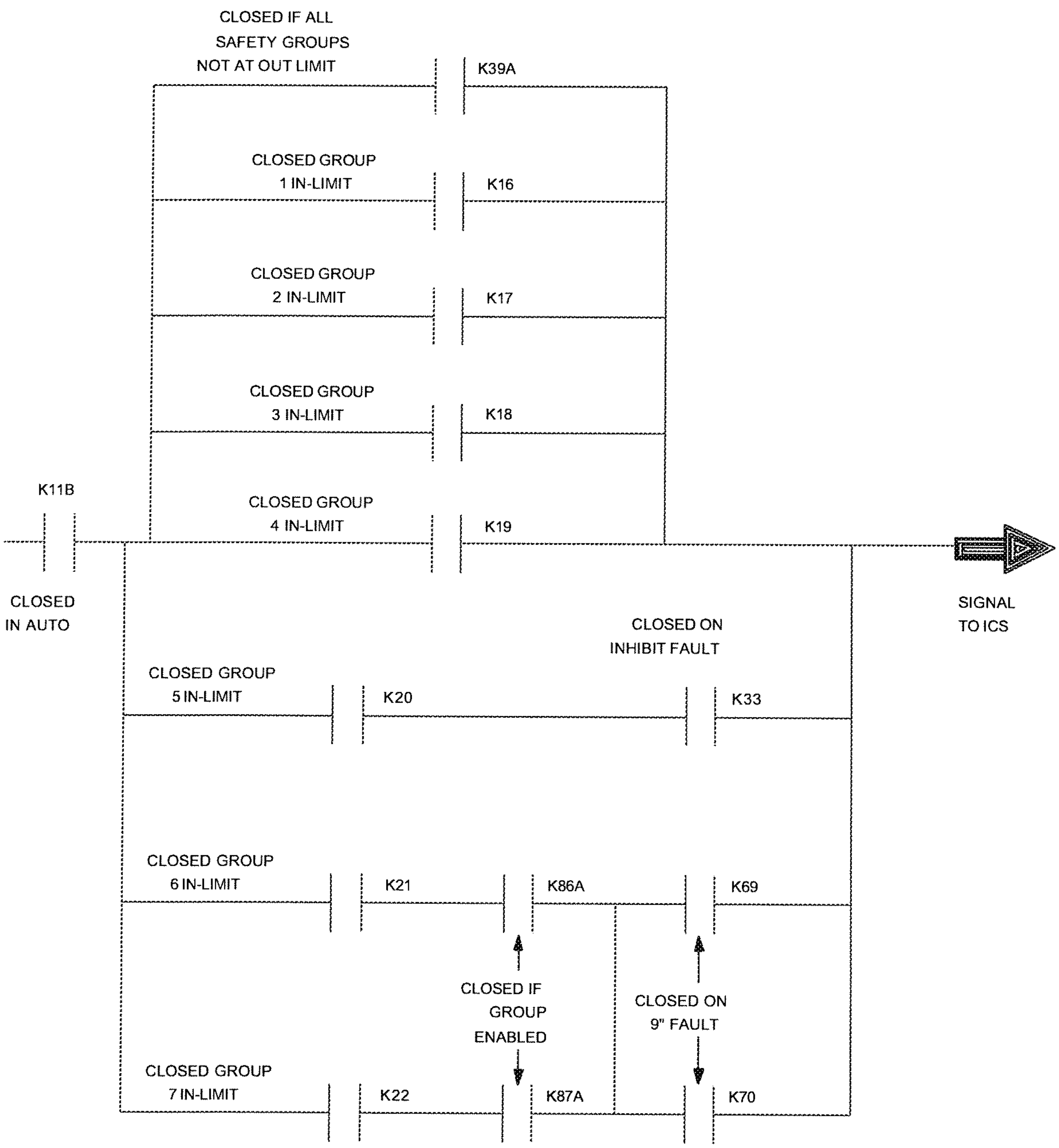
References:

STM 1-02, Rev. 5

History:

Direct from regular exam bank.
Selected for 2005 RO exam.

FIGURE 02.60: ASYMMETRIC ROD CIRCUIT



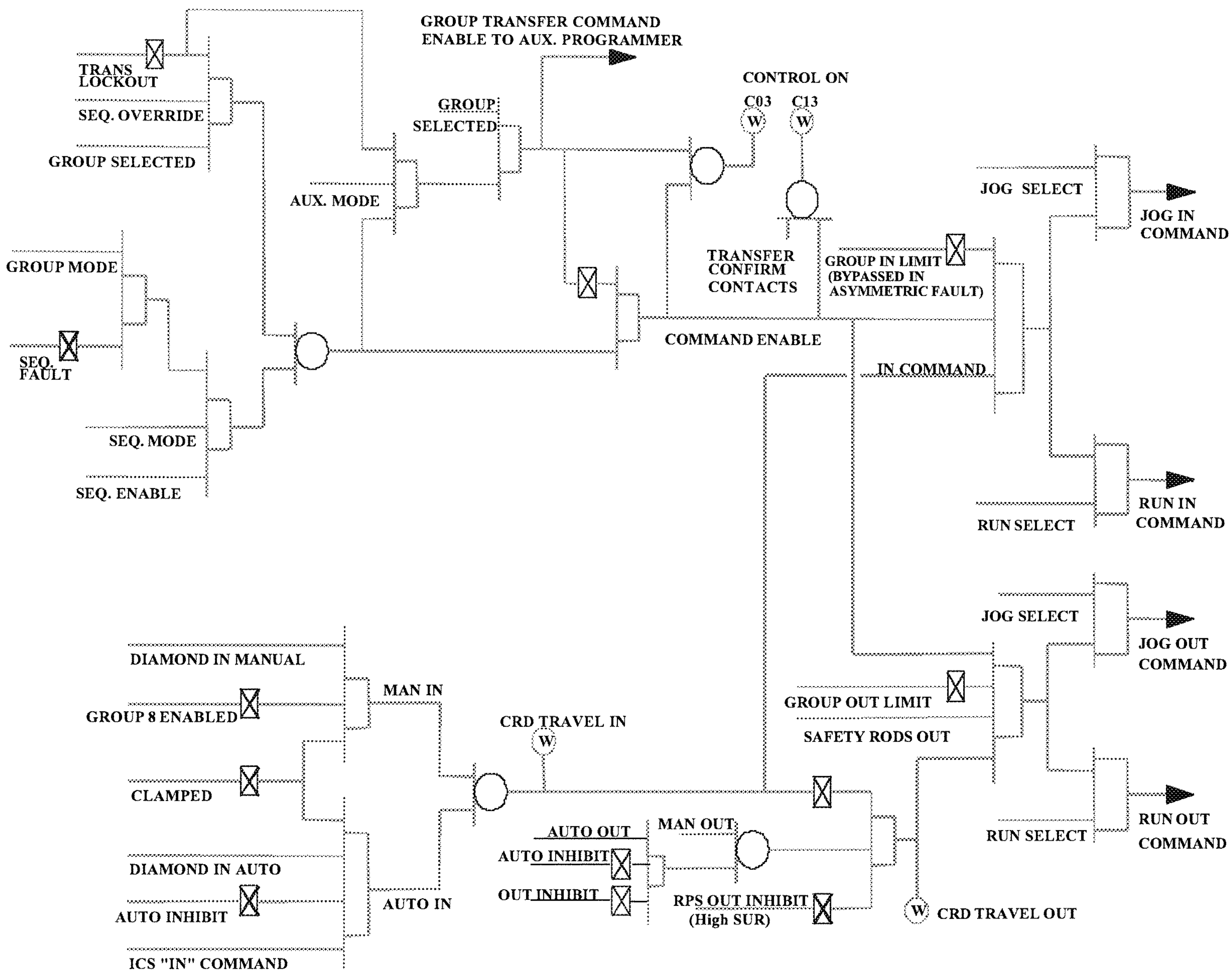


FIGURE 02.65: COMMAND LOGIC GROUP 5, 6 OR 7

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

QID: 0495 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Repeat **Originator:** NRC

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

Section: 4.2 **Type:** Generic APE

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

Description: Ability to operate and/or monitor the following as they apply to the Emergency Boration: Boric acid controlller.

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

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

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

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

Given:

- Reactor is tripped from 100% power
- Four CRDM's fail to insert according to indications in the control room
- The CRS directs you to initiate Emergency Boration in accordance with RT-12 "Emergency Boration"

You should set the INITIAL batch setting of the boric acid controller to:

A. The batch size required to maintain make up tank level between 55 and 86 inches while maintaining pressurizer level >100 inches.

B. The batch size required in order to obtain a shutdown margin of 1.5% delta K/K as determined by a reactivity balance calculation.

C. The maximum batch size setting and commence adding boric acid to the make up tank.

D. The batch size determined by the plant computer boron program to offset the reactivity worth of the four stuck rods.

Answer:

C. The maximum batch size setting and commence adding boric acid to the make up tank.

Notes:

"C" is the correct answer. RT-12 instructs the operator to commence emergency boration by setting the batch controller to the maximum batch size (999999 gals) and to begin adding boric acid via the batch controller if a boric acid pump is available. Therefore, answer "C" is correct. Answers "B" and "D" describe actions to determine the exact batch size after commencing emergency boration. The question is asking for the initial setting of the batch controller. Answer "A" uses a variety of setpoints associated with emergency boration incorrectly.

References:

1202.012 (Rev 004-03-0), Repetitive Tasks, RT-12, Emergency Boration.

History:

Developed by NRC. Modified QID 005 from ANO-1 NRC Exam Bank.
(QID 005 was used on the 1998 RO exam and the 2001 RO/SRO exam.)
Used on 2004 RO/SRO Exam.
Selected for 2005 RO exam

12. Emergency Boration:

A. **IF** Boric Acid pump (P39A or B) and Batch Controller are available, **THEN** perform the following:

1) Set Batch Controller for maximum batch size (999999).

2) Verify Condensate to Batch Controller (CV-1251) closed.

3) Open Batch Controller Outlet (CV-1250).

4) Verify both Letdown Filters in service (F-3A and B).

5) Record initial BAAT (T-6) level ____________ in.

6) Start available Boric Acid Pump(s) (P-39A or B or both).

7) Start Batch Controller by depressing RUN key.

8) Adjust Batch Controller Flow CNTRL VLV (CV-1249) to 100% open.

9) Adjust Pressurizer Level Control Setpoint to 220".

10) Open BWST Outlet to OP HPI Pump (CV-1407 or 1408).

11) **WHEN** PZR level is ≥ 100", **THEN** establish maximum Letdown flow.

12) 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) **WHEN** MU Tank level is lowered to desired level, **THEN** 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).

(12. CONTINUED ON NEXT PAGE)

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

QID: 0188 **Rev:** 0 **Rev Date:** 11/18/98 **Source:** Direct **Originator:** L. Kilby

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

Section: 4.2 **Type:** Generic AOP

System Number: 036 **System Title:** Fuel Handling Incidents

Description: Knowledge of the interrelations between the Fuel Handling Incidents and the following: Radiation monitoring equipment (portable and installed).

K/A Number: AK2.02 **CFR Reference:** 41.7 / 45.7

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

Group: 2 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** C

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

Given:

- Fuel handling is in progress in the Reactor Building.
- A fuel assembly is in the Main Bridge mast.
- Reactor Building Fuel Handling Area (RI-8017) area radiation monitor has a Failure alarm in solid.

What actions should be taken by the refueling team?

a. Place the assembly in the nearest core location and then secure refueling activities until repairs are made to RI-8017.

b. Continue refueling activities and notify Health Physics to perform area surveys of the Main Bridge every 15 minutes.

c. Continue refueling activities as long as two subcritical core neutron flux monitors are available.

d. Secure fuel handling activities until RI-8017 is operable or a suitable portable survey instrument is obtained.

Answer:

d. Secure fuel handling activities until RI-8017 is operable or a suitable portable survey instrument is obtained.

Notes:

(a.) is incorrect. In the event of a severe weather condition this action would be taken, but for RI-8017 failure the fuel movement should stop until it is restored or a suitable instrument is in place.
(b.) is incorrect. Refueling activities should be stopped and periodic surveys does not satisfy tech specs.
(c.) is incorrect. These are necessary to perform refueling activities but RI-8017 must also be operable or a suitable replacement must be in place.
(d.) is the correct answer.

References:

TRM 3.9.1

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO exam

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 entereed and the Required Action to place a portable survey instrument of appropriate range and sensitivity in service have been performed. (TRM 3.9.2).

5.3 One source range neutron flux monitor shall be operable in Mode 6. Two source range neutron flux monitors shall be operable during core alterations (TS 3.9.2).

5.4 One decay heat removal loop shall be operable and in operation in Mode 6 with water level ≥23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.4 for contingencies and exceptions.

5.5 Two decay heat removal loops shall be operable, and one loop shall be in operation in Mode 6 with the water level <23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.5 for contingencies and exceptions.

{4.3.1}

NOTE

The Refueling Boron Concentration specified by Reactivity Balance Calculation (1103.015) Worksheet 5 or 6 provides a shutdown margin of 5% as required by NRC commitment P 205. This concentration also satisfies TS 3.9.1.

5.6 Boron concentration of the RCS and Refueling canal shall be maintained within the limits specified in the COLR when the refueling canal is connected to the RCS (TS 3.9.1).

5.7 Direct communications between the control room and the refueling personnel in the reactor building shall exist during movement of irradiated fuel assemblies in the reactor building (TRM 3.9.4).

TRM 3.9 REFUELING OPERATIONS

TRM 3.9.1 Fuel Handling - Reactor Building

TRO 3.9.1 Radiation levels shall be monitored by RE-8017.

APPLICABILITY: During movement of fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RE-8017 inoperable.	A.1	Monitor area with portable survey instrument of appropriate range and sensitivity.	Immediately
B. Required Action and associated Completion Time not met.	B.1	Cease movement of fuel into reactor core	Immediately
	<u>AND</u>		
	B.2	Cease activities that might increase the reactivity of the core.	Immediately

TEST REQUIREMENTS

SURVEILLANCE	FREQUENCY
None.	

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

QID: 0216 **Rev:** 0 **Rev Date:** 11/18/98 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-EOP06 **Objective:** 16 **Point Value:** 1

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

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

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

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

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

Group: 2 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** C

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

Given:

- A 15 gpm Steam Generator tube leak cooldown is in progress.
- Normal cooldown limits are being used with the good OTSG.
- RCS pressure is 1000 psig, Tave is 405°F.

The CBOR is maintaining the RCS at about 140°F subcooled.

Why are the CBOR's actions incorrect for this accident?

a. Tube to shell Delta T limits are being exceeded.

b. A high primary to secondary Delta P is increasing primary coolant loss.

c. Excessive thermal stresses are being imposed on the Rx vessel.

d. Overfill could cause the ruptured SG main steam safeties to lift.

Answer:

b. A high primary to secondary DP is increasing primary coolant loss.

Notes:

[b] is correct. The operators are directed to maintain RCS pressure low within the limits of Figure 3. This will result in subcooling margin close to the limit and as low as possible primary to secondary differential pressure to prevent loss of unrecoverable primary coolant.
[a] is incorrect, there is not enough information given to determine if tube to shell DT limits are being exceeded.
[c] is incorrect, excessive thermal stresses are not being imposed without a high pressure condition.
[d] is incorrect, overfill could not cause the safeties to lift at 405°F.

References:

1202.006 Chg. 007-04-0, step 14
120.013, Rev. 4, Fig. 3

History:

Developed for A. Morris 98 RO Re-exam.
Used in 2001 RO/SRO Exam.
Selected for 2005 RO exam

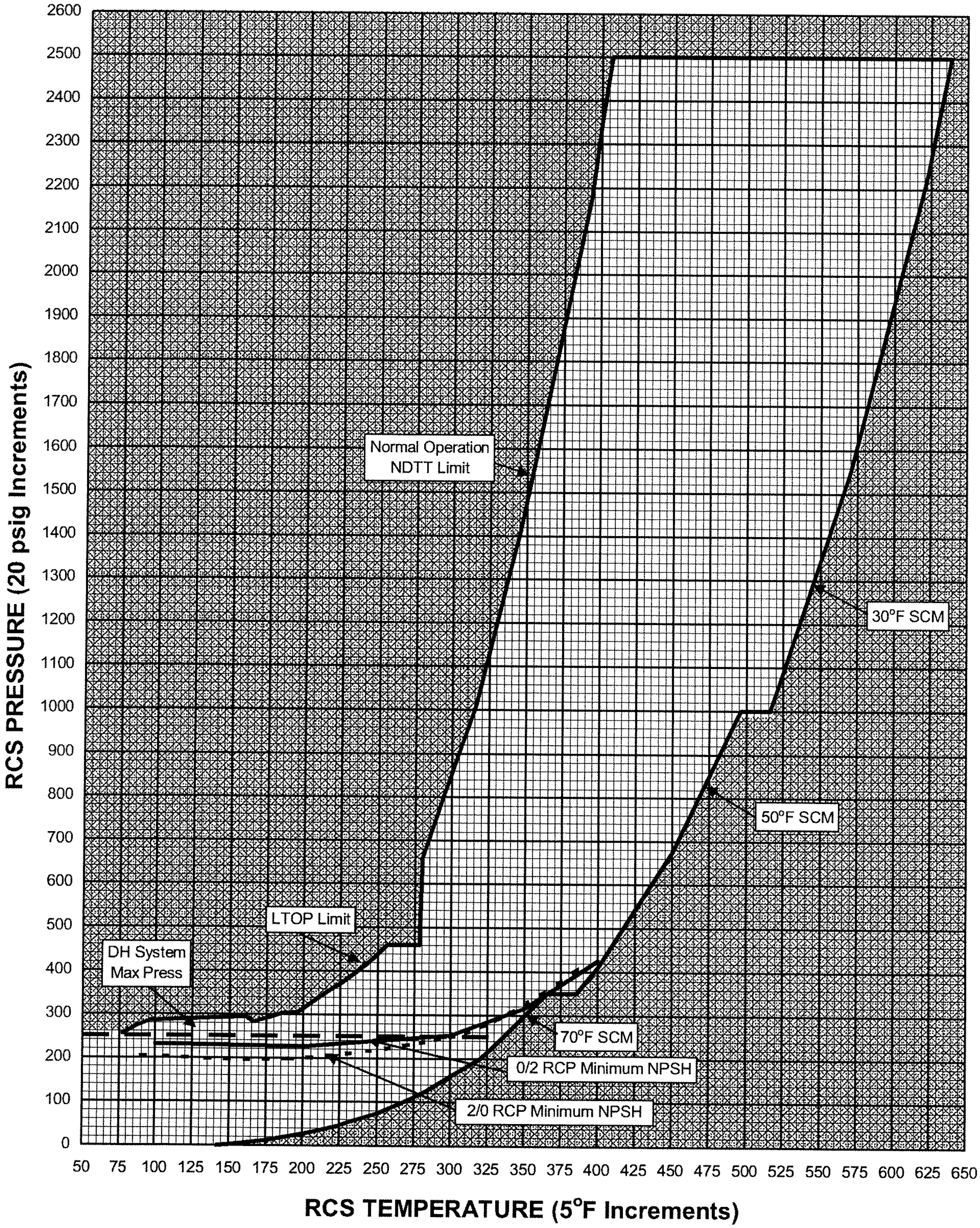
INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

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

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

FIGURE 3
RCS Pressure vs Temperature Limits



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

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

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of abnormal condition procedures.

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

Tier:	1	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- Plant is operating at 60% power
- E-11A North Waterbox is OOS for maintenance
- Condenser vacuum is degrading rapidly

Choose the appropriate operator actions:

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

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

c. Trip the turbine if vacuum falls below 26.5 inches Hg.

d. Trip the turbine if vacuum falls below 24.5 inches Hg.

Answer:

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

Notes:

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

References:

1203.016 Chg. 011-05-0
1202.001 Chg. 028-03-0

History:

Developed for 1998 RO/SRO Exam.
Modified for use in 2001 RO/SRO Exam
Selected for 2005 RO 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.

1.0 SYMPTOMS

1.1 Condenser vacuum degrading.

1.2 Any of the following annunciators in alarm:

- VACUUM PUMP AUTO START (K05-B3)
- CONDENSER VACUUM LO (K05-B2)
- TURBINE TRIP (K04-A3)
- TURBINE LO VACUUM TRIP (K05-A2)

2.0 IMMEDIATE ACTION

NONE

3.0 FOLLOW-UP ACTIONS

3.1 Commence reducing turbine load to stabilize vacuum.

- IF MWe is >270 and vacuum is <24.5" Hg, THEN verify the turbine has tripped.

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

NOTE

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

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

3.3 Refer to Rapid Plant Shutdown (1203.045).

NOTE

Condenser Inleakage Check Procedure (1106.029) may be helpful in determining and correcting the source of leakage.

3.4 Verify all available Condenser Waterbox Inlets open:

- E-11A South (CV-3630)
- E-11A North (CV-3626)
- E-11B South (CV-3622)
- E-11B North (CV-3618)

3.5 Verify available Circ Water Pumps (P-3A thru P-3D) running on C12.

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

QID: 0272 **Rev:** 1 **Rev Date:** 4/4/05 **Source:** Direct **Originator:** D. Slusher

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste Release: Isolation of the auxiliary building ventilation.

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

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	K

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

When a high radiation condition occurs in the Waste Gas Discharge Header, the radiation monitor will cause what combination of automatic action(s) to occur?

1. The Station Vent Discharge Valve CV-4830 opens.
2. The Aux. Building Vent Header diverts to the Waste Gas Surge Tank.
3. The Waste Gas Decay Tank effluent control valve (CV-4820) shuts.
4. The Aux. Building Vent Header diverts to the Waste Gas Decay Tank in service.

a. 1 and 2

b. 2 and 3

c. 3 and 4

d. 1 and 4

Answer:

b. 2 and 3

Notes:

"b" is correct
"a", "c", and "d" are incorrect because the station discharge valve closes automatically and isn't re-opened until later. The ABVH is diverted to the Waste Gas Surge Tank and not the in service Wast Gas Decay Tank.

References:

1203.006, Chg. 010-02-0, step 2

History:

Used in 1999 exam.
Direct from ExamBank, QID# 1399
Used in 2001 RO/SRO Exam.
Selected for 2005 RO exam

INSTRUCTIONS

1. **Secure any radioactive gas release or venting in progress.**

2. **Verify following valves closed:**

   - Station Vent Discharge Valve (CV-4830)
   - Gaseous Radwaste Hdr. Isolation Valve (CV-4820)

3. **Verify ABVH Diversion Valve to Surge Tank (CV-4806) open.**

4. **IF venting operation was in progress,**
   THEN perform the following:

   A. **IF** desired to continue venting operations,
   THEN align the waste gas system per Gaseous Radwaste System (1104.022), "Venting of High Activity Systems".

> **NOTE**
>
> Steps 1-6 may be re-performed as needed to purge the piping upstream of RE-4830 if it trips during the performance of this section.

   B. **IF** venting will be terminated and it is desired to align the vent header to the Station Vent Plenum,
   THEN perform the following:

      1. Reset RE-4830.

         A. **IF** RE-4830 will not reset due to current radiation levels being above the setpoint,
         THEN perform 1305.001 Supplement 5 to raise the setpoint of RE-4830 and allow reset of the detector.

      2. Open CV-4830.

      3. Re-establish N2 flow to previous amount or as desired to be consistent with system conditions.

      4. Close CV-4806.

      5. Verify C-9A and C-9B in AUTO.

         A. Verify C-9A and C9B shut down when T-17 ≤ 15.2 psia.

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

QID: 0558 **Rev:** 0 **Rev Date:** 4/4/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Control Room Evacuation: AFW emergency pump.

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

Tier:	1	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.5	**SRO Select:**	No	**Taxonomy:**	C

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

Given:
- An immediate Control Room evacuation is in progress due to a fire in the Cable Spreading Room.
- All Control Room actions have been completed.
- It is approximately 40 minutes into the event.

How is feedwater being supplied to the OTSGs?

A. Motor driven EFW pump P-7B is initially used and then automatic control of steam driven EFW pump P-7A is established.

B. Steam driven EFW pump P-7A with automatic control of P-7A flow control valves.

C. Motor driven EFW pump P-7B with automatic control of P-7B flow control valves.

D. Motor driven EFW pump P-7B is initially used and then local control of steam driven EFW pump P-7A is established.

Answer:

D. Motor driven EFW pump P-7B is initially used and then local control of steam driven EFW pump P-7A is established.

Notes:

"D" is correct, EFW is automatically actuated, then P-7B is secured and P-7A speed is controlled locally.
"A" is incorrect, automatic control of P-7A is not used.
"B" and "C" are incorrect, automatic control of flow control valves is not used.

References:

1203.002, 015-07-0

History:

New for 2005 RO exam

ATTACHMENT 9

DISCUSSION CONTINUATION

1.0 DESCRIPTION

Alternate Shutdown capability is being provided to comply with 10 CFR 50, Appendix R, and to mitigate consequences of significant fire in either the Control Room or the Cable Spread Room because fires in these areas can result in loss of controls and instrumentation required for safe shutdown. Following are some of the more significant assumptions considered in development of this procedure:

NOTE

This procedure will be used even if a coincident loss of off-site power has not occurred.

A. A coincident loss of off-site electrical power has occurred.

B. No coincident design basis accident occurs, only accidents that are a direct result of the fire.

C. Control circuits (for valves, motors, etc.) located in the fire area are assumed to cause activation of the component to the least desirable condition unless preventative action is taken.

2.0 SYSTEMS

The following are the safe shutdown system components utilized by the Alternate Shutdown procedure and a brief description of how they are used.

2.1 Emergency Feedwater

Local control of steam-driven EFW Pump (P-7A) is established with the EFW Turb K3 Trip/THROT VLV (CV-6601A). EFW P-7A to SG-B ISOL (CV-2620) is manually controlled to control SG-B level. These actions are accomplished by RO #2.

RO #1 will manually control EFW P-7A to SG-A ISOL (CV-2627) to control the SG-A level.

Steam release will be initially controlled by the mechanical steam relief valves and, as additional operators become available, by manual control of the ADVs.

The EFW CST (T-41B) will provide the initial source of water to the EFW system with the CST (T-41) and service water, the backup sources, if necessary.

As time permits, the P-7B train of EFW will be made available for backup purposes.

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

QID: 0276 **Rev:** 0 **Rev Date:** 9/2/99 **Source:** Direct **Originator:** D Slusher

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

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

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

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

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

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2.5

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

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

Given:

- A loss of offsite power has occurred.
- Annunciator K01-B1, "EDG 1 BRKR AUTO CLOSE FAILURE", is in alarm.

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

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

b. Depress EDG #1 start push-button.

c. Reset A1 Lockout relay.

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

Answer:

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

Notes:

(a) is correct, taking HS to PTL will reset anti-pump relays and allow breaker to auto-close.
(b) is incorrect, this will accomplish nothing because an auto-close would not exist unless EDG was running.
(c) is incorrect, resetting Lockout Relay will have no effect on EDG output breaker .
(d) is incorrect since breaker cannot be closed manually from C-10 unless the sync switch is ON.

References:

1203.012A, Chg. 034-03-0

History:

Developed for 1999 exam.
Selected for 2005 exam

Location: C10

Device and Setpoint: see next page.

EDG 1 BRKR
AUTO CLOSE
FAILURE

Alarm: K01-B1

1.0 OPERATOR ACTIONS

1. If bus A3 is deenergized, verify the following breakers are open:

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

2. Attempt to reset breaker anti-pump feature as follows:

   A. Place control switch for DG1 Output Breaker (A-308) in pull-to-lock and release.

3. If breaker did not close, turn Synchronize switch ON for DG1 Output (A-308).

4. Attempt to close A-308 from C10.

5. If A-308 fails to close from C10, close locally.

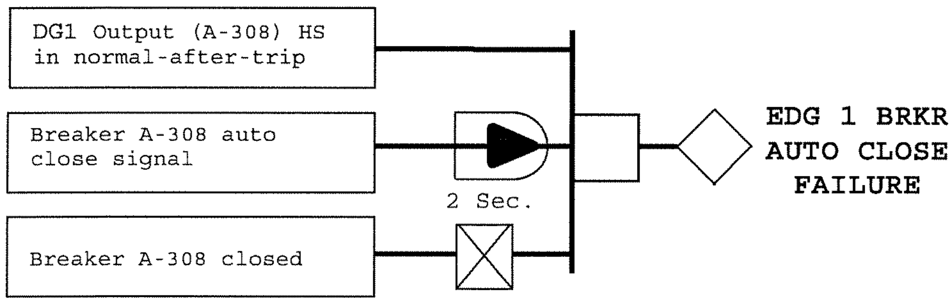
6. To clear alarm, remove A-308 HS from normal-after-trip position.

2.0 PROBABLE CAUSES

1. Breaker A-308 tripped with an auto close signal.

3.0 REFERENCES

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



DG1 Output (A-308) HS in normal-after-trip
Breaker A-308 auto close signal
2 Sec.
Breaker A-308 closed
EDG 1 BRKR AUTO CLOSE FAILURE

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

QID: 0421 **Rev:** 1 **Rev Date:** 4/5/05 **Source:** Modified **Originator:** S.Pullin

TUOI: A1LP-RO-EOP01 **Objective:** 5 **Point Value:** 1

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

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

Description: Ability to determine and interpret the following as they apply to the Inadequate Subcooling Margin): 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:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	Ap

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

Given a Rx trip has occurred and one minute later the following conditions are observed:

- RCS pressure is stable 1700 psig.
- CET average temperature is 600 degrees F.

Which Emergency Operating Procedure contains mitigating actions for this event?

a. Loss of Subcooling Margin (1202.002)

b. ESAS (1202.010)

c. Overheating (1202.004)

d. Inadequate Core Cooling (1202.005)

Answer:

a. Loss of Subcooling Margin (1202.002)

Notes:

Answer "a" is the right procedure and the rest are incorrect.

References:

1202.013, Rev. 4, Figure 1
1202.002, Chg. 004-02-0

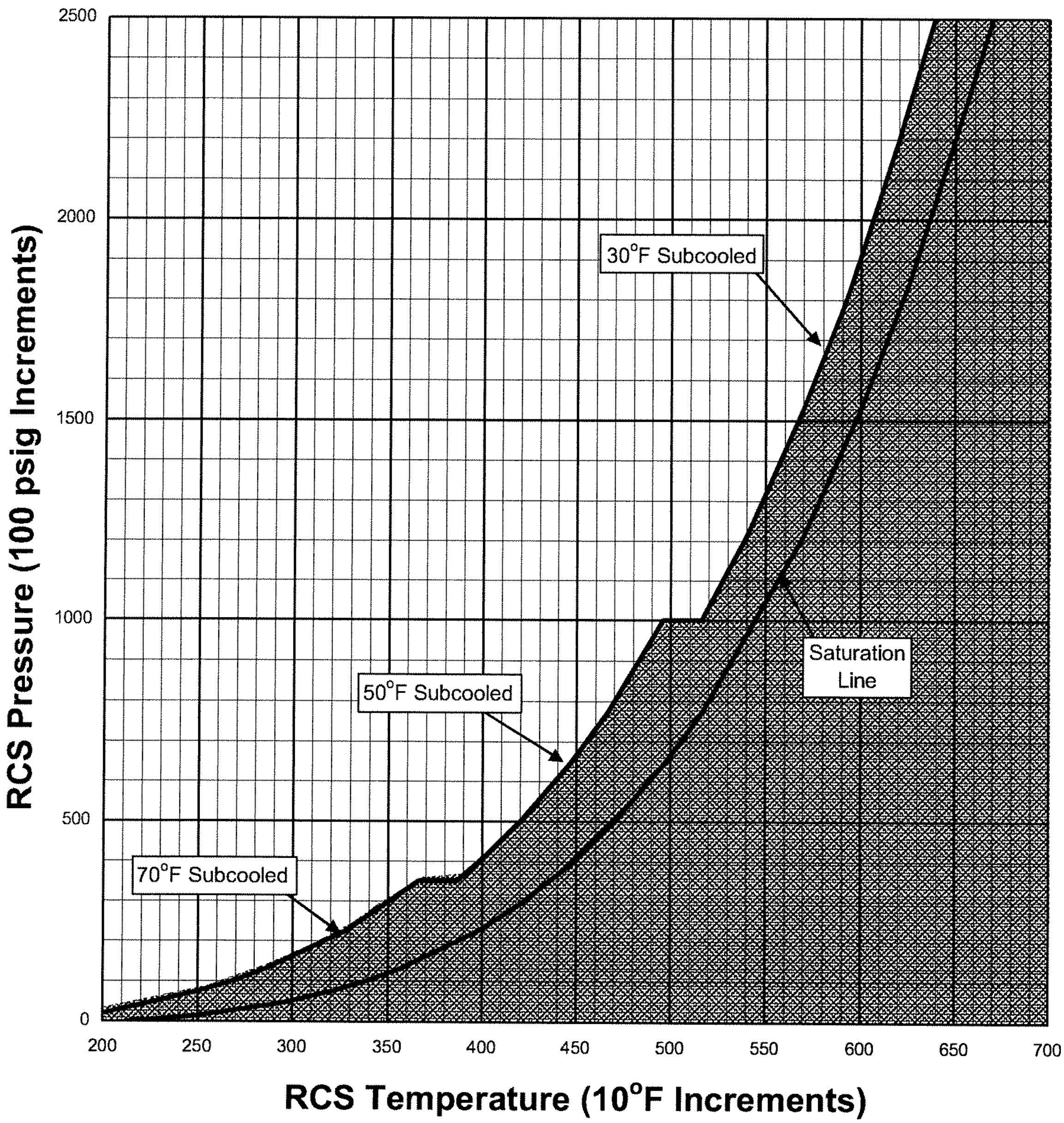
History:

Direct from regular exambank QID 2928.
Selected for use in 2002 RO/SRO exam.
Modified for use in 2005 RO exam

ENTRY CONDITIONS

- Loss of adequate SCM following a Reactor trip,
- Loss of adequate SCM while attempting to correct overcooling,
- Loss of adequate SCM following ESAS actuation
  AND
  RCS press stabilizes >150 psig (LPI pump discharge pressure).

FIGURE 1
Saturation and Adequate SCM



RCS Pressure	Adequate SCM
>1000 psig	≥30°F
350 to 1000 psig	≥50°F
<350 psig	≥70°F

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

QID: 0559 **Rev:** 0 **Rev Date:** 4/6/05 **Source:** New **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 System

Description: Knowledge of the effect of a loss or malfunction of the following will have on the RCPS: Starting requirements.

K/A Number: K6.14 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
- Plant heatup in progress from refueling outage.
- P-32A, P-32C, P-32D 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.

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

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

B. Seal injection flow is low.

C. Non-nuclear ICW to RCP motor cooling flow is low.

D. RCS cold leg temps are low.

Answer:

D. RCS cold leg temps are low.

Notes:

"D" is correct, RCS cold legs must be greater than 375°F to start the fourth RCP.
"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, non-nuclear ICW to RCPS is greater than 250 gpm.

References:

1103.006, Chg. 025-04-0

History:

New for 2005 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 >375°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).

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 $\Delta T \leq 50°F$.

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 >375°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 (VALIDATION)

QID: 0432 **Rev:** 0 **Rev Date:** 4/30/2002 **Source:** New **Originator:** S.Pullin

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

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

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

Description: Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: PZR LCS.

K/A Number: K3.05 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	C

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

Unit One is operating at 100% power.
An ICW problem causes Letdown temperature to rise to 148°F.

What is the effect on the Pressurizer level control system during this transient?

A. PZR level will continue to drop during this event.

B. PZR level will continue to rise during this event.

C. Makeup flow will rise and restore Pressurizer level to setpoint.

D. Makeup flow will drop and restore Pressurizer level to setpoint.

Answer:

B. PZR level will continue to rise during this event.

Notes:

"B" is correct as the high letdown temperature will cause letdown to isolate, makeup flow will drop but seal injection will continue to provide 8 gpm per RCP makeup to the RCS, so PZR level will continue to rise.
"A" is incorrect, PZR level will not drop.
"C" & "D" are incorrect, PZR level will not be restored to setpoint.

References:

STM 1-04, Rev. 7

History:

New for 2005 RO exam

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.

- Two normal makeup isolation valves (CV-1233 & CV-1234)
- Connection to HPI line on discharge of RCP "D" in loop "A" of the RCS
- A ~10-20GPM constant flow bypass for thermal stress mitigation

The pressurizer level control valve is controlled from the Non Nuclear Instrumentation system (NNI) by the pressurizer level control program. The signal received by this valve is dependent on both the setpoint and the deviation from setpoint of level in the pressurizer. During normal conditions the setpoint is 220". CV-1235 will open and/or close to maintain this level.

1.3.5 RCP Seal Injection and Seal Return.

Seal injection supplies filtered water to the RCP seals. Most of this water goes into the RCS. Some bleed-off from the seals is returned via seal return. Seal injection and return consists of:

- Supply flow element (FE-1239)
- Control valve and bypass (CV-1207)
- Seal injection filter (F-2)
- Seal injection isolation valve (CV-1206)
- Stop check valves, throttle valves and flow elements on the individual supply lines in containment.
- Motor operated valves on each seal return line inside containment.
- Flow indicating transmitter
- Motor operated isolation valve outside containment CV-1274
- An alternate path to the quench tank and its isolation valves
- Seal return coolers
- Connection to other systems

1.3.6 Connections to other systems

The makeup and purification system interfaces with various systems to perform its intended functions. These systems include:

Clean Liquid Radwaste System

During normal operation, letdown flow is directed into the Makeup Tank. However, flow can be diverted, using CV-1248, to the CZ system to be processed by the vacuum degasifier for storage in the T-12's or for return to the makeup tank. Piping is also provided to return T-12 liquid to makeup and purification for RCS filling if desired. With the vacuum degasifier manually bypassed, flow diverted from letdown will go directly to a T12. This would be done during plant heatup or chemical control feed-and-bleed operations. Also directed to the makeup tank are RCP seal bleedoff and HPI pump minimum recirculation flows.

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

QID: 0091 **Rev:** 0 **Rev Date:** 7/12/98 **Source:** Direct **Originator:** JCork

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

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

System Number: 005 **System Title:** Residual Heat Removal System (RHRS)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Heatup/cooldown rates.

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

Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** C

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

Given:

- Plant cooldown in progress
- "A" DH Removal pump in service
- "A" and "C" RCPs running

What is the maximum allowable cooldown rate in this condition?

a. 5°F/hr

b. 25°F/hr

c. 50°F/hr

d. 100°F/hr

Answer:

c. 50°F/hr

Notes:

Per 1102.010, the maximum cooldown rate is 50°F/hr when the RCS is <280°F and >150°F, "c" is correct.
DH cannot be placed in service unless the RC temp is <280°F, therefore "D" is incorrect.
The RCP's are removed from service while a PZR steam bubble is still present and specifically when RC temp is between 166°F and 270°F, therefore "B" is incorrect.
"A" is fictitious.

References:

1102.010, Plant Shutdown and Cooldown, Chg. 053-10-0

History:

Developed for 1998 RO/SRO exam
Used in 2005 RO Exam

NOTE

- Maximum cooldown rates are as follows:

  A. RCS >280°F: 100°F/hr (1.67°F/min)
  B. RCS 280-150°F: 50°F/hr (0.83°F/min)
  C. RCS <150°F: 25°F/hr (0.41°F/min)
  D. Pressurizer: 100°F/hr (1.67°F/min)
  E. If RCPs are operating between 180°F to 150°F, maximum RCS cooldown rate is limited to 2°F/hr.

- If available, computer indications may be used for manual plotting of RC and Pressurizer temperatures versus time.

- All graphs used during cooldown shall include the following:

  A. Incremental value
  B. Time
  C. Date
  D. Initial
  E. Parameter, including instrument number

- If available, SPDS cooldown display may be used to monitor RCS and PZR cooldown.

- The following pressure instruments are the most accurate for their ranges during all RCP combinations and are preferred for plotting pressure versus temperature.

  A. 0 to 500 psig: Low Range RC Pressure indicator (PI-1010)
  B. 500 to 1700 psig: SPDS points P1041 and P1042
  C. 1700 psig and above: SPDS points P1021, P1023, & P1039

  Instrument error on SPDS graphics displays may cause data points to be outside the limits on the SPDS PT curve.

- With RCPs inservice, the lowest WR RCS Cold Leg Temperature instrument which has a RCP operating in the same loop should be used for plotting RC temperature. If no RCPs are operating, use lowest.

Loop A	Loop B	Loop C	Loop D
T1047 PMS & SPDS	T1045 PMS	T1017 PMS	T1016 PMS & SPDS
T1147 SPDS	T1144 SPDS	T1117 SPDS	T1115 SPDS

- Due to inaccuracies, Loop A Wide Range Pressure PT-1020 (SPDS point P1020) should NOT be used for plotting pressure versus temperature.

- Plant Computer points T1406 and T1407 are the most accurate indication of DH cooler outlet temperature and should be used when available.

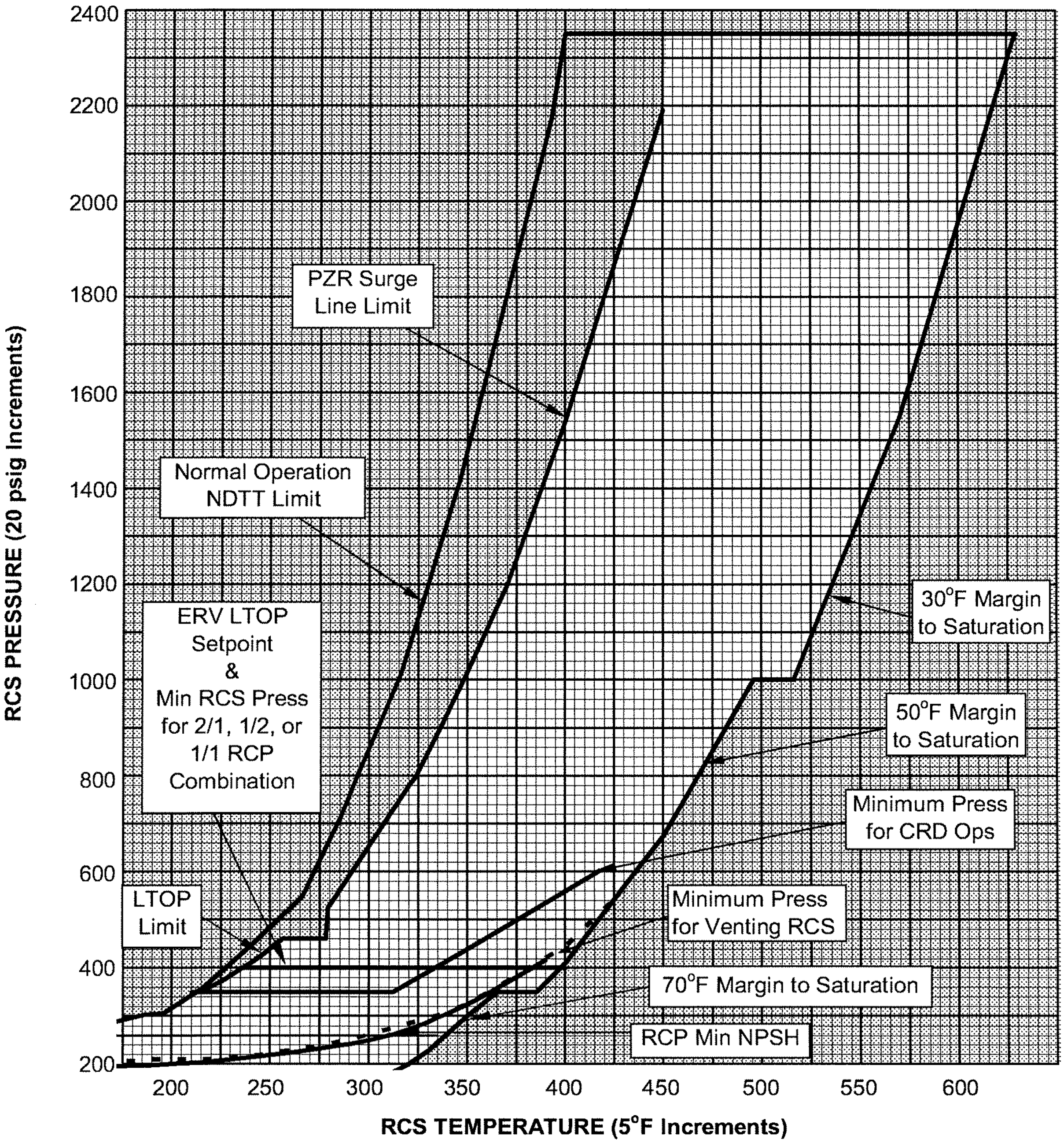
- If CET data from ICC Train A or Train B on C19 is unavailable, obtain SPDS CET value from TCETA1 or TCETB1 or one of the inputs NOT designated (ICC) in the SPDS data base listing.

- When both Decay Heat Removal systems are in service on Decay Heat, use the lower DH cooler outlet temperature for plotting cooldown rate.

- SG Tube-to-Shell ΔT may be monitored on SPDS PSHT or SGTR display.

ATTACHMENT B Page 1 of 2

{4.3.4} **Allowable RCS Pressure vs. Temperature During Cooldown for all RCP combinations EXCEPT P-32A and B running, C and D off (31 EFPY)**



NOTE: RCS Temperature	Maximum Cooldown Rate
≥ 280°F	100°F/hr
280°F to 150°F	50°F/hr
< 150°F	25°F/hr

Attachment A is to be used below 500 psig and 350°F

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

QID: 0560 **Rev:** 0 **Rev Date:** 4/5/05 **Source:** Direct **Originator:** J.Cork

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

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

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

Description: Ability to manuall operate and/or monitor in the control room: Heat exchanger bypass flow control.

K/A Number: A4.02 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	K

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

Given the following plant conditions:

---Plant shutdown in progress.

---RCS temperature 240 degrees F.

---RCS pressure 225 psig.

Identify the correct valve alignment for starting a Decay Heat pump:

A. Cooler outlet valve 50% open, cooler bypass valve shut, injection block valve 100% open.

B. Cooler outlet valve shut, cooler bypass valve 75% open, injection block valve shut.

C. Cooler outlet valve shut, cooler bypass valve 75% open, injection block valve 100% open.

D. Cooler outlet valve 50% open, cooler bypass valve shut, injection block valve 50% open.

Answer:

C. Cooler outlet valve shut, cooler bypass valve 75% open, injection block valve 100% open.

Notes:

"C" contains the correct alignment per 1104.004, the rest are incorrect alignments.

References:

1104.004, Chg. 071-05-0

History:

Direct from regular exam bank QID #1771.
Selected for use in 2005 RO exam

10.0 Shifting Operating Decay Heat Loops

CAUTION

- Severe water hammer can damage DH system pipe if DH pump is started with high flow when RCS is drained.
- Decay Heat system design pressure can be exceeded if DH pump is dead-headed with RC pressure >150 psig (DH pump suction pressure >200 psig).
- Decay Heat pump discharge relief setpoint is 445 ±13.35 psig. Discharge pressure should be maintained <400 psig to prevent challenging the relief.

{4.3.6}

NOTE

Rules for starting LPI (Decay Heat) pump:

- If RC pressure is <150 psig, always start the Decay Heat pump with the Decay Heat Cooler Outlet and Cooler Bypass valves closed or the LPI Block Valve closed.
- If RC pressure is >150 psig and RCS is not drained, the Decay Heat pump must be started with the Decay Heat Cooler Outlet valve closed, Decay Heat Cooler Bypass valve ~ 75% open and the LPI Block Valve Open.
- **If RCS is drained and RC pressure is >150 psig (as a result of extended loss of DHR for example), the RCS must be depressurized to <150 psig, then the Decay Heat pump started with the Decay Heat Cooler Outlet and Cooler Bypass valves closed or LPI Block valve closed.**

10.1 IF placing Low Pressure Injection (Decay Heat) pump P-34A into service,
THEN perform the following:

10.1.1 Establish a Plant Computer Programmable Annunciator alarm for P-34A discharge pressure of 400 psig.

CAUTION

- With RC pressure >75 psig and DH suction piping aligned to the RCS, exceeding design pressure of suction piping from BWST will occur if Decay Heat P-34A Suction from BWST (CV-1436) and Decay Heat P-34A Suction from RCS (CV-1434) are open at the same time.
- The interlock to prevent the suction from RCS valves (CV-1434 and CV-1435) opening unless suction from BWST valves (CV-1436 and CV-1437) are closed does not prevent CV-1436 and CV-1437 from opening after CV-1434 and CV-1435 are open.

10.1.2 Close Decay Heat P-34A Suction from BWST (CV-1436).

10.1.3 Open Decay Heat P-34A Suction from RCS (CV-1434).

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

QID: 0197 **Rev:** 1 **Rev Date:** 4/5/05 **Source:** Modified **Originator:** R. Walters

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

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

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

Description: Knowledge of the effect of a loss or malfunction of the following will have on the ECCS: Core flood tanks (accumulators).

K/A Number: K6.02 **CFR Reference:** 41.7 / 45.7

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

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

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

The plant is operating at 100% power. The Core Flood system is properly aligned with the following CFT parameters:

T-2A level	-	12.9 feet	T-2B level	-	13.1 feet
T-2A pressure	-	569 psig	T-2B pressure	-	620 psig

The Core Flood system parameters are unacceptable because?

A. Levels may preclude having sufficient N2 pressure to fully inject the CFT contents into the vessel.

B. High N2 pressure could cause RCS inventory to be lost out of the break in the event of a LOCA.

C. Levels may not be sufficient to reflood the vessel following a LOCA.

D. N2 pressure may not be sufficient to fully inject the CFT contents into the vessel during a LOCA.

Answer:

D. N2 pressure may not be sufficient to fully inject the CFT contents into the vessel during a LOCA.

Notes:

"D" is correct. N2 pressure for A CFT is less than 572 psig TS limit from 1104.001. Both CFTs are required per the TS bases to fully inject CFT contents and reflood the vessel following a LOCA.
"A" is incorrect. This would be the case if either level were out of spec high, however, the levels are within specs.
"B" is incorrect. N2 pressure for B CFT is within the Tech Spec limits but A is low.
"C" is incorrect. Both levels are with the Tech Spec limits.

References:

1104.001, Chg. 033-07-0
T.S. Bases for 3.5.1

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for use in 2002 RO/SRO exam.
Modified for use in 2005 RO exam

5.0 LIMITS AND PRECAUTIONS

5.1 With the exception of ASME Section XI testing and when the CFT is depressurized, during plant cooldown the CFT discharge valves (below) shall be closed and the circuit breakers for the motor operators opened before depressurizing the RCS below 600 psig or ≤262°F (TS 3.4.11).

- Core Flood Tank T-2A Outlet Isol (CV-2415), breaker (B-5661)
- Core Flood Tank T-2B Outlet Isol (CV-2419), breaker (B-5545)

5.2 When the RCS is depressurized with DH in service and either CFT is pressurized, the potential exists for personnel harm, equipment damage, and loss of DH flow in the event the core flood tank isolation valves are opened.

5.3 CFT NDTT limits are as follows:

5.3.1 CFT metal temperature shall not be less than 65°F anytime the tank is pressurized above 140 psig.

5.3.2 N_2 temperature at the point of injection shall be ≥65°F anytime the tank is pressurized above 140 psig.

5.3.3 If N_2 temperature at the point of injection is ≥ 100°F below the tank metal temperature, CFT pressure shall be less than 25 psig.

NOTE

TS limits where the CFT will be declared inoperable for CFT level (volume) and pressure are listed below and contain the maximum allowance for instrument uncertainty.

5.4 Prior to reaching RCS pressure of 800 psig, each CFT shall be at least as follows (TS 3.5.1):

5.4.1 CFT level

- Admin limit - 12.7 to 13.3 feet
- Tech Spec limit - 12.6 to 13.4 feet

5.4.2 CFT Pressure

- Admin Limit - 580 to 620 PSIG
- Tech Spec Limit - 572 to 628 PSIG

5.4.3 CFT boron concentration ≥ 2270 ppmB (Tech Spec limit).

APPLICABLE SAFETY ANALYSES

The CFTs are credited in both the large and small break LOCA analyses at full power (Ref. 1). The CFT line break analysis credits only one CFT, since the tank with the broken line is assumed to empty out the break. These Design Basis Accident (DBA) analyses establish the acceptance limits for the CFTs. In performing the LOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. In addition, a loss of offsite power is considered to ensure worst case conditions are postulated. In the early stages of a limiting large break LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS. This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver flow until the diesel generators (DGs) start and go through their timed loading sequence.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. No credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 35 seconds after the RCS pressure decreases to the ESAS actuation pressure. No operator action is assumed during the blowdown stage of a large break LOCA (Ref. 1).

The small break LOCA analysis also assumes a time delay after ESAS actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria for the ECCS established by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature of 2200°F;

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

APPLICABLE SAFETY ANALYSES (continued)

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to cover the core to the 3/4 point even assuming no liquid remains in the reactor vessel following a LOCA (Ref. 1). The downcomer then remains flooded until the HPI and LPI systems start to deliver flow for limiting large break LOCAs.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection and ensure the ability of the CFTs to fully discharge. The limiting safety analysis volume requirement is 1040 ± 70 ft^3. This volume corresponds to CFT levels of ≥ 11.95 ft and ≤ 14.00 ft. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The minimum nitrogen cover pressure requirement of 560 psig ensures that the contained gas volume will generate discharge flow rates during injection that satisfy the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The maximum nitrogen cover pressure limit of 640 psig will affect the amount and timing of CFT inventory discharged while the RCS depressurizes. Limiting the maximum pressure will therefore limit the CFT inventory lost through the break and assure that the CFT inventory injected into the RCS at the proper time is bounded by that predicted by the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum CFT level, the reactor will remain adequately shutdown in the cold condition following mixing of the CFT and Reactor Coolant System (RCS) water volumes. This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

In MODE 1, the CFTs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 2 and MODE 3 with RCS pressure > 800 psig, the CFTs satisfy Criterion 4 of 10 CFR 50.36.

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

QID: 0071 **Rev:** 0 **Rev Date:** 7/7/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-MU **Objective:** 4.C **Point Value:** 1

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

System Number: 006 **System Title:** Emergency Core Cooling System (ECCS)

Description: Knowledge of symtom based EOP mitigation strategies.

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

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 4

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

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

Given:

- HPI pump P-36B out of service
- LOCA in progress
- RCS pressure 1250 psig
- 4160v bus A3 de-energized
- HPI flows: A = 90 gpm
  B = 100 gpm
  C = 250 gpm
  D = 75 gpm

Which of the following actions is required?

a. Throttle "C" flow to within 20 gpm of "B"

b. Throttle "C" flow to within 20 gpm of "D"

c. Throttle all to within 20 gpm of each other

d. Throttle "C flow to within 20 gpm of "A"

Answer:

a. Throttle "C" flow to within 20 gpm of "B"

Notes:

"a" is correct, if only one HPI pump available and RCS press > 600 psig, then highest flow (C) must be throttle to within 20 gpm of the next highest flow.

References:

1202.012, Chg. 004-03-0, RT-2

History:

Used in 1998 RO/SRO exam
Modified QID 2840
Selected for 2005 RO exam

2. (Continued).

F. **IF** only **one** train of HPI is available
AND
RCS press is >600 psig,
THEN throttle the HPI Block valve with the highest flow to within 20 gpm of the next highest flow.

G. **IF** leakage into the RB is indicated, **THEN** maximize RB cooling:

1) Verify all four RB Cooling Fans running (VSF1A - D).

2) Open RB Cooling Coils Service Water Inlet and Outlet valves (CV-3812, 3813, 3814 and 3815).

3) Unlatch key-locked Chiller Bypass Dampers (SV-7410, 7412, 7411, 7413).

H. Verify the following sample valves closed

- Pressurizer Steam Space (CV-1814)
- Pressurizer Water Space (CV-1816)
- Hot Leg Sample (SV-1840)

I. **Unless** directed otherwise, verify the following High Point Vents closed.

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

(2. CONTINUED ON NEXT PAGE)

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

QID: 0561 **Rev:** 0 **Rev Date:** 4/5/05 **Source:** New **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 Pressurizer.

K/A Number: K5.02 **CFR Reference:** 41.5 / 45.5

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

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

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

A plant startup is in progress with a steam bubble being drawn in the Pressurizer.
Initial Quench Tank pressure is 2 psig.

Which of the following assures that venting and steam bubble formation is complete in the Pressurizer?

A. RCS pressure 55 psig, Pressurizer temperature 290°F, Quench Tank pressure 4.5 psig after a 3 minute blow of the ERV.

B. RCS pressure 65 psig, Pressurizer temperature 312°F, Quench Tank pressure 2.5 psig after a 3 minute blow of the ERV.

C. RCS pressure 45 psig, Pressurizer temperature 283°F, Quench Tank pressure 6.7 psig after a 3 minute blow of the ERV.

D. RCS pressure 35 psig, Pressurizer temperature 250°F, Quench Tank pressure 8.3 psig after a 3 minute blow of the ERV.

Answer:

B. RCS pressure 65 psig, Pressurizer temperature 312°F, Quench Tank pressure 2.5 psig after a 3 minute blow of the ERV.

Notes:

"B" is correct with PZR temp at saturation for RCS pressure and Quench Tank pressure rise less than or equal to 1 psig.
All other choices contain low PZR temperatures for the pressures given and greater than 1 psig pressure rise which indicates nitrogen is still being vented to the Quench Tank.

References:

1103.005, Chg. 030-04-0
Steam Tables

History:

New for 2005 RO exam

7.1.11 N_2 system aligned to Quench Tank per N_2 System (1104.009), Attachment A.

{4.3.6}

7.1.12 **If elevated airborne activity is expected as a result of PZR operations, personnel Hatch interlock enabled or Personnel Hatch watch stationed to ensure only one door opened at a time.**

7.2 Steam Bubble Formation

CAUTION

Pressurizer level should be verified ≤ 275" prior to steam bubble formation, except as directed by Emergency Operating Procedures (1202.XXX).

7.2.1 Monitor Quench Tank pressure, temperature and level. Maintain Quench Tank conditions per "Quench Tank (T-42) Operation" section of this procedure.

NOTE

- To maintain the RCS in a non water solid condition, TS 3.4.11 Bases limits pressurizer level with RCS temperature <272°F and the reactor vessel head in place to the following:
  - ≤105" at RCS pressures >100 psig
  - ≤150" at RCS pressures ≤100 psig
- Annunciator "LTOP Trouble" (K09-E7) alarms at the following pressurizer level setpoints with RCS temperature <272°F:
  - 95" at RCS pressures >100 psig
  - 140" at RCS pressures ≤100 psig

7.2.2 WHEN RCS temperature is <272°F,
AND reactor vessel head is in place,
THEN maintain pressurizer level as follows:

- ≤140" if RCS press is ≤100 psig
- ≤95" if RCS press is >100 psig

7.2.3 Energize pressurizer heaters to raise pressurizer temperature.

NOTE

- If the RCS has been opened, the pressurizer sample lines will likely contain air and other gases which require purging before representative samples of the pressurizer will be obtained.
- Venting the sample piping at locations such as E-30 Outlet Drain Isol (SS-1012B) in the primary sample room has been shown to greatly enhance sample line purging. This venting should be done prior to 200F in order to be able to use standard red rubber hoses to flush to the floor drains.

7.2.4 Notify Chemistry to begin flushing pressurizer sample lines.

WARNING

Opening the ERV causes a localized steam release at the pilot valve vent. This is a radiation and safety hazard.

7.2.5 Verify no one is at the vicinity of the ERV.

CAUTION

- Pressurizer heatup rate limit is ≤ 100°F/hr.
- DH system maximum pressure is ≤ 250 psig.

{4.3.4}

7.2.6 When RC pressure reaches ~ 70 psig, open following to vent nitrogen from PZR to Quench Tank (T-42):

A. ERV Isolation Valve (CV-1000)
B. ERV (PSV-1000, HS-1014 on C04)

7.2.7 Re-close ERV before RCS pressure falls to 30 psig.

7.2.8 <u>If</u> this is a heatup following a refueling outage, <u>or</u> the ERV has not been exercised during cold shutdown within the last 92 days, perform Supplement I to document Tech Spec and ASME section XI required exercising of the ERV.

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 Allow pressure to rise again to near 70 psig
<u>and</u>
repeat steps 7.2.5 through 7.2.7 as necessary until bubble forms.

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

QID: 0294 **Rev:** 0 **Rev Date:** 9/4/99 **Source:** Direct **Originator:** D Walls

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining Quench Tank water level within limits.

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

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

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

Why is a minimum water level maintained in the Quench Tank?

a. Ensure adequate NPSH for the transfer pump.

b. Provide sufficient cooling-quench water during pressurizer operations.

c. Maintain a reference water level for level indication.

d. Maintain a loop seal on the relief lines.

Answer:

b. Provide sufficient cooling-quench water during pressurizer operations.

Notes:

"a" is incorrect because the NPSH requirement is lower than the minimum water level.
"c" is incorrect because the transmitter reference leg is external to the tank.
"d" is incorrect because the relief valves relieve to a header that is aligned in the bottom of the tank.

References:

1103.005, Chg. 030-04-0

History:

Used in 1999 exam
Direct from ExamBank, QID #2180 used in class exam
Selected for 2005 RO exam

5.6 Both Pressurizer Code Safety Valves (PSV-1001, PSV-1002) shall be operable when the reactor is critical (TS 3.4.10).

5.7 When the RCS is >262°F, at least one pressurizer code safety valve shall be operable, except for hydrostatic tests (TS 3.4.10).

5.8 Maximum allowable temperature difference between pressurizer water and T-hot shall not exceed 410°F except for startup.

5.9 Maximum allowable pressurizer level at any time the reactor is critical is 290 inches.

5.10 Minimum water level during normal power (> 15%) operation is 190 inches.

5.11 The pressurizer spray valve shall not be opened with nitrogen in the pressurizer.

5.12 During heatup 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, unless required for plant safety.

5.13 Maintain reactor coolant system pressure between 30 and 50 psig while purging nitrogen from the pressurizer to maintain a solid condition in the reactor coolant system hot legs.

5.14 With a steam bubble in the pressurizer, Quench Tank level shall be maintained > 4000 gallons and < 8300 gallons to provide sufficient quench-cooling volume for pressurizer transients.

5.15 Do not allow Quench Tank pressure to approach bursting point of its Rupture Diaphragm (PSE-1051), 100 psig.

5.16 Opening the ERV causes a localized steam release at the pilot vent valve. This is a radiation and safety hazard.

5.17 Adding N_2 to the Quench Tank via RB N_2 Supply Penetration Isol (N_2-47) circumvents the double-valve isolation design of the penetration. When RCS is ≤ 200°F, Containment Closure Control, Attachment G of Decay Heat Removal and LTOP System Control (1015.002) applies.

5.18 To maintain the RCS in a non water solid condition, TS 3.4.11 Bases limits pressurizer level with RCS temperature <272°F and the reactor vessel head in place to the following:

- ≤105" at RCS pressures >100 psig
- ≤150" at RCS pressures ≤100 psig

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

QID: 0562 **Rev:** 0 **Rev Date:** 4/5/05 **Source:** Direct **Originator:** J.Cork

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

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

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

Description: Knowledge of the bus power supplies to the following: CCW pump, including emergency backup.

K/A Number: K2.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	K

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

Which of the following identifies the correct power supplies to the Intermediate Cooling Water Pumps (P-33A, P-33B, P-33C)?

A. P33A and P33B are powered from B-12 while P33C is powered from B-22.

B. P33A is powered from B-12 while P33B and P33C are powered from B-22.

C. P33A, P33B and P33C are powered from B-12, B-22 and B32 respectively.

D. P33A, P33B and P33C are powered from B-11, B-12 and B-13 respectively.

Answer:

B. P33A is powered from B-12 while P33B and P33C are powered from B-22.

Notes:

"B" lists the correct power supplies, the other choices do not.

References:

STM 1-43, Rev. 8

History:

Direct from regular exam bank QID#4674
Selected for 2005 RO exam

Obj. 5

To allow P-33B to supply system flow for either loop, a suction and discharge cross-connect valve is provided for P-33B. System operating pressure differences will cause leakage through the closed discharge cross-connect. This would cause the surge tank with the lowest pressure to overflow and drain to the floor drain if the ICW Surge Tank Cross Connect Islolation Valve (ICW-165) was not opened. Additional information on P-33B suction and discharge cross connect valves will be covered in the following section.

2.4 ICW PUMPS

(Refer to Figure 43.01)

Obj. 6

The three ICW pumps, P-33A/B/C are used to supply cooling water to components cooled by the ICW system. Net positive suction head is provided by the ICW surge tanks, discussed in the previous section. The ICW pumps are located in the Main Chiller room, elevation 354' of the Turbine Bldg.

Each ICW pump is a single-stage, centrifugal pump rated for 2500 gpm with a discharge head of 125 feet. The ICW pumps rotate at 1750 rpm driven by a 100 HP motor rotating at 1775 rpm. Power to the ICW pumps is provided from non-vital MCC's, B12 and B22. ICW pumps are controlled using handswitches located on panel C09. Associated MCC, breaker and HS for each pump are listed below:

Pump	Handswitch	MCC	Breaker
P-33A	HS-2230	MCC-B12	B-1264
P-33B	HS-2231	MCC-B22	B-2214
P-33C	HS-2232	MCC-B22	B-2264

The three pumps are connected in parallel with two normally in operation and one in standby. Suction and discharge cross-connect lines allow for any two of the three ICW pumps to be in service supplying cooling water flow to the two ICW loops. Both the suction and discharge cross-connect lines utilize air-operated butterfly valves to provide system / loop separation. Each air-operated butterfly valve is provided air from the Instrument Air system through a pressure control valve and solenoid valve. The solenoid valves powered from Y02 breaker 8.

The suction and discharge cross-connect valves are controlled by handswitches located on panel C09 in the control room. Suction and discharge cross-connect valves are interlocked with the pump handswitches and discharge pressure switches for re-alignment on standby pump auto-start.

The two suction cross-connect isolations are 12-inch butterfly valves while the discharge cross-connect line utilizes two 10-inch butterfly valves. P-33B suction supply line and pump discharge line tie into the system between the two suction and discharge cross-connect isolation valves.

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

QID: 0072 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-AOP **Objective:** 4.2 **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:
PORV failures.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 3

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

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

Given:

- RCS pressure 1950 psig and decreasing
- "RELIEF VALVE OPEN", K09-A1, in alarm
- ERV indicates closed
- Acoustic monitor indicates ERV is leaking

You immediately close ERV Isolation valve, CV-1000.
RCS pressure continues to drop with all PZR heaters ON.

What should your next action be?

a. Begin plant runback

b. Cycle the ERV

c. Initiate full HPI

d. Trip the reactor

Answer:

d. Trip the reactor

Notes:

RCS pressure continuing to decrease following isolation of the ERV requires a reactor trip per AOP actions, therefore (d) is the only correct response.

References:

1203.015 Chg. 011-00-0

History:

Developed for the 1998 RO/SRO Exam.
Selected for 2005 RO exam.

SECTION 1 -- ELECTROMATIC RELIEF VALVE (PSV-1000) SYSTEM FAILURE OR LEAK

1.0 SYMPTOMS

One or more of the following:

- SPDS alarm on T1025 at 200°F: ERV PSV-1000 OUTLET TEMP
- Quench Tank (T-42) temperature, level, or pressure rising.
- Rise on acoustic Relief Valve Monitor (VYI-1000A).
- Annunciator alarm RELIEF VALVE OPEN (K09-A1).
- ERV indicates open on C04.
- Pressurizer ERV Isolation Valve (CV-1000) inoperable.
- ERV (PSV-1000) inoperable.
- Failure of both ERV acoustic monitors.

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

NOTE

CV-1000 torque switch can be overridden in the OPEN or CLOSE direction by holding the hand switch in the respective position.

3.1 Close Pressurizer ERV Isolation Valve (CV-1000).

3.2 If ERV leakage with CV-1000 closed exceeds capability to maintain RC pressure, trip reactor and refer to Emergency Operating Procedure series (1202.XXX).

3.3 If closing CV-1000 stops leak, perform the following:

3.3.1 Continue power operations with ERV isolated.

3.3.2 Notify Ops Manager.

3.3.3 Log in station log and on plant status board.

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

QID: 0306 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J. Cork

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

Section: 3.7 **Type:** Instrumentation

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

Description: Knowledge of the operational implications of the following concepts as they apply to the RPS: DNB.

K/A Number: K5.01 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2.5
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	C

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

Which of the following conditions would result in the Reactor Protection System initiating a reactor trip designed to protect the fuel clad from DNB?

a. Ejected rod accident during startup

b. Loss of both Main Feedwater Pumps at 100 % power

c. Boron dilution accident while operating at 100% power

d. Reactor Coolant Pump trip at 95% power

Answer:

d. Reactor Coolant Pump trip at 95% power.

Notes:

"a" is incorrect because it will result in a high power trip. The high power trip is intended to prevent damage from fast reactivity excursions.
"b" is incorrect because it results in a high pressure trip and protects RCS piping.
"c" is incorrect because reactivity addition rate is slow enough that ICS should compensate for the reactivity addition.

References:

Technical Specifications 3.3, Bases

History:

Developed for 1999 exam.
Selected for 2005 exam

APPLICABLE SAFETY ANALYSES (continued)

the accident analysis calculations for small break loss of coolant accidents (LOCAs). The Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because either the trip will actuate prior to degraded conditions being reached or the equipment response will be conservative.

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to the system parameters of pressure and temperature exceeding the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the reactor outlet temperature expressed in degrees Fahrenheit within the range specified by the Reactor Outlet High Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure Allowable Value is selected to initiate a trip prior to temperature and pressure exceeding the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the unit safety analysis. Therefore, the Allowable Value does not account for errors induced by a harsh RB environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. Even in the case where this trip is a backup for other RPS trips for LOCA or MSLB, it is assumed to occur before degraded building conditions have an appreciable effect on RB High Pressure trip components. Therefore, determination of the Allowable Value does not account for errors induced by a harsh environment.

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

APPLICABLE SAFETY ANALYSES (continued)

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing may be insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline temperature SLs.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides protection for the loss of a pump or pumps which would result in both pumps in a single steam generator loop being tripped.

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least one RCP is operating in each loop. RCP status is monitored by power transducers associated with each pump. These relays indicate a loss of an RCP on underpower. The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. <u>Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE</u>

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the reactor core SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the limiting loss of flow transient which is the loss of two RCPs from four pump operation. The imbalance portion of the trip is credited for steady state protection only.

The power to flow ratio of the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system is operating with two or three pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs prior to core power, axial power peaking, and reactor coolant flow conditions reaching DNB or fuel centerline temperature limits. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

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

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

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

Section: 3.7 **Type:** Instrumentation

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

Description: Ability to manually operate and/or monitor in the control room: Bistanble trips, reset and test switches.

K/A Number: A4.04 **CFR Reference:** 41.7 / 45.5 to 45.8

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

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

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

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

A. Prevent that channel from tripping.

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

C. Lock out the other channels' test switches.

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

Answer:

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

Notes:

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

References:

STM 1-63, Rev. 6

History:

Direct from regular exam bank QID#1999
Selected for 2005 RO exam

2.4 Special Design Features

Special design features have been provided within the reactor protection system to allow for module removal / in test, shutdown bypass capability and high startup rate rod withdrawal inhibit.

2.4.1 Module Removal / In Test Interlock

(Refer to Figure 63.22)

The module removal / in test interlock trip relay is provided to initiate a trip signal when either of two circumstances occur:

- Removal of a critical module
- Placing a test module in other than the operate position

Removal of a module breaks the continuity of the module interlock circuit. Each critical reactor protection system module has an internal jumper in series with the internal jumper of the next module in the module interlock string. When all vital modules are plugged in, the circuit will be completed from a -15 volt bus to the reactor trip module test / interlock trip relay.

If a test switch is placed in other than operate, the test / interlock trip relay will be de-energized. The test switches of critical test modules are wired in parallel with the coil of the relay so that, when a switch is placed in a "test" position, the relay coil is shorted out.

The module test / interlock relay for each channel is located in the reactor trip module. The relay operates a contact that de-energizes the associated channel trip relay when open.

2.4.2 High Startup Rate Hold Interlock

(Refer to Figure 63.03 and 63.23)

To limit the rate of reactivity addition to the core, a high startup rate rod hold is developed through the reactor protection system and supplied to the control rod drive system. This occurs when the rate of change of reactor power exceeds two decades per minute in the source range or three decades per minute in the intermediate range. The circuitry consists of bistable contacts and contacts operated by relays in auxiliary relay modules.

Four contacts in parallel provide the rod hold signal, if needed and if not inhibited by other contacts in the circuit. One contact is provided for each source range instrument and one contact for each intermediate range instrument. Any of these four contacts will provide the hold signal when closed. The source range contacts are auxiliary relay actuated by a remote signal from the source range instrumentation while the intermediate range contacts are bistable actuated.

The source range high startup rate hold circuit can be disabled by two means. The first will occur when the contact in series with the source range High SUR opens when intermediate range detectors indicate ~ 1×10^{-9} amps. This disable feature is provided by a bistable in each of the intermediate range detectors. Both intermediate range detectors bistables must open for the source range high SUR to be disabled. A backup signal is provided in case a failure of a bistable to trip or intermediate range detector failure occurs. This backup feature will operate the same contacts if the following logic is satisfied by the power range instruments:

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

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

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

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

System Number: 013 **System Title:** Engineered Safety Features Actuation System(ESFAS)

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

K/A Number: K1.06 **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:** 4.4 **SRO Select:** No **Taxonomy:** C

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

Given:

- All controls are in automatic
- RCS pressure 1255 psig, slowly dropping
- Reactor Building pressure is 49 psia
- "A" and "B" OTSG levels at 415 inches

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

a. EFW pumps and LPI pumps

b. RB spray pumps and LPI pumps

c. RB spray pumps and EFW pumps

d. HPI pumps and RB spray pumps

Answer:

d. HPI pumps and RB spray pumps

Notes:

"a" is incorrect because RCS pressure exceeds the shutoff head of the LPI pumps.
"b" is incorrect because RCS pressure exceeds the shutoff head of the LPI pumps.
"c" is incorrect because OTSG level is above the level for reflux boiling.
"d" is correct

References:

1105.003, Chg. 011-02-0
STM1-65, Rev. 3

History:

Used in 1999 exam.
Direct from ExamBank, QID# 1780 used in class exam
Selected for 2005 RO exam

3.5 Summary of ESAS trips and functions:

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

4.0 REFERENCES

4.1 REFERENCES USED IN PROCEDURE PREPARATION

4.1.1 Unit 1 Technical Specifications

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

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

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

4.1.5 Integrated ES System Test (1305.006)

4.1.6 CR-1-96-0359 ESAS Setpoint Change

4.1.7 DCP 95-5010-D101 SW Valve Replacement

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

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

4.2 REFERENCES USED IN CONJUNCTION WITH THIS PROCEDURE

4.2.1 Plant Startup (1102.002)

4.2.2 Plant Shutdown and Cooldown (1102.010)

4.2.3 Decay Heat Removal Operating Procedure (1104.004)

4.2.4 Emergency Feedwater Initiation and Control (1105.005)

STM 1-65

Engineered Safeguards Actuation System

1.0 INTRODUCTION AND DESCRIPTION

This System Training Manual provides information on the Engineered Safeguards Actuation System. It provides a system description, component description; operational response and other system related material, as well as interrelations with other systems. Respective system STM's provide detailed information on the other systems and their associated components.

2.0 SYSTEM FUNCTIONS

2.1 PRIMARY FUNCTIONS

The Engineered Safeguards Actuation System (ESAS) monitors two parameters for indications of a major transient that challenges fuel cladding integrity and reactor building integrity. The system design transient is a loss of coolant accident (LOCA).

1. State the functions of the Engineered Safeguards Actuation System

The parameters monitored are reactor coolant system (RCS) pressure and reactor building (RB) pressure. During a LOCA, these two parameters provide early and positive evidence of the transient. ESAS would then actuate engineered safeguards (ES). Engineered safeguards are high quality, highly reliable components, systems and equipment intended for mitigation of a loss of coolant accident.

ESAS has adequate redundancy to ensure the integrity of the reactor building. This serves to keep public exposure below the federal limits of 10 CFR 100

Actuation of engineered safeguards for a LOCA provides:

- Protection of the fuel cladding by injection of coolant at both high and low pressures.
- Maintenance of the reactor building integrity through building isolation and cooling of the building atmosphere.
- Control of fission products in the RB atmosphere; accomplished using RB spray. (As well, the spray serves cool the RB atmosphere to limit and reduce RB pressure.)

2. State the ESAS trip setpoints.

The low RCS pressure setpoint is 1590 psig. Actuation of systems to inject coolant into the reactor to protect the fuel occurs at this setpoint. One of the two RB pressure setpoints is at 4 psig (18.7 psia). Actuation of injection of coolant into the

reactor results at the 4-psig RB setpoint as well. Isolation of some reactor building penetrations takes place at either 4-psig RB pressure or 1590 RCS pressure. The term used to describe these components is "diverse containment isolation" (See Discussion box 65-1).

In addition, exceeding 4 psig results in:

- Isolation of RB penetrations to contain the radioactive fluids released by the break.
- Operation of components to provide emergency cooling to the reactor building.

Diverse Containment Isolation-

Releases from the reactor building to the auxiliary building occurred during the accident at Three Mile Island Unit 2. The ESAS design at TMI called for isolation of the release flowpaths on high RB pressure alone. ESAS actuated during the accident, but only on low RCS pressure. The potential existed for similar release paths under similar circumstances for ANO Unit 1. This could have occurred during a small break loss of coolant accident where reactor building pressure did not exceed the 4-psig (18.7 psia) setpoint. To alleviate this problem, some components actuated by digital channels 5 or 6 (actuated on high building pressure only) were moved to digital channels 1 through 4 (actuated by either low reactor coolant system pressure or high reactor building pressure).

Discussion Box 65-1:

Reactor building spray actuation (including sodium hydroxide injection) occurs at the second high reactor building pressure setpoint of 30 psig (44.7 psia).

2.1.1 Design Requirements Affecting Function

General Design Criteria 20 requires installation of a system capable of sensing accident conditions in addition to actuating safety systems and their associated components to ensure fuel design limits are not exceeded. ESAS is that system, actuating the emergency core cooling systems (ECCS) at the applicable RCS pressure or RB pressure setpoint.

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

QID: 0564 **Rev:** 0 **Rev Date:** 4/7/05 **Source:** New **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:** Yes **Difficulty:** 3

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

Question: **RO:** 41 **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)?

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. Declare P-35B Spray pump inoperable and refer to T.S. 3.6.5.

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 plant is in cold shutdown and RB spray pumps are not required per T.S. Regardless, there is not enough information to determine operability.

References:

1203.012J, Chg. 035-00-0
1305.006, Chg. 020-04-0

History:

New for 2005 RO exam

Location: C18

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 is open, perform the following:

   A. If RB Press recorder on C18 indicates >30 psig, verify that ES CH 7 is actuated and observe that RB Spray Pump (P-35A) has started.

   B. Determine cause of P-35B failure.

2. Check RB Spray P-35B Flow gauge on C16. If flow is low, perform the following:

   A. If RB Press recorder on C18 indicates >30 psig, verify that ES CH 7 is actuated and observe 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. To clear alarm, perform either of the following:

   - Clear and reset ES CH 8.

   - Close breaker A-404 and raise RB spray flow to >1050 gpm.

2.0 PROBABLE CAUSES

1. Pump P-35B failure to auto start

3.0 REFERENCES

Schematic Diagram Annunciator K11 (E-461, sheets 1-3)

8.6 Align Service Water Pump (P-4B) to operate from bus A4 as follows:

8.6.1 Verify P-4B is off. _______

8.6.2 Verify or place P-4B BUS SELECT MOD CONTROL in MOD TO BUS A4 position. _______

8.6.3 Verify MOD to bus A4 is closed (may require manual operation of MOD). _______

8.6.4 Start P-4B to verify operation. _______

8.6.5 Stop P-4B AND leave hand switch in NORMAL-AFTER-STOP. _______

8.7 Align RB Spray Pump (P-35B) for start in recirc mode as follows:

8.7.1 Verify Test & Recirc Header <u>not</u> in use. _______

8.7.2 Verify following valves closed:

- P-35A Disch to DH Recirc & Test Line (BS-2A) _______
- P-34A Disch to DH Recirc & Test Line (DH-8A) _______
- P-34B Disch to DH Recirc & Test Line (DH-8B) _______
- SF System Disch to DH Recirc. & Test Line (SF-38) _______

8.7.3 Position the following valves as indicated:

A. Unlock and open P-35B Disch to DH Recirc & Test Line (BS-2B). _______

B. Open RB Spray Sys Disch to DH Recirc & Test Line (BS-3). _______

C. Open Throttle Valve Bypass around DH-10 (DH-9). _______

8.7.4 Verify RB Spray Block Valve (CV-2400) is closed. Hold hand switch in CLOSE position for at least 10 seconds after valve indicates closed. _______

A. At MCC B61 open breaker for CV-2400, (B-6171). _______

NOTE

A Partial Clearance will be required for RB Spray NaOH Addition, T-10 Outlet (CV-1617), to allow performance of the next step.

8.7.5 Close breaker for CV-1617 (B-6193). _______

8.7.6 If HPI is not required for RCS inventory makeup per Decay Heat Removal and LTOP System Control (1015.002)
AND
RCS level is less than BWST level, unlock and close BWST Supply to MU Pump P-36C Suction (BW-2). ______

8.7.7 If BW-2 is left open, verify HPI Pump Recirc Block Valve (CV-1301) closed to prevent gravity fill of MU Tank (T-4) from BWST. ______

8.7.8 Open BWST Outlet (CV-1408). ______

8.7.9 Open P-35B Vent (ABV-12B) as necessary to fully vent P-35B casing. ______

8.7.10 When venting is complete, close ABV-12B. ______

8.7.11 Vent P-35B discharge piping as follows:

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. ______

8.7.12 Start RB Spray Pump (P-35B). ______

8.7.13 Adjust DH-9 to obtain ~1500 GPM spray flow. ______

A. If necessary to obtain desired flow, throttle open DH Test & Recirc Isol (DH-10). ______

8.7.14 Stop P-35B AND leave hand switch in NORMAL-AFTER-STOP. ______

8.7.15 Close CV-1408. ______

8.8 Align DH Pump (P-34B) for start in DH mode as follows:

8.8.1 Close P-34B Suction From BWST (CV-1437). ______

8.8.2 Open P-34B Suction From RCS (CV-1435). ______

8.8.3 Unlock and close B DH Cooler SW Outlet Isol (SW-22B). ______

8.8.4 Verify LPI Block Valve (CV-1400) closed. ______

8.8.5 Verify Decay Heat Cooler Outlet (CV-1429) open. ______

8.8.6 Verify Decay Heat Cooler Bypass (CV-1432) closed. ______

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

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:**	No	**Taxonomy:**	K

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

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 7 & 8, 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 1 & 2, 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 return 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. 6

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO 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-D) has a single blade, butterfly damper (CV-7470-7473) at the discharge of the fan that opens when the fan starts. These 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 (VALIDATION)

QID: 0444 **Rev:** 0 **Rev Date:** 5/1/2002 **Source:** Repeat **Originator:** S.Pullin

TUOI: A1LP-WCO-RBS **Objective:** 1 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray

Description: Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Iodine scavenging via the CSS.

K/A Number: K4.06 **CFR Reference:** 41.7

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

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

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

In post accident conditions, the RB Spray System (both trains) will provide what percentage of the required RB cooling and iodine removal?

a. 100% Cooling/100% Iodine

b. 200% Cooling/100% Iodine

c. 100% Cooling/200% Iodine

d. 200% Cooling/200% Iodine

Answer:

c. 100% Cooling/200% Iodine

Notes:

References:

STM 1-08, Rev. 8

History:

Direct from regular exambank QID 1012.
Selected for use in 2002 RO exam.
Repeated for use in 2005 RO exam

on site electric power system operation (assuming off site power is not available) the system safety function can be accomplished, assuming a single failure. Single failure criteria is simply: no single failure will cause or prevent a protection system from fulfilling its design function.

The Reactor Building Spray and Emergency Cooling Units provide cooling to the RB atmosphere following a LOCA event. One Emergency Cooling Unit consists of two RB cooling fans (VSF-1A-1D) and their associated SW cooling coils (VCC-2A-D). VSF-1A & VSF-1B and their associated cooling coils (VCC-2A & VCC-2B) makeup one unit while VSF-1C & VSF-1D and their associated cooling coils (VCC-2C & VCC-2D) makeup the other. Both emergency cooling units have a combined heat removal capability of 100%. The two separate trains of RB Spray together are capable of providing 100% of the design cooling required from the system.

The following combinations are allowed by Tech Spec 3.3 for RB heat removal.

- Both Trains of RB Spray.
- Both Emergency Cooling Units.
- One RB Spray and one Emergency Cooling Unit.

2.8 RB Spray System Design

The Reactor Building Spray system has two major functions. The first of these functions is to reduce post accident containment temperature and pressure to nearly atmospheric. By reducing the pressure and temperature, the driving force for leakage will be reduced and thereby stay below 10CFR100 limits at the site boundary during a Design Based Accident (DBA). 10CFR100 limits are based on a person located at the site boundary for a two-hour period immediately following a LOCA event. The limits are 300 Rem to the Thyroid & 25 Rem Whole body.

The accident that would result in the highest containment pressure is a $5ft^2$ hot leg rupture. The DBA for the Reactor is a 14-ft2 rupture of the hot leg. The $5ft^2$ hotleg rupture results in a higher containment pressure due to the RCS will blowdown to the RB atmosphere for a longer period of time.

The second function of the Reactor Building Spray system is to remove iodine from the containment atmosphere after a loss of coolant accident that raises containment pressure to 30 psig. Iodine released from damaged fuel to the containment atmosphere during a LOCA could be released to the outside environment if it was not removed, especially during the elevated containment pressure of a post accident condition. Iodine released to the atmosphere has a tendency to be absorbed in the Thyroid. Reducing the iodine level greater reduces the thyroid dose to plant personnel and the general public should a containment breech occur.

Each individual Reactor Building Spray train will provide 100% of the design iodine removal capability. Iodine has a strong affinity for water and is "stripped" from the containment atmosphere when the Reactor Building Spray system is operating. To enhance the

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

QID: 0314 **Rev:** 0 **Rev Date:** 9/5/99 **Source:** Direct **Originator:** J Cork

TUOI: A1LP-RO-EFIC **Objective:** 31 **Point Value:** 1

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

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

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main Steam pressure.

K/A Number: A1.06 **CFR Reference:** 41.5 / 45.5

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

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

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

Given:

- A loss of offsite power
- No failures exist other than those which caused the loss of offsite power condition
- EDG's supplying vital buses

Ten (10) minutes into this event at what pressure will the OTSG's be controlled?

a. 895 psig

b. 995 psig

c. 1020 psig

d. 1050 psig

Answer:

c. 1020 psig

Notes:

Following a loss of offsite power, condenser vacuum will be non-existent and the Atmospheric Dump Valves will be used to control OTSG pressure at the nominal setpoint of 1020 psig, therefore "c" is correct.
"a" is incorrect, this is the normal setpoint for the Turbine Bypass Valves.
"b" is incorrect, this is the setpoint for the Turbine Bypass Valves with the 100 psig bias applied (to limit cooldown) following a Reactor Trip.
"d" is incorrect, this is the lowest setpoint for a Main Steam Safety Valve.

References:

1105.005, Chg. 027-02-0

History:

Developed for 1999 exam.
Selected for 2005 RO exam.

6.0 SETPOINTS

6.1 Initiation Setpoints

- EFW low level initiate ~ 13.5". (delayed 2 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.

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

QID: 0565 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** Modified **Originator:** J.Cork

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

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

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

Description: Knowledge of the physical and/or cause/effect relationships between the MFW and the following systems: ICS.

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

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

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

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

Given:

- 100% power
- ICS in full automatic

The CBOR places the ICS Delta T-Cold Hand Auto Station meter selection switch in "POS" (position). The meter reads 46%.

What does this mean in terms of ICS control of main feed water?

A. The average of feedwater loop A and feedwater loop B demand is 46%.

B. Feedwater loop B demand is greater than feedwater loop A demand.

C. The feedwater loop A demand is being boosted by a 4 °F Delta T-Cold error.

D. Feedwater loop A demand is greater than feedwater loop B demand.

Answer:

B. Feedwater loop B demand is greater than feedwater loop A demand.

Notes:

A reading <50% indicates that loop B demand is > loop A demand, therefore (B) is the correct response. (A) is incorrect because the meter does not indicate average demand, (D) is an opposite response, (C) applies to looking at the MV reading (for which it would still be incorrect).

References:

STM 1-64, Rev. 9

History:

Developed for the 1998 RO/SRO Exam.
Selected for use in 2002 RO/SRO exam.
QID #63 used on 2004 RO/SRO Exam.
Modified for 2005 RO exam.

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 meter. This provides the ΔTc setpoint for automatic operation. The setpoint may be varied from 0% to 100% which corresponds to -10°F to +10°F. The normal value is 50% (0°F).

When position is selected on this station, the ΔTc correction signal is indicated on the meter. If the meter indicates 50%, the correction signal is zero (loop "A" multiplier set at .5) and loop demand signals are equal. If the meter indication is above 50%, then loop "A" demand is > loop "B" demand. The opposite is true if the indication is < 50%.

When measured variable is selected on this station, the difference between the actual ΔTc and the ΔTc setpoint (ΔTc error) is indicated on the meter. ΔTc = "A" Loop Tc - "B" Loop Tc. The meter scale is ±10°F. Positive reading means that "A" loop is hotter. A bumpless transfer from hand to auto may take place when the ΔTc error equals zero (50% on meter). If the ΔTc does not equal zero, adjustment to zero may be accomplished by adjusting the manual output of the station or by changing the ΔTc setpoint.

If both loop demand stations are placed in hand, this station rejects to hand and can not be placed in auto.

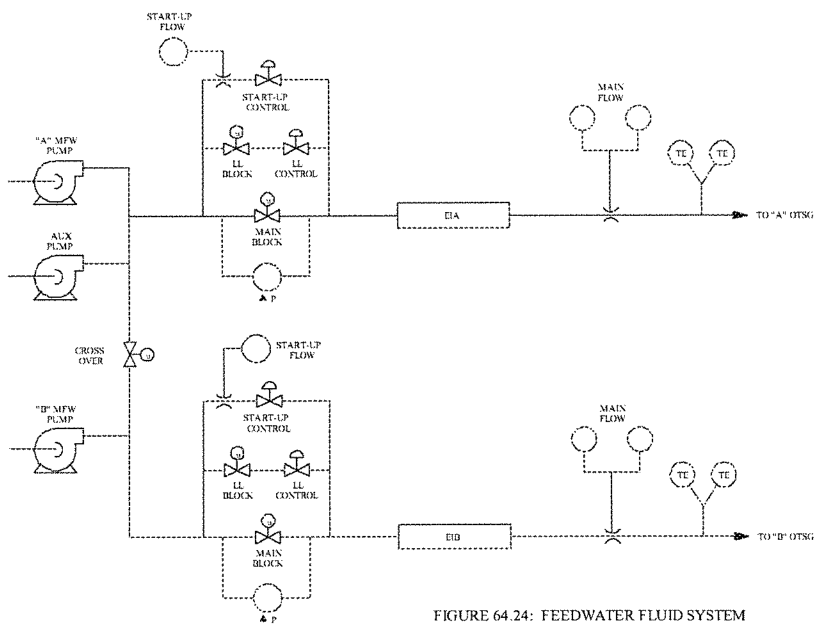


FIGURE 64.24: FEEDWATER FLUID SYSTEM

2.6.3 Feedwater Flow Control

The method of flow control used by the feedwater system is dependent upon the plant power level. (refer to figure 64.24) At low power feedwater flow is controlled by the startup and low load control valves with the main feedwater block valve shut and the feedwater pumps operating to maintain 70 psid across the feed valves. The valves are sequenced into operation so that the startup valve opens first followed by the low load control valves then the main FW block valves. As plant load is increased, feedwater flow control will be shifted from the valves to the pumps. This is

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

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: Turbine driven feed pump.

K/A Number: A3.04 **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:** No **Taxonomy:** K

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

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. 8

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO exam

2.3.1 Instrumentation

Feed pump suction pressures, temperature and flow instrumentation is provided for indication, alarms and control features associated with operation of P-1A (B). MFP suction temperature is sent to the plant computer for indication only. Suction temperature for P-1A is provided by TE-2840 and TE-2832 provides P-1B indication.

2.3.1.1 Suction / Recirc Flow Indication

(Refer to Figure 19.04)

Feedwater suction flow is used to control the MFP recirc control valve to maintain a minimum flow through the pump. Low flow through the MFP should only occur when placing the MFP in service. Flow element FE-2843 and flow transmitter FT-2833 provide a flow signal to the recirc valve controller. Recirc flow or suction flow can be read on the recirc valve controller. Recirc flow control valve CV-2874 provides additional controlling functions besides minimum flow through the MFP. Recirc valve operation, controller and controlling functions will be covered in the discharge header discussion section.

Low suction flow through the MFP or low recirc flow alarm condition is provided by flow switch FS-2843. Annunciator K07-E7 "A MFP Flow Lo" will alarm when suction flow indicates less than 1600 gpm. P-1B flow instrumentation and recirc flow CV information is provided in the following table.

Flow Element FE-2832	Flow Transmitter FT-2834
Flow Switch FS-2832 (K07-E8)	Recirc Valve CV-2876

2.3.1.2 Suction Pressure Indications

(Refer to Figure 19.04)

Feed pump suction pressure is used for indication, controlling functions, alarms and MFP trip signal. Suction pressure is sent to the plant and SPDS computers by PT-2842. MFP suction pressure local indication, PI-2842 is provided on rack 21 near E-1 FW heaters.

MFP gland seal cooling system uses suction pressure and cooling water supply pressure to maintain desired differential pressure between gland seal cooling and suction pressure. Additional information will be provided in the MFP section covering the gland seal cooling system.

MFP suction pressure trip signals and alarms are provided by PS-2841 & PS-2842. PS-2842 provides annunciator alarms for Lo and Lo-Lo suction pressure conditions. K07-C7 "A MFP Suct Press Lo" will alarm when PS-2842 indicates suction pressure less than 280 psig. PS-2842 will reset when pressure indicates greater than ~ 320 psig.

K07-B7 "A MFP Suct Press Lo-Lo" will alarm when PS-2842 indicates suction pressure $\leq$ 230 psig for greater than 5 seconds. Reset pressure for Lo-Lo- alarm is ~ 255 psig. Lo-Lo pressure signal from PS-2842 provides one of the two suction pressure trip signals required to trip the MFP.

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.

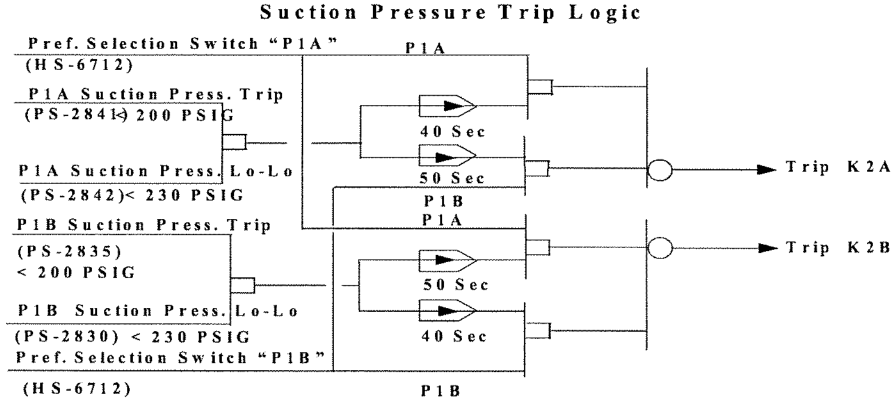
2.3.1.3 Suction Pressure Trip Logic

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 (VALIDATION)

QID: 0566 **Rev:** 0 **Rev Date:** 5/1/05 **Source:** New **Originator:** S.Pullin

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

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

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

Description: Knowledge of bus power supplies to the following: AFW electric drive pump.

K/A Number: K2.02 **CFR Reference:** 41.7

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

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

Given:
- A Degraded Power condition exists on both units.
- #1 EDG did not start.
- P-7A tripped and will not reset.

Why does the EOP direct cross-tieing of the A3 and A4 buses?

A. To run two HPI pumps for HPI cooling per RT-4.

B. To have atmospheric dump control on both SGs.

C. To start EFW Pump P-7B.

D. To restore Instrument Air for ADV control.

Answer:

C. To start EFW Pump P-7B.

Notes:

"C" is correct since P-7B is powered from A3 and power must be restored to regain EFW.
"A" is incorrect since one HPI pump can be the source of HPI cooling.
"B" is incorrect, A3 must have power to restore power to "A" ADV isolation, this is not the purpose of this step.
"D" is incorrect, ADVs are placed in manual to conserve air and have air reservoirs for this purpose. Recovering Inst. Air is a concern but this is not the key purpose for cross-tieing in this condition.

References:

1202.007, Chg. 006-01-0

History:

New for 2005 RO exam.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

This section is for correcting overheating.

53. IF any of the following criteria is met before overheating is corrected, THEN GO TO step 55.

- ERV opens in AUTO
- RCS press ≥2450 psig
- RCS press approaches NDTT Limit (Figure 3)

54. Re-verify proper EFW actuation and control (RT 5).

54. IF EFW fails to actuate, THEN perform the following:

A. Place EFW CNTRL valves in HAND **AND** close:

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

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

C. Place EFW Pump P7B in PULL-TO-LOCK.

D. IF A3 is de-energized **AND** P7A is unavailable, **THEN** restore power to P7B as follows:

1) Energize A3 using ES Electrical System Operation (1107.002).

a) IF another DG **OR** Off-site power becomes available, **THEN** restore buses to normal using 1107.002.

2) Start P7B **AND GO TO step 62.**

E. Dispatch an operator to restore EFW using Emergency Feedwater Pump Operation (1106.006).

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

QID: 0140 **Rev:** 0 **Rev Date:** 05/13/93 **Source:** Direct **Originator:** J. Haynes

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

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** A. C. Electrical Distribution

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the AC Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect on plant of de-energizing a bus.

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

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

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

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

Initial conditions:

- 100% power
- P-36C is the operating makeup pump
- ICW pumps P-33A and P-33C in service

What RCP support system would be most affected by a loss of bus A4?

a. Seal Injection

b. Motor Cooling

c. Seal Bleedoff

d. Oil Lift Pressure

Answer:

a. Seal Injection

Notes:

(a) is correct. Loss of A4 results in a loss of the running HPI pump.
(b) is incorrect. P-33 will remain in service which provides motor cooling.
(c) & (d) are incorrect. Seal bleedoff is not affected by the loss of A2 nor is RCP lift oil pressure.

References:

1203.026, Chg. 009-05-0

History:

Taken from Exam Bank QID # 3714
Used in A. Morris 98 RO Re-exam
Selected for use in 2005 RO exam.

SECTION 1 -- LOSS OF HPI PUMP

1.0 SYMPTOMS

1.1 Annunciator alarms:

- HPI PUMP TRIP (K10-A6)
- RCP SEAL INJ FLOW LO (K08-A7)
- MU TANK LEVEL HI/LO (K10-B7)
- MU TANK PRESS HI/LO (K10-B8)

1.2 Loss of or erratic makeup flow and seal injection flow.

1.3 Loss of or erratic makeup (HPI) pump discharge header pressure.

2.0 IMMEDIATE ACTION

2.1 None.

3.0 FOLLOW-UP ACTIONS

> **NOTE**
> Indications of loss of HPI suction are:
> - Erratic flow, and
> - Erratic discharge pressure, and
> - Control valves stable

3.1 IF HPI pump has lost suction,
THEN stop the HPI pump.

3.2 Isolate letdown by performing either of the following:

- Close Letdown Coolers Outlet (CV-1221),
- Close Letdown Cooler Outlets (RCS) (CV-1214 and 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.

3.3 Verify RC pump seals are being cooled by ICW.

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

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

QID: 0567 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** New **Originator:** S.Pullin

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

Section: 3.6 **Type:** Electrical

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

Description: Ability to manual operate and/or monitor in the control room: all breakers (including available switchyard).

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

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

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

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

Given:
- Loss of breaker position indicating lights for plant buses on left side of C10.
- "Trip Solenoid Power Available" light on C01 not lit.

What associated annunciators would alarm with this condition?

A. D01 UNDERVOLTAGE (K01-A7)

B. DO2 UNDERVOLTAGE (K01-A8)

C. D21 LOSS OF VOLTAGE (K01-B8)

D. TURBINE TRIP (K04-A3)

Answer:

A. D01 UNDERVOLTAGE (K01-A7)

Notes:

"A" is correct, D01 supplies left side of C10 indicating lights and Turbine trip solenoid power.
"B" and "C" are on the opposite electrical train.
"D" will not alarm since generator lockout relays will not open output breakers, generator will motorize, and turbine trip solenoid is de-energized. EOS will not trip turbine since generator is motoring and thus will not overspeed.

References:

1203.036, chg. 005-04-0

History:

New for 2005 RO exam.

DISCUSSION (In back of procedure)

SECTION 1 -- **Loss of D01**

1.0 SYMPTOMS

1.1 Low DC voltage alarms:
(alarms inoperable if generator output breakers open).

- D01 UNDERVOLTAGE (K01-A7)
- D11 LOSS OF VOLTAGE (K01-B7)
- RA1 LOSS OF VOLTAGE (K01-C7)
- D01 TROUBLE (K01-D7)

- H1 DC CONTROL POWER OFF (K02-B4)
- A1 DC CONTROL POWER OFF (K02-C6)
- A3 DC CONTROL POWER OFF (K02-D6)
- SU 1 L.O. RELAY DC FAILURE (K02-D1)

- SU 2 L.O. RELAY DC FAILURE (K02-E3)
- GENERATOR L.O. RELAY DC FAILURE (K04-D8)
- TURBINE L.O. RELAY DC FAILURE (K04-B5)
- EOS SYSTEM TROUBLE (K04-C5)

1.2 Loss of breaker position indicator lights for plant buses on left side of C10.

1.3 "Trip Solenoid Power Available" light on C01 **not** lit.

2.0 IMMEDIATE ACTION

NONE

3.0 FOLLOW-UP ACTIONS

3.1 At C10, transfer D11 to EMERG SUPPLY D02.

3.2 **IF** reactor trips, **THEN** perform the following:

3.2.1 **IF** SG pressure is <900 psig,
THEN actuate MSLI and EFW for both SGs.

3.2.2 Perform Emergency Operating Procedures (1202.XXX) in conjunction with this procedure.

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

3.4 **IF** transfer of D11 is **not** successful,
THEN attempt local transfer of D11 to D02, while continuing.

3.5 **IF** reactor is **not** tripped, **THEN** **GO TO step 7.0.**

3.6 **IF** transfer of D11 is successful, **THEN** **GO TO step 4.0.**

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

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:** No **Taxonomy:** C

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

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. 012-12-0

History:

Developed for 1998 RO exam
Used in A. Morris 98 RO Re-exam
Selected for use in 2005 RO 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	If in use, place Inverter Y11 on alternate source.
D01-23	Supply to Panel RA1	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. If MSIVs are closed, verify instrument air is isolated.
D01-24	Emer. Supply to Panel D21	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	If in use, place Inverter Y13 on alternate source.
D01-53A	DC Power Supply to Inverter Y15	Loss of Inverter Y15	If in use, place Inverter Y15 on alternate source.

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

QID: 0568 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** New **Originator:** S.Pullin

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

Section: 3.6 **Type:** Electrical

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

Description: Ability to monitor automatic operation of the ED/G system, including: Number of starts available with an air compressor.

K/A Number: A3.04 **CFR Reference:** 41.7 / 45.5

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

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

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

Given:

- #1 EDG has one Air Start Compressor and it's associated Air Receiver Tanks tagged out.
- The remaining Air Start Compressor on #1 EDG trips while running.
- The Air Receiver Tanks' pressure is 176 psig.

What is the maximum number of start attempts assured with the above #1 EDG conditions?

A. One

B. Three

C. Five

D. Seven

Answer:

C. Five

Notes:

"C" is correct per the TS bases, the others are incorrect choices.

References:

TS SR 3.8.3.3 and Bases

History:

New for 2005 RO exam.

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 $\geq$ 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 $\geq$ 175 psig.	31 days
SR 3.8.3.4	Check for and remove accumulated water from each fuel oil storage tank.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.2 (continued)

Following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-81 (Ref. 4) are met for new fuel oil when tested in accordance with ASTM D975-81 (Ref. 4), except that the analysis for sulfur may be performed in accordance with ASTM D1552-90 (Ref. 4) or ASTM D2622-87 (Ref. 4). These additional analyses are required by Specification 5.5.13, "Diesel Fuel Oil Testing Program," to be performed within 31 days following sampling and addition. This 31 days is intended to assure: 1) that the sample taken is not more than 31 days old at the time of adding the fuel oil to the storage tank, and 2) that the results of a new fuel oil sample (sample obtained prior to addition but not more than 31 days prior to) are obtained within 31 days after addition. For circumstances where multiple fuel oil additions are made within a short period of time, the samples taken for each batch added to the storage tank can be composited for a single follow-up analysis. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-88, Method A (Ref. 4). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each tank is considered and tested separately.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.3

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

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

QID: 0089 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

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

Section: 3.7 **Type:** Instrumentation

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

Description: Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint.

K/A Number: K4.01 **CFR Reference:** CFR: 41.7

Tier: 2 **RO Imp:** K4.01 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 52 **SRO:**

Given:

- Treated Waste Monitor Tank, T16A, release in progress
- "PROC MONITOR RADIATION HIGH", K10-B2, in alarm
- Liquid Radwaste Process Monitor, RI-4642, in alarm

What should your Immediate Action be?

a. Verify no flow on Discharge to Flume, FI-4642

b. Trip the running Radwaste Transfer pump, P-53A/B

c. Close Liquid Waste to Flume valve, CV-4642

d. Reset RI-4642 to verify alarm is valid

Answer:

a. Verify no flow on Discharge to Flume, FI-4642

Notes:

Answer (a) is the only immediate action per the AOP. (b) and (d) are follow up actions, (c) is an automatic action.

References:

1203.007, Rev. 8

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam.

1.0 SYMPTOMS

1.1 PROC MONITOR RADIATION HI (K10-B2) annunciated.

1.2 Liquid waste Discharge Flow to Flume (FI-4642) drops off, (C19).

1.3 Alarm on Liquid Radwaste Process Monitor (RI-4642), (C-25, Bay 2).

2.0 IMMEDIATE ACTIONS

None

3.0 FOLLOW-UP ACTIONS

3.1 Verify no release in progress at Disch Flow to Flume (FI-4642) on C19.

3.2 Stop the running radwaste transfer pump (P-53A/B, P-47A/B, or P-45).

3.3 Close Liquid Waste to Flume Valve (CV-4642) if not already closed.

3.4 <u>If</u> CV-4642 can not be closed, close applicable upstream manual valve:

3.4.1 Treated Waste Discharge to Circ. Water Flume (CZ-58).
3.4.2 Filtered Waste Monitoring Tank Disch. to CW Flume (DZ-25).
3.4.3 Laundry Drain Pump P-45 Discharge to Flume (LZ-5).

3.5 Verify proper valve lineup for applicable release path:

3.5.1 Dirty Liq. Waste & Drain Processing (1104.014), Attachment D.
3.5.2 Laundry Waste Processing (1104.015), Attachment B.
3.5.3 Clean Waste System Operation (1104.020), Attachment B.

3.6 Verify proper setting and operation of Liquid Radwaste Process Monitor (RE-4642) per the release permit.

<u>NOTE</u>

1. An unplanned release is defined as the unintended discharge of a volume of liquid or airborne radioactivity to the environment.

2. An expanded "Definition of Unplanned Releases" is contained in several Chemistry procedures, including Analysis of Liquid Waste (1604.017), Attachment 4.

3. Unplanned releases require a condition report and are reportable per ODCM App. 1, L3.2.1.B

3.7 <u>If</u> condition was caused by a problem in step 3.5 <u>or</u> a problem other than an unplanned release, re-establish release.

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

QID: 0211 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** R. Fuller

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

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

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

Description: Knowledge of bus power supplies to the following: Service water.

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

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

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

Given:

- The unit is operating at 100% power.
- Service water pumps P-4A and P-4B are in service.
- A loss of P-4A occurs.

What is required of the operator to restore service water to the required Technical Specifications configuration?

a. Swap P-4B MOD (A6) to the A3 supply and start P-4C.

b. Start P-4C, stop P-4B, swap P-4B MOD (A6) to the A4 supply and start P-4B.

c. Stop P-4B, swap P-4B MOD (A6) to the A3 supply and start P-4B and P-4C.

d. Start P-4C, stop P-4B, swap P-4B MOD (A6) to the A3 supply and start P-4B.

Answer:

d. Start P-4C, stop P-4B, swap P-4B MOD (A6) to the A3 supply and start P-4B.

Notes:

(a.) is incorrect. MOD should not be swapped under load.
(b.) is incorrect. P-4B must be powered from A3 to comply with Tech Specs since P-4C is powered from A4.
(c.) is incorrect. Stopping P-4B first would cause a loss of all service water.
(d.) is correct.

References:

STM 1-42, Rev. 9

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO 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.

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

QID: 0227 **Rev:** 1 **Rev Date:** 5/2/05 **Source:** Modified **Originator:** J. Cork

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

Section: 3.8 **Type:** Plant Services Systems

System Number: 078 **System Title:** Instrument Air System (IAS)

Description: Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems.

K/A Number: K4.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	C

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

Instrument Air pressure has dropped to 58 psig.
Field operators can not find an Inst. Air leak on Unit One.

Which of the following is the appropriate response for the given plant conditions to restore or conserve Instrument Air pressure?

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

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

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

d. If ICW available, isolate Seal Injection by closing CV-1206.

Answer:

b. Close Unit 1 to Unit 2 Instrument Air cross-connect.

Notes:

Per 1203.024, the U1 to U2 cross connect should be closed first, so [b] is correct.
[a] is incorrect, this does not occur until pressure is at 50 psig.
[c] is incorrect, this would not be done until pressure was less than 35 psig.
[d] is incorrect, this would not be done unless necessary to maintain PZR level <290".

References:

1203.024, Chg. 010-08-0

History:

Developed for 1998 RO/SRO Exam QID 0102.
Modified for A. Morris 98 RO Re-exam
Modified for 2005 RO exam.

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

DISCUSSION

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

1.0 SYMPTOMS

1.1 IA header pressure dropping.

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

1.3 INST AIR COMPRESSOR TROUBLE (K12-C3) alarm

1.4 M-1/F-8 ΔP (K21-5) alarm

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

> **NOTE**
> IA HDR Pressure can be monitored using PMS point P5409.

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

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

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

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

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

3.4.2 **GO TO** step 3.7.

> **CAUTION**
> If either Unit 1 or 2 has a significant IA leak, crossconnecting Unit 1 and 2 IA systems can result in low IA pressure in both units.

3.5 Direct Unit 2 control room operators to crossconnect Unit 2 IA system to Unit 1 IA system.

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

QID: 0569 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** Direct **Originator:** J.Cork

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Knowledge of the effect that a loss or malfunction of the Containment System will have on the following: Loss of containment integrity under normal operations.

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

Tier: 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

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

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

With the plant operating at power, what type of occurrence would make it necessary to use AOP 1203.005, Loss of Reactor Building Integrity?

A. Failure to perform a LLRT on personnel hatch within 12 hours after opening.

B. An entry into RB to add oil to a RCP motor.

C. "A" LPI RB sump suction is inoperable, closed, and de-energized.

D. The interlocks jam on the personnel hatch and both doors are open for < 5 minutes with Ops Manager's permission.

Answer:

D. The interlocks jam on the personnel hatch and both doors are open for < 5 minutes with Ops Manager's permission.

Notes:

Only "D" meets the conditions specified in 1203.005.

References:

1203.005, Chg. 011-05-0

History:

Direct from regular exam bank, QID #737.
Selected for 2005 RO exam.

NOTE

- Reactor building shall be operable whenever the unit is in Modes 1, 2, 3, or 4 (Tech Spec 3.6.1).
- This procedure assumes that reactor building integrity is required.
- During periods of frequent containment entries, the 7 day criterion for performance of airlock LLRT can be extended by Engineering Programs to 30 days.

1.0 SYMPTOMS

1.1 Failure to meet the requirements of SR 3.6.1.1.

- Reactor building is not operable due to visual examination results.
- Air leakage is in excess of Reactor Building Leakage Rate Testing Program acceptance criteria.

1.2 Either reactor building air lock not operable per TS 3.6.2.

1.3 Reactor building equipment hatch not closed and sealed (TS 3.6.1 Bases).

1.4 Inoperable reactor building isolation manual or automatic power operated valve per TS 3.6.3.

1.5 Failure to perform local leak rate testing (LLRT) on personnel lock or emergency lock within 7 days after opening (SR 3.6.2.1).

2.0 IMMEDIATE ACTION

None.

LCO

The LCO establishes the minimum conditions required to ensure that the CFTs are available to accomplish their core cooling safety function following a LOCA. Both CFTs are required to function in the event of a large break LOCA. If the entire contents of both tanks are not injected during the blowdown phase of a large break LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. For a CFT to be considered OPERABLE, the isolation valve must be fully open with power removed, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 800 psig, the CFT OPERABILITY requirements are based on full power operation. Although cooling requirements may decrease as power decreases, the CFTs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

In MODE 3 with RCS pressure ≤ 800 psig, and in MODES 4, 5, and 6, the CFT motor operated isolation valves may be closed to isolate the CFTs from the RCS. This allows RCS cooldown and depressurization without discharging the CFTs into the RCS or requiring depressurization of the CFTs.

In addition, LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)," requires that in MODE 4 when any RCS cold leg temperature is ≤ 262°F, MODE 5, and MODE 6 when the reactor vessel head is on, each CFT whose pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," be isolated.

ACTIONS

A.1

If the boron concentration of one CFT is not within limits, the ability to maintain subcriticality may be reduced, but the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of two CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

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

QID: 0262 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D Slusher

TUOI: ASLP-RO-RXT14 **Objective:** 20 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

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

Description: Ability to monitor automatic operation of the CRDS, including: Boration/dilution

K/A Number: A3.07 **CFR Reference:** 41.7/45.13

Tier: 2 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 3.5

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

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

Given:

- The plant is at 100 %.
- CRDs are at the normal rod index.
- The EHC controller is in manual.
- RCS boron concentration is 812 ppm.
- 1 ppm RCS boration requires 7.8 gallons of Boric Acid.

The CBOR is making a RCS addition with no concentration change and adds 92 gallons of boric acid and 8 gallons of DI water.

What effect will this have, without any further operator action?

a. Rods go full out, Tave stays the same, power goes down.

b. Rods go in ~10%, Tave stays the same, power goes down.

c. Rods go in ~10%, Tave goes down, power stays the same.

d. Rods go full out, Tave goes down, power stays the same.

Answer:

d. Rods go full out, Tave goes down, power stays the same.

Notes:

(a) is incorrect because rods go full out and Tave decreases to maintain power the same
(b) & (c) are is incorrect because rods go out when boric acid is added.
(d) is correct. With reactor power is maintained by Tave going down and rods moving out.

References:

Generic Fundamentals Reactor Theory Chapter 14 Rev 2

History:

Developed for 1999 exam.
Selected for 2005 RO exam.

MAINTAINING ACCEPTABLE POWER DISTRIBUTION

To prevent large distortions in the axial power profile which could lead to peaking factors outside of design limits, there are strict technical specification (tech specs) requirements that limit the allowed axial flux difference (ΔI). Each facility has plant technical specification requirements in this regard. The following three sections present an overview of power distribution requirements for B&W, CE, and Westinghouse plants.

BABCOCK &WILCOX POWER DISTRIBUTION REQUIREMENT

To prevent large distortions in the axial power profile that could lead to peaking factors outside of design limits, there are strict technical specification requirements that limit the allowed axial power imbalance. Figure 8-20 compares four upper power range detector outputs to four lower range detector outputs, looking at the axial power imbalance.

$$\text{Axial Power Imbalance} = \frac{\phi_{top} - \phi_{bottom}}{\phi_{\text{equiv. of 100\% power}}}$$

Equation 8-9

The operating crew is required to operate the reactor within the "Permissible Region" shown under the curve. There are different curves derived based on of fuel burnup (core life). The curve in Figure 8-20 represents middle-of-life (MOL) conditions.

The allowed axial flux imbalance for a B&W reactor facility must be maintained according to tech specs as represented by this MOL axial power imbalance curve.

Normally, this is accomplished by maintaining control rods almost fully withdrawn from the core. Minimizing control rod movement during power operation minimizes axial flux shifts that can start xenon oscillations. However, some B&W facilities employ axial power shaping rods with 6 feet of active absorber positioned near the center of the core to also provide flux (power) shaping.

Because most nuclear facilities are "base loaded," the load dispatcher tries to ensure nuclear units on the grid do not change power.

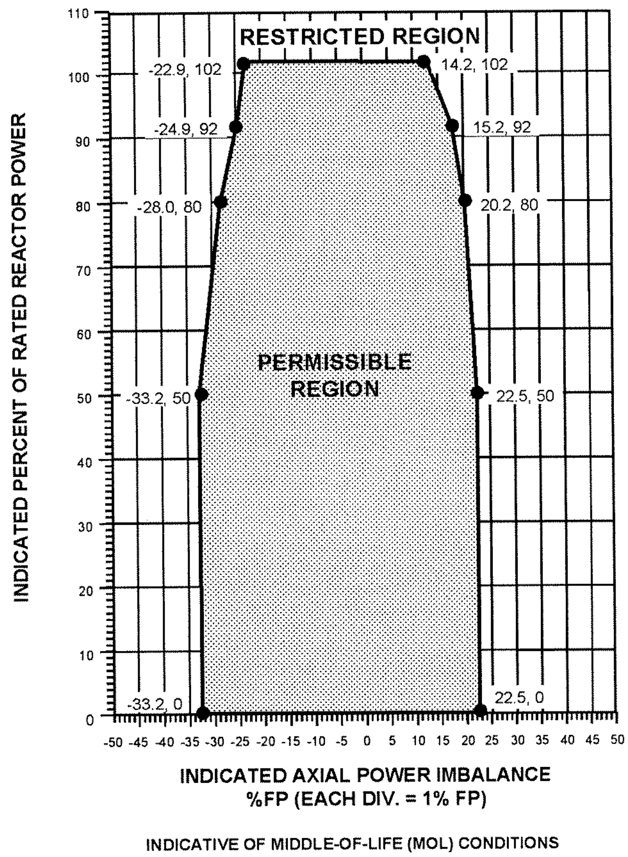


Figure 8-20 Indicated Axial Power Imbalance for a B&W Station

However, if load follow operations are warranted, then reactivity changes during load follow operations are made by changing the RCS boron concentration. This helps keep the control rods out of the core except where axial power shaping rods (APSRs) are used. As reactor power changes, however, the axial flux

tends to shift because of moderator temperature reactivity effects.

Changing reactor power upsets equilibrium conditions and induces transient xenon behavior. APSRs are most useful for handling these axial power imbalances. However, where no APSRs are used, boron concentration changes are necessary to move rods in or out.

The reactor operator must be aware of the direction of the resulting reactivity change and compensate with boron concentration adjustments to maintain RCS temperature. Rod movement in or out of the core for reactivity control is undesirable because it distorts the natural axial distribution of the flux and may initiate a xenon transient.

For negative power imbalances, boration is required to move control rod assemblies (CRAs) out. For positive power imbalances, dilution is required to move CRAs in.

Core imbalance must be monitored at a minimum of once every two hours during power operation above 40% of rated power. Except for physics tests, corrective measures (reduction of imbalance by APSR movement if allowed, and/or reduction in reactor power) must be taken to maintain operation within the envelope defined by Figure 8-20. If the imbalance is not within the envelope defined by this figure, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power must be reduced until imbalance limits are met according to Figure 8-20.

Control Rod and APSR Recommendations (B&W Procedural Examples)

Movement of the control rods and/or adjustment of the boron concentration control core power. Control rods should be maintained within the steady-state operating rod position range (Control Rod Group- 7 > 90% withdrawn) whenever steady load conditions exist for more than one-two hours or during load changes less than 0.5% per minute. This action will minimize imbalance changes. Greater rod insertion is used during all transients over 0.5% per minute and whenever power changes greater than 15% are expected.

The APSRs should be maintained within the position limits established by the plant operations director via the lead nuclear engineer (Control Rod Group 8 > 30% withdrawn) within eight hours after obtaining steady-state conditions. If these position limits cannot be maintained, the lead nuclear engineer is notified per station procedures.

COMBUSTION ENGINEERING POWER DISTRIBUTION REQUIREMENTS

To prevent large distortions in the axial power profiles that could lead to peaking factors outside CE plant design limits, Combustion Engineering power stations place strict requirements on the axial shape index in the plant technical specifications. The axial shape index (ASI) is calculated using the excore nuclear instruments and is an indication of core power distribution.

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

QID: 0604 **Rev:** 0 **Rev Date:** 6/30/05 **Source:** New **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:**	0
Group:	2	**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	

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

A reactor trip has occurred and the CRS is directing actions per 1202.001, Reactor Trip.

The CBOR reports that Pressurizer level has fallen to 50" and continuing to drop slowly.
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:

C. Isolate Letdown by closing Letdown Cooler Outlet (CV-1221).

Notes:

Answer "C" is correct per 1202.001, step 26 contingency action.
Answer "A" is incorrect, this is not done until level is < 20".
Answer "B" is incorrect, this was done early in the procedure.
Answer "D" is incorrect, CV-1235 is operating properly in Auto, taking it to hand would not help.

References:

1202.001, Chg. 028-03-0

History:

New for 2005 RO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
26. Check Pressurizer Level Control valve (CV-1235) maintains PZR level > 55".	26. Perform the following: A. **<u>IF</u>** CV-1235 fails to respond in AUTO, **<u>THEN</u>** operate CV-1235 in HAND to control PZR level 90 to 110". B. **<u>IF</u>** PZR level is < 55" with **<u>no</u>** indication of recovery, **<u>THEN</u>** isolate Letdown by closing either: Letdown Cooler Outlet (CV-1221), **<u>OR</u>** Letdown Cooler Outlets (CV-1214 and 1216). C. **<u>IF</u>** PZR level drops below 55", **<u>THEN</u>** verify Pressurizer Heaters off. D. **<u>IF</u>** PZR level drops below 30", **<u>THEN</u>** initiate HPI (RT 2).

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

QID: 0571 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** Direct **Originator:** S.Pullin

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

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation System

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

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

Tier:	2	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.3	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 58 **SRO:**

If at 100% power, NI channel 7 failed full upscale (to 125% power), which one of the following would occur?

A. ICS would withdraw control rods and runback feedwater.

B. RPS channel C would trip.

C. ICS would insert control rods and increase feedwater.

D. SASS would select NI channel 8.

Answer:

B. RPS channel C would trip.

Notes:

"B" is correct, NI 7 inputs into C RPS with high flux trip setpoint of 104.9%.
"A", "C", "D" are incorrect, the highest of channels 5 & 6 input into ICS, not 7 & 8.

References:

1105.001, Chg. 020-05-0

History:

Direct from regular exambank, QID#1793.
Selected for 2005 RO exam.

6.0 SETPOINTS

6.1 RPS Trip Setpoints.

High RCS temperature	618°F (T-Hot)
High RCS pressure	2355 psig (1720 psig when in shutdown bypass)
High Rx power	104.9% (5% when in shutdown bypass)
Low RCS pressure	1800 psig (bypassed in shutdown bypass)
High R. B. pressure	18.7 psia
High Ø/ΔØ/flow	COLR Figure 9 (Bypassed in shutdown bypass)
Variable low pressure	COLR Figure 11 (Bypassed in shutdown bypass)
High power/pumps	1 pump per loop ... 55% or 0 pumps in one loop 0% (Bypassed in shutdown bypass)
Turbine trip	Rx power >43% and turbine is tripped
Both MFWPs trip	Rx power >9% and both MFWPs tripped

7.0 Channel Checks of NI and RPS

7.1 Channel checks of NI and RPS shall be conducted by performing CBO Turbine Logsheet (OPS-A6) as follows:

7.1.1 At a minimum of once per shift above cold shutdown.

7.1.2 Upon reaching Mode 3, >525°F conditions during plant startup.

7.2 IF any parameter is identified outside the MAX or MIN values but within Tech Spec operability limit,
THEN immediately notify Shift Manager and initiate corrective action.

7.3 IF any parameter is identified outside Tech Spec operability limit,
THEN declare that component inoperable, write a Condition Report, immediately notify the Shift Manager and initiate corrective action. Reference applicable TS for required actions.

7.4 It shall be the responsibility of the Shift Manager to review the completed log sheet and verify that all channels required for the applicable mode of operation are operable.

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

QID: 0572 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** New **Originator:** J.Cork

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

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-core Temperature Monitor System

Description: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

K/A Number: 2.1.33 **CFR Reference:** 43.2 / 43.3 / 45.3

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

Group: 2 **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 59 **SRO:**

Which one of the following is required for Post Accident Monitoring per Technical Specification 3.3.15?

A. "A" and "B" OTSG Startup level

B. Reactor Building Sump level

C. ICCMDS Core Exit Thermocouples

D. PMS Self Powered Neutron Detectors

Answer:

C. ICCMDS Core Exit Thermocouples

Notes:

"C" is the correct answer since the ICCMDS CETs are the qualified instruments required per 3.3.15.
"A" is incorrect, EFIC low and high range SG level are required, not startup.
"B" is incorrect, RB flood level is required, not sump level.
"D" is incorrect, Gamma metrics wide range NI is required, not the SPNDs.

References:

T.S. 3.3.15, table 3.3.15-1

History:

New for 2005 RO exam

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 (VALIDATION)

QID: 0573 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** Direct **Originator:** J.Cork

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

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

System Number: 033 **System Title:** Spent Fuel Pool Cooling System (SFPCS)

Description: Knowledge of the physical connections and/or cause-effect relationships between the Spent Fuel Cooling System and the following systems: RHRS.

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

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

Group: 2 **SRO Imp:** 2.7 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 60 **SRO:**

What is the flowpath for the purification of the Fuel Transfer Canal and Reactor Vessel when using the Spent Fuel Cooling System?

A. Taking a suction on DHR, through the purification loop and then discharging to the Spent Fuel Pool.

B. Taking a suction on the Fuel Transfer Canal, through the purification loop and then discharging to the DHR system.

C. Taking a suction on the Spent Fuel Pool, through the purification loop and then discharging to the DHR system.

D. Taking a suction on DHR, through the purification loop and then discharging to the Fuel Transfer Canal.

Answer:

B. Taking a suction on the Fuel Transfer Canal, through the purification loop and then discharging to the DHR system.

Notes:

"B" is the correct alignment per 1104.006.
The rest are incorrect lineups per 1104.006.

References:

1104.006, Chg. 032-06-0

History:

Direct from regular exambank, QID#1977.
Selected for 2005 RO exam.

21.0 Fuel Transfer Canal and Reactor Vessel Purification **WITHOUT** Provisions for Emergency Boration

NOTE

During fuel handling, fuel transfer canal can be purified using the SF purification loop. The loop is aligned to take suction from the fuel transfer canal and discharge to the suction of the Decay Heat Pump (P-34A or P-34B). Both Spent Fuel Pool CLG Pumps (P-40A and P-40B) are available for SF Pool cooling as desired.

21.1 Initial Conditions

21.1.1 Reactor vessel head removed.

21.1.2 Fuel transfer canal filled to refueling level.

21.1.3 Fuel XFER Canal Fill/DRN ISOL Spectacle flange (M-307) rotated to OPEN position.

21.1.4 DH system in DH removal mode per Decay Heat Removal Operating Procedure (1104.004), "Decay Heat Removal During Refueling" section.

21.1.5 Spent Fuel Purification Loop secured.

CAUTION

Changes in system configuration or operational status with RCS flooded up and fuel transfer canal open to SFP could result in changes of RCS level and SFP level.

21.1.6 IF RCS is flooded up with fuel transfer canal open to SFP AND SFP level indication in the Control Room is NOT available,
THEN station an operator for local observation of SFP level (LI-2005) with direct communications to the Control Room.

NOTE

Aligning a flowpath through RB penetration P-19 is a breach of containment and requires tracking on Containment Closure Breach List 1015.002D.

21.2 Align P-66 for fuel transfer canal purification per Attachment J, "Fuel Transfer Canal Purification **WITHOUT** Emerg. Boration".

21.3 Make appropriate entry on Containment Closure Breach List 1015.002D.

21.4 Align SF purification per Attachment B, with the following exception:

21.4.1 SF to DH Suction Header (SF-20) open.

21.4.2 Close F-4A & B Discharge to SF Pool (SF-25).

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

QID: 0081 **Rev:** 1 **Rev Date:** 5/4/05 **Source:** Modified **Originator:** JCork

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

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

System Number: 035 **System Title:** Steam Generator System (S/GS)

Description: Knowledge of S/GS design feature(s) and/or interlock(s) which provide for the following: S/G level control.

K/A Number: K4.01 **CFR Reference:** 41.7

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

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

Question: **RO:** 61 **SRO:**

Which of the following are performed by Rapid Feedwater Reduction (RFR)?

1) Closes MFW block valves in fast speed
2) Inserts a zero demand signal into both FW loop demands
3) Inserts a negative error into the S/U and Low Load valve controllers
4) Shifts MFW pump control to high FW loop DP
5) Closes the low load block valve
6) MFW pump control is taken away from the operator (both lights lit on H/A stations)

a. 2, 3, and 6

b. 1, 2, and 5

c. 1, 3, and 6

d. 2, 4, and 5

Answer:

a. 2, 3, and 6

Notes:

The RFR circuit inserts a zero demand signal just below each loop feedwater demand station and an error signal is inserted into the startup and low load control valves, causing them to close. If an MSLI signal is present it will hold the startup and low load valve controllers to a zero output. Choices 2, 3 and 6 are the only correct responses.

References:

STM 1-64, Rev. 9

History:

Modified QID 4413 for 1998 RO Exam.
Modified for use in 2005 RO exam.

pump's ICS Hand/Auto control station. This signal is then scaled 0 to 10 v, passed through a signal limiter and sent to Lovejoy.

2.6.9 Rapid Feedwater Reduction Circuit

This circuit actuates on a reactor trip to reduce feedwater to each OTSG as rapidly as possible. It performs the following:

a. A zero demand signal is inserted just below each loop feedwater demand station running the loop demands to zero.

b. MFW pump control is taken away from the operator, as indicated by both lights lighted on the MFW pump H/A station, and both pump control stations are run to zero demand which would place the pumps at minimum speed.

c. The start up and low load control valves are closed by the insertion of a error signal into their controller.

Control of the feedwater pumps is restored when OTSG level is ≤ 45 inches. When control is returned to the operator, the MFW pump H/A station will be in auto. If OTSG level were to increase to >45 inches RFR would again take control of the feed pumps to run them down to minimum speed.

Control of the start up and low load control valves is restored when OTSG level decreases to ≤ 40 inches. When control is returned, the valves are latched to normal controls and the RFR signal is blocked. An exception to this action occurs if a Main Steam Line Isolation (MSLI) signal is present. An MSLI will hold the respective generator's start up and low load valve controllers to a zero output until the MSLI is cleared.

The rapid feedwater reduction circuit can be manually defeated. An RFR override switch is located on C03 between the two main feed pump H/A stations.

To limit the OTSG level undershoot on a reactor trip a 5" bias is applied to the low level limit setpoint. This bias circuit must first be armed by OTSG level being >40 inches and then an RFR signal will insert the bias. The bias decays to 0 in ~10 minutes so that normal post trip OTSG level control at 30 inches is maintained.

2.6.10 Main and Low Load Block Valve Control

Each loop's low load block valve (CV-2624 and CV-2674) is automatically controlled by the position of its associated startup control valve. Limit switches on the startup control valve will open the block at 80% and close the block at 50% when going closed. (Refer to figure 64.26) In addition, the following will also cause the low load block valves to go closed: reactor trip, both main feedwater pumps trip, or all reactor coolant pumps tripped. However, manual control may be taken by the operator. This is accomplished by repositioning the ICS Control Override switch on panel C03 from "normal" to "override" and holding the valve control switch to the open or the closed position as desired. In override the block valve acts like a modulating valve. The override control does not prevent the valve from going closed on a reactor trip. The close signal on a reactor trip comes from the Reactor Trip Confirm (RTC) circuitry of the control rod drive system which is independent of the ICS.

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

QID: 0574 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** New **Originator:** S.Pullin

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

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

System Number: 041 **System Title:** Steam Dump System and Turbine Bypass Control

Description: Knowledge of the operational implications of the following concepts as they apply to the SDS: Reactivity feedback effects.

K/A Number: K5.07 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 62 **SRO:**

Given:

- Normal startup with reactor power at 10%.
- Main Turbine is in Operator Auto.
- A turbine bypass valve (TBV) fails open.

With NO operator action, which of the following will be the final reactor power?

A. ~12%

B. ~14%

C. ~16%

D. ~18%

Answer:

B. ~14%

Notes:

"B" is correct, 4 TBVs have a total capacity of 15%, so one TBV is 3.75% steam flow and reactor power will match steam flow.
The other distractors are simple even numbered answers close to 14%.

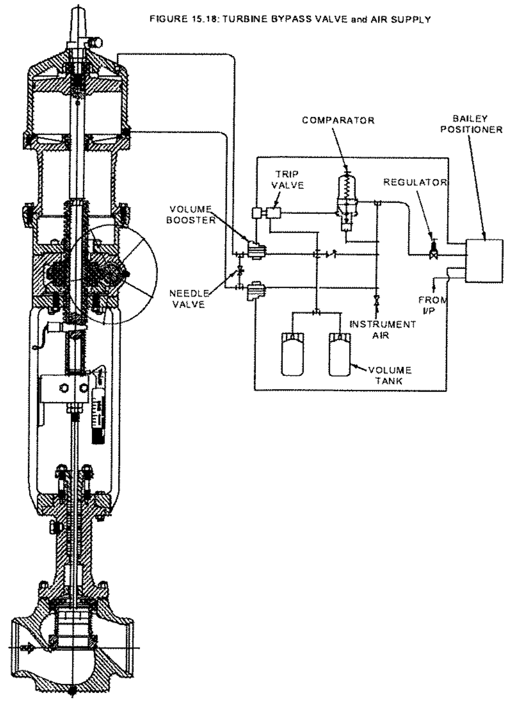
References:

STM 1-15, Rev. 7

History:

New for 2005 RO exam.

FIGURE 15.18: TURBINE BYPASS VALVE and AIR SUPPLY



(Refer to Figure 15.18)

The TBV's are designed to provide a normal means of pressure control and heat removal from the steam generators during normal heatup, cooldown and prior to synchronizing the turbine generator to the grid. After the turbine generator is synchronized to the grid and load is greater than or equal to 15%, the turbine governor valves control header pressure. Turbine Bypass Valve capacity is 418,500 lbm/hr each. This allows all four of the TBV's to pass a total steam flow equivalent to 15% of main feedwater flow.

The inlet piping to the TBV's is a 6-inch line, while the outlet is a 10-inch line. The TBV's receive ICS signals in order to operate. The TBV's are electro-pneumatically-controlled globe valves. They have a stroke time of less than 3 seconds. These valves can also be manually operated.

If either of the main condensers' vacuum lowers below 21" Hg during normal operation, all four TBVs' will close. Both steam header ADVs are automatically unisolated and an alarm will sound--"Vacuum Low ADV Control Actuated".

Once condenser vacuum is restored above 24" Hg, the operator can reset the low vacuum signal block to the TBV's via 2 push-buttons (one for each condenser) on C02 (Low Vacuum Reset). Additionally, if it is desired to override the low vacuum signal block to the TBV's, a handswitch is provided on C02 (Turbine Bypass Low Vacuum Override). This handswitch allows the operator to select "Auto" for normal operation, or "Cond" to override the signal block. If this is done, excessive pressure can be developed in the condenser, which would be relieved by the condenser rupture discs.

TBV's have a sophisticated air supply system. IA supply is supplied to the valve positioner and provides supply air for valve operation. In addition to supplying motive air for operation it also maintains the two volume tanks pressurized to maintain the TBV closed on loss of instrument air.

The positioner is supplied with regulated air at 65 psig. With this supply of air, the positioner aligns to either the open volume booster or close volume booster through the trip valve. The instrument air supply through the volume booster then operates the valve. Operation of the TBV volume booster is identical to the ones used in the ADV's. Refer back to page 14 for explanation of volume booster operation.

The comparator is a spring-loaded valve that monitors air pressure continuously. If instrument air header pressure drops to 55 psig, the comparator's spring forces the internal valve down allowing the trip valve to realign the air accumulators to the turbine bypass valves closed volume booster thus maintaining the valve closed. Once air pressure in the two volume tanks is depleted, for whatever reason, system pressure can open the turbine bypass valve.

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

QID: 0575 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** New **Originator:** S.Pullin

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

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

System Number: 045 **System Title:** Main Turbine Generator System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G System controls including: Expected response of secondary plant parameters following a T/G trip.

K/A Number: A1.06 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 63 **SRO:**

Given:

- All ICS H/A stations are in automatic.
- Reactor power is 100%.
- Main Turbine trips.
- Normal post-trip response.

What is the expected OTSG pressure and RCS Tavg?

A. 895 psig and 532°F

B. 945 pisg and 538°F

C. 995 psig and 547°F

D. 1020 psig and 550°F

Answer:

C. 995 psig and 547°F

Notes:

"C" is correct, a 100 psig bias is applied to the TBVs following a trip to limit RCS cooldown to 547°F.
"A" is incorrect, these parameters would be correct without the 100 psig bias.
"B" is incorrect, these parameters would be correct with the 50 psig bias applied when turbine above 15%.
"D" is incorrect, these parameters would be correct with the ADVs in control.

References:

STM 1-64, Rev. 9

History:

New for 2005 RO exam.

A bias of +100 psig will be added when the Reactor trips to limit the RCS cooldown. A pressure setpoint of 895 psig corresponds to a RCS Tave of 532^{0}F. Without the 100 psig bias the bypass valves would control pressure to 895 psig thus cooling the primary down from the normal operating Tave of 579^{0}F to 532^{0}F. With the 100 psig bias, the bypass valves would control to 995 psig which corresponds to ~547^{0}F Tave thus limiting the cooldown of the primary and the resultant drop in pressurizer level.

Each pair of turbine bypass valves can be manually controlled from a hand/automatic station. In position the meter indicates the hand demand signal and has a span of 0% open to 100% open. This signal is the demanded positions and not the actual valve positions. Actual valve position indications are provided above each hand/auto station by 0% to 100% indicators. Selection of measured variable on these stations indicates the biased turbine header pressure error signal. A 50% indication means that biased turbine header pressure error equals zero. If the 50 psi bias has been applied, and header pressure is at 895 psig, an indication of ~40% would be seen in measured variable.

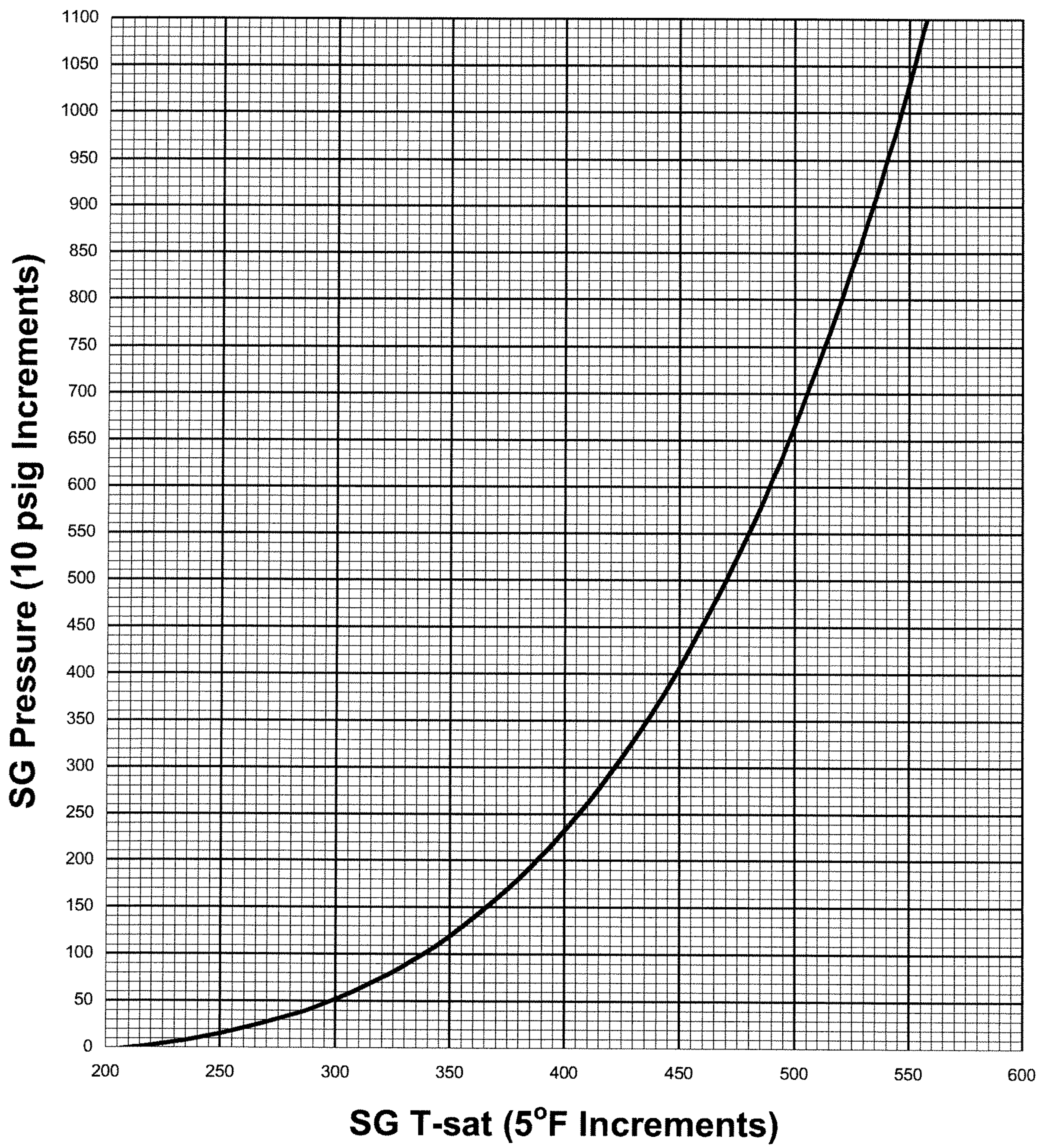
2.5.3.1 Turbine Bypass Valve Control Selector Switch

The signals to the turbine bypass valves may be affected by the position of a selector switch located on Panel C02. This switch has two positions, "Auto" and "Cond", which may be selected by the operator. The "Auto" position is normally used. In this position the control signal will always be passed to the turbine bypass valves unless a low condenser vacuum condition (<21") exists. In the event of low vacuum, the bypass valves receive a signal to keep them fully closed. If condenser vacuum increases to >23", the operator may return control to the valves by depressing two reset pushbuttons located on C02. If the operator selects the "Cond" position, the normal control signal will go to the bypass valves regardless of condenser vacuum value.

2.5.4 Calibrating Integral

During steady-state operation in the integrated control mode, the reactor and steam generator should be producing the proper amount of steam for the demanded load. If one of the following occurs, a megawatt error would develop: (1) the turbine generator efficiency changes, (2) the enthalpy of the steam to the turbine changes, or (3) an error occurs in the feedwater flow measurement. The megawatt error will, through a calibrating integral, be applied to alter the demand to both the steam generators (feedwater control) and reactor. (Refer to figure 64.20) This control action eliminates megawatt error by recalibrating the reactor and feedwater demand with respect to the unit load demand. With no megawatt error, the reactor and steam generator are producing the proper amount of steam for the demanded load.

FIGURE 2
SG Pressure vs T-sat



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

QID: 0196 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** R. Fuller

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

Section: 3.9 **Type:** Radioactive Release

System Number: 071 **System Title:** Waste Gas Disposal System

Description: Ability to manually operate and/or monitor in the control room: Setting of process radiation monitor alarms, automatic functions, and adjustment of setpoints.

K/A Number: A4.25 **CFR Reference:** 41.7 / 45.5 to 45.8

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

Group: 2 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 64 **SRO:**

While performing a reactor building purge evolution, the operator notes all four RB Purge Isolation Valves go closed. What is the most likely cause?

a. An ESAS actuation of channels 1 and 2 has closed the valves.

b. A loss of load center B-5 has occurred causing the valves to fail closed.

c. A high radiation setpoint has been exceeded on SPING 1.

d. RB Purge Exhaust Fan (VEF-15) has tripped causing the valves to close.

Answer:

c. A high radiation setpoint has been exceeded on SPING 1.

Notes:

(a.) is incorrect. Channel 1 and 2 of ESAS does not cause the valves to close.
(b.) is incorrect. A loss of B-5 affects RB cooling and RB hydrogen recombiners but not RB purge.
(c.) is correct.
(d.) is incorrect. VEF-15 is interlocked with the RB supply fan but not the RB purge valves.

References:

STM 1-09 rev. 6

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO exam.

2.2.1.4 Purge Supply Containment Isolation

The fan discharges into the containment through an exterior isolation valve (CV-7402) and an interior isolation valve (CV-7404). CV-7402 is an air-to-open/spring-to-close butterfly valve supplied by instrument air and controlled by solenoid valve SV-7402 through a hand switch on panel C16. Position indication for CV-7402 is also on panel C16. On an ES actuation from Ch. 4, or high radiation signal from Sping 1, CV-7402 automatically closes to meet containment isolation requirements. SV-7402 is powered from breaker RA2-11. Interior isolation valve CV-7404 is a air-operated air-to-open/spring-to-close butterfly valve supplied by instrument air and controlled by solenoid valve SV-7404 through a hand switch on panel C18. Position indication for this valve is also on panel C18. Power to SV-7404 is RA1-9. On an ES actuation from Ch. 3, or high radiation signal from Sping 1, CV-7404 closes to meet containment isolation requirements. Purge Supply air is ducted to various areas in the lower half of the RB.

The Containment Purge Isolation Valves are expected to close within 5 seconds of receipt of an ES signal or an signal from Sping 1 by spring pressure. The interior isolation valves are air-operated valves sized at 24". The valves are 'Tricentric' which are designed to provide a positive seal by wedging the disk into the seat. An instrument airline goes through the penetration for the Purge lines to supply air to the two valves in the Reactor Building. Until a Tech. Spec. change is approved for the use of these valves above Cold Shutdown they will remain closed until shutdown with the conditions for Reactor Building Integrity no longer required.

CV-7401/CV-7402/CV-7403/CV-7404	
Valve MFG.	Atwood & Morrill
Type	Tricentric, Metal seated, Butterfly
Size	24"
Valvop MFG.	Bettis
Type	pneumatic, NT-520

2.2.2. RB Purge Exhaust Fan VEF-15

RB Purge Exhaust, VEF-15 is a centrifugal fan that takes suction on a common exhaust line from four ducts located in the upper portion of the Reactor Building. Feeding into the common exhaust line is one duct taking suction from the dome at 545' elevation and three other ducts taking suction from around the 445' elevation. VEF-15 is powered from B3164.

VEF-15	Clarage Fan
Flow	40KCFM
Type	Centrifugal, Backward Inclined
Motor	GE, 480VAC, 60 HP

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

QID: 0090 **Rev:** 1 **Rev Date:** 11/4/98 **Source:** Direct **Originator:** JCork

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

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

System Number: 086 **System Title:** Fire Protection System (FPS)

Description: Knowledge of the effect of a loss or malfunction of the Fire Protection System will have on the following:
Fire, smoke, and heat detectors.

K/A Number: K6.04 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 4

Group: 2 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** An

Question: RO: 65 SRO:

Given:

- The yellow trouble LED is ON for #1 EDG flame detector module on C463 and has been acknowledged on C463.
- Subsequently, a #1 EDG smoke detector spuriously actuates.

Which of the following annunciators would alarm due to the smoke detector actuation?

a. "FIRE WATER FLOW", K12-A2

b. "FIRE WATER PRESSURE LOW", K12-B1

c. "FIRE", K12-A1

d. "FIRE PROT SYSTEM TROUBLE", K12-D1

Answer:

c. "FIRE", K12-A1

Notes:

The only expected annunciator is (c). The actuation of the smoke detector will cause a FIRE alarm.
"A" and "B" are incorrect, the deluge system will not flow water since there is no actual fire to melt the fusible link sprinkler heads.
"D" is not correct, a detector actuation will not cause the Trouble annunciator to re-flash.

References:

1203.009, Chg. 022-04-0

History:

Developed for the 1998 RO/SRO Exam.
Revised after 9/98 exam analysis review.
Used in 2001 RO/SRO Exam.
Selected for 2005 RO exam.

Location: C19

Device and Setpoint:

Actuation in a zone monitored by Pyrotronics C463-1 thru C463-3 or Notifier C463

FIRE

Alarm: K12-A1

1.0 OPERATOR ACTIONS

1. Check Pyrotronics and Notifier C463 panels for red alarm LEDs.

2. IF alarm on Pyrotronics, THEN perform the following:

NOTE

If Alarm/Silence is not depressed, subsequent alarms will not reflash.

A. Depress Alarm/Silence button on module (red alarm lamp will flash and a clicking noise will begin).

B. Verify that red time cycle complete LED on module C3-4 in Fire Detection Panel (C463-3) is off (otherwise reflash is overridden).

3. IF alarm on Notifier, THEN perform the following:

NOTE

- If multiple fire alarms have occurred, the first alarm will be displayed.
- If a fire alarm and a trouble alarm have occurred, the fire alarm will always display first.
- The audible alarm will be silenced when all alarms are acknowledged.

A. Depress ACK STEP.

1. After 2-second time delay, verify fire alarm is acknowledged.

2. IF multiple alarms, THEN continue to depress ACK STEP for each alarm.

3. Continue to depress ACK STEP as desired to step through the previously acknowledged alarms to view info as desired.

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

QID: 0038 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** GGiles

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

Section: 4.1 **Type:** Generic EPEs

System Number: 038 **System Title:** Steam Generator Tube Rupture (SGTR)

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

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

Tier:	3	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 66 **SRO:**

The plant is shutdown and a cooldown is in progress due to a 200 gpm tube leak in the A OTSG.

- RCS temperature is 520 °F and lowering
- BWST level is at 35 ft and lowering
- A OTSG level is 290 inches and rising
- Dose rates at site boundary are normal

Which of the following RCS cooldown limits apply?

a. Less than or equal to 50 °F/hour.

b. Less than or equal to 100 °F/hour.

c. Less than or equal to 240 °F/hour.

d. Less than or equal to 520 °F/hour.

Answer:

b. Less than or equal to 100 °F/hour.

Notes:

(b) is correct in accordance with 1202.006, Tube Rupture. (a) is incorrect, it is the limit if RCS temp is between 300 and 170 °F. (c) is incorrect, it is the emergency cooldown limit which only applies if the affected steam generator level is at 410 inches or off site dose rates reach alert criteria, (d) is incorrect, it is the pressurizer cooldown limit.

References:

1202.006, Chg. 007-04-0

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
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:**	17. **GO TO step 18.**
A. For good SG, place TURB BYP valves in HAND **AND** adjust to maintain cooldown rate ≤240°F/hr.	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. (see table below) 1) **IF** both SGs are bad, **THEN** steam both SGs.
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.	C. **IF** both SGs are bad, **THEN** steam both SGs.

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

INSTRUCTIONS	CONTINGENCY ACTIONS
18. <u>IF</u> emergency cooldown rate is <u>not</u> required <u>OR</u> RCS T-hot is ≤500°F, <u>THEN</u> establish RCS cooldown rate of ≤100°F/hr as follows:	
A. For good SG, place TURB BYP valves in HAND **<u>AND</u>** adjust to maintain cooldown rate ≤100°F/hr.	A. <u>**IF**</u> TURB BYP valves are **<u>not</u>** available, **<u>THEN</u>** operate ATM Dump Control System for good SG in HAND to maintain cooldown rate ≤100°F/hr. SG A: CV-2676 (ATM DUMP ISOL), CV-2668 (ATM DUMP CNTRL); SG B: CV-2619 (ATM DUMP ISOL), CV-2618 (ATM DUMP CNTRL) 1) <u>IF</u> both SGs are bad, **<u>THEN</u>** steam both SGs.
B. **<u>When</u>** RCS press is <1700 psig, **<u>THEN</u>** bypass ESAS.	
C. <u>IF</u> only one SG is bad, **<u>THEN</u>** 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.	C. <u>IF</u> both SGs are bad, **<u>THEN</u>** steam both SGs.

Table in contingency action A:

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

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

QID: 0079 **Rev:** 1 **Rev Date:** 5/9/05 **Source:** Modified **Originator:** S.Pullin

TUOI: A1LP-WCO-PRMS **Objective:** 2 **Point Value:** 1

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

System Number: 029 **System Title:** Containment Purge System (CPS)

Description: Ability to execute procedure steps.

K/A Number: 2.1.20 **CFR Reference:** 41.10 / 43.5 / 45.12

Tier: 3 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 67 **SRO:**

Given:

- Unit is in Mode 5
- Reactor Building pressure is 15.9 psia and stable
- No other abnormal conditions exist

Which of the following procedural actions is required to establish RB purge with these conditions?

a. Open RB purge inlets first, then open outlets.

b. Vent RB via H2 sample lines.

c. Vent RB via RB leak detector.

d. Open RB purge outlets first, then open inlets.

Answer:

d. Open RB purge outlets first, then open inlets.

Notes:

"D" is correct due to positive pressure in RB to prevent reverse flow through filters.
"A" is incorrect since this would induce reverse flow through filters.
"B" and "C" are used at power to lower RB pressure when RB integrity is required.

References:

1104.033, Rev 060-10-0

History:

Developed for the 1998 RO/SRO Exam.
Used in 2001 RO/SRO Exam.
Modified for 2005 RO exam.

ATTACHMENT B

REACTOR BUILDING PURGE GASEOUS RELEASE PERMIT

CAUTION

The following sequence prevents reverse flow through RB purge exhaust filters. Reverse flow can cause damage to filters.

NOTE

SPING 1 is interlocked to close all four RB Purge Isolation Valves (CV-7401, CV-7402, CV-7403, CV-7404) on a high radiation signal.

5.7 Open RB purge dampers as follows:

5.7.1 IF RB pressure is negative, THEN perform the following:

A. Open RB Purge Inlets (CV-7402 and CV-7404). _______

B. WHEN RB pressure equals atmospheric pressure, THEN open RB Purge Outlets (CV-7401 and CV-7403). _______

5.7.2 IF RB pressure is positive, THEN perform the following:

A. Open RB Purge Outlets (CV-7401 and CV-7403). _______

B. WHEN RB pressure equals atmospheric pressure, THEN open RB Purge Inlets (CV-7402 and CV-7404). _______

5.7.3 IF RB pressure equals atmospheric pressure, THEN open inlets and outlets. _______

- CV-7401
- CV-7402
- CV-7403
- CV-7404

5.7.4 Verify RB Purge Valves open. _______

- CV-7401
- CV-7402
- CV-7403
- CV-7404

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

QID: 0576 **Rev:** 0 **Rev Date:** 5/9/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-SRO-TS **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to determine Mode of Operation.

K/A Number: 2.1.22 **CFR Reference:** 43.5 / 45.13

Tier:	3	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 68 **SRO:**

Which one of the following conditions is required by Unit 1 Technical Specifications in order to consider the reactor in Mode 3?

A. The reactor must be subcritical by at least 1.5% Delta k/k.

B. RCS temperature is between 200 °F and 280 °F.

C. K effective is >0.99.

D. RCS temperature is 300 °F.

Answer:

D. RCS temperature is 300 °F.

Notes:

Only "D" is correct per T.S. definition of mode 3.
The other choices are for other modes.

References:

T.S. table 1.1-1

History:

New for 2005 RO exam.

Table 1.1-1

MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 280
4	Hot Shutdown[(b)]	< 0.99	NA	$280 > T_{avg} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

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

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:** No **Taxonomy:** K

Question: **RO:** 69 **SRO:**

During an INITIAL approach to criticality, if criticality is NOT achieved within _________ of the ECP, 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
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
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. 019-03-0

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 2005 RO 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 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. ______

9.9 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: ______

9.9.1 Inform Reactor Engineering. ______

9.9.2 Verify boron concentrations. ______

9.9.3 Verify ECC calculation. ______

9.9.4 Verify position of all control rods by comparing API to zone or limit position switches. ______

9.9.5 WHEN cause of ECC error is determined, AND cause corrected, THEN re-perform AND re-initial applicable steps of this procedure. ______

9.10 Record time reactor is made critical _________. ______

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

QID: 0577 **Rev:** 0 **Rev Date:** 5/9/05 **Source:** Direct **Originator:** J.Cork

TUOI: ELP-OPS-CLR **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of tagging and clearance procedures.

K/A Number: 2.2.13 **CFR Reference:** 41.10 / 45.13

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

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

Question: **RO:** 70 **SRO:**

Equal responsibility for a tagout is shared by the preparer and the ___________.

A. Verifier

B. Reviewer

C. Approver

D. Taggout Holder

Answer:

B. Reviewer

Notes:

"B" is the correct position, the other choices are legitimate tagging positions but do not share this responsibility.

References:

OP-102, Rev. 0

History:

Direct from Unit Two exam bank QID#ANO-OpsUnit2-09269.
Selected for 2005 RO exam.

4.5 OPERATIONS SUPERVISOR

[1] Authorizes the installation and removal of all tagouts.

[2] Ensures Technical Specification compliance.

[3] Verifies application / removal of tagouts will not adversely affect the unit.

[4] Assigns tasks to Tagger/Verifier.

[5] Authorize operation of equipment under Test & Maintenance Tag control in the case of an emergency response for which the equipment is needed to mitigate the consequences of the event.

4.6 PROPOSER

[1] Provide detailed information on work scope.

[2] Recommends isolation boundaries to provide for equipment protection and personnel safety.

4.7 PREPARER

[1] Reviews each work order that will be covered by a tagout to determine the total work scope.

[2] Determines boundary isolation requirements for work being performed to ensure personnel and plant safety.

[3] Prepares the Tagout.

4.8 REVIEWER

[1] Reviews the tagout has been properly developed for the planned work.

[2] Shares equal responsibility with the preparer.

4.9 TAGGER

[1] Reviews tagout detail of the tagout.

[2] Ensures components are positioned in accordance with the Tag Hang sheet.

[3] Restores components to their normal positions in accordance Tags To Be Removed sheet.

[4] Hangs and removes Tags.

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

QID: 0390 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Direct **Originator:** S. Pullin

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

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

System Number: 2.2 **System Title:** Control Rod Drive

Description: Knowledge of Control Rod programming.

K/A Number: 2.2.33 **CFR Reference:** 43.6

Tier: 3 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 71 **SRO:**

Given:

- A startup is in progress.
- All safety groups are 100% withdrawn.
- The group 6 rods' RPI were not re-zeroed prior to startup.
- The CBOR is continuing with the startup with group 5 rods.

Which of the following results?

a. Auto inhibit.

b. Asymmetric rod alarm.

c. Out inhibit.

d. Sequence inhibit.

Answer:

d. Sequence inhibit.

Notes:

Answer [d] is correct, input to sequence inhibit is from RPI.
Answer [a] is incorrect, input from safety group out limit.
Answer [b] is incorrect, input from API (reed switches).
Answer [c] is incorrect, input from High SUR and API.

References:

1105.009, Chg. 019-00-0

History:

Modified regular exambank QID # 3153 for use in 2001 RO exam.
Selected for 2005 RO exam.

the panel may best be described by identifying and explaining the function of each indicator and/or control.

2.8.1 Diamond Panel Alarms

2.8.1.1 Trip Confirmed Lamp

"Trip Confirmed" lamp (amber): When lighted, indicates that power to the CRDM's has been interrupted allowing rods to drop into the core. The lamp is turned on when the following conditions are met:

- "A" and "B" AC breakers open or
- "A" AC breaker and "D" DC breakers open with "F" electronic trips or
- "B" AC breaker open and "C" DC breakers open with "E" electronic trips or
- DC breakers "C" and "D" open with "E" and "F" electronic trips.

In addition to the Diamond Panel indication, the trip confirm circuit (Refer to figure 2.41) will trip the Control Rod Drive Control System (CRDCS) to manual, transfer the Sequence/Sequence Override circuit to the Sequence mode, and transfers the Group/Auxiliary Mode to Group.

A trip confirm will be annunciated "Reactor Trip" on control room annunciator panel K08. The electronic trip signal alone will be annunciated via the plant computer to K02 "Plant Computer Critical Alarm".

2.8.1.2 Asymmetric Rods Lamp

Asymmetric Rods Lamp (amber): When on, indicates that one or more rods within a group are more than 9 inches out of alignment with the group average position. More on Asymmetric Rods later.

2.8.1.3 Out Inhibit Lamp

Out Inhibit Lamp (amber): Indicates that control rods will not respond to any out command. Refer to figure 2.42. The following conditions will result in an "Out Inhibit":

- High startup rate signal from the RPS of 2 DPM in source range and 3 DPM in intermediate range. The high startup rate signals are bypassed in the RPS when greater than 10% power.
- If the Diamond is in automatic and greater than 40% reactor power, a loss of any safety group out limit, (refer to Safety Rods Out relay logic figure 2.43) or a 9" asymmetric rod fault. If either of these conditions occur they will "seal in" and the "Fault Reset" switch must be used to reset the circuit.

2.8.1.4 Sequence Inhibit Lamp

Sequence Inhibit Lamp (Amber): This lamp, when lighted, indicates excessive overlap between regulating groups (> 25%).

The signal for sequence inhibit (Sometimes referred to as sequence fault) is developed by one of two sequence monitor circuits. Refer to figure 2.44. Input to the sequence monitor circuits is group

average from relative position indication (RPI). RPI is utilized in order to provide the monitoring without incurring a sequence fault in the event an asymmetric rod condition alters the group average.

The same relay which causes the sequence inhibit will de-energize the sequence light in the "SEQ/SEQ OR" pushbutton.

Sequence inhibit will reject the Diamond Panel to manual

2.8.1.5 Auto Inhibit Lamp

Auto Inhibit Lamp (amber) indicates that the Diamond cannot be placed in automatic because:

- The safety groups are not at the out limit.
- The neutron error signal (demand versus actual) exceeds ± 1.00%.
- ICS power not available.

If the Diamond panel is in automatic, a loss of ICS Power will bring in this alarm and reject the Diamond to manual (refer to figure 2.45).

2.8.1.6 APSR Overlap Fault Lamp

APSR Overlap Fault Lamp (amber) Indicates that the group 6 lower poison section is less than three inches from group 8 upper poison section. This alarm is used for indication only.

2.8.2 Diamond Panel Indication Lights

2.8.2.1 Out Limit Lamps

Out Limit Lamps Groups 1 - 8 (red): Indicates that at least one rod out of its respective group is at the out limit. This will stop rod withdrawal for all rods within that group. On group 7, out limit occurs at 91.4% withdrawn unless the group 7 "91.4% key/bypass switch" is in bypass, then group 7 will actually be 100% withdrawn.

2.8.2.2 Control On Lamps

White lights that indicate a particular group has been selected (enabled for either automatic or manual command, or is selected for transfer. The corresponding control on lamps on the PIP must also be energized to allow insert/withdraw command.

Control on lamps for groups 1 - 4 indicate that the transfer logic is setup for that group to be transfer between DC Hold and the Auxiliary Power Supply. (If the white control on lamp on the Position Indication Panel is lit, the group is on the Auxiliary Power Supply.)

Control on Lamps Group 5 - 8 (refer to figure 2.46) indicate that a specific group has been selected (enabled) for either automatic or manual command or is selected for transfer.

2.8.2.3 In Limit Lamps

In Limit Lamp Group 1 - 8 (green): Indicates that at least one rod in that group is at the "in limit". This will stop rod movement in

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

QID: 0605 **Rev:** 0 **Rev Date:** 6/30/05 **Source:** New **Originator:** Pullin/Cork

TUOI: ASLP-RO-RADPRO **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

K/A Number: 2.3.4 **CFR Reference:** 43.4 / 45.10 / 41.12

Tier: 3 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 72 **SRO:**

Which of the following is a correct Routine Annual Administrative Guidelines for Entergy employees?

A. TEDE 2000 mrem per year and LDE 12 rem

B. TEDE 4500 mrem per year and LDE 12 rem

C. TEDE 5000 mrem per year and LDE 15 rem

D. TEDE 2000 mrem per year and LDE 15 rem

Answer:

A. TEDE 2000 mrem per year and LDE 12 rem

Notes:

Answer "A" contains the correct routine annual admin guidelines for TEDE and LDE.
Answer "B" contains the correct maximum annual admin guidelines for TEDE and LDE.
Answer "C" contains the correct maximum annual regulatory limits for TEDE and LDE.
Answer "D" contains the correct routine annual admin guideline for TEDE but the regulatory limit for LDE.

References:

ENS-RP-201, Rev. 3

History:

New for 2005 RO exam.

5.2 Limits / Guidelines

5.2.1 Annual Regulatory Limits

5.2.1.1 TEDE = 5 rem

5.2.1.2 SDE, WB = 50 rem

5.2.1.3 SDE, ME = 50 rem

5.2.1.4 LDE = 15 rem

5.2.1.5 TODE = 50 rem

5.2.1.6 Declared Pregnant TEDE = 50 mrem/month, 500 mrem/gestation period

5.2.1.7 Minors = 10% of any Regulatory Limit

5.2.1.8 Unmonitored Individuals = 10% of any Regulatory Limit

5.2.1.9 Members of the General Public TEDE = 100 mrem/year

5.2.2 Maximum Annual Administrative Guidelines

5.2.2.1 TEDE = 4.5 rem

5.2.2.2 SDE, WB = 40 rem

5.2.2.3 SDE, ME = 40 rem

5.2.2.4 LDE = 12 rem

5.2.2.5 TODE = 40 rem

5.2.2.6 Declared Pregnant TEDE = 50 mrem/month, 500 mrem/gestation period

5.2.2.7 Minors TEDE = 100 mrem

5.2.2.8 Unmonitored individual TEDE = 100 mrem

5.2.2.9 Members of the General Public TEDE = 100 mrem/year

5.2.3 Routine Annual Administrative Guidelines

5.2.3.1 TEDE = The lessor of:

- 2000 mrem per year

  Or

- 5000 mrem – (1250 mrem x UQ per year)

  Where UQ = the number of undocumented quarters for the current year

  Except when Lifetime DDE$\geq$ to individuals age x 1 rem in which case the guideline will be set to 1 rem.

5.2.3.2 SDE, WB = 40 rem

5.2.3.3 SDE, ME = 40 rem

5.2.3.4 LDE = 12 rem

5.2.3.5 TODE = 40 rem

5.2.3.6 Declared Pregnant TEDE = 50 mrem/month, 400 mrem/gestation period

5.2.3.7 Minors TEDE = 50 mrem

5.2.3.8 Unmonitored Individuals TEDE = 50 mrem/month, 100 mrem/year

5.2.3.9 Members of the General Public TEDE = 50 mrem

5.3 Instructions for Extending Routine Administrative Guidelines

5.3.1 Extend a Radiation Workers' administrative TEDE guidelines to the guidelines described in the following table, after obtaining the indicated approvals.

5.3.2 Document the authorization of the increased radiation exposure.

NOTE
Responsible individuals may be designated to authorize dose extensions.

Exposure Guideline	Requirements	Authorizations (Note)
Greater than 2000 mrem and less than or equal to 3000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends RP Supervision approves
Greater than 3000 mrem and less than or equal to 4000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends Radiation Protection Manager approves Plant General Manager concurs
Greater than 4000 mrem per year for Radiation Workers.	No undocumented quarters in the current year	Radiation Protection Manager recommends Plant General Manager concurs Site Vice President approves
Greater than 1000 mrem and less than or equal to 2000 mrem for individuals whose lifetime exposure ≥ 1 n where n = age	No undocumented quarters in the current year	Individual's Supervisor recommends RP Supervision approves

5.4 Handling of Workers Receiving Radio-pharmaceutical Treatments

5.4.1 Individuals should notify Radiation Protection of any medical Radiopharmaceutical Treatments prior to entering the protected area of the plant.

5.4.2 Examples of procedures utilizing Radiopharmaceuticals include: cardiac stress tests and thyroid treatments. This list is not all inclusive and any other test requiring radiopharmaceuticals need to be mentioned as well.

5.4.3 Dosimetry evaluates any necessary radworker restrictions.

5.5 Program Reviews

5.5.1 Periodically, at least annually, document a review of plant isotopic composition (WBC library, passive monitoring basis, etc).

5.5.2 Periodically, at least annually, document a review of plant average beta energy.

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

QID: 0579 **Rev:** 0 **Rev Date:** 5/9/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-EOP10 **Objective:** 5 **Point Value:** 1

Section: 3.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

K/A Number: 2.3.10 **CFR Reference:** 43.4 / 45.10

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

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

Question: **RO:** 73 **SRO:**

During performance of the ESAS procedure with BWST level at 8 ft., which of the following actions is performed specifically to reduce plant personnel exposure?

A. Notify RP to begin monitoring BWST suction line for back-leakage.

B. Throttle RB Spray flow 1050 to 1200 gpm per train.

C. Aligning HPI to provide PZR Aux Spray.

D. Removing all but C & D condensate polishers from service.

Answer:

C. Aligning HPI to provide PZR Aux Spray

Notes:

"C" is performed prior to BWST level reaching 6 ft. so the operator performing this alignment is not exposed to unknown dose from fluid being pumped from the RB through the AB.
"A" is performed following the swap to RB recirc but is done to prevent an unmonitored release to offsite personnel via the BWST vent.
"B" is performed just prior to the swap to RB recirc but is done for NPSH concerns for the RB and LPI pumps.
"D" is performed to prevent personnel exposure during a SGTR but not during the ESAS procedure.

References:

1203.010, Chg. 005-03-0

History:

New for 2005 RO exam.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

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

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

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

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

QID: 0580 **Rev:** 0 **Rev Date:** 5-9-05 **Source:** New **Originator:** S.Pullin

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

Section: 2.0 **Type:** Generic K/As

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

Description: Knowledge of EOP terms and definitions.

K/A Number: 2.4.17 **CFR Reference:** 41.10 / 45.13

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

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

Question: **RO:** 74 **SRO:**

Given:

- Reactor trip from 100% power.
- Normal post-trip plant parameters.

The CRS asks for critical parameters.

Which of the following is NOT a critical parameter for the above conditions?

A. Subcooling margin

B. RCS pressure

C. Pressurizer level

D. RCS Tavg

Answer:

D. RCS Tavg

Notes:

Only "D" does not meet the definition of a critical parameter per 1015.043, 4.11. The RCS temperature which is a critical parameter is CET which is used to evaluate core cooling.

References:

1015.043, Chg. 000-03-0

History:

New for 2005 RO exam.

4.9 CONDITIONAL STATEMENTS AND LOGIC SEQUENCES

Used in EOPs/AOPs to describe a set of conditions or express complex combinations of conditions. Logic terms include IF, WHEN, THEN, AND, OR and NOT.

4.10 CONTINGENCY ACTION STEPS

Action steps performed if the instruction or expected response is not achieved.

4.11 CRITICAL PARAMETERS

Parameters which are closely monitored to provide early detection of upsets in heat transfer. Critical parameters include but are not limited to:

- Subcooling Margin
- CET temperature
- RCS pressure

- Pressurizer Level
- A/B SG levels
- A/B SG pressures

4.12 DUAL-COLUMN FORMAT

A procedure format using two parallel columns. The left column is designated "**INSTRUCTIONS**". This column contains the required operator action and the expected plant response. The right column is designated "**CONTINGENCY ACTIONS**". This column contains actions to be taken when the expected response is not obtained or required operator action cannot be performed.

4.13 EMERGENCY OPERATING PROCEDURE (EOP)

A plant specific document based on Emergency Procedure Guidelines which contains the steps needed to take the plant from a reactor trip to a safe, stable condition. Emergency Operating Procedures use a specific format for clarity of operator actions, Control Room personnel interactions, and compatibility with the design of the Control Room.

4.14 ENTRY CONDITIONS

Conditions that are written to explicitly identify those conditions that should exist for the user to enter an EOP/AOP.

4.15 FLOATING STEPS

Steps which apply at all times when the associated procedure is in use. They are located on foldout pages at the end of the procedure.

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

QID: 0358 **Rev:** 0 **Rev Date:** 9/7/99 **Source:** Direct **Originator:** L Maggard

TUOI: ANO-S-LP-EP-A0082 **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and abilities

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

Description: Knowledge of the emergency plan.

K/A Number: 2.4.29 **CFR Reference:** 43.5 / 45.11

Tier: 3 **RO Imp:** 2.6 **RO Select:** Yes **Difficulty:** 2

Group: G **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 75 **SRO:**

During a declared emergency, which of the responsibilities for the person with Emergency Direction and Control can NOT be delegated?

1. Initial Accountability of on-site personnel
2. The decision to notify offsite authorities
3. Dispatching Emergency Response Teams
4. Recommending protective actions to the Arkansas Department of Health.

a. 1, 2 and 3 are correct.

b. 1, 2, and 4 are correct.

c. 2 and 4 are correct.

d. 1 and 3 are correct.

Answer:

c. 2 and 4 are correct.

Notes:

"c" is correct. Per 1903.064, the decision to notify offsite authorities and the recommendation of PARs can never be delegated.
"a" and "b" are incorrect since they include Initial Accountability. "b" and "d" are incorrect, they include the dispatch of ERTs which can be delegated.

References:

1903.064, Chg. 007-03-0

History:

Used in 1999 exam
Direct from Eplan ExamBank, QID# 63 used in class exam
Selected for 2005 RO exam.

6.3 TURNOVER

6.3.1 Shift Manager

A. The Shift Manager of the affected unit shall have responsibility and authority for Emergency Direction and Control of the incident response until relieved by the EOF Director or TSC Director.

[B. The Shift Manager SHALL NOT delegate the responsibility for making offsite Protective Action Recommendations (PARS) or for making decisions to notify offsite authorities while responsible for Emergency Direction and Control.]

C. The Shift Manager must turn over responsibilities to a qualified individual before leaving the Control Room when he has responsibility for Emergency Direction and Control.

D. The responsibility for Emergency Direction and Control will normally be transferred from the Shift Manager to the EOF Director within 60 - 90 minutes of an Alert, or higher, emergency class. However, if the situation dictates, the TSC Director may relieve the Shift Manager of this responsibility.

E. The EOF Director shall notify the Shift Manager when he is prepared to assume the responsibility and authority for Emergency Direction and Control of the incident.

F. The Shift Manager shall promptly turn over responsibility and authority for the overall response as requested by the EOF Director.

G. The Shift Manager shall announce the turnover to the Initial Response Staff (IRS) personnel and report this turnover to the Support Manager located in the EOF.

H. It is the responsibility of the Shift Manager to ensure that the Command and Control Status Board in the Control Room is updated as turnover occurs in the ERO.

6.3.2 Control Room Staff

A. Emergency Response personnel in the Control Room who must leave their assigned location temporarily must inform their immediate superior of their absence, destination, and estimated time of return (with the exception of the Shift Manager as outlined in Section 6.3.1.C).

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

QID: 0581 **Rev:** 0 **Rev Date:** 5/11/05 **Source:** New **Originator:** S. Pullin

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.

K/A Number: 2.4.9 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.3	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	An

Question: RO: SRO: 76

Given:
- Unit is in Mode 5.
- DH has been isolated from RCS due to a LOCA in RCS.
- RCS CET temps are 175°F and rising.
- RCS pressure is 110 psig.
- RCS is intact with loops NOT filled.
- Time to Steam Release is 90 minutes.

Which of the following portions of 1203.028, Loss of Decay Heat Removal, should be in use?

A. Section 1, Loss of Inventory

B. Attachment G, Containment Closure

C. Section 2, DH Removal System Leak

D. Section 8, Loss of Both DH Systems -- RCS Pressure Boundary Intact

Answer:

A. Loss of Inventory

Notes:

"A" is correct, Section 1 should be in use with the given conditions.
"B" is incorrect, while it would be used with a LOCA, this section would not be in use until RCS pressure >150 psig and loops not filled, or time remaining to steam release is <60 min. and DH removal cannot be immediately restored, or RCS pressure > DH system maximum limit.
"C" is incorrect, while it is true this will combat a leak, this section is used for a leak in the DH System and not in the RCS.
"D" is incorrect, while it is true the RCS intact and DH has been isolated, this section would not be used until RCS temperature was >280°F.

References:

1203.028, Chg. 016-03-0

History:

New for 2005 SRO exam.

TABLE OF CONTENTS

SECTION 1 Loss of Inventory

ENTRY CONDITIONS

One or more of the following:

- DECAY HEAT FLOW HI/LO (K09-A8) alarm.
- Train A RCS LEVEL LO (K09-C7) alarm.
- Train B RCS LEVEL LO (K09-D7) alarm.
- Reduced level in the following:
  - Pressurizer
  - Hot leg
  - Tygon RCS level indication
  - **IF** Spent Fuel Transfer Tube (SF-45) open,
    THEN SF pool
- Possible rising level in the following:
  - RB sump
  - Aux. building sump
  - Aux. Building Equipment Drain Tank (T-11)
  - Makeup Tank (T-4)
  - Dirty Waste Drain Tank (T-20A/B)

SECTION 1 - Loss of Inventory

INSTRUCTIONS

1. **Stop the running DH pump(s).**

2. **Close at least one Decay Heat Suction valve:**

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

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

4. **IF due to Fuel Transfer Canal seal plate gasket failure,
   THEN GO TO Refueling Abnormal Operations (1203.042), "Transfer Canal Seal Plate Gasket Failure" section.**

> **NOTE**
>
> - IF DH system is isolated from RCS, THEN:
>   - Continued drop in RCS level indicates leak from the RCS.
>   - Continued drop in DH pump suction press with stable RCS level indicates leak from the DH Removal system.
> - IF DH system is NOT isolated from the RCS, THEN leak identification is by direct observation OR possibly process monitor indication of DH cooler tube leak per "DH Removal System Leak" section of this procedure.

5. **Determine whether leak is from RCS or DH Removal system.**

   A. IF from DH Removal system,
   THEN GO TO "DH Removal System Leak" section of this procedure.

   B. IF from RCS,
   THEN proceed with this section.

6. **IF maintenance activities in the Reactor Building could be affected by RCS level rise during refill,
   THEN perform local evacuation of affected areas.**

7. **IF RCS temp exceeds 280°F,
   THEN GO TO applicable "Loss of Both DH Systems" section of this procedure.**

(continued)

SECTION 1 - Loss of Inventory

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.

8. **IF any of the following conditions occur, THEN initiate containment closure per Attachment G of this procedure, while continuing with this section.**

   A. Time remaining to steam release is OR becomes <1 hour AND DH removal can NOT be immediately restored.

   B. RCS press >150 psig AND RCS loops NOT filled.

   C. RCS press > Decay Heat Sys. Max. Pressure limit of Plant Shutdown and Cooldown (1102.010), Attachment A.

NOTE

Minimum Height of Water to Avoid Vortex Formation vs. Decay Heat Flow is provided by Decay Heat Removal Operating Procedure (1104.004), Attachment B.

9. **Using one of the RCS makeup methods listed on Attachment H of this procedure, establish AND maintain RCS hot leg level per one of the following limits:**

   - ≥375'
   - Within "Operating Region" of 1104.004, Attachment B

(continued)

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

QID: 0331 **Rev:** 0 **Rev Date:** 9-6-99 **Source:** Repeat **Originator:** J.Cork

TUOI: A1LP-RO-EOP06 **Objective:** 5 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to determine or interpret the following as they apply to a SGTR: Plant conditions, from a survey of control room indications.

K/A Number: EA2.07 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	4.4	**RO Select:**	No	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.8	**SRO Select:**	Yes	**Taxonomy:**	A

Question: **RO:** **SRO:** 77

Following a controlled plant shutdown per 1202.006, Tube Rupture, the following conditions exist:

- RCS temperature is 460 °F
- RCS pressure is 900 psig
- Both OTSGs have ruptured tubes
- Both OTSGs have been isolated.
- BWST level drops below 23'

Which Emergency Operating Procedure should the CRS be using?

A. 1202.002, Loss of Subcooling Margin

b. 1202.010, ESAS

c. 1202.011, HPI Cooldown

d. 1202.006, Tube Rupture

Answer:

c. 1202.011, HPI Cooldown

Notes:

"a" is incorrect, Loss of Subcooling Margin is entered only if a Loss of SCM occurred prior to entry into the Tube Rupture EOP and conditions do not indicate a Loss of SCM. "b" is incorrect, ESAS is not entered from Tube Rupture. The ESAS would have had to occur prior to the Tube Rupture for entry into ESAS. "c" is correct, neither OTSGs is suitable for heat removal since both are ruptured and have been isolated. BWST level <23 feet with these conditions is an entry condition for 1202.011, HPI Cooldown. "d" is incorrect since 1202.006, Tube Rupture, provides guidance to transition to HPI Cooldown for the given conditions.

References:

1202.006, Chg. 007-04-0

History:

Used in 1999 exam -
Direct from Exam Bank, QID# 551 used in class exam.
Selected for 2002 SRO exam.
Repeated for use in 2005 SRO exam.

INSTRUCTIONS

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

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

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

CONTINGENCY ACTIONS

A. Perform the following:

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

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

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

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

(3) **GO TO step 53.A.2).**

b) **IF** ERV **cannot** be opened,
THEN adjust HPI as necessary to maintain RCS press/temp within limits of Figure 3.

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

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

(53. CONTINUED ON NEXT PAGE)

INSTRUCTIONS | **CONTINGENCY ACTIONS**

53. (Continued).

2) Verify bad SG Main Feedwater Isolation valve closed:

SG A	SG B
CV-2680	CV-2630

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

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

4) **IF** RCS press is >950 psig,
THEN reduce RCS press to ≤950 psig, while maintaining adequate SCM by any or all of the following:

a) Maintain emergency cooldown rate of ≤240°F/hr to 500°F.

b) Raise AUX Pressurizer Spray flow.

c) Maximize Letdown flow.

d) Throttle HPI.

e) Open High Point Vents:

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

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

(53. CONTINUED ON NEXT PAGE)

INSTRUCTIONS

53. (Continued).

54. **Before RCS temp drops to 390°F, verify no more than 3 RCPs running.**

55. **WHEN RCS temp drops to 300°F, THEN reduce RCS cooldown rate to ≤50°F/hr.**

CONTINGENCY ACTIONS

5) Verify bad SG ATM Dump ISOL open **AND** ATM Dump CNTRL valve in AUTO **AND** closed:

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

   a) **IF** ATM Dump CNTRL is open **AND** SG press is <1020 psig, **THEN** verify associated ATM Dump ISOL in MANUAL **AND** close.

6) Close bad SG MSIV:

SG A	SG B
CV-2691	CV-2692

   a) **IF** both SGs are isolated, **THEN GO TO 1202.011, "HPI COOLDOWN"** procedure.

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

QID: 0584 **Rev:** 0 **Rev Date:** 5/20/05 **Source:** New **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:** 3.1 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 78

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.

References:

1202.012, chg. 004-03-0, RT-14

History:

New for 2005 SRO exam.

NOTE

- PTS limits apply if <u>any</u> 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 **<u>OR</u>** SGTR is in progress, PZR cooldown rate limits **<u>do</u>** **<u>not</u>** apply.

14. Control RCS press within limits of Figure 3.

A. <u>IF</u> PTS limits apply or RCS leak exists,
<u>THEN</u> maintain RCS press <u>low</u> within limits of Figure 3.

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

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

1) Throttle makeup flow.

2) <u>IF</u> SCM is adequate, **<u>THEN</u>** throttle HPI flow.

3) Raise Letdown flow.

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

4) Verify ERV Isolation open (CV-1000) **<u>AND</u>** cycle ERV (PSV-1000).

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

1) Throttle makeup flow.

2) <u>IF</u> HPI is in service **<u>AND</u>** SCM is adequate **<u>AND</u>** any of the following conditions is met,
<u>THEN</u> throttle HPI flow:

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

a) <u>IF</u> ESAS has actuated **<u>AND</u>** HPI must be throttled,
<u>THEN</u> override **<u>AND</u>** throttle HPI.

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

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

5) Place Pressurizer Heaters in OFF.

(14. CONTINUED ON NEXT PAGE)

14. (Continued).

6) Raise Letdown flow.

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

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

E. **IF** RCS press is low, **THEN** raise press using one or more of the following:

1) Raise makeup flow.

2) Raise or initiate HPI flow (RT 2).

3) **IF** RCP is running, **THEN** verify Pressurizer Spray (CV-1008) closed.

4) Reduce Letdown flow.

5) Place Pressurizer Heaters in MANUAL.

CAUTION

If HPI cooling is in progress, ERV Isolation (CV-1000) must be left open until HPI cooling is no longer required.

6) Verify ERV (PSV-1000) or ERV Isolation closed.

CAUTION

With RCS solid, 1°F temp change can cause 100 psig press change.

F. **IF** PZR is solid, **THEN** RCS press may also be controlled by varying RCS temperature.

- Raise RCS temp to raise RCS press
- Lower RCS temp to lower RCS press

NOTE

Adjusting Pressurizer Level Control setpoint and HPI as necessary to maintain normal makeup flow on-scale will allow CV-1235 to automatically compensate for small changes in RCS leak rate and cooldown rate.

G. **IF** normal makeup is in service
AND
HPI is in service,
THEN adjust Pressurizer Level Control setpoint and HPI as necessary to maintain normal makeup flow on-scale.

END

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

QID: 0586 **Rev:** 0 **Rev Date:** 5/31/05 **Source:** Direct **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 029 **System Title:** Loss of Offsite Power

Description: Ablity to apply Technical Specifications for a system.

K/A Number: 2.1.12 **CFR Reference:** 43.2 / 43.5 / 45.3

Tier: 1 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 4

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question: RO: SRO: 79

Given:
- Plant is at 100% power with no failed equipment.
- A loss of the 161 KV ring bus occurs (de-energized).
- Autotransformer is energized from the 500 KV ring bus.

Providing the 161 KV ring bus remains de-energized, when is the plant required to be in Mode 3?

A. Within 24 hours

B. Within 36 hours

C. Within 72 hours

D. Within 84 hours

Answer:

D. Within 84 hours

Notes:

Answer "D" is correct, the time for Required Action A.3 must be added to Required Action F.1 to arrive at the correct time limit.
Answer "A" is incorrect, 24 hours is the completion time for Required Action A.2.
Answer "B" is incorrect, 36 hours is the completion time for Required Action A.2 added to the time for Condition F.
Answer "C" is incorrect, 72 hours is the completion time for Required Action A.3 alone.

References:

T.S. 3.8.1

This reference must be included in the student's exam handout!!!

History:

Direct from regular exambank, QID #3073
Selected for 2005 SRO exam

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit.	1 hour AND Once per 12 hours thereafter
	AND	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 ------------NOTE---------------- Startup Transformer No. 2 may be removed from service for up to 30 days for preplanned preventative maintenance. This 30 day Completion Time may be applied not more than once in any 10 year period. The provisions of LCO 3.0.4 are not applicable to Startup Transformer No. 2 during this 30 day preventative maintenance period. ------------------------------------ Restore required offsite circuit to OPERABLE status.	 72 hours <u>AND</u> 10 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s). <u>AND</u> B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable. <u>AND</u> B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure. <u>OR</u>	1 hour <u>AND</u> Once per 12 hours thereafter 4 hours from discovery of Condition B concurrent with inoperablity of redundant required feature(s) 24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG. <u>AND</u> B.4 Restore DG to OPERABLE status.	24 hours 7 days <u>AND</u> 10 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. <u>AND</u> C.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours
D. One required offsite circuit inoperable. <u>AND</u> One DG inoperable.	-------------------NOTE------------------- Enter applicable Conditions and Required Actions of LCO 3.8.6, "Distribution Systems – Operating," when Condition D is entered with no AC power source to any train. --- D.1 Restore required offsite circuit to OPERABLE status. <u>OR</u> D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	----------------------------NOTE---------------------------- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. --- Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves "ready-to-load" conditions.	31 days
SR 3.8.1.3	----------------------------NOTES---------------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and follow, without shutdown, a successful performance of SR 3.8.1.2. --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.	31 days

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

QID: 0587 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** New **Originator:** J.Cork

TUOI: A1LP-RO-TS **Objective:** 5 **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: AC instrument bus alarms for the inverter and alternate power source.

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

Tier: 1 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 4

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

Question: **RO:** **SRO:** 80

Given:
- Plant is at 100% power.
- Inverter Y-15 is tagged out for maintenance.

Subsequently, the annunciator K01-B5 "RS3 INVERTER TROUBLE" alarms.
A local investigation reveals the local alarms are "LOW INVERTER VOLTAGE" and "STATIC SWITCH TRANSFER".

Which of the following Technical Specification conditions should be entered?

A. 3.8.7 Condition "A"

B. 3.8.7 Condition "D"

C. 3.8.9 Condition "B"

D. 3.8.9 Condition "F"

Answer:

A. 3.8.7 Condition "A"

Notes:

Answer "A" is correct, the alarm is due to a failure of Y13, therefore 3.8.7.A entry is required.
Answer "B" is incorrect, although the conditions show that two inverters are out of service, the conditions do not reveal a problem with Y-11 or the green train inverters.
Answer "C" is incorrect, although there is an alarm on RS3, as long as it is powered from the alternate source, it is still considered operable for 3.8.9.
Answer "D" is incorrect, although one might assume a problem exists with two 120 AC buses since Y-13 and Y-15 have problems, as long as RS3 is powered from the alternate source, only 3.8.7.A applies.

References:

T.S. 3.8.7 and 3.8.9

These references must be included in the student's exam handout!!!

History:

New for 2005 SRO exam.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 The following inverters shall be OPERABLE.

a. Two Red Train inverters (Y11 and Y13, Y11 and Y15, or Y13 and Y15),

b. Two Green Train inverters (Y22 and Y24, Y22 and Y25, or Y24 and Y25), and

c. Inverter Y28

---NOTE---------------------------------------
One of the four inverters requried by LCO 3.8.7.a and LCO 3.8.7.b may be disconnected from its associated DC bus for ≤ 2 hours to perform load transfer to or from the swing inverter, provided:

a. The associated 120 VAC bus is energized from its alternate AC source; and

b. The other three 120 VAC buses are energized from their associated OPERABLE inverters.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	A.1 --------------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any of the 120 VAC buses RS1, RS2, RS3, or RS4 de-energized. ----------------------------------- Restore inverter to OPERABLE status.	24 hours <u>AND</u> 96 hours from discovery of failure to meet LCO

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Inverter Y28 inoperable.	B.1 --------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with 120 VAC bus C540 de-energized. -------------------------------------- Restore inverter to OPERABLE status.	72 hours <u>AND</u> 96 hours from discovery of failure to meet LCO
C. Inverter Y28 inoperable. <u>AND</u> One of the two Red Train inverters required by LCO 3.8.7.a inoperable.	C.1 Restore one inverter to OPERABLE status.	2 hours
D. Required Action and associated Completion Time not met. <u>OR</u> Two or more of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to associated 120 VAC buses RS1, RS2, RS3, RS4, and C540.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Two AC, DC, and 120 VAC electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystem(s) inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One or more 120 VAC electrical power distribution subsystem(s) (RS1, RS2, RS3, RS4) inoperable.	B.1 Restore 120 VAC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
C. 120 VAC electrical power distribution subsystem C540 inoperable.	C.1 Enter applicable Conditions and Required Actions of LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation," LCO 3.3.15, "Post Accident Monitoring (PAM) Instrumentation," and LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage."	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DC electrical power distribution subsystem(s) inoperable.	D.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	12 hours 36 hours
F. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	7 days

Location: C10

Device and Setpoint: N/A

RS3 INVERTER TROUBLE

Alarm: K01-B5

1.0 OPERATOR ACTIONS

1. Check SPDS ACDC display for indication of problem.

2. Go to local alarm panel K1621 or K1654 on Inverter Y13 or Y15 depending on which one is in service.

3. Refer to Attachment F for further instructions.

4. Reference TS 3.8.7, TS 3.8.8, TS 3.8.9 and TS 3.8.10 for operability requirements.

2.0 PROBABLE CAUSES

NOTE

This annunciator has reflash capability. If the K01-B5 window is lit solid due to one K1621 or K1654 alarm and another K1621 or K1654 alarm actuates, K01-B5 will go into fast flash with an audible alarm.

1. Low DC Input Voltage
2. Hi DC Input Voltage
3. Low Inverter Voltage
4. Hi Inverter Voltage

5. Inverter Failure
6. Out of Sync
7. Fan Failure
8. Static Switch Transfer

9. Hi Temp
10. Alternate Source Trouble
11. Low System Voltage
12. Hi System Voltage

3.0 REFERENCES

1. Schematic Diagram Annunciator K01 (E-451)
2. Inverter Y11, Y13, Y22, Y24 Local Reflash Annunciator Schematic Diagram (E-418, Sheet 9)

ATTACHMENT F

Location: Inverter Y13 or Y15

Device and Setpoint: 118V AC

o	LOW INVERTER VOLTAGE

Alarm: K1621 or K1654-03

1.0 OPERATOR ACTIONS

NOTE

- TS 3.8.7 allows transfer between inverters without entry into TS 3.8.7 Condition A for ≤2 hours provide:
  - 120 VAC vital bus remains energized from alternate source.
  - All other 120 VAC vital buses are powered from operable inverters.
- Any condition other than normal transfer which results in a 120 VAC vital bus being powered from inverter on alternate source requires entry into TS 3.8.7 Condition A which includes 24 hour time clock.
- Failure to exit TS 3.8.7 Condition A within 24 hours requires plant shutdown in accordance with TS 3.8.7 Condition D.

1. If Alternate Source Supplying Load light is on, transfer Manual Selector Switch to ALT SOURCE TO LOAD position.
2. Investigate and correct cause of alarm, as necessary.
3. Restore Y13 or Y15 to normal operation per Inverter and 120V Vital AC Distribution (1107.003).

2.0 PROBABLE CAUSES

1. Inverter failure

3.0 REFERENCES

1. Schematic Diagram Inverter Y11, Y13, Y22, Y24 Local Reflash Annunciator (E-418 sh 9)

ATTACHMENT F

Location: Inverter Y13 or Y15

Device and Setpoint: N/A

o	STATIC SWITCH TRANSFER

Alarm: K1621 or K1654-08

1.0 OPERATOR ACTIONS

NOTE

- TS 3.8.7 allows transfer between inverters without entry into TS 3.8.7 Condition A for ≤2 hours provide:
  - 120 VAC vital bus remains energized from alternate source.
  - All other 120 VAC vital buses are powered from operable inverters.
- Any condition other than normal transfer which results in a 120 VAC vital bus being powered from inverter on alternate source requires entry into TS 3.8.7 Condition A which includes 24 hour time clock.
- Failure to exit TS 3.8.7 Condition A within 24 hours requires plant shutdown in accordance with TS 3.8.7 Condition D.

1. Transfer Manual Selector Switch to ALT SOURCE TO LOAD position.
2. Investigate and correct cause of alarm, as necessary.
3. Restore Y13 or Y15 to normal operation per Inverter and 120V Vital AC Distribution (1107.003).

2.0 PROBABLE CAUSES

1. Inverter failure
2. Static Switch in Alternate Source to Load.

3.0 REFERENCES

1. Schematic Diagram Inverter Y11, Y13, Y22, Y24 Local Reflash Annunciator (E-418 sh 9)

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

QID: 0588 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** New **Originator:** J.Cork
TUOI: A1LP-RO-EOP04 **Objective:** 11 **Point Value:** 1

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

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

Description: Ability to determine and interpret the following as they apply to the (Inadequate Heat Transfer): 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:** 3.2 **RO Select:** No **Difficulty:** 3
Group: 1 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 81

Given:

- Reactor tripped due to a loss of both MFWPs approximately 20 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. 004-02-0

History:

New for 2005 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.

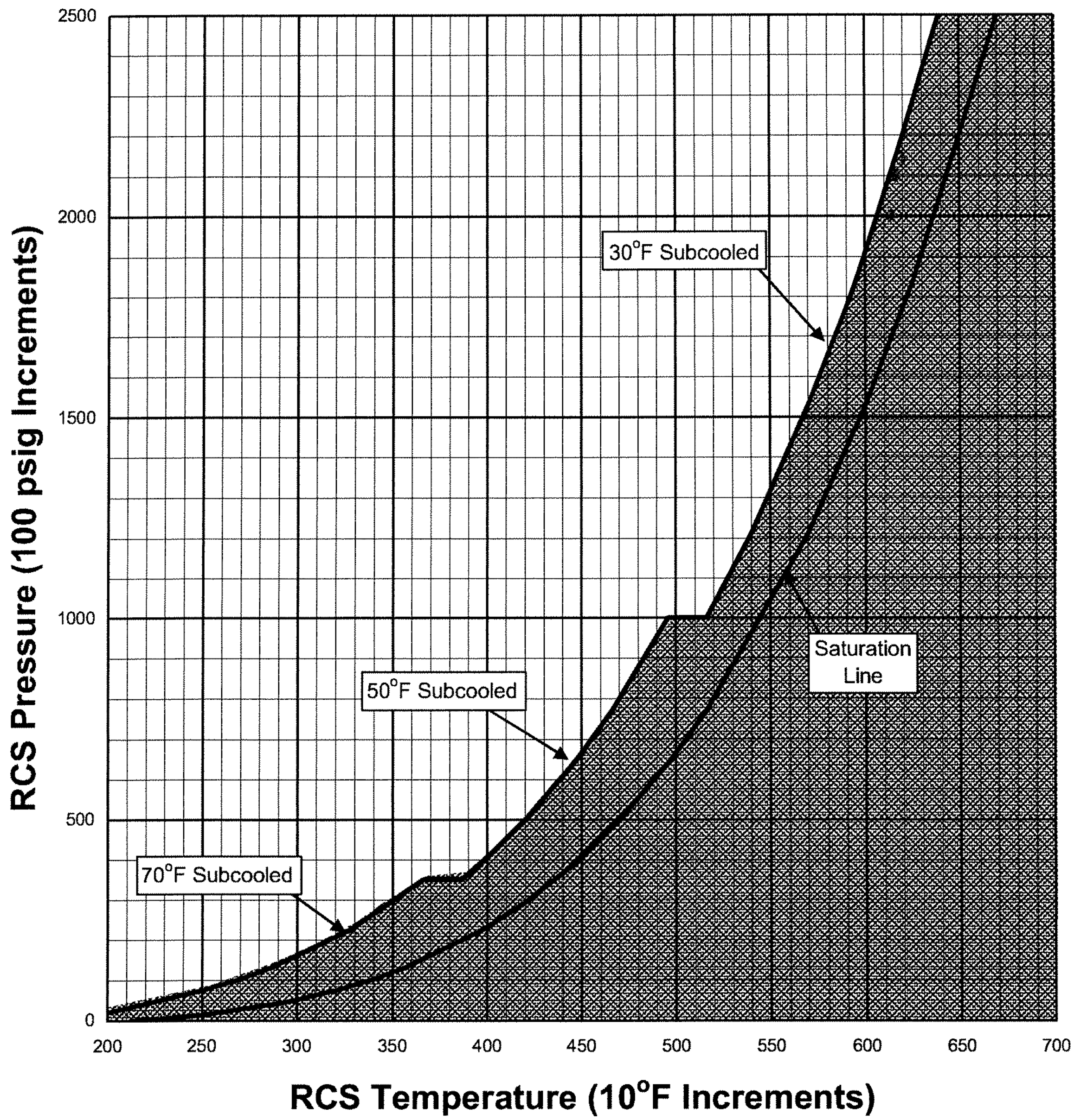
- CET temp rising above 610°F

  AND

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

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

FIGURE 1
Saturation and Adequate SCM



RCS Pressure	Adequate SCM
>1000 psig	≥30°F
350 to 1000 psig	≥50°F
<350 psig	≥70°F

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

QID: 0589 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** New **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

What are the required action(s) per Technical Specifications for the following conditions?

- 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.

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

Do not include this spec in the student handout!!!

History:

New for 2005 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 <u>AND</u> Once per 12 hours thereafter
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2.1 Restore CONTROL ROD alignment.	2 hours
	<u>OR</u>	
	A.2.2.1 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours
	<u>AND</u>	
	A.2.2.2 Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	72 hours
	<u>AND</u>	

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
	<u>OR</u>		
	C.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	C.2	Be in MODE 3.	6 hours

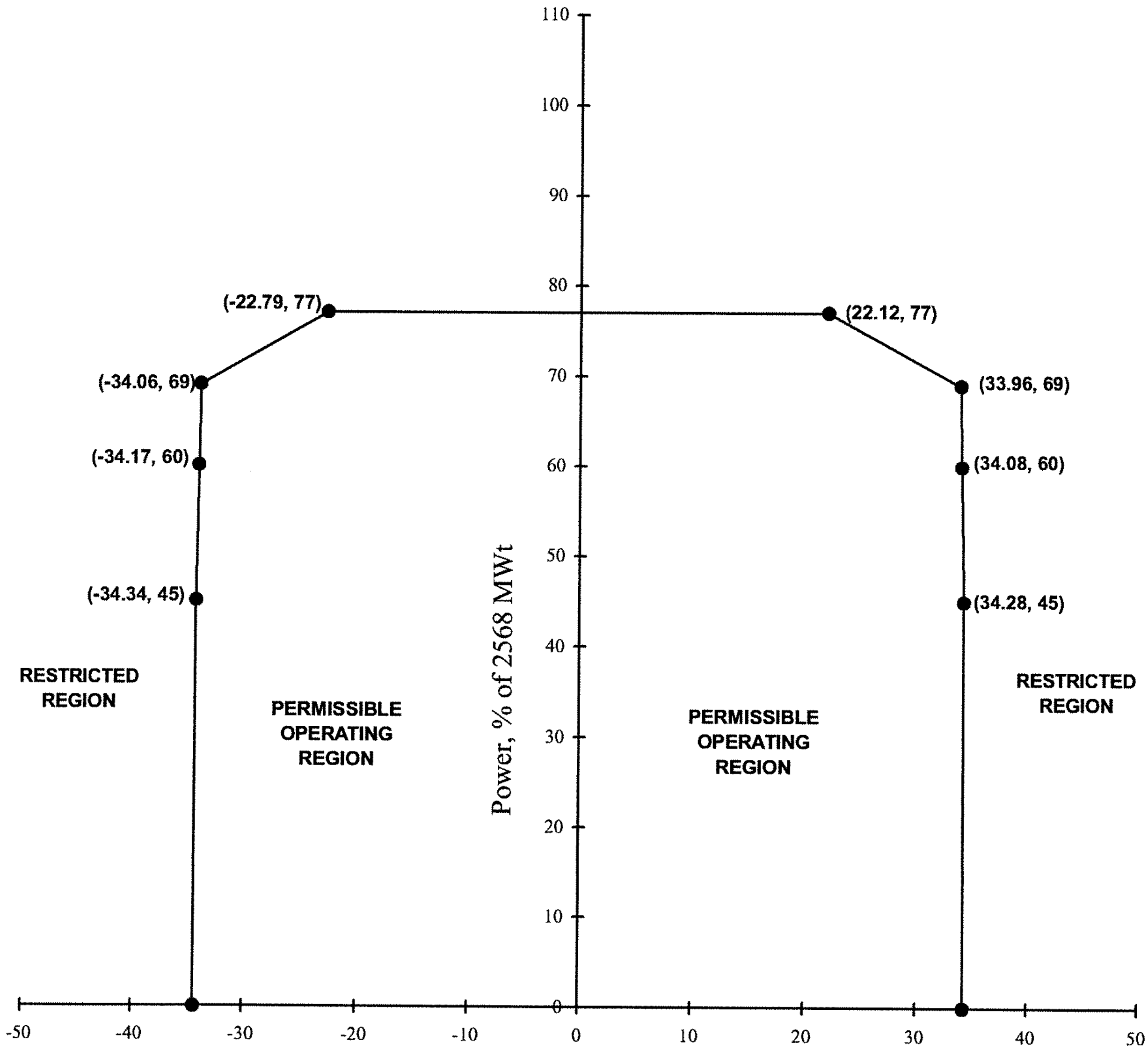
SURVEILLANCE REQUIREMENTS

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

FIGURE 7-A

AXIAL POWER IMBALANCE Setpoints for Full In-Core Conditions for Three-Pump Operation

(Figure is referred to by Technical Specification 3.2.3)



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

QID: 0590 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** New **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Tech-spec limits if both intermediate-range channels have failed.

K/A Number: AA2.10 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 3.1 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 83

Given:

- Plant startup in progress.
- Channel 1 Source Range, NI-501 at 9 E4 cps
- Channel 2 Source Range, NI-502 at 1 E5 cps
- Reactor power Wide Range recorder, NR-502, is operable and at 5 E -2% power
- Intermediate Range Channel NI-3 at 2 E -11 amps
- Intermediate Range Channel NI-4 at 5 E -11 amps
- Power Range Channels NI-5 thru 8 at 0%

Which of the following actions are procedurally required and ensure compliance with Technical Specification 3.3.10, Intermediate Range Neutron Flux?

A. Trip the reactor immediately and refer to 1202.001, Reactor Trip.

B. Immediately suspend positive reactivity additions and initiate a shutdown so that all CRD breakers are open within one hour.

C. Lower power until Source Range is on scale and corrective maintenance is performed on appropriate IR channel(s).

D. Since NR-502 is operable, continue with plant operations until Power Range channels come on-scale.

Answer:

B. Immediately suspend positive reactivity additions and initiate a shutdown so that all CRD breakers are open within one hour.

Notes:

Answer "B" is IAW 1203.021 and TS 3.3.10 since both IR channels are inoperable.
Answer "A" is incorrect, this would be applicable only if no on-scale neutron flux existed.
Answer "C" is incorrect, this is the required action for improper overlap.
Answer "D" is incorrect, this would be the action if only one IR channel was inoperable.

References:

T.S. 3.3.10
1203.021, Chg. 008-01-0
STM 1-67, Fig. 67-1

History:

New for 2005 SRO exam.

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One intermediate range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODE 2
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	------------------NOTE------------------ Plant temperature changes are allowed provided the temperature change is accounted for in the SDM calculations. --	
	A.1 Suspend operations involving positive reactivity changes.	Immediately
	<u>AND</u>	
	A.2 Open CRD trip breakers.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.10.2	Perform CHANNEL FUNCTIONAL TEST.	31 days

SECTION 2

LOSS OF ONE OR MORE INTERMEDIATE RANGE NI CHANNELS IN MODE 2

1.0 SYMPTOMS

1.1 Intermediate range indication reading incorrectly.

1.2 RPS TROUBLE (K08-C3) alarm.

1.3 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 refer to Emergency Operating Procedure (1202.XXX Series).

CAUTION

Reactor condition must be such that delays which are necessary to perform the following steps will not result in reactor going critical due to xenon, etc.

3.2 Check the failed intermediate range channel(s) power supply voltages at detector P.S. modules in associated RPS cabinets:

- Detector power supply: 595 to 605 VDC
- Auxiliary power supply: 17 to 23 VDC

3.2.1 IF normal voltage is not indicated, THEN reset power supplies as follows:

A. Check detector power supply toggle switch ON.

B. Check auxiliary power supply toggle switch ON.

SECTION 2

LOSS OF ONE OR MORE INTERMEDIATE RANGE NI CHANNELS IN MODE 2
(Continued)

NOTE
Resetting intermediate range channel power supplies will cause voltage spikes and neutron power signal spikes, which may alarm annunciators CRD WITHDRAWAL INHIBITED (K08-A2) and HI SUR ROD HOLD (K08-B2).

C. Depress both toggle switches to RESET position.

D. IF intermediate range indication returns to normal, THEN continue plant operations.

3.3 IF only one intermediate range channel is operable, OR 2 of 4 power range instrument channels indicate >5% power, THEN continue plant operations (TS 3.3.10).

3.4 IF all three of the following conditions are met,

- Both intermediate range channels have failed,
- 3 of 4 power range instruments indicate ≤5%,
- Reactor Power Wide Range Recorder (NR-502) is available,

THEN perform the following:

NOTE
Plant temperature changes which result in positive reactivity additions are allowed provided the temperature change is accounted for in the Shutdown Margin calculations (TS 3.3.10 Condition A).

- Refer to TS 3.3.10.
- Immediately suspend operations involving positive reactivity changes.
- Initiate a shutdown in order to have all CRD trip breakers open (Mode 3) within 1 hour.
- IF reactor power is ≥2%, THEN perform applicable steps of Power Reduction and Plant Shutdown (1102.016).
- Concurrently with reactor shutdown, monitor reactor power using NR-502.

  A. WHEN ~1E-4 log reactor power on NR-502 is reached, THEN observe that source range indicators come on scale.

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

STM 1-67

Nuclear Instrumentation

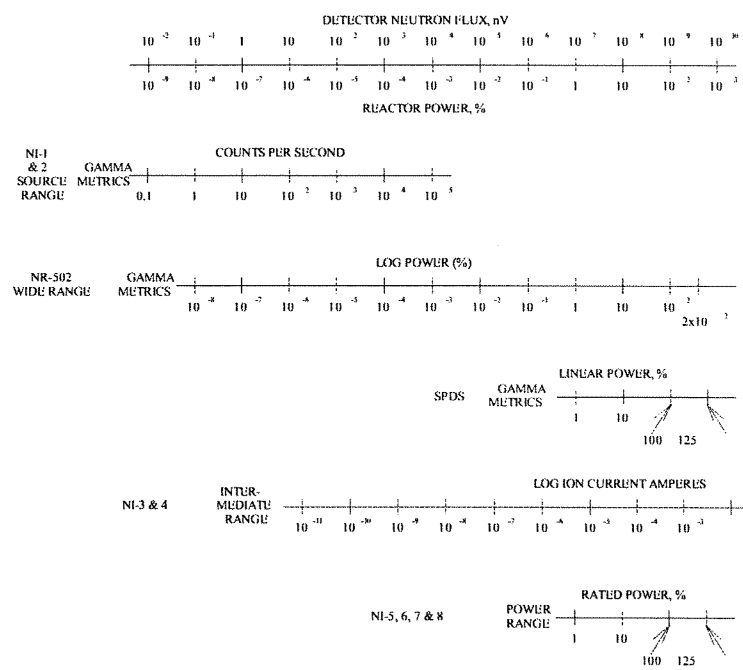
1.0 Introduction

This STM contains information on the Excore (Out of Core) Nuclear Instrument System (NIs) for ANO Unit 1. It includes operational theory of detectors, component locations in the plant and normal and abnormal operations and equipment conditions. The effect nuclear instruments have on plant operation, and the effect plant operations have on the Nuclear Instruments is discussed. Additional information on theory of detector operation is found in STM 1-62, Radiation Monitoring.

1.1 System Function

The Nuclear Instrumentation (NI) System is designed to measure over twelve decades of neutron flux using ten channels of out of core neutron detectors and instrumentation. (Refer to Figure 67.01) The full range of indications are displayed to the Reactor Operator and are supplied to the Reactor Protection and Integrated Control systems. Measurement ranges are designed to overlap to provide complete and continuous information of the full operating range of the reactor.

FIGURE 67.01: NUCLEAR INSTRUMENTATION FLUX RANGES



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

QID: 0591 **Rev:** 0 **Rev Date:** 6/6/05 **Source:** New **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.

INSTRUCTIONS	CONTINGENCY ACTIONS

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

INSTRUCTIONS	CONTINGENCY ACTIONS
2. **IF any combination of both NNI X and NNI Y power is lost, THEN perform the following:**	2. **RETURN TO step 1.**
A. Trip the Rx **AND** perform **1202.001, "REACTOR TRIP"** in conjunction with this procedure.	
B. Manually actuate EFW **AND** 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. **IF** TURB BYP valves are **not** available, **THEN** 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:** New **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 Cooldown

Description: Knowledege of EOP implementation hierarchy and coordination with other support procedures.

K/A Number: 2.4.6 **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 shutdown using 1203.039, Excess RCS Leakage, and 1203.045, Rapid Plant Shutdown, due to RCS leak into 2 RCP seal coolers.
- "C" & "B" RCPs are running
- RCS pressure 1720 psig and lowering slowly
- HPI flow 150 gpm
- A & B SG pressure 910 psig
- RCS cooldown rate 35°F per hour
- All Turbine bypass valves closed

Which procedure should be in use?

A. 1202.001, Reactor Trip

B. 1203.041, Small Break LOCA cooldown

C. 1203.040, Forced Flow cooldown

D. 1202.010, ESAS

Answer:

B. 1203.041, Small Break LOCA cooldown

Notes:

Answer "B" is correct with an uncontrolled cooldown continuing due to break/HPI flow, regardless of SG status.
Answer "A" is incorrect, Reactor Trip EOP would be transitioned to if cause of ES actuation had been present and was corrected.
Answer "C" is incorrect, although RCPs are running, there is no control of the cooldown.
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. 007-03-0

History:

New for 2005 SRO exam.

H. (continued)

CAUTION

Closing the seal bleed off path on a running RCP will cause seal damage due to overheating.

NOTE

Closing the seal bleed off valves after the RCP is tripped limits the heatup rate of the seal.

5. For any RCP tripped due to inadequate seal cooling, place associated RCP Seal Bleed off (Alternate Path to Quench Tank) in CLOSE and close associated RCP Seal Bleed off (Normal):

RCP	RCP Seal Bleed off (Alternate Path to Quench Tank)	RCP Seal Bleed off (Normal)
P-32A	SV-1273	CV-1273
P-32B	SV-1272	CV-1272
P-32C	SV-1271	CV-1271
P-32D	SV-1270	CV-1270

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

6. 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)

2) IF any RCP is running,
THEN perform Forced Flow Cooldown (1203.040)

3) IF all RCPs are off,
THEN perform Natural Circulation Cooldown (1203.013).

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

QID: 0594 **Rev:** 0 **Rev Date:** 6/8/05 **Source:** New **Originator:** J.Cork

TUOI: A1LP-RO-DH **Objective:** 27 **Point Value:** 1

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown.

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

Tier: 2 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 4

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

Question: **RO:** **SRO:** 86

Given:

- RCS pressure is 200 psig.
- RCS temperature is 150°F.
- "A" DH pump is in service.
- I&C is performing "A" RPS channel pressure calibration.

Subsequently RCS pressure is observed to be dropping rapidly.

Which of the following procedures should be used to mitigate this transient?

A. 1203.015, Pressurizer Systems Failure

B. 1203.012H, K09-D8 "DECAY HEAT VORTEX WARNING"

C. 1203.028, Loss of Decay Heat Removal

D. 1203.012H, K09-B7 "CV-1050 AUTO CLOSE"

Answer:

A. 1203.015, Pressurizer Systems Failure

Notes:

Answer "A" is correct, calibration of "A" RPS pressure will lift the ERV and entry into 1203.015 will lead the operators to isolate the ERV.
Answer "B" is incorrect, Decay Heat flow will change somewhat but his will not be sufficient to bring in this alarm.
Answer "C" is incorrect, even though inventory is being lost this procedure will not lead to isolating the ERV.
Answer "D" is incorrect, this pressure channel will not cause auto isolation of DH.

References:

1203.015, Chg. 011-00-0
STM 1-69, Rev. 8

History:

New for 2005 SRO exam.

SECTION 1 -- ELECTROMATIC RELIEF VALVE (PSV-1000) SYSTEM FAILURE OR LEAK

1.0 SYMPTOMS

One or more of the following:

- SPDS alarm on T1025 at 200°F: ERV PSV-1000 OUTLET TEMP
- Quench Tank (T-42) temperature, level, or pressure rising.
- Rise on acoustic Relief Valve Monitor (VYI-1000A).
- Annunciator alarm RELIEF VALVE OPEN (K09-A1).
- ERV indicates open on C04.
- Pressurizer ERV Isolation Valve (CV-1000) inoperable.
- ERV (PSV-1000) inoperable.
- Failure of both ERV acoustic monitors.

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

NOTE

CV-1000 torque switch can be overridden in the OPEN or CLOSE direction by holding the hand switch in the respective position.

3.1 Close Pressurizer ERV Isolation Valve (CV-1000).

3.2 If ERV leakage with CV-1000 closed exceeds capability to maintain RC pressure, trip reactor and refer to Emergency Operating Procedure series (1202.XXX).

3.3 If closing CV-1000 stops leak, perform the following:

3.3.1 Continue power operations with ERV isolated.

3.3.2 Notify Ops Manager.

3.3.3 Log in station log and on plant status board.

SECTION 1 -- ELECTROMATIC RELIEF VALVE (PSV-1000) SYSTEM FAILURE OR LEAK (continued)

3.4 If closing CV-1000 does not stop leak, perform the following:

3.4.1 Initiate leak-rate determination per "RCS Leakage Monitoring" section of RCS Leak Detection (1103.013).

3.4.2 If total RC leakage exceeds limit allowed by Tech Spec, perform Rapid Plant Shutdown (1203.045).

A. Refer to Emergency Action Level Classification (1903.010) to determine emergency class.

3.4.3 If total PZR steam space leakage is >1 gpm, initiate a Condition Report and perform an operability determination within 24 hours.

A. If leakage is evaluated as unsafe, commence plant shutdown per Power Reduction and Plant Shutdown (1102.016), and Plant Shutdown and Cooldown (1102.010).

3.4.4 Monitor Quench Tank (T-42) pressure, level, and temperature. Maintain quench tank parameters within limits described in Pressurizer Operation (1103.005).

3.5 If ERV (PSV-1000), ERV Isolation Valve (CV-1000), or both ERV acoustic monitors are inoperable, perform the following:

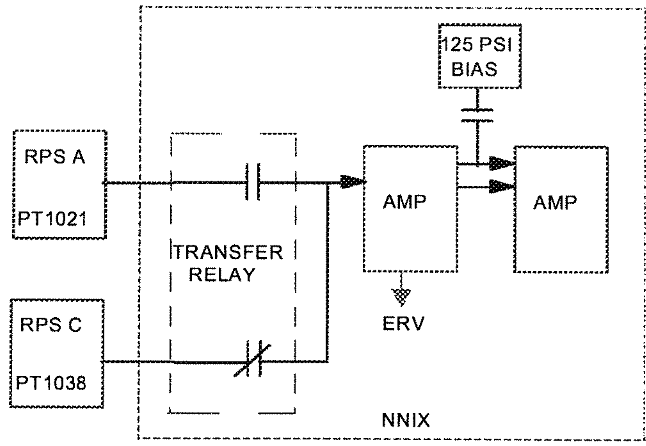
3.5.1 Close ERV Isolation.

3.5.2 Maintain ERV vent path closed. Refer to TRM 3.4.2.

3.5.3 Place caution tag on CV-1000 hand switch stating "Use as required by EOP and then only as a last resort".

3.5.4 Initiate a Condition Report.

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. With the handswitch in the AUTO position, the signal monitor opens and closes the ERV. When either handswitch is placed in the OPEN position, the ERV solenoid is energized and ERV is opened.

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

QID: 0595 **Rev:** 0 **Rev Date:** 6/8/2005 **Source:** New **Originator:** S.Pullin

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

Section: 3.5 **Type:** Containment Integrity

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

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

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

Tier: 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 87

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 2.1.2 and bases

History:

New for 2005 SRO exam.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5×10^{-3} x (Burnup, MWD/MTU)°F) for TACO2 applications and ≤ 4642 - (5.8×10^{-3} x (Burnup, MWD/MTU)°F) for TACO 3 applications.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation.

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the Core Operating Limits Report, so that the safety limits are met.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits AND be in MODE 3 within 1 hour.

2.2.3 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits AND be in MODE 3 within 1 hour.

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection (Ref. 1). Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. One safety valve is required for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 2). The required lift pressure is 2500 psig + 1%, - 3%. The safety valves discharge steam from the pressurizer to a quench tank located in the reactor building. The discharge flow is indicated by acoustic flow monitoring devices, by an increase in temperature downstream of the safety valves, and by an increase in the quench tank temperature, pressure, and level.

The upper and lower as-left pressure limits are based on the ± 1% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

The overpressure protection analysis (Ref. 3) is based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). One pressurizer code safety valve is capable of preventing overpressurization in MODE 3 and in MODE 4 with RCS temperature > 262°F since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat (Ref. 1 and 4). These valves must accommodate pressurizer insurges that

APPLICABLE SAFETY ANALYSES (continued)

could occur during a startup, rod withdrawal, or ejected rod event. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at low power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

In MODES 1 and 2, pressurizer safety valves satisfy Criterion 3 of the 10 CFR 50.36 (Ref. 5). In MODE 3 and MODE 4 above the LTOP enable temperature, the pressurizer safety valves satisfy Criterion 4 of 10 CFR 50.36.

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The upper and lower as-left pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 2) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

The LCO is modified by four Notes. Note 1 states that in MODE 3 and MODE 4 with RCS temperature above 262°F, only one pressurizer safety valve is required to be OPERABLE. In this condition, one pressurizer safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than the sum of the available heat sources.

Note 2 allows entry into MODE 3, and into MODE 4 with RCS temperature > 262°F, with the lift settings potentially outside the limits. This permits testing of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

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

QID: 0596 **Rev:** 0 **Rev Date:** 6/8/05 **Source:** New **Originator:** J.Cork

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

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to analyze the affect of maintenance activities on LCO status.

K/A Number: 2.2.24 **CFR Reference:** 43.2 / 45.13

Tier: 2 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 88

Given:

- Plant is at 100% power.
- RB Cooling Supply Fan VSF-1B is tagged out due to a faulty motor.

During the surveillance on EDG #2, the diesel's output breaker failed to close.

Regarding Reactor Building Spray and Cooling Systems, what are the required actions in Tech Specs?

A. 3.6.5 Required Action B.1

B. 3.6.5 Required Action C.1

C. 3.6.5 Required Action E.1

D. 3.6.5 Required Action G.1

Answer:

B. 3.6.5 Required Action C.1

Notes:

Answer "B" is correct, both RB cooling trains are inoperabe due to the red train having one inoperable fan and the green train's emergency power supply is inoperable (redundant required feature per 3.8.1.B).
Answer "A" is incorrect, more than one RB cooling train is inoperable due to 3.8.1.B.
Answer "C" is incorrect, more than one RB cooling train is inoperable due to 3.8.1.B, plus the plant is in Mode 1.
Answer "D" is incorrect, the RB Spray trains are operable.

References:

T.S. 3.6.5

This reference must be included in the student's exam handout!!!

History:

New for 2005 SRO exam.

3.6 REACTOR BUILDING SYSTEMS

3.6.5 Reactor Building Spray and Cooling Systems

LCO 3.6.5 Two reactor building spray trains and two reactor building cooling trains shall be OPERABLE.

---NOTE---
Only one train of reactor building spray and one train of reactor building cooling are required to be OPERABLE during MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One reactor building spray train inoperable in MODE 1 or 2.	A.1 Restore reactor building spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One reactor building cooling train inoperable in MODE 1 or 2.	B.1 Restore reactor building cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
C. Two reactor building cooling trains inoperable in MODE 1 or 2.	C.1 Restore one reactor building cooling train to OPERABLE status.	72 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours
E. One required reactor building spray train inoperable in MODE 3 or 4. <u>OR</u> One required reactor building cooling train inoperable in MODE 3 or 4.	E.1 Restore required inoperable train to OPERABLE status.	36 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 5.	36 hours
G. Two reactor building spray trains inoperable in MODE 1 or 2. <u>OR</u> Any combination of three or more trains inoperable in MODE 1 or 2. <u>OR</u> One required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4.	G.1 Enter LCO 3.0.3.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 ------------NOTE----------------- Startup Transformer No. 2 may be removed from service for up to 30 days for preplanned preventative maintenance. This 30 day Completion Time may be applied not more than once in any 10 year period. The provisions of LCO 3.0.4 are not applicable to Startup Transformer No. 2 during this 30 day preventative maintenance period. ------------------------------------- Restore required offsite circuit to OPERABLE status.	 72 hours AND 10 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s). AND B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable. AND B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure. OR	1 hour AND Once per 12 hours thereafter 4 hours from discovery of Condition B concurrent with inoperablity of redundant required feature(s) 24 hours

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

QID: 0597 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-FH **Objective:** 31 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Montioring

Description: Knowledge of new and spent fuel movement procedures.

K/A Number: 2.2.28 **CFR Reference:** 43.7 / 45.13

Tier: 2 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 89

Given:
- Reactor in cold shutdown.
- SPING #3 detector is in alarm showing high stack activity.
- Local radiation monitors alarming on 404' in Aux. Bldg.
- Core off load is in progress.
- Bubbles near upender in Aux. Bldg.

Which procedures contain guidance to combat the above conditions?

A. Control of Unit 1 Refueling, 1502.004, and High Activity in Reactor Coolant, 1203.019.

B. Refueling Abnormal Operations, 1203.042, and Annunciator Corrective Action, 1203.012

C. Control of Fuel and Control Component Handling in Unit 1 Spent Fuel Area, 1502.010, and Annunciator Corrective Action, 1203.012

D. Fuel and Control Component Handling, 1506.001, and Moderator Dilution, 1203.017

Answer:

B. Refueling Abnormal Operations, 1203.042, and Annunciator Corrective Action, 1203.012

Notes:

Answer "B" lists the correct procedures for a fuel handling accident.
Answer "A" is incorrect, 1502.004 is used for normal fuel handling activities. 1502.004 directs one to 1203.042.
Answer "C" is incorrect, 1502.010 is used specifically to guide fuel handling in the Spent Fuel pool but contains no actions for a fuel handling accident.
Answer "D" is incorrect, 1506.001 is used specifically for operation of the bridge but contains no actions for a fuel handling accident.

References:

1203.042, Chg. 005-03-0
1203.012I, Chg. 040-06-0

History:

Direct from regular exam bank QID#3026
Selected for 2005 SRO exam.

SECTION 1 -- FUEL HANDLING ACCIDENT

ENTRY CONDITIONS

Report of any of the following:

- Failure of electrical and/or mechanical interlocks on fuel handling equipment which may have caused damage to a spent fuel assembly.
- Bubbles emerging from a submerged fuel assembly which has been dropped or damaged.
- Visual inspection reveals bent, twisted, or warped spent fuel assembly.
- High or abnormal air activity indications on installed or portable air monitor.
- Observed parameters on the fuel handling bridge indicate a spent fuel assembly has been dropped.
- Dry Fuel Storage cask drop event.
- Any event during transfer of spent fuel to the dry fuel storage system where the integrity of the spent fuel is compromised or threatened.

Location: C16

Device and Setpoint:

RDACS computer in alarm caused by:

- EAL Report generated
- 10 minute average wind speed >50 mph
- Loss of communication between "A" and "B" RDACs servers

RDACS RADIATION HI

Alarm: K10-A1

1.0 OPERATOR ACTIONS

1. If EAL Report generated, notify Nuclear Chemistry to evaluate release per Offsite Dose Projections -- RDACS Computer Method (1904.002) and Emergency Action Level Classification (1903.010).

2. If high wind speed, refer to "High Wind/Tornado/Thunderstorm" section of Natural Emergencies (1203.025).

3. Notify Unit 2 Shift Manager of the condition.

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.

1. RDAC server communication problem
2. EAL Report generated
3. 10 minute average wind speed >50 mph
4. High radiation on Channel 5 from any SPING

3.0 REFERENCES

Schematic Diagram Annunciator K10 (E-460, sheets 1 - 3)
CR-2-97-0288-04, software provides reflash

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

QID: 0598 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-R-AOP **Objective:** 6 **Point Value:** 1

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the SWS System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 /45.13

Tier:	2	**RO Imp:**	3.5	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 90

Unit at 100%.

While performing Loss of Service Water AOP 1203.030, Service Water Loop 1 has been declared inoperable.

What is the required action with the shortest time duration per the affected Tech Spec actions?

A. Perform action for TS 3.7.7, condition B.1

B. Perform SR 3.8.1.2

C. Perform SR 3.8.1.1

D. Perform actions for TS 3.4.6, condition A.1

Answer:

C. Perform SR 3.8.1.1

Notes:

"C" is correct since it must be performed within one hour.
"A" is incorrect, this is performed within 80 hours.
"B" is incorrect this is performed within 24 hours.
"D" is incorrect, this says immediately and is referred to by 3.7.7 but is for RCS and DH loops and is not applicable.

References:

T.S. 3.8.1

Note: T.S. 3.4, 3.7, & 3.8 must be in students' handout.

History:

Direct from regular exam bank, QID#ANO-OpsUnit1-05954
Selected for 2005 SRO exam.

3.7 PLANT SYSTEMS

3.7.7 Service Water System (SWS)

LCO 3.7.7 Two SWS loops shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS loop inoperable.	A.1 --------------NOTES------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by SWS. 2. Enter Applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal made inoperable by SWS. ----------------------------------- Restore SWS loop to OPERABLE status.	 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----------------------------NOTE----------------------------- Isolation of SWS flow to individual components does not render the SWS inoperable. --- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.7.3	Verify each required SWS pump starts automatically on an actual or simulated signal.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit. AND A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable. AND	1 hour AND Once per 12 hours thereafter 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 ------------NOTE----------------- Startup Transformer No. 2 may be removed from service for up to 30 days for preplanned preventative maintenance. This 30 day Completion Time may be applied not more than once in any 10 year period. The provisions of LCO 3.0.4 are not applicable to Startup Transformer No. 2 during this 30 day preventative maintenance period. ------------------------------------- Restore required offsite circuit to OPERABLE status.	 72 hours AND 10 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s). AND B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable. AND B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure. OR	1 hour AND Once per 12 hours thereafter 4 hours from discovery of Condition B concurrent with inoperablity of redundant required feature(s) 24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG. AND B.4 Restore DG to OPERABLE status.	24 hours 7 days AND 10 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND C.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours
D. One required offsite circuit inoperable. AND One DG inoperable.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.6, "Distribution Systems – Operating," when Condition D is entered with no AC power source to any train. -- D.1 Restore required offsite circuit to OPERABLE status. OR D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	12 hours 36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	------------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. --- Verify each DG starts from standby conditions and, in $\leq$ 15 seconds achieves "ready-to-load" conditions.	31 days
SR 3.8.1.3	-----------------------------NOTES----------------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and follow, without shutdown, a successful performance of SR 3.8.1.2. --- Verify each DG is synchronized and loaded and operates for $\geq$ 60 minutes at a load $\geq$ 2475 kW and $\leq$ 2750 kW.	31 days

SURVEILLANCE		FREQUENCY
SR 3.8.1.4	Verify each day tank contains ≥ 160 gallons of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage tanks to the day tank.	31 days
SR 3.8.1.7	-----------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. --- Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.	 18 months
SR 3.8.1.8	-----------------------------NOTE------------------------------ All DG starts may be preceded by an engine prelube period. --- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves “ready-to-load” conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes.	 18 months

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	-----------------------------NOTE----------------------------- All DG starts may be preceded by an engine prelube period. --- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in $\leq$ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected emergency loads through load sequencing timers, and 4. supplies connected loads for $\geq$ 5 minutes.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops – MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and one OPERABLE loop shall be in operation.

---------------------------------------NOTE---------------------------------------
All reactor coolant pumps (RCPs) and DHR pumps may be removed from operation for ≤ 1 hour provided:

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

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

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status. AND A.2 -------------NOTE--------------- Only required if DHR loop is OPERABLE. ------------------------------------ Be in MODE 5.	Immediately 24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable. OR Required loop not in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND	
	B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required DHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	---------------------------NOTE-------------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power available to each required pump.	7 days

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

QID: 0570 **Rev:** 0 **Rev Date:** 5/3/05 **Source:** New **Originator:** J.Cork

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

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

System Number: 002 **System Title:** Reactor Coolant System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of forced circulation.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier: 2 **RO Imp:** 4.1 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 4.3 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** [ ] **SRO:** [91]

The plant was at 80% power when only one failure caused the following:

- Plant running back
- Tavg Control swaps to Loop "A"
- "A" Main Feedwater flow rapidly rising
- "B" Main Feedwater flow rapidly dropping
- Feedwater pumps' discharge crosstie valve shut

Which one of the following procedures would be used in response to the above conditions?

A. 1203.022, Reactor Coolant Pump Trip

B. 1203.020, Load Rejection

C. 1203.047, Loss of NNI Power

D. 1203.027, Loss of Steam Generator Feed

Answer:

A. 1203.022, Reactor Coolant Pump Trip

Notes:

"A" is correct, indications are indicative of FW re-ratioing in response to an RCP trip.
"B" is incorrect, although the plant is running back, FW flows should be matched.
"C" is incorrect, a loss of NNI power would cause both FW flows to be rapidly rising.
"D" is incorrect, these could be the indications of a "B" MFW pump control problem but "A" FW would be dropping due to mis-matched T-cold temps.

References:

1203.022, Chg. 009-05-0

History:

New for 2005 SRO exam.

ENTRY CONDITIONS

- **Annunciator Alarms**

  - RCP TRIP (K08-A6)
  - RCS FLOW LO (K09-D2)
  - UNIT MASTER IN TRACK (K07-A1)
  - LOSS OF RCP RUNBACK IN EFFECT (K07-B1)

  - FW RERATIO ON RC-FLOW ENABLED (K07-E1)
  - A OTSG BTU LIMIT (K07-E2)
  - B OTSG BTU LIMIT (K07-E3)

- **RCS flow dropping**

- **ICS running back**

INSTRUCTIONS

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

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

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

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

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

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

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

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

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

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

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

(continued)

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: 2.1.12 **CFR Reference:** 43.2 / 43.5 / 45.3

Tier: 2	**RO Imp:** 2.9	**RO Select:** No	**Difficulty:** 4
Group: 2	**SRO Imp:** 4.0	**SRO Select:** Yes	**Taxonomy:** Ap

Question: **RO:** **SRO:** 92

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

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.

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

-----NOTES-----

1. LCO 3.0.4 is not applicable
2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to prepare and submit a Special Report.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.15-1 for the channel.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.15.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	 18 months

Table 3.3.15-1
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Wide Range Neutron Flux	2	E
2. RCS Hot Leg Temperature	2	E
3. RCS Hot Leg Level	2	F
4. RCS Pressure (Wide Range)	2	E
5. Reactor Vessel Water Level	2	F
6. Reactor Building Water Level (Wide Range)	2	E
7. Reactor Building Pressure (Wide Range)	2	E
8. Penetration Flow Path Automatic Reactor Building Isolation Valve Position	2 per penetration flow path[(a)(b)]	E
9. Reactor Building Area Radiation (High Range)	2	F
10. Deleted		
11. Pressurizer Level	2	E
12. a. SG "A" Water Level – Low Range	2	E
b. SG "B" Water Level – Low Range	2	E
c. SG "A" Water Level – High Range	2	E
d. SG "B" Water Level – High Range	2	E
13. a. SG "A" Pressure	2	E
b. SG "B" Pressure	2	E
14. Condensate Storage Tank Level	2	E
15. Borated Water Storage Tank Level	2	E
16. Core Exit Temperature (CETs per quadrant)	2	E
17. a. Emergency Feedwater Flow to SG "A"	2	E
b. Emergency Feedwater Flow to SG "B"	2	E
18. High Pressure Injection Flow	2	E
19. Low Pressure Injection Flow	2	E
20. Reactor Building Spray Flow	2	E

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

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

QID: 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: 2.2.26 **CFR Reference:** 43.5 /45.13

Tier:	2	**RO Imp:**	2.5	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 93

Given:

- Plant is in a Refueling outage.
- Core re-load is in progress.
- Approximately 90% of the core is in the Reactor vessel.

The Main Fuel Handling Bridge has a once-burned fuel assembly and is in the process of indexing over the specified core location when NI-502 fails to 0.1 cps.

What action should be taken?

A. No action necessary because with NI-501 operating, Tech Spec NI requirements for operablility are met.

B. Establish communications between the Control Room and the refueling personnel in the Reactor Building and continue fuel load.

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" and "D" are incorrect, these simply list requirements for refueling.

References:

1502.004, Chg. 034-02-0
T.S. 3.9.2

History:

Direct from regular exam bank QID#3178
Selected for 2005 SRO exam.

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 entereed and the Required Action to place a portable survey instrument of appropriate range and sensitivity in service have been performed. (TRM 3.9.2).

5.3 One source range neutron flux monitor shall be operable in Mode 6. Two source range neutron flux monitors shall be operable during core alterations (TS 3.9.2).

5.4 One decay heat removal loop shall be operable and in operation in Mode 6 with water level ≥23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.4 for contingencies and exceptions.

5.5 Two decay heat removal loops shall be operable, and one loop shall be in operation in Mode 6 with the water level <23 feet above the top of the irradiated fuel seated in the reactor pressure vessel. Refer to TS 3.9.5 for contingencies and exceptions.

{4.3.1}

NOTE

The Refueling Boron Concentration specified by Reactivity Balance Calculation (1103.015) Worksheet 5 or 6 provides a shutdown margin of 5% as required by NRC commitment P 205. This concentration also satisfies TS 3.9.1.

5.6 Boron concentration of the RCS and Refueling canal shall be maintained within the limits specified in the COLR when the refueling canal is connected to the RCS (TS 3.9.1).

5.7 Direct communications between the control room and the refueling personnel in the reactor building shall exist during movement of irradiated fuel assemblies in the reactor building (TRM 3.9.4).

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2

a. One source range neutron flux monitor shall be OPERABLE, and

b. One additional source range neutron flux monitor shall be OPERABLE during CORE ALTERATIONS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable during CORE ALTERATIONS.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. No OPERABLE source range neutron flux monitor.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

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

QID: 0601 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Modified **Originator:** J.Cork

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

Section: 2 **Type:** Generic

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:** 43.5 / 45.12 / 45.13

Tier: 3 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 3

Group: **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 94

The plant computer is unavailable and excore nuclear instrumentation is being used to calculate quadrant tilt. The following conditions exist:

- Core EFPD is 200
- Reactor power is 80%
- Calculated quadrant tilt is 4.96% in quadrant WX.
- The COLR limit is 1.96%.

If these indications continue, which of the following actions should be taken in accordance with Techincal Specification 3.2.4?

A. Reduce applicable RPS trip setpoints 2%.

B. Reduce applicable RPS trip setpoints 4%.

C. Reduce applicable RPS trip setpoints 6%.

D. Reduce applicable RPS trip setpoints 8%.

Answer:

C. Reduce applicable RPS trip setpoints 6%

Notes:

Answer "C" is correct per T.S. 3.2.4 action A.1.2.2.
The other answers are simply mathematical possiblities.

References:

T.S. 3.2.4

History:

Modified regular exam bank question QID#ANO-OPS1-6624.
Selected for 2005 SRO exam.

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT (QPT)

LCO 3.2.4 QPT shall be maintained less than or equal to the steady state limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPT greater than the steady state limits specified in the COLR.	A.1.1 Perform SR 3.2.5.1. <u>OR</u>	Once per 2 hours
	A.1.2.1 Reduce THERMAL POWER ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. <u>AND</u>	2 hours <u>OR</u> 2 hours after last performance of SR 3.2.5.1
	A.1.2.2 Reduce nuclear overpower based on Reactor Coolant System flow and AXIAL POWER IMBALANCE trip setpoint ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. <u>AND</u>	10 hours <u>OR</u> 10 hours after last performance of SR 3.2.5.1

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1.2.3 Reduce the regulating group insertion limits given in the COLR ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours <u>OR</u> 10 hours after last performance of SR 3.2.5.1
	<u>AND</u>	
	A.1.2.4 Reduce the Operational Power Imbalance Setpoints given in the COLR ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours <u>OR</u> 10 hours after last performance of SR 3.2.5.1
	<u>AND</u>	
	A.2 Restore QPT to less than or equal to the steady state limit.	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 < 60% of the ALLOWABLE THERMAL POWER.	2 hours
	<u>AND</u>	
	B.2 Reduce nuclear overpower trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER.	10 hours
C. Required Action and associated Completion Time for Condition B not met.	C.1 Reduce THERMAL POWER to ≤ 20% RTP.	4 hours
D. QPT greater than the maximum limit specified in the COLR.	D.1 Reduce THERMAL POWER to ≤ 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify QPT is within limits as specified in the COLR.	7 days AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP

QUADRANT POWER TILT LIMITS AND SETPOINTS

(Figure is referred to by Technical Specification 3.2.4)

From 0 EFPD to EOC

Measurement System	Steady State Value (%)		Maximum Value (%)
	≤ 60% RTP	> 60% RTP	
Full In-core Detector System Setpoint	6.83	4.31	25.0
Minimum In-core Detector System Setpoint	2.78*	1.90*	25.0
Ex-core Power Range NI Channel Setpoint	4.05	1.96	25.0
Measurement System Independent Limit	7.50	4.92	25.0

* Assumes that no individual long emitter detector affecting the minimum in-core tilt calculation exceeds 73% sensitivity depletion. The setpoint must be reduced to 1.50% (power levels > 60% FP) and to 2.19% (power levels ≤ 60% FP) at the earliest time-in-life that this assumption is no longer valid.

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

QID: 0252 **Rev:** 1 **Rev Date:** 8-31-99 **Source:** Direct **Originator:** S.Pullin

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

Section: 2 **Type:** Generic

System Number: 2.1 **System Title:** Conduct Of Operations

Description: Knowledge of less than one hour technical specification action statements for systems.

K/A Number: 2.1.11 **CFR Reference:** 43.2 / 45.13

Tier: 3 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 2

Group: G **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 95

A normal plant startup is in progress. Critical data is being obtained.

Which Tech Spec condition would require a plant shutdown to Mode 3 within 30 minutes?

A. One pressurizer code safety valve is declared inoperable.

B. Reactor coolant temperature is below 525 °F.

C. Shutdown margin not within specification.

D. Pressurizer heater power from ES busses is limited to 120 KW.

Answer:

b. Reactor coolant temperature goes below 525 °F.

Notes:

"B" is correct, if temperature drops below 525, then the action is to go to Mode 3 within 30 minutes.
"A" is incorrect because Tech Specs requires the code safety to be restored to operable status within 15 minutes or be in Mode 3 within 6 hours.
"C" is incorrect because SDM is restored by initiating boration within 15 minutes.
"D" is incorrect because while out of spec level would lead to a trip (and thus be in Mode 3), heater power from ES busses per Tech Specs is allowed 72 hours to restore capacity.

References:

Tech Spec 3.4.2

History:

Developed for 1999 exam.
Selected for 2005 SRO exam with minor modifications.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 The RCS average temperature (T_{avg}) shall be ≥ 525°F.

APPLICABILITY: MODE 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} not within limit.	A.1 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} ≥ 525°F.	12 hours

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

QID: 0409 **Rev:** 1 **Rev Date:** 6/27/05 **Source:** Modified **Originator:** JCork

TUOI: ASLP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of process for making changes in the facility as described in the Safety Analysis Report.

K/A Number: 2.2.5 **CFR Reference:** 43.3 / 45.13

Tier:	3	**RO Imp:**	1.6	**RO Select:**	No	**Difficulty:**	2
Group:		**SRO Imp:**	2.7	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 96

Which of the following would require a 10 CFR 50.59 Review per ENS-LI-101, 10 CFR 50.59 Review Program?

A. A correction to the software for SPDS Alternate Shutdown display, A/S-G.

B. A change to the table of contents for 1203.049, Fires in Areas Affecting Safe Shutdown.

C. A change in the title of Shift Superintendent to Shift Manager.

D. A drawing change to correct a HPI injection valve number on a P&ID used in the SAR.

Answer:

A. A correction to the software for SPDS Alternate Shutdown display, A/S-G.

Notes:

Answer [a] is correct, any time safety related software is changed, a 50.59 review is required.
Answers [b], [c], [d] are excluded from 50.59 evaluations as listed in ENS-LI-101.

References:

ENS-LI-101, Rev. 7

History:

Created for 2001 SRO Exam.
Modified for use in 2005 SRO exam.

In general, the key aspect of 50.59 with regard to applicability to a proposed change is "as described in the FSAR", or a test or experiment "not described in the FSAR". Additional guidance on performing activities with this procedure is contained in Attachments 9.1 and 9.2.

[3] The 50.59 Review process consists of a Screening, an Environmental Screening, a Security Plan Screening, an ISFSI Screening, a 50.59 Evaluation Exemption, and/or a 50.59 Evaluation. The 50.59 Review is documented on the 50.59 Review Form.

[4] The 50.59 Review Form (NMM Form LI-101-01) is maintained in the IDEAS Reference Library (ECH site) under "Nuclear Management Manual/NMM Forms". Since it is controlled as a separate document, the 50.59 Review Form may be a different revision than this procedure.

NOTE

RBS: Changes to procedures may be processed in accordance with TRM Section 5.4.3.

[5] Changes, tests, and experiments as delineated below shall require a 50.59 Review unless excluded in accordance with Section 5.2[8].

(a) Changes to any existing SSC as described in the FSAR or new SSCs added by design changes regardless of the safety classification of the affected portion of the facility.

(b) Temporary alterations in accordance with associated procedures.

(c) Tests and experiments per Section 3.0[62].

(d) New procedures (both safety related and non-safety related and temporary procedures)

(e) Revisions (including temporary changes) to procedures/directives, special instructions and other procedures (e.g., startup test procedures)

(f) Safety-related plant software changes

(g) Changes to methods of evaluation

[6] Preparers and Reviewers shall be:

(a) Knowledgeable of the subject being evaluated and any pertinent site-specific processes and licensing basis documents.

(b) Qualified in accordance with Section 5.7.

[7] Preparers and Reviewers should ensure their qualifications are current prior to beginning any 50.59 Review activities.

[19] Editorial Change – Licensee-Controlled Licensing Basis Documents: A change that does not affect the result, requirements, or performance of the document including:

(a) Changes to text, tables, or figures (e.g., changes to table of contents, page numbers, grammar, misspelled words)

(b) Revisions to figures that:

(1) Split, redraw, or consolidate existing drawings, provided no information is added, deleted, or changed on the drawing.

(2) Change references to other drawings or revise reference arrows that have the wrong coordinates.

(3) Add or revise identification numbers to components already shown on the drawing (without changing the safety class or material of the component).

(4) Redraw to show subcomponents to equipment already shown on the drawing, provided that there is no change to equipment operation or function.

(5) Involve valve lineups other than those used for full power operation where the alternate lineups are acceptable based on current system design and within established procedural controls (e.g., multiple trains within a system; shutdown configurations).

(6) Correct errors on drawings for valve types that do not change system function or operating characteristics.

(7) Reflect minor changes to FSAR drawings involving piping interfaces/relocations that would not alter system function or operation.

(c) Changes to position titles, corporate names, manufacturer's component type, or other similar changes when no responsibilities or functions have changed.

(d) Corrections to:

(1) Mistakes made when incorporating requested changes (a 50.59 Review having been performed that continues to provide sufficient justification)

(2) Inconsistencies between sections, tables, or figures within the FSAR where there is documentation supporting accurate information in another section, table, or figure.

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

QID: 0119 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

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

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

Tier: 3 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 97

Given the following "A" Core Flood Tank (CFT) parameters with the plant is at 100% power.

- Level transmitter out of service
- CFT level is 12.5 feet
- Boric acid concentration is 2280 ppm
- CFT pressure is 580 psig

Which of the above parameters makes the "A" CFT inoperable per Tech Specs?

a. Level transmitter

b. CFT level

c. Boric acid concentration

d. CFT pressure

Answer:

b. CFT level

Notes:

T.S. 3.5.1 requires the following:
1/2 level channels and 1/2 pressure channels
Level of 12.6 to 13.4 ft
Boric acid concentration of at least 2270 ppm
Pressure of 575 to 625 psig
therefore "B" is correct.

References:

Technical Specifications, 3.5.1

History:

Used in 1998 SRO exam
Used in NRC developed SRO exam, no.23, 2/6/95
Selected for 2005 SRO exam

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Two CFTs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤ 800 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each CFT isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each CFT is ≥ 970 ft^3 and ≤ 1110 ft^3.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is ≥ 560 psig and ≤ 640 psig.	12 hours

APPLICABLE SAFETY ANALYSES (continued)

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to cover the core to the 3/4 point even assuming no liquid remains in the reactor vessel following a LOCA (Ref. 1). The downcomer then remains flooded until the HPI and LPI systems start to deliver flow for limiting large break LOCAs.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection and ensure the ability of the CFTs to fully discharge. The limiting safety analysis volume requirement is 1040 ± 70 ft^3. This volume corresponds to CFT levels of ≥ 11.95 ft and ≤ 14.00 ft. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The minimum nitrogen cover pressure requirement of 560 psig ensures that the contained gas volume will generate discharge flow rates during injection that satisfy the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The maximum nitrogen cover pressure limit of 640 psig will affect the amount and timing of CFT inventory discharged while the RCS depressurizes. Limiting the maximum pressure will therefore limit the CFT inventory lost through the break and assure that the CFT inventory injected into the RCS at the proper time is bounded by that predicted by the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum CFT level, the reactor will remain adequately shutdown in the cold condition following mixing of the CFT and Reactor Coolant System (RCS) water volumes. This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

In MODE 1, the CFTs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 2 and MODE 3 with RCS pressure > 800 psig, the CFTs satisfy Criterion 4 of 10 CFR 50.36.

5.0 LIMITS AND PRECAUTIONS

5.1 With the exception of ASME Section XI testing and when the CFT is depressurized, during plant cooldown the CFT discharge valves (below) shall be closed and the circuit breakers for the motor operators opened before depressurizing the RCS below 600 psig or ≤262°F (TS 3.4.11).

- Core Flood Tank T-2A Outlet Isol (CV-2415), breaker (B-5661)
- Core Flood Tank T-2B Outlet Isol (CV-2419), breaker (B-5545)

5.2 When the RCS is depressurized with DH in service and either CFT is pressurized, the potential exists for personnel harm, equipment damage, and loss of DH flow in the event the core flood tank isolation valves are opened.

5.3 CFT NDTT limits are as follows:

5.3.1 CFT metal temperature shall not be less than 65°F anytime the tank is pressurized above 140 psig.

5.3.2 N_2 temperature at the point of injection shall be ≥65°F anytime the tank is pressurized above 140 psig.

5.3.3 If N_2 temperature at the point of injection is ≥ 100°F below the tank metal temperature, CFT pressure shall be less than 25 psig.

NOTE

TS limits where the CFT will be declared inoperable for CFT level (volume) and pressure are listed below and contain the maximum allowance for instrument uncertainty.

5.4 Prior to reaching RCS pressure of 800 psig, each CFT shall be at least as follows (TS 3.5.1):

5.4.1 CFT level

- Admin limit - 12.7 to 13.3 feet
- Tech Spec limit - 12.6 to 13.4 feet

5.4.2 CFT Pressure

- Admin Limit - 580 to 620 PSIG
- Tech Spec Limit - 572 to 628 PSIG

5.4.3 CFT boron concentration ≥ 2270 ppmB (Tech Spec limit).

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

QID: 0120 **Rev:** 1 **Rev Date:** 6/27/05 **Source:** Modified **Originator:** JCork
TUOI: ASLP-SRO-RADP **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As
System Number: 2.3 **System Title:** Radiation Control
Description: Knowledge of 10CFR: 20 and related facility radiation control requirements.

K/A Number: 2.3.1 **CFR Reference:** 41.12/ 43.4 / 45.9 / 45.10

Tier:	3	**RO Imp:**	2.6	**RO Select:**	No	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.0	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question: **RO:** **SRO:** 98

Given:

- A Site Area Emergency has been declared on Unit 1.
- An Emergency Medical Team member must enter a 100 REM/hr area to rescue a critically injured employee.

Which of the following is the MAXIMUM time an individual team member can stay in this area?

A. 3 minutes

B. 6 minutes

C. 15 minutes

D. 30 minutes

Answer:

C. 15 minutes

Notes:

The limit for life saving is 25 rem TEDE. 100R/hr means a 15 minute stay time, therefore "C" is correct. "A" and "B" are the limits for all acitivities and protecting valuable property, respectively. "D" is just double the correct answer.

References:

1903.033, Chg. 018-01-0

History:

Modified for use in 1998 SRO exam
Modified question from NRC developed SRO exam 2/6/95, no. 94
Modified for use in 2005 SRO exam.

Dose limit* (rem TEDE)	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Life saving or protection of large populations	Lower dose not practicable
>25	Life saving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved (refer to Attachment 1 of this procedure for health risks).

* Workers performing services during emergencies should limit dose to the lens of the eye to three times the listed value and doses to any other organ (including skin and body extremities) to ten times 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 Immediate Actions

A. **IF** exposure to significant radioiodine concentrations is possible,
THEN refer to procedure 1903.035, "Administration of Potassium Iodide" for guidance.

B. Rescue/repair and damage control teams shall be briefed using Form 1903.033B, "OSC Team Briefing Form". This form serves as an emergency RWP and Work Order. Instructions on conducting re-entry team briefings are contained in Attachment 3.

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

QID: 0123 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of the process for perfoming a planned gaseous radioactive release.

K/A Number: 2.3.8 **CFR Reference:** 43.4 / 45.10

Tier:	3	**RO Imp:**	2.3	**RO Select:**	No	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.2	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** **SRO:** 99

Given:

- Plant shutdown and cooldown in progress
- RCS Tave 185°F
- RB Purge in progress to lower RB atmospheric activity
- RB Purge projected release duration is 7 hours
- RB Purge release commenced two (2) hours ago
- Nuclear Chemistry stated gaseous releases projected to exceed quarterly objectives by 30%
- Release report preliminary release rate 4.1 E4 cfm
- Design flow rate 3.6 E4 cfm to 4.1 E4 cfm

Which of the following would violate the RB Purge permit and require RB Purge termination?

A. Actual (stable) RB purge flow rate of 3.9 E4 cfm.

B. RB Atmosphere Gaseous Detector slowly trending upward.

C. SPING 1 indicates stack activity approaching NUE criteria over next 5 hours.

D. Loss of Decay Heat Removal results in RCS temperature at 220°F.

Answer:

D. Loss of Decay Heat Removal results in RCS temperature at 220°F.

Notes:

"D" is correct, RB Purge may only be performed at Cold Shutdown, RCS temp >200°F requires purge termination.
"A" is incorrect, this only requires notifying Nuclear Chemistry to adjust release data.
"B" is incorrect, only large changes require termination.
"C" is incorrect, SPING 1 trending to NUE criteria over one hour is the termination criteria.

References:

1104.033, Chg. 060-10-0

History:

Developed for 1998 SRO exam
Selected for 2005 SRO exam.

4.1.30 CR-C-98-080-02 Notify Chemistry upon initiation and completion of Reactor Building Purge

4.1.31 Offsite Dose Calculation Manual (ODCM)

4.1.32 CR-1-00-031-10 Failure to implement contingent sampling requirements when required by Tech Spec.

4.1.33 CR-1-2001-0199, SPING 1 Error Versus Local Flow Readings

4.1.34 ER-ANO-2003-0221 Manual Isolation of PASS Boundary Valves

4.2 REFERENCES USED IN CONJUNCTION WITH THIS PROCEDURE

4.2.1 P&ID M-261, Sheets 1 & 2

4.2.2 P&ID M-210

4.2.3 Chilled Water System (1104.026)

4.2.4 Plant Heating System (1104.039)

4.2.5 Containment H_2 Control (1104.031)

4.2.6 Radiation Monitoring System Check and Test (1305.001)

4.2.7 Eberline Radiation Monitoring System (1604.051)

4.3 NRC COMMITMENTS

None

5.0 LIMITS AND PRECAUTIONS

5.1 Maximum allowable reactor building pressure with RCS >200°F is 3 psig.

5.2 Minimum allowable reactor building pressure with RCS >200°F is -1 psig.

5.3 A reactor building purge shall not be initiated unless a properly documented Reactor Building Purge Gaseous Release Permit, Attachment B to this procedure, has been issued.

5.4 Do not allow temperature in RB Purge Supply Fan (VSF-2) heating coil to drop below 33°F.

5.5 WHEN reactor building operability is required, (RCS Tave > 200°F), THEN RB purge system shall be isolated with the following valves closed and key switch keys removed.

A. RB Purge Inlet (CV-7404)
B. RB Purge Inlet (CV-7402)
C. RB Purge Outlet (CV-7403)
D. RB Purge Outlet (CV-7401)

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

QID: 0602 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Modified **Originator:** J.Cork

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

Section: 2 **Type:** Generic KA's

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

Description: Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

K/A Number: 2.4.22 **CFR Reference:** 43.5 / 45.12

Tier:	3	**RO Imp:**	3.0	**RO Select:**	No	**Difficulty:**	4
Group:		**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 100

Given:

- A shutdown was in progress for a SGTR.
- A loss of offsite power caused a reactor trip.
- Two control rods are stuck out.
- #2 EDG did not start.
- Neither channel of EFIC actuated.
- Pressurizer level is offscale low.

Which of the following EOP Repetitive Tasks should be the number one priority?

A. RT-2, Initiate HPI

B. RT-5, Verify proper EFW actuation and control

C. RT-12, Emergency Boration

D. RT-14, Control RCS pressure low within limits of Figure 3

Answer:

C. RT-12, Emergency Boration

Notes:

Answer "C" is correct, step 25 establishes Reactivity Control as the highest priority by virtue of it being the first check made following a trip.
The other answers are correct RTs to perform for the conditions given but are not performed until later.

References:

1202.006, Chg. 007-04-0

History:

Modified from QID#498 (used in 2004 SRO exam) for 2005 SRO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
1. IF Reactor or Turbine has tripped OR trips during plant runback, THEN GO TO step 25.	
2. Advise Shift Manager to perform BOTH of the following: A. Notify Nuclear Chemistry to begin off-site dose projections. AND B. Implement Emergency Action Level Classification (1903.010).	
3. Open BWST Outlet to OP HPI pump (CV-1407 or 1408).	
4. Verify Pressurizer Level Control valve maintains PZR level >200" (CV-1235).	4. Perform the following, while continuing with this procedure: A. Place RCP Seal INJ Block (CV-1206) in OVRD. B. Operate HPI Block associated with OP HPI pump to restore PZR level ≥200" (CV-1220 or 1285). C. Isolate Letdown by closing: Letdown Coolers Outlet (CV-1221) **OR** Letdown Cooler Outlets (CV-1214 and 1216). D. IF CV-1220 or 1285 associated with OP HPI pump cannot maintain PZR level ≥200" **THEN** initiate HPI (RT-2). E. IF PZR level drops below 100" **AND** Reactor is critical, **THEN** trip the Reactor **AND** **GO TO step 25.**

INSTRUCTIONS	CONTINGENCY ACTIONS
25. <u>IF</u> Reactor or Turbine trips, <u>THEN</u> perform the following:	
A. Manually Trip Rx.	
1) Verify all rods inserted **AND** Reactor power dropping.	1) Perform the following: a) **IF** Rx fails to trip, **THEN** depress CRD Power Supply Breaker Trip PBs on C03 (A-501 and B-631). b) **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. c) **IF** more than one rod fails to fully insert **OR** Rx power is **not** dropping, **THEN** perform Emergency Boration (RT 12).
B. Manually trip Turbine.	B. Perform the following:
1) Verify Turbine tripped.	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** Header press is <900 psig, **THEN** perform the following: a) Actuate MSLI and EFW **AND** verify proper actuation and control (RT 6). b) **GO TO 1202.003, "OVERCOOLING"** procedure.

(25. CONTINUED ON NEXT PAGE)

SECOND (POST-COMPROMISE) VERSION
OF THE
DRAFT RO WRITTEN EXAM

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier1 /Group1(RO / SRO)		Form ES-401-2
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
00007 (BW/E02 & E10; CE/E02) Reactor Trip – Stabilization – Recovery / 1		**X**					EK3.01 – Knowledge of the reasons for the following as they apply to Reactor Trip: Actions contained in EOP for reactor trip. **Changed to EK2.02 – Breakers, relays, and disconnects.**	4.0 **2.6**	1
00008 Pressurizer Vapor Space Accident / 3			**X**				AK2.01 – Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves. **Changed to AK3.03 - Actions contained in EOP for PZR vapor space accident/ LOCA.**	2.7* **4.1**	2
00009 Small Break LOCA / 3	**X**						EA2.01 – Ability to determine or interpret the following as they apply to a Small Break LOCA: Actions to be taken, based on RCS temperature and pressure, saturated and superheated. **Changed to EK1.01 – Natural circulation and cooling, including reflux boiling.**	4.2 **4.2**	3
000011 Large Break LOCA / 3		X					EK2.02 – Knowledge of the interrelations between the Large Break LOCA and the following: Pumps.	2.6*	4
000015/17 RCP Malfunctions / 4		X					AK2.07 – Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP seals.	2.9	5
000022 Loss of Rx Coolant Makeup / 2			X				AK3.04 – Knowledge of the reasons for the following responses as they apply to Loss of Rx Coolant Makeup: isolating letdown.	3.2	6
000025 Loss of RHR System / 4						X	2.4.10 - Knowledge of annunciator response procedures.	3.0	7
000026 Loss of Component Cooling Water / 8					X		AA2.03 – Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition. **Changed to AA2.06 – The length of time after the loss of CCW flow to a component before that component may be damaged.**	2.6 **2.8**	8
000027 Pressurizer Pressure Control System Malfunction / 3		X					AK2.02 – Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Sensors and detectors. **Justification for K/A <2.5**: Knowledge of interrelationship between sensors/detectors and control systems is important to a Reactor Operator's duties of monitoring the control panels.	2.4	9
000029 ATWS / 1							Not selected.		
000038 Steam Gen. Tube Rupture / 3				X			EA1.32 – Ability to operate and monitor the following as they apply to SGTR: Isolation of a ruptured S/G.	4.6	10
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4	X						AK1.01 – Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: Consequences of PTS.	4.1	11
000054 (CE/E06) Loss of Main Feedwater / 4							Not selected.		
000055 Station Blackout / 6						X	2.4.1 Knowledge of EOP entry conditions and immediate action steps.	4.3	12
000056 Loss of Off-site Power / 6	X						AK1.01 – Knowledge of the operational implications of the following concepts as they apply to Loss of Off-site Power: Principle of cooling by natural convection.	3.7	13

000057 Loss of Vital AC Inst. Bus / 6				X			AA1.01 – Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual inverter swapping.	3.7	14
000058 Loss of DC Power / 6			X				AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of DC power.	4.0	15
000062 Loss of Nuclear Svc Water / 4						X	2.4.9 - Knowledge of low power/shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.	3.3	16
000065 Loss of Instrument Air / 8					X		AA2.05 - Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing.	3.4*	17
W/E04 LOCA Outside Containment / 3							Not applicable to this unit.		
W/E11 Loss of Emergency Coolant Recirc. / 4							Not applicable to this unit.		
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4				X			EA1.2 - Ability to operate and/or monitor the following as they apply to the (Inadequate Heat Transfer): Operating behavior characteristics of the facility.	3.4	18
K/A Category Totals:	3	4	3	3	2	3	Group Point Total:		18

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

QID: 0608 **Rev:** 0 **Rev Date:** 8/5/2005 **Source:** New **Originator:** S.Pullin

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

Section: 4.1 **Type:** Generic EPEs

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

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

K/A Number: EK2.02 **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:** No **Taxonomy:** C

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

With the plant at 100% power, what is the relationship between a Reactor Trip signal and the closing of the SU #1 feeder breakers for the A1/A2 and H1/H2 buses?

A. Reactor trip causes a Generator Lockout which closes SU #1 feeder breakers as long as the SU #2 feeder breakers are open.

B. Reactor trip confirm signal opens the Unit Aux. feeder breakers, the SU #1 feeder breakers close because the Unit Aux. breakers are open.

C. Reactor trip causes a Unit Aux Lockout which opens the Unit Aux. feeder breakers, the SU #1feeder breakers close due to undervoltage on the buses.

D. Reactor trip confirm signal closes the SU #1 feeder breakers, with both feeder breakers closed simultaneously the Unit Aux. feeder breakers will open after a short time delay.

Answer:

A. Reactor trip causes a Generator Lockout which closes SU #1 feeder breakers as long as the SU #2 feeder breakers are open.

Notes:

Answer "A" is correct, since the conditions given are part of the logic that closes the SU #1 breakers. Answer "B", "C", and "D" are incorrect since a Generator Lockout or SU #2 Lockout is needed to satisfy the auto close logic for the SU #1 feeder breakers.

References:

STM 1-32, Rev. 27

History:

New for 2005 RO exam, replacement question.

3.3 Breaker Logic

3.3.1 Unit Auxiliary Feeder (A-112, A-212, H-14, H-24)

(Refer to Figure 32.62)

The unit auxiliary 6900V and 4160V breakers supply A1, 2/H1, 2 during power operations from the output of the main generator. There is no auto close function for these breakers. The breaker may be closed from the control room as long as:

* "Remote" selected
* SYNC select switch "On"
* No main generator L.O. relay trip
* No bus L.O. relay trip

The Unit Auxiliary feeder breakers to A and H buses will trip when:

* *6900/4160V bus locks out
* *Main generator lock out
* 4160V A1 bus only ES ch. 1
* 4160V A2 bus only ES ch. 2
* 6900/4160V undervoltage

If selected to remote, the breaker will trip on manual transfer to SU-1 or SU-2 when:

- SU-1/SU-2 (C10) hand switch in normal after position
- SU-1/SU-2 synch selector switch on
- SU-1/SU-2 feeder breaker to bus closed

*Circuit contains a test switch to defeat protective function.

3.3.2 SU1 Feeder Breaker (A-113, A-213, H-15, H-25)

(Refer to Figure 32.63)

SU1 transformer is the normal supply to A and H buses when the main generator is not operating. A and H buses may be supplied from SU-1 (either automatically or manually) provided the transformer is available. The SU-1 transformer is available if:

* Breaker control selected to remote
* Normal voltage on SU-1 transformer secondary.
* NO SU1 lock out relay tripped.
* The breaker control switch on C-10 is not in pull-to-lock.

With the transformer available and the bus lockout relay not tripped, the A and H buses may be manually supplied from SU-1 transformer by turning the sync selector switch on and placing the C-10 control switch to close. The A and H buses will automatically transfer to SU-1 (SU-1 bus feeder breaker closes) if the main generator lockout relay trips and SU-2 feeder breaker is open or SU-2 lock out relay trips and the Unit Aux feeder breaker is open.

The following conditions must be met for this transfer to occur:

* SU-1 must be available as described above
* No associated bus lock outs
* SU-1 selected or SU-2 not available
* Synch check relay determining SU-1 and associated bus are in phase or
* Associated bus undervoltage relay tripped.

The SU-1 transformer feeder breakers to the A and H buses will trip when:

* *SU-1 transformer secondary output undervoltage
* *Associated bus lock out relay tripped
* *SU-1 transformer lock out relay tripped

If selected to remote, the breaker will trip on manual transfer to Unit Aux or SU-2 when:

- Unit Aux/SU-2 (C10) hand switch in normal after position
- Unit Aux/SU-2 synch selector switch on
- Unit Aux/SU-2 feeder breaker to bus closed

*Circuits contain a test switch to defeat protective function.

3.3.3 SU 2 Feeder Breaker (A-111, A-211, H-13, H-23)

(Refer to Figure 32.64)

SU-2 transformer is the back-up supply to A and H buses when the main generator is not producing power and SU-1 transformer is not available.

The relaying for breaker manual/automatic closure and breaker trip is essentially the same as Startup Transformer No. 1. However, it should be noted that SU-2 can only be selected to feed the Unit 1 A-1 and Unit 2 2A-1 buses at the same time.

3.3.4 ES Bus Feeder Breaker (A-309, A-409)

(Refer to Figure 32.66)

The 4160V ES bus feeder breaker supplies power to A3/A4 from A1/A2 respectively at all times when off site power is available. There is no auto close or ES functions associated with these breakers. The breakers can be manually closed from C-10 if:

* Breaker control selected to remote
* No A1/2 bus locks out
* No A3/4 bus locks out
* Synch selector switch On
* A309/409 synch check relay in synch or A3/4 dead or A1/2 dead

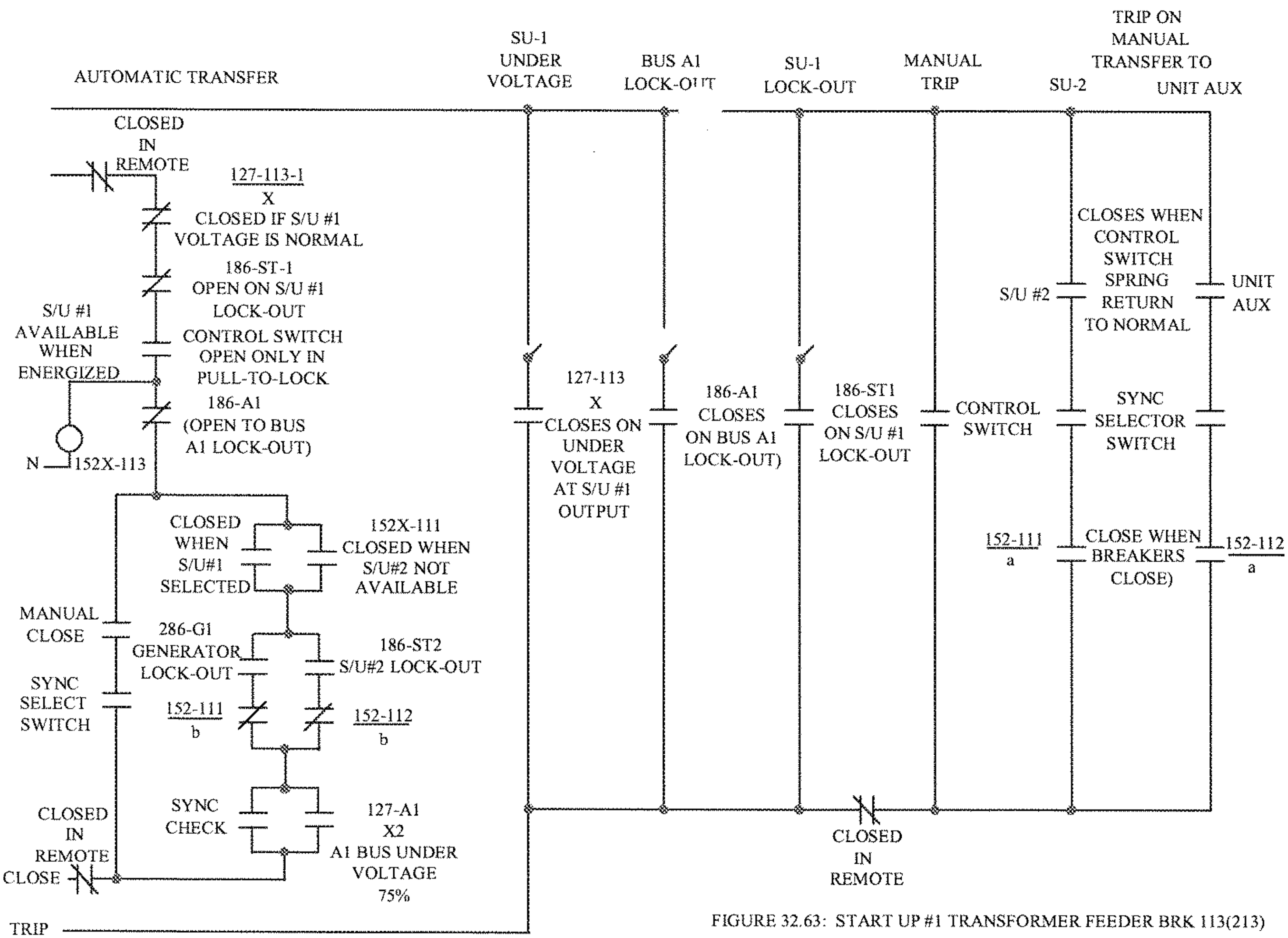


FIGURE 32.63: START UP #1 TRANSFORMER FEEDER BRK 113(213)

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

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

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

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

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

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

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

Tier:	1	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	An

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

Given:

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

1202.012, Chg. 004-03-0

History:

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

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

QID: 0324 **Rev:** 0 **Rev Date:** 9-6-99 **Source:** New **Originator:** J. Cork

TUOI: ANO-1-LP-RO-EOP02 **Objective:** 4 **Point Value:** 1

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

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

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

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

Tier:	1	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	3.2
Group:	2	**SRO Imp:**	4.6	**SRO Select:**	Yes	**Taxonomy:**	An

Question:

Given:

PARENT Question

Reactor tripped on low pressure.
RCS Tave 545 °F and stable
Pressurizer level off-scale high
RCS pressure 1850 psig and rising rapidly
RB sump level 55% and rising

During this transient, which of the following methods will be used to limit the RCS pressure rise, in accordance with RT-14?

a. Cycle ERV as required

PARENT Question

b. Secure steaming OTSGs

c. Raise makeup flow

d. Lower letdown flow

Answer:

a. Cycle ERV as required

Notes:

"a" is the correct answer since the conditions given are representative of a steam space leak and the RCS is in a "solid" condtion.
"b", "c", and "d" are incorrect as these would increase pressure during solid plant conditions.

References:

1202.012 Rev 4

History:

Developed for 1999 exam.

NOTE

- PTS limits apply if <u>any</u> 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 **<u>OR</u>** SGTR is in progress, PZR cooldown rate limits **<u>do</u>** **<u>not</u>** apply.

14. Control RCS press within limits of Figure 3.

A. <u>IF</u> PTS limits apply or RCS leak exists,
<u>THEN</u> maintain RCS press <u>low</u> within limits of Figure 3.

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

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

1) Throttle makeup flow.

2) <u>IF</u> SCM is adequate, **<u>THEN</u>** throttle HPI flow.

3) Raise Letdown flow.

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

4) Verify ERV Isolation open (CV-1000) **<u>AND</u>** cycle ERV (PSV-1000).

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

1) Throttle makeup flow.

2) <u>IF</u> HPI is in service **<u>AND</u>** SCM is adequate **<u>AND</u>** any of the following conditions is met,
<u>THEN</u> throttle HPI flow:

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

a) <u>IF</u> ESAS has actuated **<u>AND</u>** HPI must be throttled,
<u>THEN</u> override **<u>AND</u>** throttle HPI.

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

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

5) Place Pressurizer Heaters in OFF.

(14. CONTINUED ON NEXT PAGE)

14. (Continued).

6) Raise Letdown flow.

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

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

E. **IF** RCS press is low, **THEN** raise press using one or more of the following:

1) Raise makeup flow.

2) Raise or initiate HPI flow (RT 2).

3) **IF** RCP is running, **THEN** verify Pressurizer Spray (CV-1008) closed.

4) Reduce Letdown flow.

5) Place Pressurizer Heaters in MANUAL.

CAUTION

If HPI cooling is in progress, ERV Isolation (CV-1000) must be left open until HPI cooling is no longer required.

6) Verify ERV (PSV-1000) or ERV Isolation closed.

CAUTION

With RCS solid, 1°F temp change can cause 100 psig press change.

F. **IF** PZR is solid, **THEN** RCS press may also be controlled by varying RCS temperature.

- Raise RCS temp to raise RCS press
- Lower RCS temp to lower RCS press

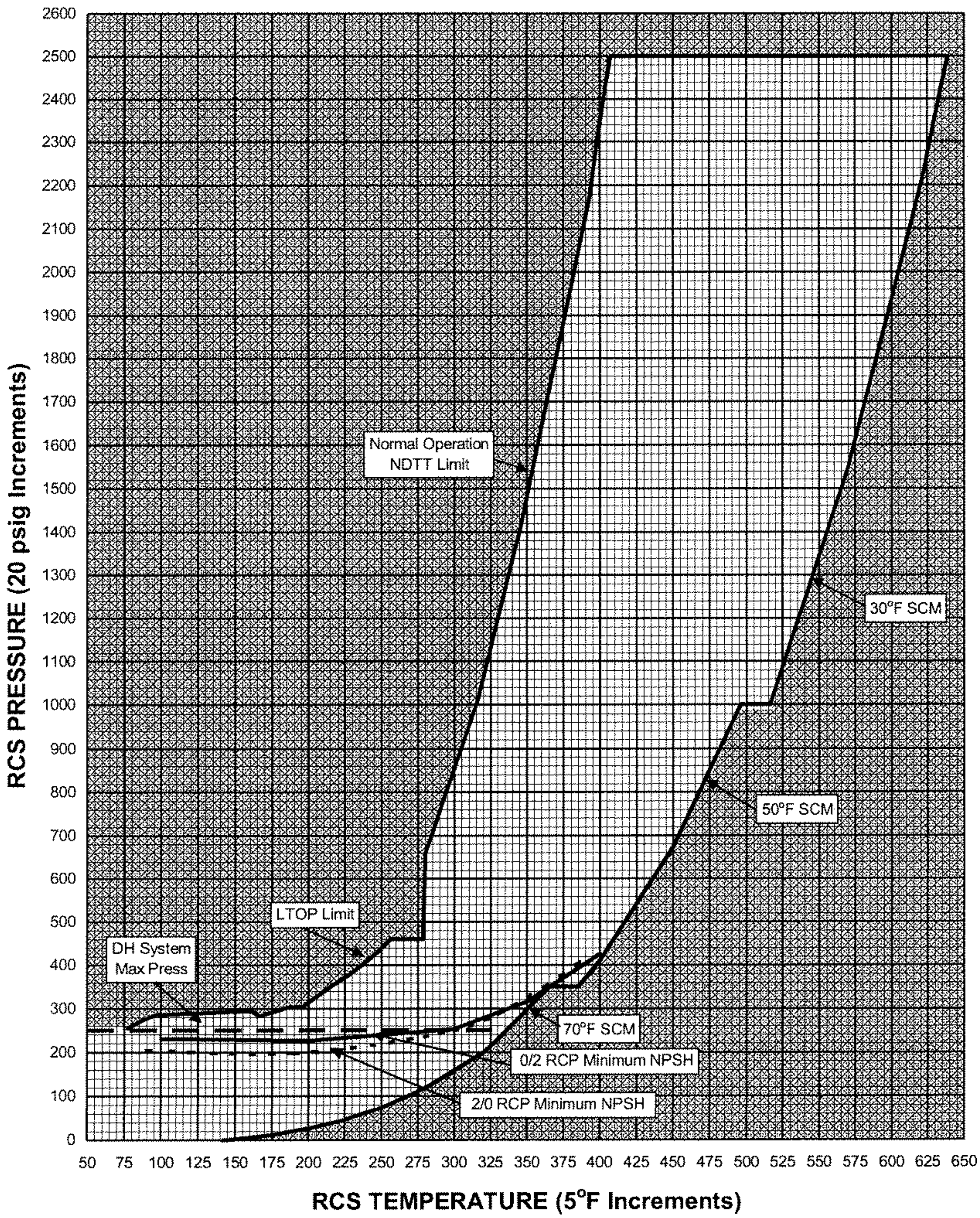
NOTE

Adjusting Pressurizer Level Control setpoint and HPI as necessary to maintain normal makeup flow on-scale will allow CV-1235 to automatically compensate for small changes in RCS leak rate and cooldown rate.

G. **IF** normal makeup is in service
AND
HPI is in service,
THEN adjust Pressurizer Level Control setpoint and HPI as necessary to maintain normal makeup flow on-scale.

END

FIGURE 3
RCS Pressure vs Temperature Limits



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

QID: 0368 **Rev:** 1 **Rev Date:** 8/8/05 **Source:** Modified **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:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	No	**Taxonomy:**	An

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

A reactor trip has occurred from 100% power.

The following conditions have existed for three minutes:

- 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.

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

QID: 0368 **Rev:** 0 **Rev Date:** 5/6/2002 **Source:** Direct **Originator:** J.Cork

TUOI: ANO-1-LP-RO-EOP02 **Objective:** 8 **Point Value:** 1

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

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

Description: Knowledge of the reasons for the following responses as they apply to the (Inadequate Subcooling Margin): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

K/A Number: EK3.1 **CFR Reference:** 41.5 / 41.10, 45.6, 45.13/ 43.5

Tier: 1 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3

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

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

A reactor trip has occurred from 100% power.

PARENT Question

One minute later the following conditions exist:

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

Which of the following operator actions will be performed?

a. Trip one (1) RCP in each loop.

b. Verify EFW flow to each Steam Generator is ~430 gpm.

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

d. Initiate 1202.001, Reactor Trip, and go to Overheating EOP.

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 one SG is available.
Answer [d] is incorrect, this would not be done since the entry condition for Overheating have not been met.

References:

1202.012, Repetitive Tasks, change 004-02-0, RT-5, page 9, step C

History:

Direct from regular exambank QID 3030.
Selected for use in 2002 SRO exam.

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)

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:**	No	**Taxonomy:**	K

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

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. 004-03-0

History:

Developed for use in A. Morris 98 RO Re-exam.
Selected for use in 2005 RO exam, replacement question.

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)

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

QID: 0609 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** New **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 seals.

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

Tier:	1	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	K

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

Which of the following is a condition that requires stopping of a Reactor Coolant Pump?

A. Seal cavity pressures oscillating >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. 016-02-0

History:

New for 2005 RO exam, replacement question.

SECTION 1
SEAL DEGRADATION

CAUTION

Operation with only 1 RCP in each loop is limited to 18 hours, after which the plant shall be placed in Mode 3 within an additional 6 hours per TS 3.4.4.

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 .08 gpm. Third stage leakage in excess of design will affect upper seal cavity pressure and seal bleed off flow.
- RCP Parameters, Attachment A of this procedure, contains the essential criteria to aid monitoring.

3. 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).

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

QID: 0610 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** New **Originator:** Cork/Pullin

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

Given:

- Plant at 100% power.
- Makeup pump P-36B has tripped.

Why does the AOP isolate Letdown for these conditions?

A. Prevent overheating of the Letdown Demeralizers.

B. Prevent overfill of the Makeup Tank.

C. Reduce heat load on Nuclear ICW system.

D. Maintain RCS inventory.

Answer:

D. Maintain RCS inventory.

Notes:

Answer "D" is correct, RCS inventory is maintained by isolating letdown.
Answer "A" is incorrect, this will be done automatically on high Letdown temperature.
Answer "B" is incorrect, the primary reason is to maintain RCS inventory.
Answer "C" is incorrect, this will be done when letdown is isolated but is not the reason for the action.

References:

1203.026, Chg. 009-04-0

History:

New for 2005 RO exam, replacement question.

SECTION 1 -- LOSS OF HPI PUMP

1.0 SYMPTOMS

1.1 Annunciator alarms:

- HPI PUMP TRIP (K10-A6)
- RCP SEAL INJ FLOW LO (K08-A7)
- MU TANK LEVEL HI/LO (K10-B7)
- MU TANK PRESS HI/LO (K10-B8)

1.2 Loss of or erratic makeup flow and seal injection flow.

1.3 Loss of or erratic makeup (HPI) pump discharge header pressure.

2.0 IMMEDIATE ACTION

2.1 None.

3.0 FOLLOW-UP ACTIONS

NOTE

Indications of loss of HPI suction are:

- Erratic flow, and
- Erratic discharge pressure, and
- Control valves stable

3.1 IF HPI pump has lost suction,
THEN stop the HPI pump.

3.2 Isolate letdown by performing either of the following:

- Close Letdown Coolers Outlet (CV-1221),
- Close Letdown Cooler Outlets (RCS) (CV-1214 and 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.

3.3 Verify RC pump seals are being cooled by ICW.

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

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

QID: 0611 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** New **Originator:** Cork/Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 025 **System Title:** Loss of Residual Heat Removal System (RHRS)

Description: Knowledge of annunciator response procedures.

K/A Number: 2.4.10 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

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

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

Given:

- Plant is in Mode 5.
- "A" Decay Heat Removal system is in service.
- RB sump level 40% and rising.

Annunciators K09-C7 "TRAIN A RCS LEVEL LO" and K09-D7 "TRAIN B RCS LEVEL LO" alarm.

Which of the following actions should be performed?

A. Start P-34B Decay Heat pump and secure P-34A.

B. Stop P-34A Decay Heat pump and close CV-1404.

C. Stop P-34A Decay Heat pump and close CV-1434.

D. Fill RCS by starting P-34B using LPI flowpath.

Answer:

B. Stop P-34A Decay Heat pump and close CV-1404.

Notes:

Answer "B" is the correct response, an RCS leak is indicated, the pump should be secured and the suction from the RCS isolated.
Answer "A" is incorrect, although this is the response to other problems with P-34A, this will not mitigate the low level.
Answer "C" is incorrect, closing CV-1434 does not isolate the RCS drop leg only the suction to P-34A.
Answer "D" is incorrect, although this would raise level, it will not mitigate the level loss.

References:

1203.028, Chg. 016-03-0

History:

New for 2005 RO exam, replacement question.

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

QID: 0007 **Rev:** 1 **Rev Date:** 8/9/05 **Source:** Direct **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

System Number: 026 **System Title:** Loss of Component Cooling Water (ICW at ANO)

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

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

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

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

The plant is operating at 100% power and "A" RCP experiences a simultaneous loss of seal cooling and seal injection flow.
This occurred at 1715.
Attempts to restore seal cooling and/or seal injection flow to the pump are unsuccessful.

What is the LATEST time at which action can be taken and still avoid damage to the pump seal due to overheating?

A. 1719

b. 1718

c. 1717

d. 1716

Answer:

c. 1717

Notes:

Answer "C" is correct. With the plant operating at 100% power, the appropriate action in accordance with the follow-up actions of 1203.031, Reactor Coolant Pump and Motor Emergencies, would be to trip the reactor and the A RCP within 2 minutes if attempts to restore seal cooling and/or seal injection flow are unsuccessful. Continued RCP operation without seal cooling and seal injection flow would result in damage due to overheating.
The other answers represent either a time which is less than two minutes or longer than two minutes.

References:

1203.031, Chg. 016-02-0

History:

Developed for 1998 RO Exam.
Selected for 2005 RO exam, replacement question.

SECTION 3
SIMULTANEOUS LOSS OF SEAL INJECTION AND SEAL COOLING FLOW

INSTRUCTIONS

CAUTION

Continued RCP operation without seal injection or seal cooling will cause seal damage due to overheating

1. **Attempt to restore seal injection OR seal cooling within 2 minutes, while continuing with this procedure.**

2. **IF at least 1 RCP in each loop is unaffected, THEN begin reducing reactor power at maximum rate using Rapid Plant Shutdown (1203.045) to within limits of unaffected RCP combination.**

3. **IF either seal injection OR seal cooling cannot be restored within 2 minutes, THEN perform the following:**

NOTE

- Flux/ΔFlux/Flow reactor trip setpoints are shown on COLR Figure 10.
- High power/pumps reactor trip setpoints are:
  - One pump per loop ≥55%
  - Zero pumps in one loop ≥0%
- Tripping 1 RCP with reactor power >92% may result in reactor trip on high power/imbalance/flow.

A. IF tripping the affected RCP(s) will result in reactor trip on high power/pumps, THEN perform the following:

1) Trip reactor.

2) Trip affected RCP(s).

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

B. IF tripping the affected RCP(s) will NOT cause a reactor trip on high power/pumps, THEN perform the following:

CAUTION

Operation with only 1 RCP in each loop is limited to 18 hours, after which the plant shall be placed in Mode 3 within an additional 6 hours per TS 3.4.4.

1) Trip affected RCP(s).

2) Verify proper ICS response.

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: Sensors and detectors.

K/A Number: AK2.02 **CFR Reference:** 41.7 / 45.7

Tier:	1	**RO Imp:**	2.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.6	**SRO Select:**	No	**Taxonomy:**	C

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

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. 009-03-0
STM 1-69, Rev. 8

History:

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

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.

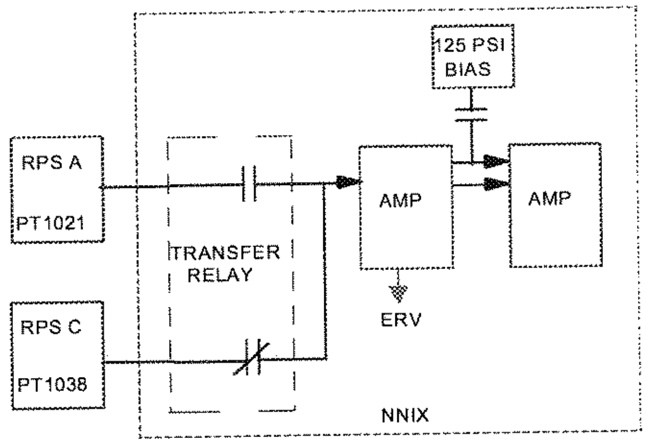
Pressure transmitters PT-1020, PT-1022, and PT-1040 provide input to A, B, and C ESAS analog channels, respectively. The pressure transmitters that supply ESAS are Rosemount differential capacitance detectors. ESAS analog channel A supplies indication on C-166 (Dasey Panel). The range of the indication 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. With the handswitch in the AUTO position, the signal monitor opens and closes the ERV. When either handswitch is placed in the OPEN position, the ERV solenoid is energized and ERV is opened.

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

QID: 0612 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** New **Originator:** J.Cork

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to operate and monitor the following as they apply to a SGTR: Isolation of a ruptured S/G.

K/A Number: EA1.32 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 4.6 **RO Select:** Yes **Difficulty:** 3

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

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

Given:
- A tube rupture has occurred on the "A" SG.
- Reactor has been tripped and cooldown started.

In accordance with the Tube Rupture EOP, which of the following parameters is the trigger point for the need to isolate the bad SG?

A. T-hot indicates 485°F

B. CETs indicate 495°F

C. T-hot indicates 505°F

D. CETs indicate 510°F

Answer:

A. T-hot indicates 485°

Notes:

Answer "A" is the correct parameter with temperature less than 490°F.
Answer "B" has the CETs vs. T-hot and the temp is too high.
Answer "C" is Tavg, T-hot could be greater than 490.
Anser "D" is T-cold, T-hot could be greater than 490.

References:

1202.006, Chg. 007-04-0

History:

New for 2005 RO exam, replacement question.

INSTRUCTIONS

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

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

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

(53. CONTINUED ON NEXT PAGE)

CONTINGENCY ACTIONS

A. Perform the following:

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

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

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

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

(3) **GO TO step 53.A.2).**

b) **IF** ERV **cannot** be opened,
THEN adjust HPI as necessary to maintain RCS press/temp within limits of Figure 3.

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

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

INSTRUCTIONS

53. (Continued).

CONTINGENCY ACTIONS

2) Verify bad SG Main Feedwater Isolation valve closed:

SG A	SG B
CV-2680	CV-2630

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

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

4) **IF** RCS press is >950 psig, **THEN** reduce RCS press to ≤950 psig, while maintaining adequate SCM by any or all of the following:

   a) Maintain emergency cooldown rate of ≤240°F/hr to 500°F.

   b) Raise AUX Pressurizer Spray flow.

   c) Maximize Letdown flow.

   d) Throttle HPI.

   e) Open High Point Vents:

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

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

(53. CONTINUED ON NEXT PAGE)

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

QID: 0551 **Rev:** 0 **Rev Date:** 3-30-05 **Source:** New **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:**	No	**Taxonomy:**	C

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

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 cooldown rate 110°F/hr with Tcold 360°F

C. RCS cooldown rate 75°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. 004-03-0

History:

New for 2005 RO exam.

NOTE

- PTS limits apply if <u>any</u> 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 **<u>OR</u>** SGTR is in progress, PZR cooldown rate limits **<u>do</u>** **<u>not</u>** apply.

14. Control RCS press within limits of Figure 3.

A. <u>IF</u> PTS limits apply or RCS leak exists,
<u>THEN</u> maintain RCS press <u>low</u> within limits of Figure 3.

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

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

1) Throttle makeup flow.

2) <u>IF</u> SCM is adequate, **<u>THEN</u>** throttle HPI flow.

3) Raise Letdown flow.

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

4) Verify ERV Isolation open (CV-1000) **<u>AND</u>** cycle ERV (PSV-1000).

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

1) Throttle makeup flow.

2) <u>IF</u> HPI is in service **<u>AND</u>** SCM is adequate **<u>AND</u>** any of the following conditions is met,
<u>THEN</u> throttle HPI flow:

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

a) <u>IF</u> ESAS has actuated **<u>AND</u>** HPI must be throttled,
<u>THEN</u> override **<u>AND</u>** throttle HPI.

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

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

5) Place Pressurizer Heaters in OFF.

(14. CONTINUED ON NEXT PAGE)

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

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

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

Section: 4.2 **Type:** Generic EPEs

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

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

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

Tier:	1	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	No	**Taxonomy:**	K

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

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

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

B. All 4160V busses de-energized

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

D. All 6900V busses de-energized

Answer:

B. All 4160V busses de-energized

Notes:

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

References:

1202.008, chg. 007-01-0

History:

New for 2005 RO exam

ENTRY CONDITIONS

NOTE

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

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

- All 4160V buses de-energized

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

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the operational implications of the following concepts as they apply to the Loss of Off-site Power: Principles of cooling by natural convection.

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

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	Ap

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

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

1202.012, chg. 004-03-0, RT-6 step J
1202.013, rev. 4, Fig. 2

History:

New for 2005 RO exam.

INSTRUCTIONS	CONTINGENCY ACTIONS
2. **Verify SW to DG1 and DG2 CLRs open to operating EDGs (CV-3806 and 3807).**	
3. **Verify OR start a Service Water pump on each operating DG, after 15-second time delay (P4A, B, C).**	3. IF both EDGs are operating **AND** only one Service Water pump can be started **AND** ESAS has not actuated, **THEN** perform the following: A. Close ACW Isolation (CV-3643). B. Verify both Service Water to ICW Coolers Supply valves open (CV-3811 and 3820).
4. **Actuate MSLI for both SGs AND verify proper actuation and control of EFW and MSLI (RT 6):** A. Operate ATM Dump CNTRL valves in HAND to minimize cycling and conserve Instrument Air. B. IF Instrument Air to ATM Dump CNTRL valves is lost, **THEN** perform the following: 1) Dispatch an operator with a radio to place ATM Dump CNTRL valves on hand jack **AND** fully open (Refer to Alternate Shutdown (1203.002), Exhibit A) 2) Establish SG press control using ATM Dump ISOL valves in MANUAL from the Control Room.	4. IF all EFW is lost, **THEN GO TO step 53.**
5. **Check RCS press remains ≥1700 psig** **AND** **PZR level remains ≥30".**	5. Initiate HPI (RT 2).

6. (Continued).

J. **IF** all RCPs are off,
THEN check primary to secondary heat transfer in progress indicated by all of the following:

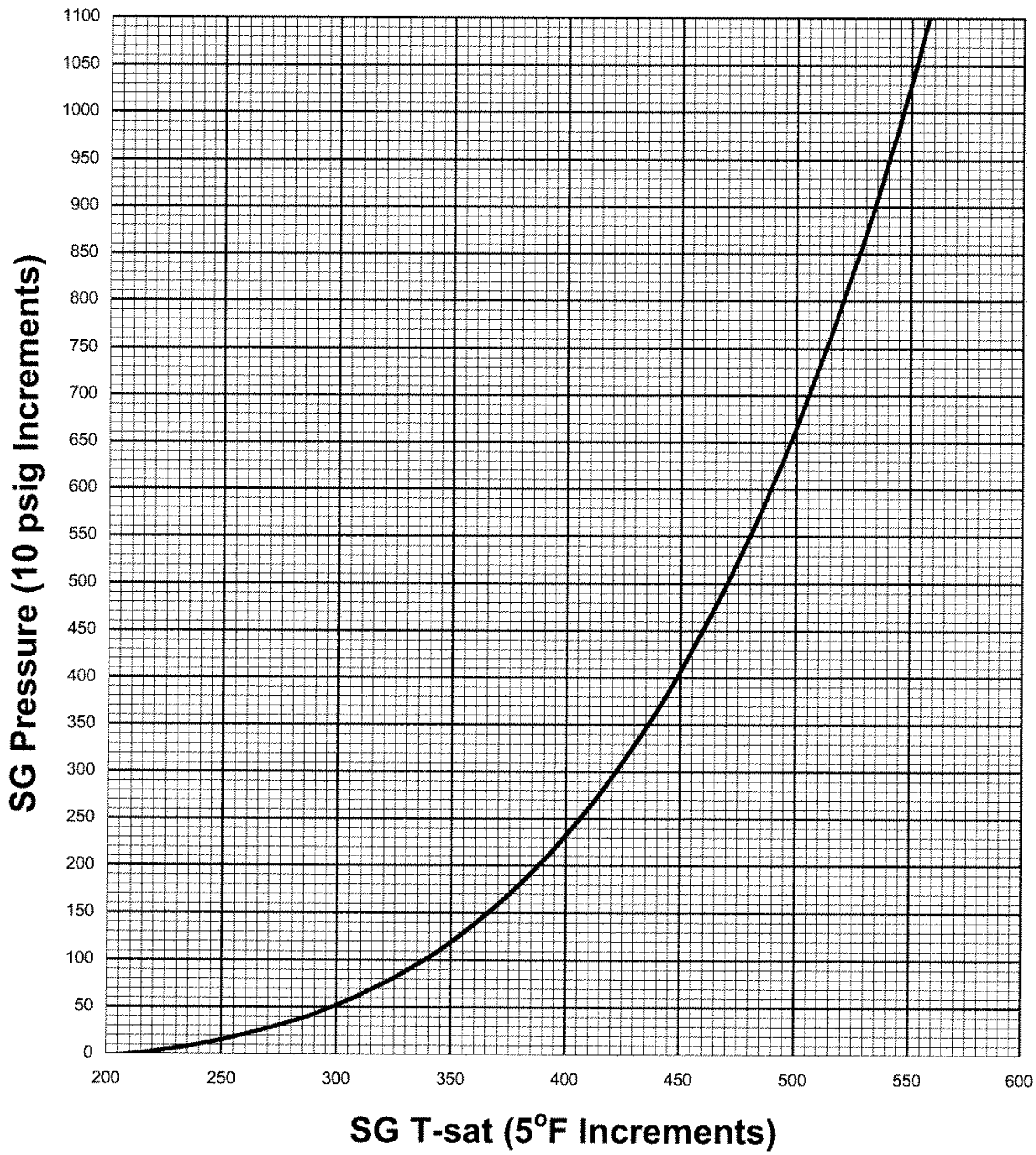
- T-cold tracking associated SG T-sat (Fig. 2)
- T-hot tracking CET temps
- T-hot/T-cold ΔT stable or dropping

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

Table 2
Examples of Less Than Adequate EFW Flow Indications
• SG level < 20" **AND** no EFW flow indicated
• All RCPs off **AND** SG level **not** tracking EFIC calculated setpoint
• All RCPs off **AND** EFIC level setpoint **not** trending toward applicable level band
Examples of Excessive EFW Flow Indications
• SG press drops ≥ 100 psig due to EFW flow induced overcooling
• SCM approaching minimum adequate due to EFW flow induced overcooling
• EFW CNTRL valve open with associated SG level > applicable setpoint level band

END

FIGURE 2
SG Pressure vs T-sat



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

QID: 0339 **Rev:** 0 **Rev Date:** 9/7/99 **Source:** Direct **Originator:** D Slusher
TUOI: A1LP-RO-ELECD **Objective:** 13C **Point Value:** 1

Section: 4.2 **Type:** Generic APE's
System Number: 057 **System Title:** Loss of Vital AC Electrical Instrument Bus
Description: Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual inverter swapping.

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

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

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

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

a. power RS-2 from Y-25.

b. de-energize RS-4.

c. parallel RS-2 and RS-4.

d. damage the Y-25 inverter.

Answer:

b. de-energize RS-4.

Notes:

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

References:

STM 1-32, Rev. 26

History:

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

Power distribution to and from the inverters is as follows:

INVERTER	AC INPUT	DC INPUT	OUTPUT TO
Y-11	51	D01	RS-1
Y-13	53	D01	RS-3
Y-15	57	D01	RS1/RS3
Y-22	65	D02	RS-2
Y-24	61	D02	RS-4
Y-25	63	D01	RS2/RS4
Y-41	56	D41	RC-1
Y-28	61	D02	C-540

4.3.1.2 Simplified Inverter Operation

(SEE FIGURE 32.54)

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

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

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

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

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

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

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

QID: 0613 **Rev:** 0 **Rev Date:** 8/9/05 **Source:** New **Originator:** Cork/Pullin

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

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

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

Description: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of DC power.

K/A Number: AK3.02 **CFR Reference:** 41.5,41.10 / 45.6 / 45.1

Tier:	1	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.2	**SRO Select:**	No	**Taxonomy:**	An

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

Given:

- Plant is at 100% power,
- D02 Undervoltage (K01-A8),
- D02 Trouble (K01-D8),
- Loss of breaker position indicator lights for plant buses on right side of C10.

Which action, with the correct reason for the action, should be performed?

A. Verify reactor is tripped due to loss of power to RCP underpower monitor circuit.

B. Trip the Generator Output Breakers to prevent the Main Generator from motoring.

C. Verify #2 EDG has automatically started and tied on to A4 ES bus due to undervoltage.

D. Line up Battery Charger D04A or D04B to the D02 Bus to restore DC power.

Answer:

A. Verify reactor is tripped due to loss of power to RCP underpower monitor circuit.

Notes:

Answer "A" is correct per the 1203.036 actions and discussion section.
Answer "B" is incorrect, although this is performed in Section 1 of 1203.036 for loss of D01, it has nothing to do with D02.
Answer "C" is incorrect, the A4 bus will be de-energized but the EDG will not start or tie on to the dead bus due to loss of control power.
Answer "D" is incorrect, there are no procedural actions to perform this. The procedure will swap D21 to D11.

References:

1203.036, Chg. 005-04-0

History:

New for 2005 RO exam, replacement question.

Control power for the generator output breakers (5114 and 5118) is from the switchyard 125V DC system, however, the breakers do not auto trip on generator lockout due to loss of power to the generator lockout circuit. Switchyard protective relays may trip the generator output breakers sensing an electrical fault. If power is restored to D11, the generator lockout circuit will actuate to trip the output breakers. The generator output breakers should not be manually tripped at this time because a loss of AC power will result as the turbine coasts down carrying the A1/A2 buses and H1/H2 buses. DG2 only will auto start if power is lost.

Restoration of power to distribution panel D11 by manually transferring to the emergency power source will allow automatic actions to occur such as turbine trip and generator lockout. This results in generator output and Exciter Field breakers tripping, then automatic transfer to a SU XFMR (SU 1 or SU 2).

If the generator output breakers were opened prior to re-energizing D11, an automatic transfer to offsite power could not take place and buses would remain energized from the turbine generator. If the turbine were tripped manually or electrically or mechanically due to overspeed or loss of vacuum, or the MSIVs are closed due to low pressure, the turbine would coast down and voltage and frequency control would be lost. The undervoltage condition would cause load shedding of even-train buses and starting of DG2. The odd train buses would become de-energized and DG1 could not start due to loss of DC control power.

If AC power is lost, inverters Y11, Y13, and Y15, cannot operate due to the loss of DC input power from D01. With a loss of alternate AC input power also, power will be lost to 120V AC panels RS1 and RS3. This causes a loss of power to two out of three ESAS analog channels, resulting in actuation of all ES even digital channels. Also, EFIC MSLI and EFW will actuate. Odd ES channels cannot actuate due to loss of power to the odd digital channels.

If AC power is **not** lost and a valid ES signal is received, the following valves will not reposition due to a loss of ES control (RA1 BKR 9).

- DH Cooler Bypass (CV-1433)
- Letdown Coolers Outlet (CV-1214 and CV-1216)
- RCP Seal Bleedoff Valves (CV-1270 thru CV-1273)

Effects of Loss of D02

A complete loss of bus D02 includes loss of 125V DC Station Battery Bank to Bus D02 (D06), loss of battery charger, and loss of distribution system. This results in the following conditions:

- If reactor power is >55%, reactor trip.
- Loss of power to even train distribution breaker control as well as other loads powered from bus D02.
- Loss of power to EOS Main Turbine Trip Solenoids (SV-6623 and SV-6624).
- Loss of power to EOS Channel B (SY-6650).

A reactor trip will result from loss of bus D02 if reactor power is >55% due to loss of power to the RCP underpower monitor circuit (RA2 BKR 16).

The reactor trip results in turbine trip and generator lockout relays (286 G1-1, 286 G1-2, and 286 G1-3) actuation, which causes automatic transfer to offsite power. With a loss of control power to the even train breakers, these breakers will not operate. This results in a loss of AC power to the even train buses. DG2 will not start due to a loss of control power.

With no AC or DC power, inverters Y22, Y24, Y25, and Y28 will be lost resulting in loss of power to 120V AC Panels RS2 and RS4. Inverters Y11, Y13, and Y15 remain in a normal mode.

Loss of power to RS2 and RS4 results in EFIC actuation of MSL and EFW. With a loss of D21, control power to EFW Pump (P-7A) is lost and the turbine will trip on overspeed. EFW control valves associated with P-7A are failed full open (loss of RA2).

Loss of power to Y02 results in closure of Purification Demineralizer Inlet and Makeup Filter Inlet valves causing letdown relief valve to lift. Letdown must be isolated by closing LD Cooler E-29A Outlet MOV (CV-1214) and LD Cooler E-29B Outlet MOV (CV-1216).

Loss of DC control power to Condenser Vacuum Pump (C-5B), if operating, causes the Seal Recirc Pump (P-31B) to stop and vacuum pump inlet valve to close. Upon restoration of DC control power, the condenser vacuum pump will trip and must be restarted or will auto start on low vacuum.

If a valid ES signal is received, DH Cooler Bypass (CV-1432) will not reposition due to a loss of ES control (RA2 BKR 11).

Effects of Loss of Both D01 and D02

A complete loss of both bus D01 and D02 includes loss of:

- 125V DC Station Battery Bank to Bus D01 (D07)
- 125V DC Station Battery Bank to Bus D02 (D06)
- Battery chargers to D07 and D06

- D01 distribution system
- D02 distribution system

This loss results in the following conditions:

- Reactor trip
- Loss of power to main turbine trip solenoids (SV-8524 and SV-8527 and XZ-8524).
- Loss of power to EOS Overspeed Trip Protection.
- Loss of EOS Main Turbine Trip Solenoids (SV-6623 and SV-6624).

- Loss of power to generator lockout relays (286 G1-1, 286 G1-2, and 286 G1-3).
- Loss of power to distribution breaker control power.

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

QID: 0554 **Rev:** 0 **Rev Date:** 3/30/05 **Source:** Direct **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.

K/A Number: 2.4.9 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	K

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

Plant heat up is in progress with RCS temperature at 215 degrees F.
Service water is lost to the in-service DH cooler.

What action is required during the re-establishment of SW flow through the DH cooler and why?

A. Establish SW slowly to prevent DH cooler water hammer.

B. Establish SW slowly to prevent SW pump runout.

C. Establish SW quickly to prevent DH cooler thermal shock.

D. Establish SW quickly to prevent RCS heat up.

Answer:

A. Establish SW slowly to prevent DH cooler water hammer.

Notes:

"A" is the correct answer per 1203.028 step 10, the others are concerns but are shown incorrectly.

References:

1203.028, chg. 016-03-0

History:

Direct from regular exam bank QID #ANO-OPS1-2788
Selected for 2005 RO exam

SECTION 4 -- LOSS OF SERVICE WATER FLOW

INSTRUCTIONS

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

2. **IF maintenance activities in the Reactor Building could be affected by RCS level rise, THEN perform local evacuation of the affected areas.**

{2}

3. **IF fuel loading is in progress, THEN terminate movement into core until DH cooling is returned to normal.**

4. **IF SW is available to idle DH cooler, THEN place respective DH loop in service for DH removal per applicable steps of Decay Heat Removal Operating Procedure (1104.004), "Decay Heat Removal During Cooldown" section.**

5. **Continue DH flow to provide mixing and minimize heatup rate of RCS.**

   A. IF RCS temp exceeds 280°F,
   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.

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 Time remaining to steam release is OR becomes <1 hour AND DH removal can NOT be immediately restored, THEN initiate containment closure per Attachment G of this procedure, while continuing with this section.**

(continued)

SECTION 4 -- LOSS OF SERVICE WATER FLOW

7. **IF RCS press approaches Decay Heat Sys. Max Pressure limit of Plant Shutdown and Cooldown (1102.010), THEN perform the following:**

   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.

8. **Investigate cause of loss of heat sink.**

CAUTION

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

9. **IF RCS >200°F AND there is NO SW flow through DH cooler, THEN perform the following:**

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

Cooler	Valve	Breaker
E-35A	CV-3822	B-5182
E-35B	CV-3821	B-6183

(continued)

SECTION 4 -- LOSS OF SERVICE WATER FLOW

10. WHEN SW is regained,
THEN restore SW flow to DH cooler as follows:

A. Station operator at cooler to listen for evidence of water hammer during next step.

> **NOTE**
> If SW side of DH cooler has reached saturation temp, slowly cutting in SW will minimize thermal shock and water hammer.

B. Slowly open applicable SW Inlet to E-35A or E-35B DH Cooler manually:

E-35A	E-35B
CV-3822	CV-3821

1) **IF** water hammer is observed,
THEN open SW Inlet more slowly.

C. **WHEN** SW valve is fully open,
THEN close associated supply breaker:

CV-3822	CV-3821
B-5182	B-6183

END

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

QID: 0555 **Rev:** 0 **Rev Date:** 4/4/05 **Source:** New **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing.

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

Tier: 1 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

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

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

A Loss of Instrument Air pressure has occurred.
Suddenly Instrument Air pressure quickly degrades to 35 psig.

Which of the following actions should be taken?

A. Open the Unit One to Unit Two Instrument Air Cross-connect.

B. Shutdown at greater than or equal to 10%/min. per 1203.045, Rapid Plant Shutdown.

C. Shutdown at greater than or equal to 5%/min. per 1203.045, Rapid Plant Shutdown.

D. Trip the reactor and go to 1202.001, Reactor Trip.

Answer:

D. Trip the reactor and go to 1202.001, Reactor Trip.

Notes:

"D" contains the proper action per 1203.024.
"A" is incorrect, the valve is closed if IA pressure dropped to <75 psig.
"B" and "C" are incorrect, the procedure is the proper procedure but the pressure is too low to continue plant operation.

References:

1203.024, chg. 010-08-0, section 2, step 3.1

History:

New for 2005 RO exam, replacement question.

SECTION 3 -- LOSS OF INSTRUMENT AIR PRESSURE (≤35 PSIG)

1.0 SYMPTOMS

1.1 IA header pressure ≤35 psig.

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

3.1 Trip reactor
<u>and</u>
perform Reactor Trip (1202.001) in conjunction with this procedure.

3.2 Fully actuate EFW and MSLI on both SGs.

3.3 Isolate Letdown by closing either:

- Letdown Coolers Outlet (CV-1221), or
- Letdown Cooler Outlets (CV-1214 and CV-1216)

3.4 Verify RCP Seal INJ Block (CV-1206) pushbutton in OVRD (OVRD light on).

NOTE

- Using HPI Block Valves CV-1220 or CV-1285 will minimize nozzle stress cycles because of the normal makeup path.
- Pressurizer Makeup Flow Control Valve (CV-1235) will fail as is, when IA pressure drops to ~45 psig.

3.5 <u>IF</u> necessary to maintain pressurizer level,
<u>THEN</u> use HPI block valves to provide makeup to RCS or take manual control of CV-1235 as follows:

3.5.1 Align stem and gear holes.

3.5.2 Install lock pin.

3.5.3 Open equalizing valve.

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

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

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

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

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

Description: Ability to determine and interpret the following as they apply to the (Inadequate Heat Transfer): Operating behavior characteristics of the facility.

K/A Number: EA1.2 **CFR Reference:** 41.7 / 45.5 / 45.6

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

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

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

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

A. 40°F (tubes hotter)

B. 60°F (tubes hotter)

C. 40°F (tubes colder)

D. 60°F (tubes colder)

Answer:

B. 60°F (tubes hotter)

Notes:

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

References:

1202.004, Chg. 004-02-0

History:

Direct from regular exambank, QID#1610.
Selected for use in 2005 RO exam, replacement question.

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

(6. CONTINUED ON NEXT PAGE)

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

Emergency and Abnormal Plant Evolutions – Tier1 /Group2 (RO / SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000001 Continuous Rod Withdrawal / 1							Not selected.		
000003 Dropped Control Rod / 1					X		AA2.02 Ability to determine and interpret the following as they apply to the Dropped Control Rod: Signal inputs to rod control system.	2.7	19
000005 Inoperable/Stuck Control Rod / 1							Not selected.		
000024 Emergency Boration / 1				X			AA1.03 Ability to operate and/or monitor the following as they apply to the Emergency Boration: Boric acid controller.	3.5	20
000028 Pressurizer Level Malfunction / 2							Not selected.		
000032 Loss of Source Range NI / 7							Not selected.		
000033 Loss of Intermediate Range NI / 7							Not selected.		
000036 (BW/A08) Fuel Handling Accident / 8							**Changed to "Not selected" (NRC comment "Not RO question"). Changed system to Plant Runback.**		
000037 Steam Generator Tube Leak / 3	X						AK1.02 Knowledge of the operational implications of the following concepts as they apply to the Steam Generator Tube Leak: Leak rate vs. pressure drop.	3.5	21
000051 Loss of Condenser Vacuum / 4						X	2.4.11 Knowledge of abnormal condition procedures.	3.4	22
000059 Accidental Liquid RadWaste Rel. / 9							Not selected.		
000060 Accidental Gaseous Radwaste Rel. / 9			X				AK3.02 Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste Release: Isolation of the auxiliary building ventilation.	3.3	23
000061 ARM System Alarms / 7							Not selected.		
000067 Plant Fire On-site / 8							Not selected.		
000068 (BW/A06) Control Room Evac. / 8				X			AA1.02 Ability to operate and/or monitor the following as they apply to the Control Room Evacuation: AFW emergency pump.	4.3	24
000069 (W/E14) Loss of CTMT Integrity / 5							Not selected.		
000074 (W/E06&E07) Inad. Core Cooling / 4							Not selected.		
000076 High Reactor Coolant Activity / 9							Not Selected.		
W/EO1 & E02 Rediagnosis & SI Termination / 3							Not applicable to this Unit.		
W/E13 Steam Generator Over-pressure / 4							Not applicable to this Unit.		
W/E15 Containment Flooding / 5							Not applicable to this Unit.		
W/E16 High Containment Radiation / 9							Not applicable to this Unit.		
BW/A01 Plant Runback / 1		**X**					**AK2.2 Knowledge of the interrelations between the Plant Runback and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.**	**3.5**	25
BW/A02&A03 Loss of NNI-X/Y / 7							Not selected.		
BW/A04 Turbine Trip / 4							Not selected.		

BW/A05 Emergency Diesel Actuation / 6	X						AK1.3 Knowledge of the operational implications of the following concepts as they apply to the (Emergency Diesel Actuation): Annunciators and conditions indicating signals, and remedial actions associated with the (Emergency Diesel Actuation).	3.8	26
BW/A07 Flooding / 8							Not selected.		
BW/E03 Inadequate Subcooling Margin / 4					X		EA2.1 Ability to determine and interpret the following as they apply to the (Inadequate Subcooling Margin): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	3.0	27
BW/E08; W/E03 LOCA Cooldown - Depress. / 4							Not selected.		
BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4							Not selected.		
BW/E13&E14 EOP Rules and Enclosures							Not selected.		
CE/A11; W/E08 RCS Overcooling - PTS / 4							Not applicable to this Unit.		
CE/A16 Excess RCS Leakage / 2							Not applicable to this Unit.		
CE/E09 Functional Recovery							Not applicable to this Unit.		
K/A Category Point Totals:	2	1	1	2	2	1			9

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

QID: 0557 **Rev:** 0 **Rev Date:** 10/23/98 **Source:** Direct **Originator:** C.Alden
TUOI: A1LP-RO-CRD **Objective:** 16 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Dropped Control Rod: Signal inputs to rod control system.

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

Tier: 1 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3
Group: 2 **SRO Imp:** 2.8 **SRO Select:** No **Taxonomy:** C

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

Given:
- The CRD system is in automatic and the plant is at 100% power.
- A dropped rod in Group 6 results in an asymmetric rod runback.

Group 6 rods will drive "in" when a Group 6 "in limit" is ON because:

A. In automatic, a Group 6 "out limit" bypasses the "in limit".

B. In automatic, an asymmetric rod runback bypasses the group "in limit".

C. The dropped rod's relative position is aligned with the group.

D. The "in limit" is only functional when the Diamond is in manual.

Answer:

B. In automatic, an asymmetric rod runback bypasses the group "in limit".

Notes:

"B" is correct per the CRD control logic.
The other choices represent actual CRD logic conditions but do not cause runbacks.

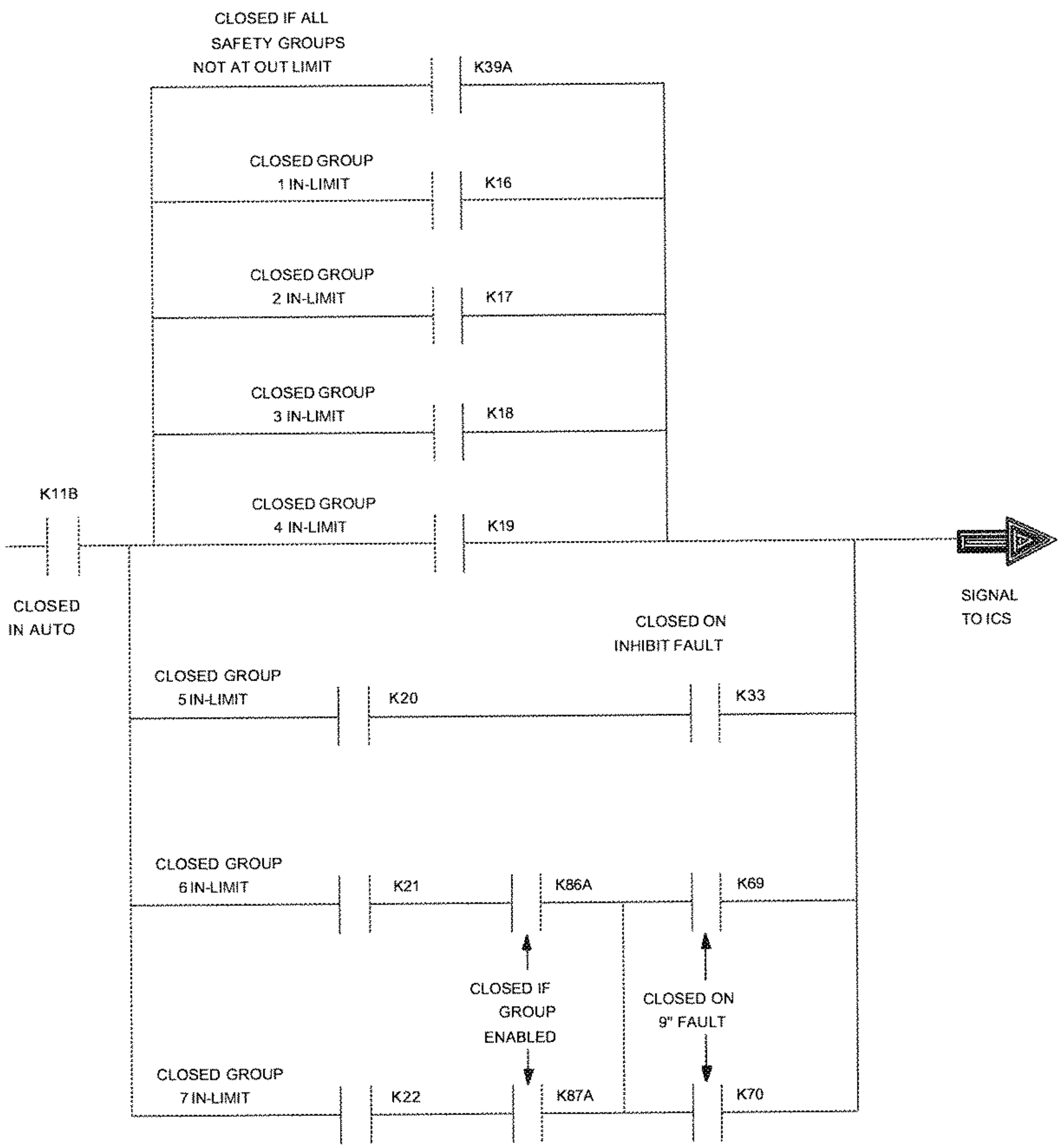
References:

STM 1-02, Rev. 5

History:

Direct from regular exam bank.
Selected for 2005 RO exam.

FIGURE 02.60: ASYMMETRIC ROD CIRCUIT



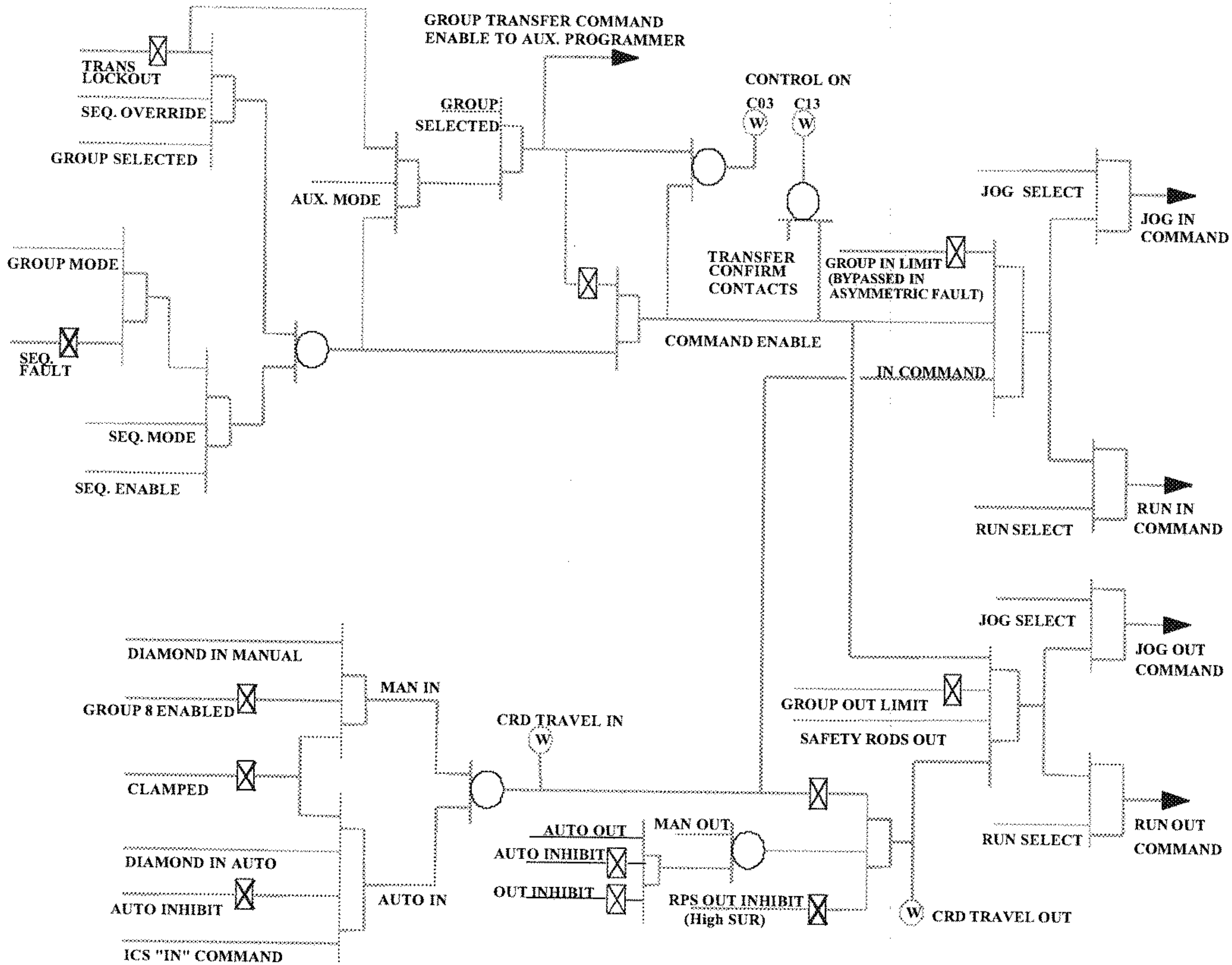


FIGURE 02.65: COMMAND LOGIC GROUP 5, 6 OR 7

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

QID: 0495 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Repeat **Originator:** NRC

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

Section: 4.2 **Type:** Generic APE

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

Description: Ability to operate and/or monitor the following as they apply to the Emergency Boration: Boric acid controlller.

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

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

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

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

Given:

- Reactor is tripped from 100% power
- Four CRDM's fail to insert according to indications in the control room
- The CRS directs you to initiate Emergency Boration in accordance with RT-12 "Emergency Boration"

You should set the INITIAL batch setting of the boric acid controller to:

A. The batch size required to maintain make up tank level between 55 and 86 inches while maintaining pressurizer level >100 inches.

B. The batch size required in order to obtain a shutdown margin of 1.5% delta K/K as determined by a reactivity balance calculation.

C. The maximum batch size setting and commence adding boric acid to the make up tank.

D. The batch size determined by the plant computer boron program to offset the reactivity worth of the four stuck rods.

Answer:

C. The maximum batch size setting and commence adding boric acid to the make up tank.

Notes:

"C" is the correct answer. RT-12 instructs the operator to commence emergency boration by setting the batch controller to the maximum batch size (999999 gals) and to begin adding boric acid via the batch controller if a boric acid pump is available. Therefore, answer "C" is correct. Answers "B" and "D" describe actions to determine the exact batch size after commencing emergency boration. The question is asking for the initial setting of the batch controller. Answer "A" uses a variety of setpoints associated with emergency boration incorrectly.

References:

1202.012 (Rev 004-03-0), Repetitive Tasks, RT-12, Emergency Boration.

History:

Developed by NRC. Modified QID 005 from ANO-1 NRC Exam Bank.
(QID 005 was used on the 1998 RO exam and the 2001 RO/SRO exam.)
Used on 2004 RO/SRO Exam.
Selected for 2005 RO exam

12. Emergency Boration:

A. IF Boric Acid pump (P39A or B) and Batch Controller are available, **THEN** perform the following:

1) Set Batch Controller for maximum batch size (999999).

2) Verify Condensate to Batch Controller (CV-1251) closed.

3) Open Batch Controller Outlet (CV-1250).

4) Verify both Letdown Filters in service (F-3A and B).

5) Record initial BAAT (T-6) level ____________ in.

6) Start available Boric Acid Pump(s) (P-39A or B or both).

7) Start Batch Controller by depressing RUN key.

8) Adjust Batch Controller Flow CNTRL VLV (CV-1249) to 100% open.

9) Adjust Pressurizer Level Control Setpoint to 220".

10) Open BWST Outlet to OP HPI Pump (CV-1407 or 1408).

11) **WHEN** PZR level is ≥ 100", **THEN** establish maximum Letdown flow.

12) 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) **WHEN** MU Tank level is lowered to desired level, **THEN** 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).

(12. CONTINUED ON NEXT PAGE)

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

QID: 0216 **Rev:** 0 **Rev Date:** 11/18/98 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-EOP06 **Objective:** 16 **Point Value:** 1

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

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

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

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

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	C

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

Given:

- A 15 gpm Steam Generator tube leak cooldown is in progress.
- Normal cooldown limits are being used with the good OTSG.
- RCS pressure is 1000 psig, Tave is 405°F.

The CBOR is maintaining the RCS at about 140°F subcooled.

Why are the CBOR's actions incorrect for this accident?

a. Tube to shell Delta T limits are being exceeded.

b. A high primary to secondary Delta P is increasing primary coolant loss.

c. Excessive thermal stresses are being imposed on the Rx vessel.

d. Overfill could cause the ruptured SG main steam safeties to lift.

Answer:

b. A high primary to secondary DP is increasing primary coolant loss.

Notes:

[b] is correct. The operators are directed to maintain RCS pressure low within the limits of Figure 3. This will result in subcooling margin close to the limit and as low as possible primary to secondary differential pressure to prevent loss of unrecoverable primary coolant.
[a] is incorrect, there is not enough information given to determine if tube to shell DT limits are being exceeded.
[c] is incorrect, excessive thermal stresses are not being imposed without a high pressure condition.
[d] is incorrect, overfill could not cause the safeties to lift at 405°F.

References:

1202.006 Chg. 007-04-0, step 14
120.013, Rev. 4, Fig. 3

History:

Developed for A. Morris 98 RO Re-exam.
Used in 2001 RO/SRO Exam.
Selected for 2005 RO exam

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

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

14. Operate Pressurizer Heaters AND Pressurizer Spray valve (CV-1008) to maintain RCS press low within limits of Figure 3.

A. **WHEN** RCS press is <1700 psig, **THEN** bypass ESAS.

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

15. Stabilize PZR level ≥55" as follows:

A. Adjust Pressurizer Level Control setpoint to 100".

B. **IF** HPI is in service, **THEN** adjust HPI flow as necessary to maintain PZR level ≥55" **AND** RCS press low within limits of Figure 3.

B. **IF** necessary to maintain PZR level ≥55", **THEN** initiate HPI (RT 2).

16. Verify OTSG N-16 monitors selected to GROSS.

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

QID: 0025 **Rev:** 0 **Rev Date:** 7/8/98 **Source:** Direct **Originator:** GGiles
TUOI: A1LP-RO-AOP **Objective:** 2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs
System Number: A04 **System Title:** Loss of Condenser Vacuum
Description: Knowledge of abnormal condition procedures.

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

Tier:	1	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	An

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

The following plant conditions existed just prior to an automatic reactor trip:

- Reactor power at 45%
- Loss of Bus H-1 due to an electrical fault
- "B" Main Feedwater Pump tripped
- Condenser vacuum at 23.5 inches Hg
- RCS pressure at 1950 psig

Which of the following was the most likely cause of the automatic reactor trip?

a. Main Turbine Anticipatory trip.

b. Power to Pumps trip.

c. Variable Low Pressure trip.

d. Main Feedwater Anticipatory trip.

Answer:

a. Main Turbine Anticipatory trip.

Notes:

Answer (a) is correct since an automatic turbine trip on low vacuum will occur at 24.5 inches Hg. (Note: The loss of condenser vacuum procedure requires manually tripping the turbine at 24.5 inches Hg if >270 MWe, but the question states an automatic trip occurred.) Answers (b), (c) and (d) are incorrect because they list other automatic trips that are not applicable to the given conditions.

References:

1203.016 Chg. 011-05-0
1202.001 Chg. 028-03-0

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam, replacement question.

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: 0606 **Rev:** 0 **Rev Date:** 8/1/2005 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste Release: Isolation of the auxiliary building ventilation.

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

Tier:	1	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	K

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

Which of the following actions with reasons will NOT occur if Gaseous Radwaste radiation monitor, RE-4830, is in a high alarm condition?

A. Gaseous Radwaste Discharge Isolation valve, CV-4830, closes to isolate any releases from the Aux. Bldg. vent header.

B. Gaseous Radwaste Discharge Flow Control valve, CV-4820, closes to isolate any Decay Tank release in progress.

C. RB Vent Header Isolation to T-17, CV-4804, closes to isolate any additions to system from RB vent header.

D. AB Vent Header Diversion to T-17 valve, CV-4806, opens to send Decay Tank release back to surge tank.

Answer:

C. RB Vent Header Isolation to T-17, CV-4804, closes to isolate any additions to system from RB vent header.

Notes:

Answer "C" is correct, the other three occur on a RE-4830 high alarm, no action occurs with CV-4804.

References:

1203.006, Chg. 010-02-0

History:

New for 2005 RO exam, replacement question.

INSTRUCTIONS

1. Secure any radioactive gas release or venting in progress.

2. Verify following valves closed:

- Station Vent Discharge Valve (CV-4830)
- Gaseous Radwaste Hdr. Isolation Valve (CV-4820)

3. Verify ABVH Diversion Valve to Surge Tank (CV-4806) open.

**4. IF venting operation was in progress,
THEN perform the following:**

A. **IF** desired to continue venting operations,
THEN align the waste gas system per Gaseous Radwaste System (1104.022), "Venting of High Activity Systems".

> **NOTE**
>
> Steps 1-6 may be re-performed as needed to purge the piping upstream of RE-4830 if it trips during the performance of this section.

B. **IF** venting will be terminated and it is desired to align the vent header to the Station Vent Plenum,
THEN perform the following:

1. Reset RE-4830.

    A. **IF** RE-4830 will not reset due to current radiation levels being above the setpoint,
    THEN perform 1305.001 Supplement 5 to raise the setpoint of RE-4830 and allow reset of the detector.

2. Open CV-4830.

3. Re-establish N2 flow to previous amount or as desired to be consistent with system conditions.

4. Close CV-4806.

5. Verify C-9A and C-9B in AUTO.

    A. Verify C-9A and C9B shut down when T-17 $\leq$ 15.2 psia.

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

QID: 0558 **Rev:** 0 **Rev Date:** 4/4/05 **Source:** New **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Control Room Evacuation: AFW emergency pump.

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

Tier:	1	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.5	**SRO Select:**	No	**Taxonomy:**	C

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

Given:
- An immediate Control Room evacuation is in progress due to a fire in the Cable Spreading Room.
- All Control Room actions have been completed.
- It is approximately 40 minutes into the event.

How is feedwater being supplied to the OTSGs?

A. Motor driven EFW pump P-7B is initially used and then automatic control of steam driven EFW pump P-7A is established.

B. Steam driven EFW pump P-7A with automatic control of P-7A flow control valves.

C. Motor driven EFW pump P-7B with automatic control of P-7B flow control valves.

D. Motor driven EFW pump P-7B is initially used and then local control of steam driven EFW pump P-7A is established.

Answer:

D. Motor driven EFW pump P-7B is initially used and then local control of steam driven EFW pump P-7A is established.

Notes:

"D" is correct, EFW is automatically actuated, then P-7B is secured and P-7A speed is controlled locally.
"A" is incorrect, automatic control of P-7A is not used.
"B" and "C" are incorrect, automatic control of flow control valves is not used.

References:

1203.002, 015-07-0

History:

New for 2005 RO exam

ATTACHMENT 9

DISCUSSION CONTINUATION

1.0 DESCRIPTION

Alternate Shutdown capability is being provided to comply with 10 CFR 50, Appendix R, and to mitigate consequences of significant fire in either the Control Room or the Cable Spread Room because fires in these areas can result in loss of controls and instrumentation required for safe shutdown. Following are some of the more significant assumptions considered in development of this procedure:

NOTE

This procedure will be used even if a coincident loss of off-site power has not occurred.

A. A coincident loss of off-site electrical power has occurred.

B. No coincident design basis accident occurs, only accidents that are a direct result of the fire.

C. Control circuits (for valves, motors, etc.) located in the fire area are assumed to cause activation of the component to the least desirable condition unless preventative action is taken.

2.0 SYSTEMS

The following are the safe shutdown system components utilized by the Alternate Shutdown procedure and a brief description of how they are used.

2.1 Emergency Feedwater

Local control of steam-driven EFW Pump (P-7A) is established with the EFW Turb K3 Trip/THROT VLV (CV-6601A). EFW P-7A to SG-B ISOL (CV-2620) is manually controlled to control SG-B level. These actions are accomplished by RO #2.

RO #1 will manually control EFW P-7A to SG-A ISOL (CV-2627) to control the SG-A level.

Steam release will be initially controlled by the mechanical steam relief valves and, as additional operators become available, by manual control of the ADVs.

The EFW CST (T-41B) will provide the initial source of water to the EFW system with the CST (T-41) and service water, the backup sources, if necessary.

As time permits, the P-7B train of EFW will be made available for backup purposes.

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

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

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

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

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

Description: Knowledge of the interrelations between the Plant Runback and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

K/A Number: AK2.2 **CFR Reference:** 41.7 / 45.7

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

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

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

Reactor power is 90% and generated megawatts is 800. After a runback for loss of one main feedwater pump, the ICS should stabilize the plant at ________________.

A. 360 Mwe

b. 340 Mwe

c. 45% Reactor Power

d. 50% Reactor Power

Answer:

a. 360 MWe

Notes:

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

References:

1105.004, Chg. 016-02-0

History:

Taken from Exam Bank QID # 4
Used in A. Morris 98 RO Re-exam
Selected for use in 2005 RO exam, replacement question.

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

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

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

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

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

- Reactor trip
- Runback in effect
- Cross limits in effect

- Breakers 5114 and 5118 open (generator output)
- SG/RX Demand in HAND
- Reactor Demand in HAND

- Rod Controller (Diamond Panel) in MANUAL
- Turbine Control in TURBINE MANUAL or OPER AUTO.
- Both Feedwater Demand Loop A and Feedwater Demand Loop B in HAND.

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

QID: 0276 **Rev:** 0 **Rev Date:** 9/2/99 **Source:** Direct **Originator:** D Slusher

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

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

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

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

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

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2.5

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

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

Given:

- A loss of offsite power has occurred.
- Annunciator K01-B1, "EDG 1 BRKR AUTO CLOSE FAILURE", is in alarm.

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

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

b. Depress EDG #1 start push-button.

c. Reset A1 Lockout relay.

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

Answer:

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

Notes:

(a) is correct, taking HS to PTL will reset anti-pump relays and allow breaker to auto-close.
(b) is incorrect, this will accomplish nothing because an auto-close would not exist unless EDG was running.
(c) is incorrect, resetting Lockout Relay will have no effect on EDG output breaker .
(d) is incorrect since breaker cannot be closed manually from C-10 unless the sync switch is ON.

References:

1203.012A, Chg. 034-03-0

History:

Developed for 1999 exam.
Selected for 2005 exam

Location: C10

Device and Setpoint: see next page.

EDG 1 BRKR
AUTO CLOSE
FAILURE

Alarm: K01-B1

1.0 OPERATOR ACTIONS

1. If bus A3 is deenergized, verify the following breakers are open:

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

2. Attempt to reset breaker anti-pump feature as follows:

   A. Place control switch for DG1 Output Breaker (A-308) in pull-to-lock and release.

3. If breaker did not close, turn Synchronize switch ON for DG1 Output (A-308).

4. Attempt to close A-308 from C10.

5. If A-308 fails to close from C10, close locally.

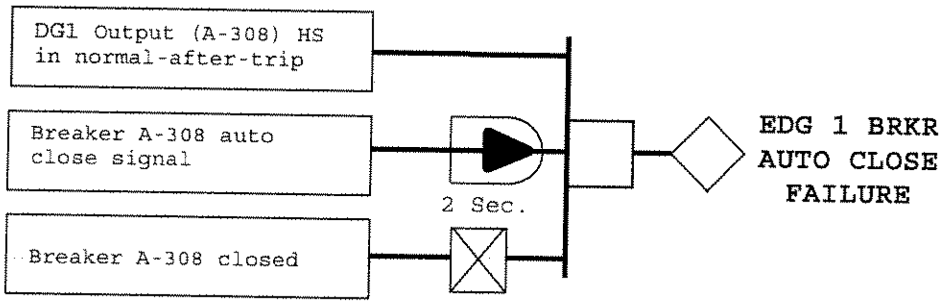
6. To clear alarm, remove A-308 HS from normal-after-trip position.

2.0 PROBABLE CAUSES

1. Breaker A-308 tripped with an auto close signal.

3.0 REFERENCES

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



DG1 Output (A-308) HS in normal-after-trip
Breaker A-308 auto close signal
2 Sec.
Breaker A-308 closed
EDG 1 BRKR AUTO CLOSE FAILURE

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

QID: 0421 **Rev:** 1 **Rev Date:** 4/5/05 **Source:** Modified **Originator:** S.Pullin

TUOI: A1LP-RO-EOP01 **Objective:** 5 **Point Value:** 1

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

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

Description: Ability to determine and interpret the following as they apply to the Inadequate Subcooling Margin): 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:** 3.0 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** Ap

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

Given a Rx trip has occurred and one minute later the following conditions are observed:

- RCS pressure is stable 1700 psig.
- CET average temperature is 600 degrees F.

Which Emergency Operating Procedure contains mitigating actions for this event?

a. Loss of Subcooling Margin (1202.002)

b. ESAS (1202.010)

c. Overheating (1202.004)

d. Inadequate Core Cooling (1202.005)

Answer:

a. Loss of Subcooling Margin (1202.002)

Notes:

Answer "a" is the right procedure and the rest are incorrect.

References:

1202.013, Rev. 4, Figure 1
1202.002, Chg. 004-02-0

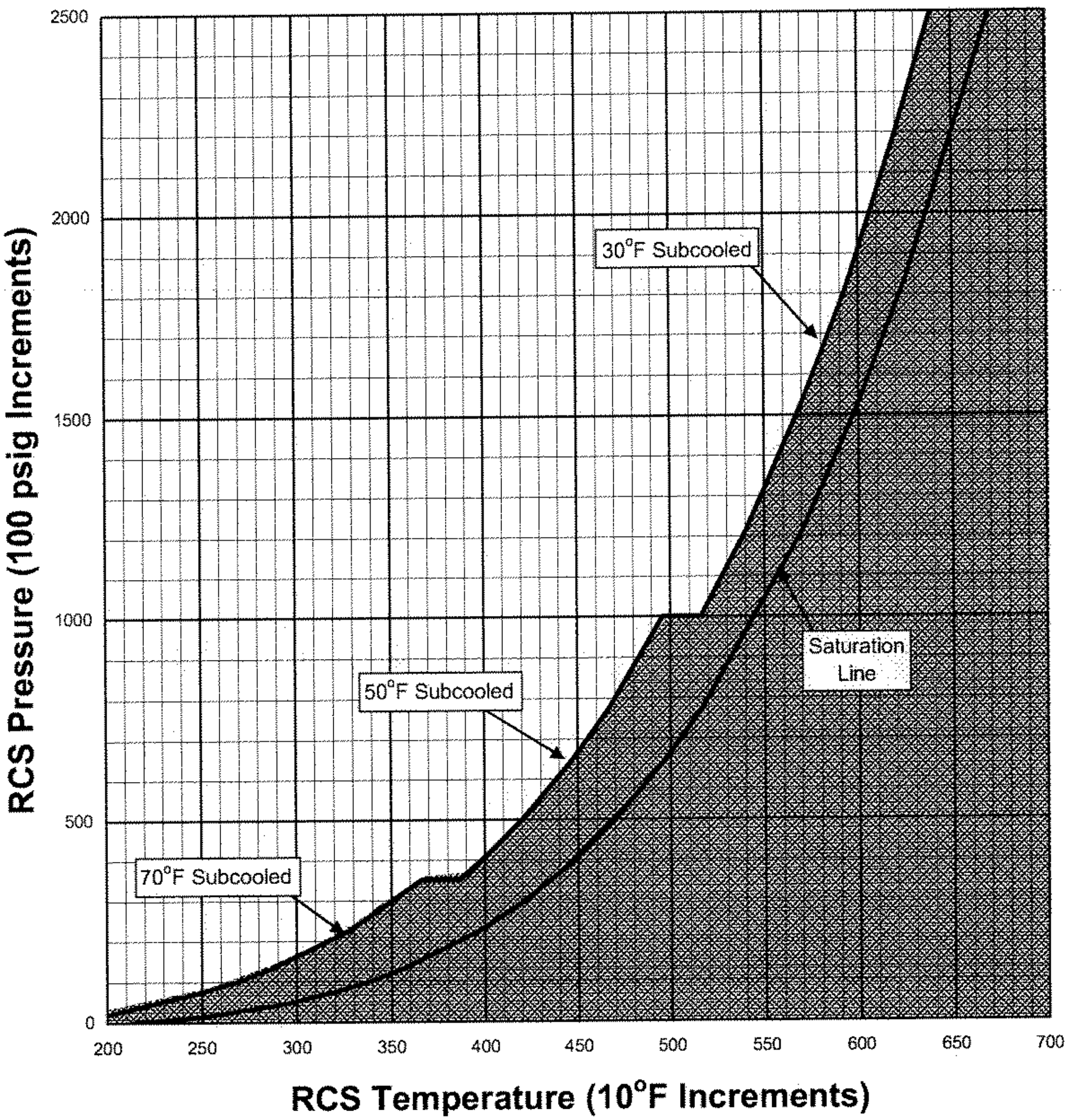
History:

Direct from regular exambank QID 2928.
Selected for use in 2002 RO/SRO exam.
Modified for use in 2005 RO exam

ENTRY CONDITIONS

- Loss of adequate SCM following a Reactor trip,
- Loss of adequate SCM while attempting to correct overcooling,
- Loss of adequate SCM following ESAS actuation
  AND
  RCS press stabilizes >150 psig (LPI pump discharge pressure).

FIGURE 1
Saturation and Adequate SCM



RCS Pressure	Adequate SCM
>1000 psig	≥30°F
350 to 1000 psig	≥50°F
<350 psig	≥70°F

ES-401	PWR Examination Outline Plant systems – Tier 2/Group 1 (RO / SRO)													Form ES-401-2

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	#
003 Reactor Coolant Pump						X						K6.14 Knowledge of the effect of a loss or malfunction of the following will have on the RCPS: Starting requirements.	2.6	28
004 Chemical and Volume Control			X									K3.05 Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: PZR LCS.	3.8	29
005 Residual Heat Removal							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: heatup/cooldown rates.	3.5	30
005 Residual Heat Removal										X		A4.02 Ability to manually operate and/or monitor in the control room: Heat exchanger bypass flow control.	3.4	31
006 Emergency Core Cooling						X						K6.02 Knowledge of the effect of a loss or malfunction of the following will have on the ECCS system: Core flood tanks (accumulators).	3.4	32
006 Emergency Core Cooling											X	2.4.6 Knowledge of symptom based EOP mitigation strategies.	3.1	33
007 Pressurizer Relief/ Quench Tank					X							K5.02 Knowledge of the operational implications of the following concepts as they apply to the PRTS: Method of forming a steam bubble in the PZR.	3.1	34
007 Pressurizer Relief/ Quench Tank							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining Quench Tank water level within limits.	2.9	35
008 Component Cooling Water		X										K2.02 Knowledge of bus power supplies to the following: CCW pump, including emergency backup.	3.0	36
010 Pressurizer Pressure Control								X				A2.03 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: PORV failures.	4.1	37
012 Reactor Protection					X							K5.01 Knowledge of the operational implications of the following concepts as they apply to the RPS: DNB.	3.3	38
012 Reactor Protection										X		A4.04 Ability to manually operate and/or monitor in the control room: Bistable trips, reset and test switches.	3.3	39
013 Engineered Safety Features Actuation	X											K1.06 Knowledge of the physical connections and/or cause-effect relationships between the ESFAS and the following systems: ECCS.	4.2	40
013 Engineered Safety Features Actuation											X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	3.3	41
022 Containment Cooling									X			A3.01 Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation.	4.1	42
025 Ice Condenser												Not applicable to this Unit.		
026 Containment Spray				X								K4.06 Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Iodine scavenging via the CSS.	2.8	43

039 Main and Reheat Steam							X					A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main Steam pressure.	3.0	44
059 Main Feedwater	X											K1.07 Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: ICS (FWCS).	3.2	45
059 Main Feedwater									X			A3.04 Ability to monitor automatic operation of the MFW, including: Turbine driven feed pump.	2.5	46
061 Auxiliary/Emergency Feedwater		X										K2.02 Knowledge of bus power supplies to the following: AFW electric drive pumps.	3.7	47
062 AC Electrical Distribution								X				A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the AC Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect on plant of de-energizing a bus.	3.1	48
062 AC Electrical Distribution										X		A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard).	3.3	49
063 DC Electrical Distribution			X									K3.02 Knowledge of the effect that a loss or malfunction of the DC Electrical Distribution System will have on the following: Components using DC control power.	3.5	50
064 Emergency Diesel Generator									X			A3.04 Ability to monitor automatic operation of the ED/G system, including: Number of starts available with an air compressor.	3.1	51
073 Process Radiation Monitoring				X								K4.01 Knowledge of PRM System design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint.	4.0	52
076 Service Water		X										K2.01 Knowledge of bus power supplies to the following: Service Water.	2.7	53
078 Instrument Air				X								K4.02 Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems.	3.2	54
103 Containment			X									K3.02 Knowledge of the effect that a loss or malfunction of the Containment System will have on the following: Loss of containment integrity under normal operations.	3.8	55
K/A Category Point Totals:	2	3	3	3	2	2	3	2	3	3	2	Group Point Total:		28

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

QID: 0615 **Rev:** 0 **Rev Date:** 8/9/2005 **Source:** New **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 System

Description: Knowledge of the effect of a loss or malfunction of the following will have on the RCPS: Starting requirements.

K/A Number: K6.14 **CFR Reference:** 41.7 / 45.5

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

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

Given:
- Plant heatup in progress from refueling outage.
- P-32B, P-32C, P-32D 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 245 gpm.
- Nuclear ICW to RCP seal cooling flow is 35 gpm.
- RCS loop A & B cold leg temps are 380°F.

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. Non-nuclear ICW to RCP motor cooling flow is low.

D. RCS cold leg temps are low.

Answer:

C. Non-nuclear ICW to RCP motor cooling flow is low.

Notes:

"C" is correct, non-nuclear ICW to RCPS is less than 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.
"D" is incorrect, RCS cold legs must be greater than 375°F to start the fourth RCP and they are.

References:

1103.006, Chg. 025-04-0

History:

New (modified new QID 559) for 2005 RO exam, replacement question.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0559 **Rev:** 0 **Rev Date:** 4/6/05 **Source:** New **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 System
Description: Knowledge of the effect of a loss or malfunction of the following will have on the RCPS: Starting requirements.

K/A Number: K6.14 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	2.6	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: SRO:

Given:
- Plant heatup in progress from refueling outage.
- P-32A, P-32C, P-32D 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.

PARENT Question

A start of RCP P-32B is attempted but is unsuccessful. Why?

A. Nuclear ICW to RCP seal cooling flow is low.

B. Seal injection flow is low.

C. Non-nuclear ICW to RCP motor cooling flow is low.

D. RCS cold leg temps are low.

Answer:

D. RCS cold leg temps are low.

Notes:

"D" is correct, RCS cold legs must be greater than 375°F to start the fourth RCP.
"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, non-nuclear ICW to RCPS is greater than 250 gpm.

References:

1103.006, Chg. 025-04-0

History:

New for 2005 RO exam, but not used. Modified version of 615.

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 >375°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).

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 $\Delta T \leq 50°F$.

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 >375°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 (VALIDATION)

QID: 0432 **Rev:** 0 **Rev Date:** 4/30/2002 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-MU **Objective:** 10 **Point Value:** 1

Section: 3.2 **Type:** RCS Inventory Control

System Number: 004 **System Title:** Chemical and Volume Control System

Description: Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: PZR LCS.

K/A Number: K3.05 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 29 **SRO:**

Unit One is operating at 100% power.
An ICW problem causes Letdown temperature to rise to 148°F.

What is the effect on the Pressurizer level control system during this transient?

A. PZR level will continue to drop during this event.

B. PZR level will continue to rise during this event.

C. Makeup flow will rise and restore Pressurizer level to setpoint.

D. Makeup flow will drop and restore Pressurizer level to setpoint.

Answer:

B. PZR level will continue to rise during this event.

Notes:

"B" is correct as the high letdown temperature will cause letdown to isolate, makeup flow will drop but seal injection will continue to provide 8 gpm per RCP makeup to the RCS, so PZR level will continue to rise.
"A" is incorrect, PZR level will not drop.
"C" & "D" are incorrect, PZR level will not be restored to setpoint.

References:

STM 1-04, Rev. 7

History:

New for 2005 RO exam

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.

- Two normal makeup isolation valves (CV-1233 & CV-1234)
- Connection to HPI line on discharge of RCP "D" in loop "A" of the RCS
- A ~10-20GPM constant flow bypass for thermal stress mitigation

The pressurizer level control valve is controlled from the Non Nuclear Instrumentation system (NNI) by the pressurizer level control program. The signal received by this valve is dependent on both the setpoint and the deviation from setpoint of level in the pressurizer. During normal conditions the setpoint is 220". CV-1235 will open and/or close to maintain this level.

1.3.5 RCP Seal Injection and Seal Return.

Seal injection supplies filtered water to the RCP seals. Most of this water goes into the RCS. Some bleed-off from the seals is returned via seal return. Seal injection and return consists of:

- Supply flow element (FE-1239)
- Control valve and bypass (CV-1207)
- Seal injection filter (F-2)
- Seal injection isolation valve (CV-1206)
- Stop check valves, throttle valves and flow elements on the individual supply lines in containment.
- Motor operated valves on each seal return line inside containment.
- Flow indicating transmitter
- Motor operated isolation valve outside containment CV-1274
- An alternate path to the quench tank and its isolation valves
- Seal return coolers
- Connection to other systems

1.3.6 Connections to other systems

The makeup and purification system interfaces with various systems to perform its intended functions. These systems include:

Clean Liquid Radwaste System

During normal operation, letdown flow is directed into the Makeup Tank. However, flow can be diverted, using CV-1248, to the CZ system to be processed by the vacuum degasifier for storage in the T-12's or for return to the makeup tank. Piping is also provided to return T-12 liquid to makeup and purification for RCS filling if desired. With the vacuum degasifier manually bypassed, flow diverted from letdown will go directly to a T12. This would be done during plant heatup or chemical control feed-and-bleed operations. Also directed to the makeup tank are RCP seal bleedoff and HPI pump minimum recirculation flows.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0091 **Rev:** 0 **Rev Date:** 7/12/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-DHR **Objective:** 11 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 005 **System Title:** Residual Heat Removal System (RHRS)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Heatup/cooldown rates.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 30 **SRO:**

Given:

- Plant cooldown in progress
- "A" DH Removal pump in service
- "A" and "C" RCPs running

What is the maximum allowable cooldown rate in this condition?

a. [illegible]°F/hr

b. 25°F/hr

c. 50°F/hr

d. 100°F/hr

Answer:

c. 50°F/hr

Notes:

Per 1102.010, the maximum cooldown rate is 50°F/hr when the RCS is <280°F and >150°F, "c" is correct.
DH cannot be placed in service unless the RC temp is <280°F, therefore "D" is incorrect.
The RCP's are removed from service while a PZR steam bubble is still present and specifically when RC temp is between 166°F and 270°F, therefore "B" is incorrect.
"A" is fictitious.

References:

1102.010, Plant Shutdown and Cooldown, Chg. 053-10-0

History:

Developed for 1998 RO/SRO exam
Used in 2005 RO Exam

NOTE

- Maximum cooldown rates are as follows:

  A. RCS >280°F: 100°F/hr (1.67°F/min)
  B. RCS 280-150°F: 50°F/hr (0.83°F/min)
  C. RCS <150°F: 25°F/hr (0.41°F/min)
  D. Pressurizer: 100°F/hr (1.67°F/min)
  E. If RCPs are operating between 180°F to 150°F, maximum RCS cooldown rate is limited to 2°F/hr.

- If available, computer indications may be used for manual plotting of RC and Pressurizer temperatures versus time.

- All graphs used during cooldown shall include the following:

  A. Incremental value
  B. Time
  C. Date
  D. Initial
  E. Parameter, including instrument number

- If available, SPDS cooldown display may be used to monitor RCS and PZR cooldown.

- The following pressure instruments are the most accurate for their ranges during all RCP combinations and are preferred for plotting pressure versus temperature.

  A. 0 to 500 psig: Low Range RC Pressure indicator (PI-1010)
  B. 500 to 1700 psig: SPDS points P1041 and P1042
  C. 1700 psig and above: SPDS points P1021, P1023, & P1039

  Instrument error on SPDS graphics displays may cause data points to be outside the limits on the SPDS PT curve.

- With RCPs inservice, the lowest WR RCS Cold Leg Temperature instrument which has a RCP operating in the same loop should be used for plotting RC temperature. If no RCPs are operating, use lowest.

Loop A	Loop B	Loop C	Loop D
T1047 PMS & SPDS	T1045 PMS	T1017 PMS	T1016 PMS & SPDS
T1147 SPDS	T1144 SPDS	T1117 SPDS	T1115 SPDS

- Due to inaccuracies, Loop A Wide Range Pressure PT-1020 (SPDS point P1020) should NOT be used for plotting pressure versus temperature.

- Plant Computer points T1406 and T1407 are the most accurate indication of DH cooler outlet temperature and should be used when available.

- If CET data from ICC Train A or Train B on C19 is unavailable, obtain SPDS CET value from TCETA1 or TCETB1 or one of the inputs NOT designated (ICC) in the SPDS data base listing.

- When both Decay Heat Removal systems are in service on Decay Heat, use the lower DH cooler outlet temperature for plotting cooldown rate.

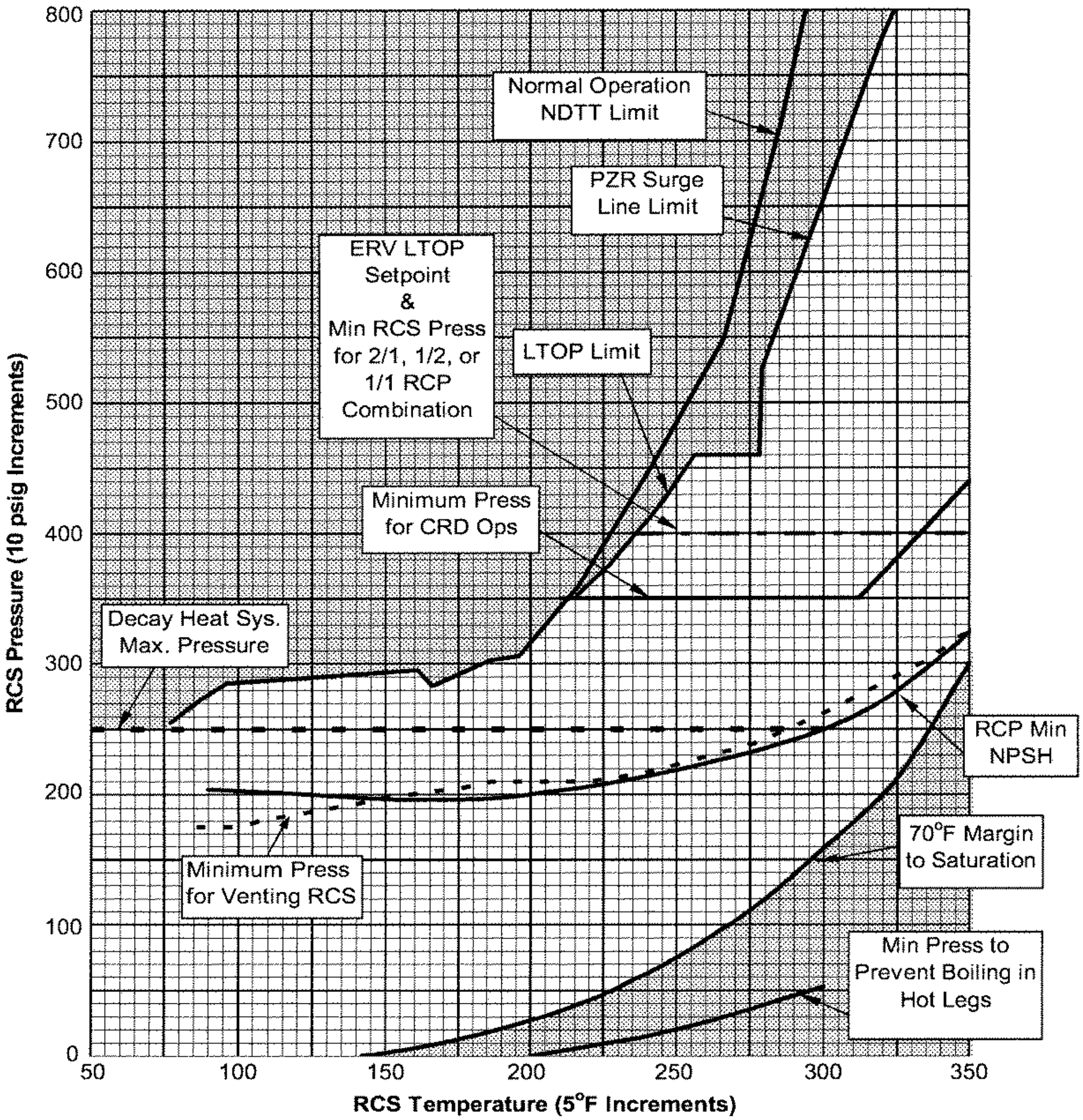
- SG Tube-to-Shell ΔT may be monitored on SPDS PSHT or SGTR display.

{4.3.4}

ATTACHMENT A

Page 1 of 2

Allowable RCS Pressure vs. Temperature During Cooldown, Low Range
For all RCP Combinations EXCEPT P-32A and B running, C and D off (31 EFPY)



NOTE

1) Maximum step change of 15°F is allowed when removing all RCPs from operation with DH system operating. (T-cold used for plotting cooldown rate prior to stopping all RCPs minus DH Cooler Outlet Temperature)
2) The appropriate ramp drop is allowed both before and after the step change.
3) Decay heat system max pressure includes 10 psig instrument accuracy.
4) RCS Temperature Maximum Cooldown Rate

≥ 280°F	100°F/hr
280°F to 150°F	50°F/hr
< 150°F	25°F/hr

5) If RCPs are operating between 180°F to 150°F, maximum RCS cooldown rate is limited to 2°F/hr.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0560 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** Modified **Originator:** J.Cork

TUOI: A1LP-RO-DH **Objective:** 6 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 005 **System Title:** Residual Heat Removal System

Description: Ability to manuall operate and/or monitor in the control room: Heat exchanger bypass flow control.

K/A Number: A4.02 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 31 **SRO:**

Given the following plant conditions:

---Plant is in Mode 5.

---RCS temperature 180 degrees F.

---RCS pressure 30 psig.

Identify the correct valve alignment for starting a Decay Heat pump:

A. Cooler outlet valve 50% open, cooler bypass valve shut, injection block valve 100% open.

B. Cooler outlet valve shut, cooler bypass valve 75% open, injection block valve shut.

C. Cooler outlet valve shut, cooler bypass valve 75% open, injection block valve 100% open.

D. Cooler outlet valve 50% open, cooler bypass valve shut, injection block valve 50% open.

Answer:

B. Cooler outlet valve shut, cooler bypass valve 75% open, injection block valve shut.

Notes:

"B" contains the correct alignment per 1104.004 for starting a DH pump when RCS pressure < 150 psig.

References:

1104.004, Chg. 071-05-0

History:

Direct from regular exam bank QID #1771.
Originally selected as Direct, later Modified as replacement question in 2005 RO exam.

10.0 Shifting Operating Decay Heat Loops

CAUTION

- Severe water hammer can damage DH system pipe if DH pump is started with high flow when RCS is drained.
- Decay Heat system design pressure can be exceeded if DH pump is dead-headed with RC pressure >150 psig (DH pump suction pressure >200 psig).
- Decay Heat pump discharge relief setpoint is 445 ±13.35 psig. Discharge pressure should be maintained <400 psig to prevent challenging the relief.

{4.3.6}

NOTE

Rules for starting LPI (Decay Heat) pump:

- If RC pressure is <150 psig, always start the Decay Heat pump with the Decay Heat Cooler Outlet and Cooler Bypass valves closed or the LPI Block Valve closed.
- If RC pressure is >150 psig and RCS is not drained, the Decay Heat pump must be started with the Decay Heat Cooler Outlet valve closed, Decay Heat Cooler Bypass valve ~ 75% open and the LPI Block Valve Open.
- **If RCS is drained and RC pressure is >150 psig (as a result of extended loss of DHR for example), the RCS must be depressurized to <150 psig, then the Decay Heat pump started with the Decay Heat Cooler Outlet and Cooler Bypass valves closed or LPI Block valve closed.**

10.1 IF placing Low Pressure Injection (Decay Heat) pump P-34A into service,
THEN perform the following:

10.1.1 Establish a Plant Computer Programmable Annunciator alarm for P-34A discharge pressure of 400 psig.

CAUTION

- With RC pressure >75 psig and DH suction piping aligned to the RCS, exceeding design pressure of suction piping from BWST will occur if Decay Heat P-34A Suction from BWST (CV-1436) and Decay Heat P-34A Suction from RCS (CV-1434) are open at the same time.
- The interlock to prevent the suction from RCS valves (CV-1434 and CV-1435) opening unless suction from BWST valves (CV-1436 and CV-1437) are closed does not prevent CV-1436 and CV-1437 from opening after CV-1434 and CV-1435 are open.

10.1.2 Close Decay Heat P-34A Suction from BWST (CV-1436).

10.1.3 Open Decay Heat P-34A Suction from RCS (CV-1434).

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0197 **Rev:** 1 **Rev Date:** 4/5/05 **Source:** Modified **Originator:** R. Walters

TUOI: A1LP-RO-TS **Objective:** 4 **Point Value:** 1

Section: 3.2 **Type:** Reactor Coolant System Inventory Control

System Number: 006 **System Title:** Emergency Core Cooling System

Description: Knowledge of the effect of a loss or malfunction of the following will have on the ECCS: Core flood tanks (accumulators).

K/A Number: K6.02 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	A

Question: RO: 32 SRO:

The plant is operating at 100% power. The Core Flood system is properly aligned with the following CFT parameters:

T-2A level	- 12.9 feet	T-2B level	- 13.1 feet
T-2A pressure	- 632 psig	T-2B pressure	- 615 psig

The Core Flood system parameters are unacceptable because:

A. Levels may preclude having sufficient N2 volume to fully inject the CFT contents into the vessel.

B. High N2 pressure could cause RCS inventory to be lost out of the break in the event of a LOCA.

C. Levels may not be sufficient to reflood the vessel following a LOCA.

D. N2 pressure may not be sufficient to fully inject the CFT contents into the vessel during a LOCA.

Answer:

B. High N2 pressure could cause RCS inventory to be lost out of the break in the event of a LOCA.

Notes:

"B" is correct, A CFT pressure is greater than the limit of 620 psig.
"A" is incorrect. This would be the case if either level were out of spec high, however, the levels are within specs.
"C" is incorrect. Both levels are within the Tech Spec limits.
"D" is correct. N2 pressure for are greater than the 572 psig TS limit from 1104.001.

References:

1104.001, Chg. 033-07-0
T.S. Bases for 3.5.1

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for use in 2002 RO/SRO exam.
Modified for use in 2005 RO exam, and again as a replacement question.

5.0 LIMITS AND PRECAUTIONS

5.1 With the exception of ASME Section XI testing and when the CFT is depressurized, during plant cooldown the CFT discharge valves (below) shall be closed and the circuit breakers for the motor operators opened before depressurizing the RCS below 600 psig or ≤262°F (TS 3.4.11).

- Core Flood Tank T-2A Outlet Isol (CV-2415), breaker (B-5661)
- Core Flood Tank T-2B Outlet Isol (CV-2419), breaker (B-5545)

5.2 When the RCS is depressurized with DH in service and either CFT is pressurized, the potential exists for personnel harm, equipment damage, and loss of DH flow in the event the core flood tank isolation valves are opened.

5.3 CFT NDTT limits are as follows:

5.3.1 CFT metal temperature shall not be less than 65°F anytime the tank is pressurized above 140 psig.

5.3.2 N_2 temperature at the point of injection shall be ≥65°F anytime the tank is pressurized above 140 psig.

5.3.3 If N_2 temperature at the point of injection is ≥ 100°F below the tank metal temperature, CFT pressure shall be less than 25 psig.

NOTE

TS limits where the CFT will be declared inoperable for CFT level (volume) and pressure are listed below and contain the maximum allowance for instrument uncertainty.

5.4 Prior to reaching RCS pressure of 800 psig, each CFT shall be at least as follows (TS 3.5.1):

5.4.1 CFT level

- Admin limit - 12.7 to 13.3 feet
- Tech Spec limit - 12.6 to 13.4 feet

5.4.2 CFT Pressure

- Admin Limit - 580 to 620 PSIG
- Tech Spec Limit - 572 to 628 PSIG

5.4.3 CFT boron concentration ≥ 2270 ppmB (Tech Spec limit).

APPLICABLE SAFETY ANALYSES

The CFTs are credited in both the large and small break LOCA analyses at full power (Ref. 1). The CFT line break analysis credits only one CFT, since the tank with the broken line is assumed to empty out the break. These Design Basis Accident (DBA) analyses establish the acceptance limits for the CFTs. In performing the LOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. In addition, a loss of offsite power is considered to ensure worst case conditions are postulated. In the early stages of a limiting large break LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS. This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver flow until the diesel generators (DGs) start and go through their timed loading sequence.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. No credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 35 seconds after the RCS pressure decreases to the ESAS actuation pressure. No operator action is assumed during the blowdown stage of a large break LOCA (Ref. 1).

The small break LOCA analysis also assumes a time delay after ESAS actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria for the ECCS established by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature of 2200°F;

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

APPLICABLE SAFETY ANALYSES (continued)

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to cover the core to the 3/4 point even assuming no liquid remains in the reactor vessel following a LOCA (Ref. 1). The downcomer then remains flooded until the HPI and LPI systems start to deliver flow for limiting large break LOCAs.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection and ensure the ability of the CFTs to fully discharge. The limiting safety analysis volume requirement is 1040 ± 70 ft^3. This volume corresponds to CFT levels of ≥ 11.95 ft and ≤ 14.00 ft. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The minimum nitrogen cover pressure requirement of 560 psig ensures that the contained gas volume will generate discharge flow rates during injection that satisfy the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The maximum nitrogen cover pressure limit of 640 psig will affect the amount and timing of CFT inventory discharged while the RCS depressurizes. Limiting the maximum pressure will therefore limit the CFT inventory lost through the break and assure that the CFT inventory injected into the RCS at the proper time is bounded by that predicted by the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum CFT level, the reactor will remain adequately shutdown in the cold condition following mixing of the CFT and Reactor Coolant System (RCS) water volumes. This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

In MODE 1, the CFTs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 2 and MODE 3 with RCS pressure > 800 psig, the CFTs satisfy Criterion 4 of 10 CFR 50.36.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0071 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** Modified **Originator:** JCork

TUOI: A1LP-RO-MU **Objective:** 4.C **Point Value:** 1

Section: 3.2 **Type:** Reactor Coolant System Inventory Control

System Number: 006 **System Title:** Emergency Core Cooling System (ECCS)

Description: Knowledge of symtom based EOP mitigation strategies.

K/A Number: 2.4.6 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	Ap

Question: **RO:** 33 **SRO:**

Given:

- HPI pump P-36B out of service
- LOCA in progress
- RCS pressure 1425 psig and dropping slowly
- 4160v bus A4 de-energized
- HPI flows: A = 190 gpm
  B = 100 gpm
  C = 105 gpm
  D = 90 gpm

Which of the following actions is required?

A. Throttle all to within 20 gpm of each other

B. Throttle "A" flow to within 20 gpm of "C"

C. Throttle "A" flow to within 20 gpm of "D"

D. Throttle "A flow to within 20 gpm of "B"

Answer:

B. Throttle "A" flow to within 20 gpm of "C"

Notes:

"B" is correct, if only one HPI pump available and RCS press > 600 psig, then highest flow (A) must be throttled to within 20 gpm of the next highest flow. The other answers are combinations of the wrong loops or throttling all.

References:

1202.012, Chg. 004-03-0, RT-2

History:

Used in 1998 RO/SRO exam
Modified QID 2840
Selected for 2005 RO exam, later modified as a replacement.

2. (Continued).

F. **IF** only **one** train of HPI is available
AND
RCS press is >600 psig,
THEN throttle the HPI Block valve with the highest flow to within 20 gpm of the next highest flow.

G. **IF** leakage into the RB is indicated, **THEN** maximize RB cooling:

1) Verify all four RB Cooling Fans running (VSF1A - D).

2) Open RB Cooling Coils Service Water Inlet and Outlet valves (CV-3812, 3813, 3814 and 3815).

3) Unlatch key-locked Chiller Bypass Dampers (SV-7410, 7412, 7411, 7413).

H. Verify the following sample valves closed

- Pressurizer Steam Space (CV-1814)
- Pressurizer Water Space (CV-1816)
- Hot Leg Sample (SV-1840)

I. **Unless** directed otherwise, verify the following High Point Vents closed.

A Loop	B Loop	Reactor Vessel	Pressurizer
SV-1081	SV-1091	SV-1071	SV-1077
SV-1082	SV-1092	SV-1072	SV-1079
SV-1083	SV-1093	SV-1073	
SV-1084	SV-1094	SV-1074	

(2. CONTINUED ON NEXT PAGE)

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0561 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** New **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 Pressurizer.

K/A Number: K5.02 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** Ap

Question: **RO:** 34 **SRO:**

A plant startup is in progress with a steam bubble being drawn in the Pressurizer.
- Initial Quench Tank pressure is 3 psig.
- RCS pressure 65 psig.
- Pressurizer temperature 312°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. 030-04-0

History:

New for 2005 RO exam, later modified for replacement.

NOTE

- If the RCS has been opened, the pressurizer sample lines will likely contain air and other gases which require purging before representative samples of the pressurizer will be obtained.
- Venting the sample piping at locations such as E-30 Outlet Drain Isol (SS-1012B) in the primary sample room has been shown to greatly enhance sample line purging. This venting should be done prior to 200F in order to be able to use standard red rubber hoses to flush to the floor drains.

7.2.4 Notify Chemistry to begin flushing pressurizer sample lines.

WARNING

Opening the ERV causes a localized steam release at the pilot valve vent. This is a radiation and safety hazard.

7.2.5 Verify no one is at the vicinity of the ERV.

CAUTION

- Pressurizer heatup rate limit is ≤ 100°F/hr.
- DH system maximum pressure is ≤ 250 psig.

{4.3.4} **7.2.6 When RC pressure reaches ~ 70 psig, open following to vent nitrogen from PZR to Quench Tank (T-42):**

A. ERV Isolation Valve (CV-1000)
B. ERV (PSV-1000, HS-1014 on C04)

7.2.7 Re-close ERV before RCS pressure falls to 30 psig.

7.2.8 <u>If</u> this is a heatup following a refueling outage, <u>or</u> the ERV has not been exercised during cold shutdown within the last 92 days, perform Supplement I to document Tech Spec and ASME section XI required exercising of the ERV.

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 Allow pressure to rise again to near 70 psig
<u>and</u>
repeat steps 7.2.5 through 7.2.7 as necessary until bubble forms.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0294 **Rev:** 0 **Rev Date:** 9/4/99 **Source:** Direct **Originator:** D Walls

TUOI: A1LP-RO-RCS **Objective:** 13 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 007 **System Title:** Pressurizer Relief Tank/Quench Tank System (PRTS)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining Quench Tank water level within limits.

K/A Number: A1.01 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 35 **SRO:**

Why is a minimum water level maintained in the Quench Tank?

a. Ensure adequate NPSH for the transfer pump.

b. Provide sufficient cooling-quench water during pressurizer operations.

c. Maintain a reference water level for level indication.

d. Maintain a loop seal on the relief lines.

Answer:

b. Provide sufficient cooling-quench water during pressurizer operations.

Notes:

"a" is incorrect because the NPSH requirement is lower than the minimum water level.
"c" is incorrect because the transmitter reference leg is external to the tank.
"d" is incorrect because the relief valves relieve to a header that is aligned in the bottom of the tank.

References:

1103.005, Chg. 030-04-0

History:

Used in 1999 exam
Direct from ExamBank, QID #2180 used in class exam
Selected for 2005 RO exam

5.6 Both Pressurizer Code Safety Valves (PSV-1001, PSV-1002) shall be operable when the reactor is critical (TS 3.4.10).

5.7 When the RCS is >262°F, at least one pressurizer code safety valve shall be operable, except for hydrostatic tests (TS 3.4.10).

5.8 Maximum allowable temperature difference between pressurizer water and T-hot shall not exceed 410°F except for startup.

5.9 Maximum allowable pressurizer level at any time the reactor is critical is 290 inches.

5.10 Minimum water level during normal power (> 15%) operation is 190 inches.

5.11 The pressurizer spray valve shall not be opened with nitrogen in the pressurizer.

5.12 During heatup 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, unless required for plant safety.

5.13 Maintain reactor coolant system pressure between 30 and 50 psig while purging nitrogen from the pressurizer to maintain a solid condition in the reactor coolant system hot legs.

5.14 With a steam bubble in the pressurizer, Quench Tank level shall be maintained > 4000 gallons and < 8300 gallons to provide sufficient quench-cooling volume for pressurizer transients.

5.15 Do not allow Quench Tank pressure to approach bursting point of its Rupture Diaphragm (PSE-1051), 100 psig.

5.16 Opening the ERV causes a localized steam release at the pilot vent valve. This is a radiation and safety hazard.

5.17 Adding N_2 to the Quench Tank via RB N_2 Supply Penetration Isol (N_2-47) circumvents the double-valve isolation design of the penetration. When RCS is ≤ 200°F, Containment Closure Control, Attachment G of Decay Heat Removal and LTOP System Control (1015.002) applies.

5.18 To maintain the RCS in a non water solid condition, TS 3.4.11 Bases limits pressurizer level with RCS temperature <272°F and the reactor vessel head in place to the following:

- ≤105" at RCS pressures >100 psig
- ≤150" at RCS pressures ≤100 psig

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0562 **Rev:** 0 **Rev Date:** 4/5/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-MSSS **Objective:** 9 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 008 **System Title:** Component Cooling Water System (CCWS)

Description: Knowledge of the bus power supplies to the following: CCW pump, including emergency backup.

K/A Number: K2.02 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 36 **SRO:**

Which of the following identifies the correct power supplies to the Intermediate Cooling Water Pumps (P-33A, P-33B, P-33C)?

A. P33A and P33B are powered from B-12 while P33C is powered from B-22.

B. P33A is powered from B-12 while P33B and P33C are powered from B-22.

C. P33A, P33B and P33C are powered from B-12, B-22 and B32 respectively.

D. P33A, P33B and P33C are powered from B-11, B-12 and B-13 respectively.

Answer:

B. P33A is powered from B-12 while P33B and P33C are powered from B-22.

Notes:

"B" lists the correct power supplies, the other choices do not.

References:

STM 1-43, Rev. 8

History:

Direct from regular exam bank QID#4674
Selected for 2005 RO exam

To allow P-33B to supply system flow for either loop, a suction and discharge cross-connect valve is provided for P-33B. System operating pressure differences will cause leakage through the closed discharge cross-connect. This would cause the surge tank with the lowest pressure to overflow and drain to the floor drain if the ICW Surge Tank Cross Connect Islolation Valve (ICW-165) was not opened. Additional information on P-33B suction and discharge cross connect valves will be covered in the following section.

Obj. 5

2.4 ICW PUMPS

(Refer to Figure 43.01)

The three ICW pumps, P-33A/B/C are used to supply cooling water to components cooled by the ICW system. Net positive suction head is provided by the ICW surge tanks, discussed in the previous section. The ICW pumps are located in the Main Chiller room, elevation 354' of the Turbine Bldg.

Obj. 6

Each ICW pump is a single-stage, centrifugal pump rated for 2500 gpm with a discharge head of 125 feet. The ICW pumps rotate at 1750 rpm driven by a 100 HP motor rotating at 1775 rpm. Power to the ICW pumps is provided from non-vital MCC's, B12 and B22. ICW pumps are controlled using handswitches located on panel C09. Associated MCC, breaker and HS for each pump are listed below:

Pump	Handswitch	MCC	Breaker
P-33A	HS-2230	MCC-B12	B-1264
P-33B	HS-2231	MCC-B22	B-2214
P-33C	HS-2232	MCC-B22	B-2264

The three pumps are connected in parallel with two normally in operation and one in standby. Suction and discharge cross-connect lines allow for any two of the three ICW pumps to be in service supplying cooling water flow to the two ICW loops. Both the suction and discharge cross-connect lines utilize air-operated butterfly valves to provide system / loop separation. Each air-operated butterfly valve is provided air from the Instrument Air system through a pressure control valve and solenoid valve. The solenoid valves powered from Y02 breaker 8.

The suction and discharge cross-connect valves are controlled by handswitches located on panel C09 in the control room. Suction and discharge cross-connect valves are interlocked with the pump handswitches and discharge pressure switches for re-alignment on standby pump auto-start.

The two suction cross-connect isolations are 12-inch butterfly valves while the discharge cross-connect line utilizes two 10-inch butterfly valves. P-33B suction supply line and pump discharge line tie into the system between the two suction and discharge cross-connect isolation valves.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0072 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-AOP **Objective:** 4.2 **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:
PORV failures.

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 4.1 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 37 **SRO:**

Given:

- RCS pressure 1950 psig and decreasing
- "RELIEF VALVE OPEN", K09-A1, in alarm
- ERV indicates closed
- Acoustic monitor indicates ERV is leaking

You immediately close ERV Isolation valve, CV-1000.
RCS pressure continues to drop with all PZR heaters ON.

What should your next action be?

a. Begin plant runback

b. Cycle the ERV

c. Initiate full HPI

d. Trip the reactor

Answer:

d. Trip the reactor

Notes:

RCS pressure continuing to decrease following isolation of the ERV requires a reactor trip per AOP actions, therefore (d) is the only correct response.

References:

1203.015 Chg. 011-00-0

History:

Developed for the 1998 RO/SRO Exam.
Selected for 2005 RO exam.

SECTION 1 -- ELECTROMATIC RELIEF VALVE (PSV-1000) SYSTEM FAILURE OR LEAK

1.0 SYMPTOMS

One or more of the following:

- SPDS alarm on T1025 at 200°F: ERV PSV-1000 OUTLET TEMP
- Quench Tank (T-42) temperature, level, or pressure rising.
- Rise on acoustic Relief Valve Monitor (VYI-1000A).
- Annunciator alarm RELIEF VALVE OPEN (K09-A1).
- ERV indicates open on C04.
- Pressurizer ERV Isolation Valve (CV-1000) inoperable.
- ERV (PSV-1000) inoperable.
- Failure of both ERV acoustic monitors.

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

NOTE

CV-1000 torque switch can be overridden in the OPEN or CLOSE direction by holding the hand switch in the respective position.

3.1 Close Pressurizer ERV Isolation Valve (CV-1000).

3.2 If ERV leakage with CV-1000 closed exceeds capability to maintain RC pressure, trip reactor and refer to Emergency Operating Procedure series (1202.XXX).

3.3 If closing CV-1000 stops leak, perform the following:

3.3.1 Continue power operations with ERV isolated.

3.3.2 Notify Ops Manager.

3.3.3 Log in station log and on plant status board.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0306 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-RPS **Objective:** 10 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 012 **System Title:** Reactor Protection System

Description: Knowledge of the operational implications of the following concepts as they apply to the RPS: DNB.

K/A Number: K5.01 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2.5
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 38 **SRO:**

Which of the following conditions would result in the Reactor Protection System initiating a reactor trip designed to protect the fuel clad from DNB?

a. Ejected rod accident during startup

b. Loss of both Main Feedwater Pumps at 100 % power

c. Boron dilution accident while operating at 100% power

d. Reactor Coolant Pump trip at 95% power

Answer:

d. Reactor Coolant Pump trip at 95% power.

Notes:

"a" is incorrect because it will result in a high power trip. The high power trip is intended to prevent damage from fast reactivity excursions.
"b" is incorrect because it results in a high pressure trip and protects RCS piping.
"c" is incorrect because reactivity addition rate is slow enough that ICS should compensate for the reactivity addition.

References:

Technical Specifications 3.3, Bases

History:

Developed for 1999 exam.
Selected for 2005 exam

APPLICABLE SAFETY ANALYSES (continued)

the accident analysis calculations for small break loss of coolant accidents (LOCAs). The Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because either the trip will actuate prior to degraded conditions being reached or the equipment response will be conservative.

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to the system parameters of pressure and temperature exceeding the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the reactor outlet temperature expressed in degrees Fahrenheit within the range specified by the Reactor Outlet High Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure Allowable Value is selected to initiate a trip prior to temperature and pressure exceeding the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the unit safety analysis. Therefore, the Allowable Value does not account for errors induced by a harsh RB environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. Even in the case where this trip is a backup for other RPS trips for LOCA or MSLB, it is assumed to occur before degraded building conditions have an appreciable effect on RB High Pressure trip components. Therefore, determination of the Allowable Value does not account for errors induced by a harsh environment.

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

APPLICABLE SAFETY ANALYSES (continued)

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing may be insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline temperature SLs.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides protection for the loss of a pump or pumps which would result in both pumps in a single steam generator loop being tripped.

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least one RCP is operating in each loop. RCP status is monitored by power transducers associated with each pump. These relays indicate a loss of an RCP on underpower. The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. <u>Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE</u>

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the reactor core SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the limiting loss of flow transient which is the loss of two RCPs from four pump operation. The imbalance portion of the trip is credited for steady state protection only.

The power to flow ratio of the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system is operating with two or three pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs prior to core power, axial power peaking, and reactor coolant flow conditions reaching DNB or fuel centerline temperature limits. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0307 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J. Cork

TUOI: A1LP-RO-RPS **Objective:** 11 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 012 **System Title:** Reactor Protection System

Description: Ability to manually operate and/or monitor in the control room: Bistable trips, reset and test switches.

K/A Number: A4.04 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 39 **SRO:**

Given:

- Plant is at 100% power.
- "B" Reactor Protection System channel is inoperable due to NI-6 failed high.
- A surveillance test on "D" Reactor Protection System channel is in progress.

What is the Reactor Protection System bistable trip logic under these conditions?

a. One out-of-two

b. One out-of-three

c. Two out-of-two

d. Two out-of-four

Answer:

a. One out-of-two

Notes:

The reactor protection system logic is normally 2 of 4 required to trip the reactor. The "D" RPS channel is placed in bypass to perform the surveillance test, therefore it cannot trip. Only one channel can be in bypass at a time. The "B" RPS channel will be tripped since NI-6 has failed high. The logic will then be 1 of 2, therefore "b", "c", and "d" are incorrect.

References:

STM-63 Rev 6

History:

Developed for 1999 exam.
Selected for use in 2005 RO exam as replacement.

1.6 System Logic

(Refer to Figure 63.01 and 63.08)

The trip logic system consists of four functionally identical reactor protection channels, each terminating in a channel trip relay within the reactor trip module. Reactor trip modules are the communications interface between the four channels and also between the reactor protection system and the Control Rod Drive system. The entire system functions as a "de-energize to trip" system. RPS logic requires any two of four channels to trip before associated CRD breakers will open.

1.6.1 Channel Trip Logic

When a variable monitored by RPS to cause a reactor trip exceeds acceptable values, contacts in the circuitry monitoring that value are opened by bistables or contact buffers. The contacts for the variables in each channel are connected in a series circuit, forming a contact string that terminates in a channel trip relay. When all variables are within acceptable values, all of the contacts in the contact string are closed maintaining the trip relay energized. If the circuit is broken by any one of the contacts opening, the channel trip relay is de-energized resulting in a channel trip. The channel trip relay for each channel is located in the reactor trip module (RT).

The RT relays are given the same letter designation as the protection channel in which they are physically located. Thus RPS channel "A" RT module trip relay is designated as KA, channel "B" KB, channel "C" KC and channel "D" KD. Four coincidence (output) logic relays are also located in each reactor trip module. The logic relays (one from each channel) are de-energized when contacts operated by the associated channel trip relay opens. The logic relays are designated based on cabinet location and associated trip relay. The logic relays located in "A" RPS cabinet are all designated as (1), channel "B" (2), channel "C" (3) and channel "D" (4). Example: the logic relay for "D" channel located in "B" reactor trip module would be designated as KD2.

1.6.2 Coincidence Logic

Coincidence Logic determines which RPS channel and which CRD breaker will trip. Each coincidence logic relay operates two contacts in the supply to the UV coils for the associated CRD breakers. A total of eight contacts are provided in each coincidence circuit, arranged in two groups of four such that any two logic relays being de-energized (four contacts open) will result in interruption of current to the UV coil and subsequent trip of the breaker(s). Refer to figure provided below or Figure 63.20.

One group of four contacts will cause the breaker(s) associated with that reactor trip module to open if called for by the combination of channel A or channel B and channel C or channel D. To complete the coincidence logic and assure a trip when called for by two of the four reactor protection system channels, a second set of four contacts is provided. This set of contacts will produce a trip when called for by channels A or C and channels B or D. Between the two sets of contacts, all possible combinations of two channels are covered and the demand by any two channels for a trip will result in a reactor trip.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0266 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Modified **Originator:** D. Slusher

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

Section: 3.2 **Type:** Reactor Coolant System Inventory Control

System Number: 013 **System Title:** Engineered Safety Features Actuation System(ESFAS)

Description: Knowledge of the physical connections and/or cause-effect relationships between the ESFAS and the following systems: ECCS.

K/A Number: K1.06 **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:** 4.4 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 40 **SRO:**

Given:

- All controls are in automatic
- RCS pressure 135 psig, slowly dropping
- Reactor Building pressure is 28 psia
- "A" and "B" OTSG levels at 415 inches

Which pair of pumps should be pumping fluid as designed (not recircing)?

A. HPI pumps and LPI pumps

B. RB spray pumps and LPI pumps

C. RB spray pumps and EFW pumps

D. HPI pumps and EFW pumps

Answer:

A. HPI pumps and LPI pumps

Notes:

"A" is correct because RCS pressure exceeds the shutoff head of the LPI pumps.
"B" and "C" are incorrect because RB pressure has not reached the RB spray actuation setpoint of 30 psia.
"D" is incorrect because OTSG level is above the level for reflux boiling.

References:

1105.003, Chg. 011-02-0
STM1-65, Rev. 3

History:

Used in 1999 exam.
Direct from ExamBank, QID# 1780 used in class exam
Selected for 2005 RO exam, modified later as a replacement.

3.5 Summary of ESAS trips and functions:

Channel No.	Action	Trip Condition	Trip Point
1 & 2	HP injection & diverse containment isolation	Low RCS pressure High RB pressure	<1590 psig >4 psig (18.7 psia)
3 & 4	LP injection, diverse containment isolation & EFW	Low RCS pressure High RB pressure	<1590 psig >4 psig (18.7 psia)
5 & 6	RB isolation & RB cooling	High RB pressure	>4 psig (18.7 psia)
7 & 8	RB spray	High RB pressure	>30 psig (44.7 psia)
9 & 10	RB Spray NAOH Addition	High RB pressure	>30 psig (44.7 psia)

4.0 REFERENCES

4.1 REFERENCES USED IN PROCEDURE PREPARATION

4.1.1 Unit 1 Technical Specifications

4.1.2 ESAS Technical Manual, Volume 3 (M-1-29)

4.1.3 Engineered Safeguards Actuation System (STM-1-65)

4.1.4 Safety Analysis Report, sections 6, 7.1.1, 7.1.3, 7.4.7 and 14.2.2.5

4.1.5 Integrated ES System Test (1305.006)

4.1.6 CR-1-96-0359 ESAS Setpoint Change

4.1.7 DCP 95-5010-D101 SW Valve Replacement

4.1.8 CR-1-02-0410 ESAS Digital Channel Logic Test Module

4.1.9 CR-ANO-1-2003-0373 Digital Channel failure during testing

4.2 REFERENCES USED IN CONJUNCTION WITH THIS PROCEDURE

4.2.1 Plant Startup (1102.002)

4.2.2 Plant Shutdown and Cooldown (1102.010)

4.2.3 Decay Heat Removal Operating Procedure (1104.004)

4.2.4 Emergency Feedwater Initiation and Control (1105.005)

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0564 **Rev:** 0 **Rev Date:** 4/7/05 **Source:** New **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:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 41 **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)?

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. Declare P-35B Spray pump inoperable and refer to T.S. 3.6.5.

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 plant is in cold shutdown and RB spray pumps are not required per T.S. Regardless, there is not enough information to determine operability.

References:

1203.012J, Chg. 035-00-0
1305.006, Chg. 020-04-0

History:

New for 2005 RO exam

Location: C18

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 is open, perform the following:

   A. _If_ RB Press recorder on C18 indicates >30 psig, verify that ES CH 7 is actuated and observe that RB Spray Pump (P-35A) has started.

   B. Determine cause of P-35B failure.

2. Check RB Spray P-35B Flow gauge on C16. _If_ flow is low, perform the following:

   A. _If_ RB Press recorder on C18 indicates >30 psig, verify that ES CH 7 is actuated and observe 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. To clear alarm, perform either of the following:

   - Clear and reset ES CH 8.
   - Close breaker A-404 and raise RB spray flow to >1050 gpm.

2.0 PROBABLE CAUSES

1. Pump P-35B failure to auto start

3.0 REFERENCES

Schematic Diagram Annunciator K11 (E-461, sheets 1-3)

8.6 Align Service Water Pump (P-4B) to operate from bus A4 as follows:

8.6.1 Verify P-4B is off. ______

8.6.2 Verify or place P-4B BUS SELECT MOD CONTROL in MOD TO BUS A4 position. ______

8.6.3 Verify MOD to bus A4 is closed (may require manual operation of MOD). ______

8.6.4 Start P-4B to verify operation. ______

8.6.5 Stop P-4B AND leave hand switch in NORMAL-AFTER-STOP. ______

8.7 Align RB Spray Pump (P-35B) for start in recirc mode as follows:

8.7.1 Verify Test & Recirc Header not in use. ______

8.7.2 Verify following valves closed:

- P-35A Disch to DH Recirc & Test Line (BS-2A) ______
- P-34A Disch to DH Recirc & Test Line (DH-8A) ______
- P-34B Disch to DH Recirc & Test Line (DH-8B) ______
- SF System Disch to DH Recirc. & Test Line (SF-38) ______

8.7.3 Position the following valves as indicated:

A. Unlock and open P-35B Disch to DH Recirc & Test Line (BS-2B). ______

B. Open RB Spray Sys Disch to DH Recirc & Test Line (BS-3). ______

C. Open Throttle Valve Bypass around DH-10 (DH-9). ______

8.7.4 Verify RB Spray Block Valve (CV-2400) is closed. Hold hand switch in CLOSE position for at least 10 seconds after valve indicates closed. ______

A. At MCC B61 open breaker for CV-2400, (B-6171). ______

NOTE

A Partial Clearance will be required for RB Spray NaOH Addition, T-10 Outlet (CV-1617), to allow performance of the next step.

8.7.5 Close breaker for CV-1617 (B-6193). ______

8.7.6 If HPI is not required for RCS inventory makeup per Decay Heat Removal and LTOP System Control (1015.002)
AND
RCS level is less than BWST level, unlock and close BWST Supply to MU Pump P-36C Suction (BW-2). ______

8.7.7 If BW-2 is left open, verify HPI Pump Recirc Block Valve (CV-1301) closed to prevent gravity fill of MU Tank (T-4) from BWST. ______

8.7.8 Open BWST Outlet (CV-1408). ______

8.7.9 Open P-35B Vent (ABV-12B) as necessary to fully vent P-35B casing. ______

8.7.10 When venting is complete, close ABV-12B. ______

8.7.11 Vent P-35B discharge piping as follows:

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. ______

8.7.12 Start RB Spray Pump (P-35B). ______

8.7.13 Adjust DH-9 to obtain ~1500 GPM spray flow. ______

A. If necessary to obtain desired flow, throttle open DH Test & Recirc Isol (DH-10). ______

8.7.14 Stop P-35B AND leave hand switch in NORMAL-AFTER-STOP. ______

8.7.15 Close CV-1408. ______

8.8 Align DH Pump (P-34B) for start in DH mode as follows:

8.8.1 Close P-34B Suction From BWST (CV-1437). ______

8.8.2 Open P-34B Suction From RCS (CV-1435). ______

8.8.3 Unlock and close B DH Cooler SW Outlet Isol (SW-22B). ______

8.8.4 Verify LPI Block Valve (CV-1400) closed. ______

8.8.5 Verify Decay Heat Cooler Outlet (CV-1429) open. ______

8.8.6 Verify Decay Heat Cooler Bypass (CV-1432) closed. ______

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

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:** No **Taxonomy:** K

Question: **RO:** 42 **SRO:**

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 7 & 8, 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 1 & 2, 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 return 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. 6

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO 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-D) has a single blade, butterfly damper (CV-7470-7473) at the discharge of the fan that opens when the fan starts. These 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 (VALIDATION)

QID: 0444 **Rev:** 0 **Rev Date:** 5/1/2002 **Source:** Repeat **Originator:** S.Pullin

TUOI: A1LP-WCO-RBS **Objective:** 1 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray

Description: Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Iodine scavenging via the CSS.

K/A Number: K4.06 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** K

Question: RO: 43 SRO:

In post accident conditions, the RB Spray System (both trains) will provide what percentage of the required RB cooling and iodine removal?

a. 100% Cooling/100% Iodine

b. 200% Cooling/100% Iodine

c. 100% Cooling/200% Iodine

d. 200% Cooling/200% Iodine

Answer:

c. 100% Cooling/200% Iodine

Notes:

References:

STM 1-08, Rev. 8

History:

Direct from regular exambank QID 1012.
Selected for use in 2002 RO exam.
Repeated for use in 2005 RO exam

on site electric power system operation (assuming off site power is not available) the system safety function can be accomplished, assuming a single failure. Single failure criteria is simply: no single failure will cause or prevent a protection system from fulfilling its design function.

The Reactor Building Spray and Emergency Cooling Units provide cooling to the RB atmosphere following a LOCA event. One Emergency Cooling Unit consists of two RB cooling fans (VSF-1A-1D) and their associated SW cooling coils (VCC-2A-D). VSF-1A & VSF-1B and their associated cooling coils (VCC-2A & VCC-2B) makeup one unit while VSF-1C & VSF-1D and their associated cooling coils (VCC-2C & VCC-2D) makeup the other. Both emergency cooling units have a combined heat removal capability of 100%. The two separate trains of RB Spray together are capable of providing 100% of the design cooling required from the system.

The following combinations are allowed by Tech Spec 3.3 for RB heat removal.

- Both Trains of RB Spray.
- Both Emergency Cooling Units.
- One RB Spray and one Emergency Cooling Unit.

2.8 RB Spray System Design

The Reactor Building Spray system has two major functions. The first of these functions is to reduce post accident containment temperature and pressure to nearly atmospheric. By reducing the pressure and temperature, the driving force for leakage will be reduced and thereby stay below 10CFR100 limits at the site boundary during a Design Based Accident (DBA). 10CFR100 limits are based on a person located at the site boundary for a two-hour period immediately following a LOCA event. The limits are 300 Rem to the Thyroid & 25 Rem Whole body.

The accident that would result in the highest containment pressure is a $5ft^2$ hot leg rupture. The DBA for the Reactor is a 14-ft2 rupture of the hot leg. The $5ft^2$ hotleg rupture results in a higher containment pressure due to the RCS will blowdown to the RB atmosphere for a longer period of time.

The second function of the Reactor Building Spray system is to remove iodine from the containment atmosphere after a loss of coolant accident that raises containment pressure to 30 psig. Iodine released from damaged fuel to the containment atmosphere during a LOCA could be released to the outside environment if it was not removed, especially during the elevated containment pressure of a post accident condition. Iodine released to the atmosphere has a tendency to be absorbed in the Thyroid. Reducing the iodine level greater reduces the thyroid dose to plant personnel and the general public should a containment breech occur.

Each individual Reactor Building Spray train will provide 100% of the design iodine removal capability. Iodine has a strong affinity for water and is "stripped" from the containment atmosphere when the Reactor Building Spray system is operating. To enhance the

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0314 **Rev:** 0 **Rev Date:** 9/5/99 **Source:** Direct **Originator:** J Cork

TUOI: A1LP-RO-EFIC **Objective:** 31 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 039 **System Title:** Main and Reheat Steam System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main Steam pressure.

K/A Number: A1.06 **CFR Reference:** 41.5 / 45.5

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 44 **SRO:**

Given:

- A loss of offsite power
- No failures exist other than those which caused the loss of offsite power condition
- EDG's supplying vital buses

Ten (10) minutes into this event at what pressure will the OTSG's be controlled?

a. 895 psig

b. 995 psig

c. 1020 psig

d. 1050 psig

Answer:

c. 1020 psig

Notes:

Following a loss of offsite power, condenser vacuum will be non-existent and the Atmospheric Dump Valves will be used to control OTSG pressure at the nominal setpoint of 1020 psig, therefore "c" is correct.
"a" is incorrect, this is the normal setpoint for the Turbine Bypass Valves.
"b" is incorrect, this is the setpoint for the Turbine Bypass Valves with the 100 psig bias applied (to limit cooldown) following a Reactor Trip.
"d" is incorrect, this is the lowest setpoint for a Main Steam Safety Valve.

References:

1105.005, Chg. 027-02-0

History:

Developed for 1999 exam.
Selected for 2005 RO exam.

6.0 SETPOINTS

6.1 Initiation Setpoints

- EFW low level initiate ~ 13.5". (delayed 2 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.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0565 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** Modified **Originator:** J.Cork

TUOI: A1LP-RO-ICS **Objective:** 13 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal

System Number: 059 **System Title:** Main Feedwater (MFW) System

Description: Knowledge of the physical and/or cause/effect relationships between the MFW and the following systems: ICS.

K/A Number: K1.07 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** An

Question: **RO:** 45 **SRO:**

Given:

- 100% power
- ICS in full automatic

The CBOR places the ICS Delta T-Cold Hand Auto Station meter selection switch in "POS" (position). The meter reads 46%.

What does this mean in terms of ICS control of main feed water?

A. The average of feedwater loop A and feedwater loop B demand is 46%.

B. Feedwater loop B demand is greater than feedwater loop A demand.

C. The feedwater loop A demand is being boosted by a 4 °F Delta T-Cold error.

D. Feedwater loop A demand is greater than feedwater loop B demand.

Answer:

B. Feedwater loop B demand is greater than feedwater loop A demand.

Notes:

A reading <50% indicates that loop B demand is > loop A demand, therefore (B) is the correct response. (A) is incorrect because the meter does not indicate average demand, (D) is an opposite response, (C) applies to looking at the MV reading (for which it would still be incorrect).

References:

STM 1-64, Rev. 9

History:

Developed for the 1998 RO/SRO Exam.
Selected for use in 2002 RO/SRO exam.
QID #63 used on 2004 RO/SRO Exam.
Modified for 2005 RO exam.

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 meter. This provides the ΔTc setpoint for automatic operation. The setpoint may be varied from 0% to 100% which corresponds to -10°F to +10°F. The normal value is 50% (0°F).

When position is selected on this station, the ΔTc correction signal is indicated on the meter. If the meter indicates 50%, the correction signal is zero (loop "A" multiplier set at .5) and loop demand signals are equal. If the meter indication is above 50%, then loop "A" demand is > loop "B" demand. The opposite is true if the indication is < 50%.

When measured variable is selected on this station, the difference between the actual ΔTc and the ΔTc setpoint (ΔTc error) is indicated on the meter. ΔTc = "A" Loop Tc - "B" Loop Tc. The meter scale is ±10°F. Positive reading means that "A" loop is hotter. A bumpless transfer from hand to auto may take place when the ΔTc error equals zero (50% on meter). If the ΔTc does not equal zero, adjustment to zero may be accomplished by adjusting the manual output of the station or by changing the ΔTc setpoint.

If both loop demand stations are placed in hand, this station rejects to hand and can not be placed in auto.

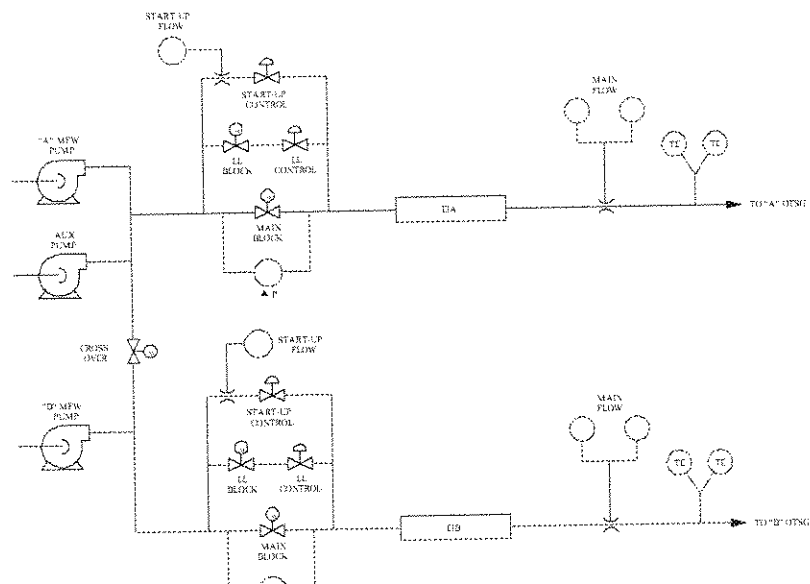


FIGURE 64.24: FEEDWATER FLUID SYSTEM

2.6.3 Feedwater Flow Control

The method of flow control used by the feedwater system is dependent upon the plant power level. (refer to figure 64.24) At low power feedwater flow is controlled by the startup and low load control valves with the main feedwater block valve shut and the feedwater pumps operating to maintain 70 psid across the feed valves. The valves are sequenced into operation so that the startup valve opens first followed by the low load control valves then the main FW block valves. As plant load is increased, feedwater flow control will be shifted from the valves to the pumps. This is

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

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: Turbine driven feed pump.

K/A Number: A3.04 **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:** No **Taxonomy:** K

Question: **RO:** 46 **SRO:**

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. 8

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO exam

2.3.1 Instrumentation

Feed pump suction pressures, temperature and flow instrumentation is provided for indication, alarms and control features associated with operation of P-1A (B). MFP suction temperature is sent to the plant computer for indication only. Suction temperature for P-1A is provided by TE-2840 and TE-2832 provides P-1B indication.

2.3.1.1 Suction / Recirc Flow Indication

(Refer to Figure 19.04)

Feedwater suction flow is used to control the MFP recirc control valve to maintain a minimum flow through the pump. Low flow through the MFP should only occur when placing the MFP in service. Flow element FE-2843 and flow transmitter FT-2833 provide a flow signal to the recirc valve controller. Recirc flow or suction flow can be read on the recirc valve controller. Recirc flow control valve CV-2874 provides additional controlling functions besides minimum flow through the MFP. Recirc valve operation, controller and controlling functions will be covered in the discharge header discussion section.

Low suction flow through the MFP or low recirc flow alarm condition is provided by flow switch FS-2843. Annunciator K07-E7 "A MFP Flow Lo" will alarm when suction flow indicates less than 1600 gpm. P-1B flow instrumentation and recirc flow CV information is provided in the following table.

Flow Element FE-2832	Flow Transmitter FT-2834
Flow Switch FS-2832 (K07-E8)	Recirc Valve CV-2876

2.3.1.2 Suction Pressure Indications

(Refer to Figure 19.04)

Feed pump suction pressure is used for indication, controlling functions, alarms and MFP trip signal. Suction pressure is sent to the plant and SPDS computers by PT-2842. MFP suction pressure local indication, PI-2842 is provided on rack 21 near E-1 FW heaters.

MFP gland seal cooling system uses suction pressure and cooling water supply pressure to maintain desired differential pressure between gland seal cooling and suction pressure. Additional information will be provided in the MFP section covering the gland seal cooling system.

MFP suction pressure trip signals and alarms are provided by PS-2841 & PS-2842. PS-2842 provides annunciator alarms for Lo and Lo-Lo suction pressure conditions. K07-C7 "A MFP Suct Press Lo" will alarm when PS-2842 indicates suction pressure less than 280 psig. PS-2842 will reset when pressure indicates greater than ~ 320 psig.

K07-B7 "A MFP Suct Press Lo-Lo" will alarm when PS-2842 indicates suction pressure ≤ 230 psig for greater than 5 seconds. Reset pressure for Lo-Lo- alarm is ~ 255 psig. Lo-Lo pressure signal from PS-2842 provides one of the two suction pressure trip signals required to trip the MFP.

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.

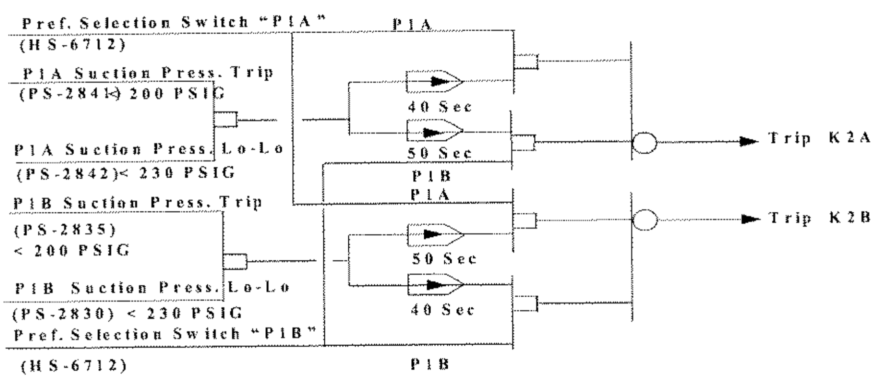
2.3.1.3 Suction Pressure Trip Logic

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 (VALIDATION)

QID: 0566 **Rev:** 0 **Rev Date:** 5/1/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-EOP **Objective:** **Point Value:** 1

Section: 3.4 **Type:** Reactor Heat Removal

System Number: 061 **System Title:** Auxiliary / Emergency Feedwater System

Description: Knowledge of bus power supplies to the following: AFW electric drive pump.

K/A Number: K2.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	C

Question: RO: 47 SRO:

Given:
- A Degraded Power condition exists on both units.
- #1 EDG did not start.
- P-7A tripped and will not reset.

Why does the EOP direct cross-tieing of the A3 and A4 buses?

A. To run two HPI pumps for HPI cooling per RT-4.

B. To have atmospheric dump control on both SGs.

C. To start EFW Pump P-7B.

D. To restore Instrument Air for ADV control.

Answer:

C. To start EFW Pump P-7B.

Notes:

"C" is correct since P-7B is powered from A3 and power must be restored to regain EFW.
"A" is incorrect since one HPI pump can be the source of HPI cooling.
"B" is incorrect, A3 must have power to restore power to "A" ADV isolation, this is not the purpose of this step.
"D" is incorrect, ADVs are placed in manual to conserve air and have air reservoirs for this purpose. Recovering Inst. Air is a concern but this is not the key purpose for cross-tieing in this condition.

References:

1202.007, Chg. 006-01-0

History:

New for 2005 RO exam.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

This section is for correcting overheating.

53. IF any of the following criteria is met before overheating is corrected, THEN GO TO step 55.

- ERV opens in AUTO
- RCS press ≥2450 psig
- RCS press approaches NDTT Limit (Figure 3)

54. Re-verify proper EFW actuation and control (RT 5).

54. **IF** EFW fails to actuate, **THEN** perform the following:

A. Place EFW CNTRL valves in HAND **AND** close:

SG A		SG B
CV-2645	P7A	CV-2647
CV-2646	P7B	CV-2648

B. Place EFW Pump Turbine (K3) Steam Admission valves in MANUAL **AND** close (CV-2613 and 2663).

C. Place EFW Pump P7B in PULL-TO-LOCK.

D. **IF** A3 is de-energized **AND** P7A is unavailable, **THEN** restore power to P7B as follows:

1) Energize A3 using ES Electrical System Operation (1107.002).

a) **IF** another DG **OR** Off-site power becomes available, **THEN** restore buses to normal using 1107.002.

2) Start P7B **AND GO TO step 62.**

E. Dispatch an operator to restore EFW using Emergency Feedwater Pump Operation (1106.006).

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0140 **Rev:** 0 **Rev Date:** 05/13/93 **Source:** Direct **Originator:** J. Haynes

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** A. C. Electrical Distribution

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the AC Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect on plant of de-energizing a bus.

K/A Number: A2.04 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 48 **SRO:**

Initial conditions:

- 100% power
- P-36C is the operating makeup pump
- ICW pumps P-33A and P-33C in service

What RCP support system would be most affected by a loss of bus A4?

a. Seal Injection

b. Motor Cooling

c. Seal Bleedoff

d. Oil Lift Pressure

Answer:

a. Seal Injection

Notes:

(a) is correct. Loss of A4 results in a loss of the running HPI pump.
(b) is incorrect. P-33 will remain in service which provides motor cooling.
(c) & (d) are incorrect. Seal bleedoff is not affected by the loss of A2 nor is RCP lift oil pressure.

References:

1203.026, Chg. 009-05-0

History:

Taken from Exam Bank QID # 3714
Used in A. Morris 98 RO Re-exam
Selected for use in 2005 RO exam.

SECTION 1 -- LOSS OF HPI PUMP

1.0 SYMPTOMS

1.1 Annunciator alarms:

- HPI PUMP TRIP (K10-A6)
- RCP SEAL INJ FLOW LO (K08-A7)
- MU TANK LEVEL HI/LO (K10-B7)
- MU TANK PRESS HI/LO (K10-B8)

1.2 Loss of or erratic makeup flow and seal injection flow.

1.3 Loss of or erratic makeup (HPI) pump discharge header pressure.

2.0 IMMEDIATE ACTION

2.1 None.

3.0 FOLLOW-UP ACTIONS

NOTE

Indications of loss of HPI suction are:
- Erratic flow, and
- Erratic discharge pressure, and
- Control valves stable

3.1 IF HPI pump has lost suction,
THEN stop the HPI pump.

3.2 Isolate letdown by performing either of the following:

- Close Letdown Coolers Outlet (CV-1221),
- Close Letdown Cooler Outlets (RCS) (CV-1214 and 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.

3.3 Verify RC pump seals are being cooled by ICW.

3.3.1 IF ICW to RCP seals is NOT available,
THEN perform Reactor Coolant Pump and Motor Emergency (1203.031), "Simultaneous Loss of Seal Injection and Seal Cooling Flow" section.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0616 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-ELECD **Objective:** 17 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** AC Electrical Distribution

Description: Ability to manually operate and/or monitor in the control room: all breakers (including available switchyard).

K/A Number: A4.01 **CFR Reference:** 41.7 / 45.5 / to 45.8

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 49 **SRO:**

The CBOT notices on C10 that all of the 161 KV ring bus breakers have opened.

Which of the following will be de-energized as a result of the above indications?

A. Startup #1 Transformer

B. Startup #2 Transformer

C. Auto transformer

D. Startup #3 Transformer

Answer:

B. Startup #2 Transformer

Notes:

Answer "B" is correct, the 161KV ring bus supplies the #2 SU transformer.
Answers "A", "C", and "D" are the remaining but incorrect transformers supplied by offsite power.

References:

STM 1-32, Rev. 27, p. 5

History:

New for 2005 RO exam, replacement question.

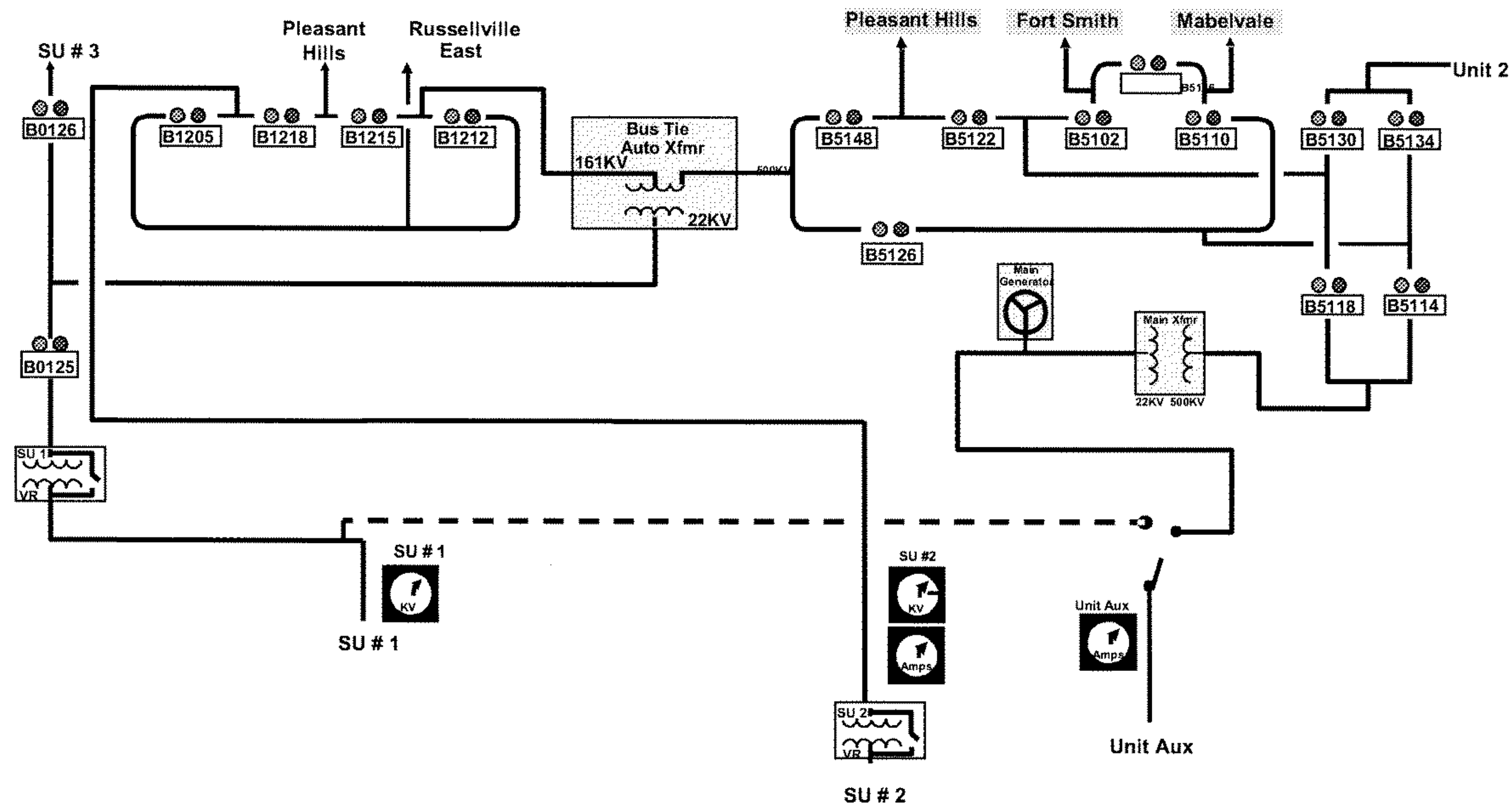


Figure 32.3A, Switchyard Breaker Layout (C-10)

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0617 **Rev:** 0 **Rev Date:** 8/10/05 **Source:** New **Originator:** J.Cork

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:**	No	**Taxonomy:**	C

Question: **RO:** 50 **SRO:**

Given:

-The plant is at 55% power.
- RCP P-32A is stopped for maintenance.

What would be the effect if DC panel RA-1 were d-energized?

A. ICS runback to 45% power.

B. RPS Channel "A" would trip.

C. P-32A could not be re-started.

D. Reactor would trip.

Answer:

D. Reactor would trip.

Notes:

Answer "D" is correct, this would de-energize RPC "A" and "C" power monitor.
Answer "A" is incorrect, although the ICS does runback on loss of one pump in each loop to 45%, de-energizing RA1 is equivalent to a loss of one pump in each loop. If power is >50%, a Rx trip occurs.
Answer "B" is incorrect, this RPS channel is powered from RS-1, not RA-1.
Answer "C" is incorrect, this would be the case if power were lost to D11.

References:

1107.004, Chg. 012-12-0

History:

New for 2005 RO exam, replacement question.

ATTACHMENT J

125V DC Panel RA1 Breakers (Continued)			
BREAKER NUMBER	DESCRIPTION	CONSEQUENCES OF OPENING	REQUIRED ACTION
RA1-12	SV-7479, RB Leak Detector Isol Ret. to RB, SV-7457, H_2 Analyzer Inlet, SV-7467, H_2 Analyzer Outlet CV-1065, Quench Tank Condensate, CV-6202, Chilled Water Isolation, CV-1667, N_2 to RB (C18), SV-1440, PASS RB Sump Return	SV-7479, SV-7457 SV-7467 and SV-1440 fail closed. Loss of redundant power supply for CV-1667, Quench Tank Condensate Isolation (CV-1065) and Chilled Water Isolation (CV-6202). Lead Penetration Room Ventilation System ES function disabled.	None
RA1-16	RCP A and C Power Monitor Control C15	Reactor Trip if >50% power or in 3 pump ops. (RPS receives a "not running" signal from A and C RC pumps.)	Verify reactor power <50% and not in 3 RCP operations.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0568 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-EDG **Objective:** 2 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators

Description: Ability to monitor automatic operation of the ED/G system, including: Number of starts available with an air compressor.

K/A Number: A3.04 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 51 **SRO:**

Given:

- #1 EDG has one Air Start Compressor and it's associated Air Receiver Tanks tagged out.
- The remaining Air Start Compressor on #1 EDG trips while running.
- The Air Receiver Tanks' pressure is 176 psig.

What is the maximum number of start attempts assured with the above #1 EDG conditions?

A. One

B. Three

C. Five

D. Seven

Answer:

C. Five

Notes:

"C" is correct per the TS bases, the others are incorrect choices.

References:

TS SR 3.8.3.3 and Bases

History:

New for 2005 RO exam.

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 $\geq$ 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 $\geq$ 175 psig.	31 days
SR 3.8.3.4	Check for and remove accumulated water from each fuel oil storage tank.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.2 (continued)

Following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-81 (Ref. 4) are met for new fuel oil when tested in accordance with ASTM D975-81 (Ref. 4), except that the analysis for sulfur may be performed in accordance with ASTM D1552-90 (Ref. 4) or ASTM D2622-87 (Ref. 4). These additional analyses are required by Specification 5.5.13, "Diesel Fuel Oil Testing Program," to be performed within 31 days following sampling and addition. This 31 days is intended to assure: 1) that the sample taken is not more than 31 days old at the time of adding the fuel oil to the storage tank, and 2) that the results of a new fuel oil sample (sample obtained prior to addition but not more than 31 days prior to) are obtained within 31 days after addition. For circumstances where multiple fuel oil additions are made within a short period of time, the samples taken for each batch added to the storage tank can be composited for a single follow-up analysis. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-88, Method A (Ref. 4). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each tank is considered and tested separately.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.3

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0089 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-AOP **Objective:** 2 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring (PRM) System

Description: Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint.

K/A Number: K4.01 **CFR Reference:** CFR: 41.7

Tier: 2 **RO Imp:** K4.01 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 4.3 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 52 **SRO:**

Given:

- Treated Waste Monitor Tank, T16A, release in progress
- "PROC MONITOR RADIATION HIGH", K10-B2, in alarm
- Liquid Radwaste Process Monitor, RI-4642, in alarm

What should your Immediate Action be?

a. Verify no flow on Discharge to Flume, FI-4642

b. Trip the running Radwaste Transfer pump, P-53A/B

c. Close Liquid Waste to Flume valve, CV-4642

d. Reset RI-4642 to verify alarm is valid

Answer:

a. Verify no flow on Discharge to Flume, FI-4642

Notes:

Answer (a) is the only immediate action per the AOP. (b) and (d) are follow up actions, (c) is an automatic action.

References:

1203.007, Rev. 8

History:

Developed for 1998 RO/SRO Exam.
Selected for 2005 RO exam.

PROC./WORK PLAN NO.	PROCEDURE/WORK PLAN TITLE:	PAGE: 1 of 2
1203.007	LIQUID WASTE DISCHARGE LINE HIGH RADIATION ALARM	REV: 8 CHANGE:

1.0 SYMPTOMS

1.1 PROC MONITOR RADIATION HI (K10-B2) annunciated.

1.2 Liquid waste Discharge Flow to Flume (FI-4642) drops off, (C19).

1.3 Alarm on Liquid Radwaste Process Monitor (RI-4642), (C-25, Bay 2).

2.0 IMMEDIATE ACTIONS

None

3.0 FOLLOW-UP ACTIONS

3.1 Verify no release in progress at Disch Flow to Flume (FI-4642) on C19.

3.2 Stop the running radwaste transfer pump (P-53A/B, P-47A/B, or P-45).

3.3 Close Liquid Waste to Flume Valve (CV-4642) if not already closed.

3.4 If CV-4642 can not be closed, close applicable upstream manual valve:

3.4.1 Treated Waste Discharge to Circ. Water Flume (CZ-58).
3.4.2 Filtered Waste Monitoring Tank Disch. to CW Flume (DZ-25).
3.4.3 Laundry Drain Pump P-45 Discharge to Flume (LZ-5).

3.5 Verify proper valve lineup for applicable release path:

3.5.1 Dirty Liq. Waste & Drain Processing (1104.014), Attachment D.
3.5.2 Laundry Waste Processing (1104.015), Attachment B.
3.5.3 Clean Waste System Operation (1104.020), Attachment B.

3.6 Verify proper setting and operation of Liquid Radwaste Process Monitor (RE-4642) per the release permit.

NOTE

1. An unplanned release is defined as the unintended discharge of a volume of liquid or airborne radioactivity to the environment.

2. An expanded "Definition of Unplanned Releases" is contained in several Chemistry procedures, including Analysis of Liquid Waste (1604.017), Attachment 4.

3. Unplanned releases require a condition report and are reportable per ODCM App. 1, L3.2.1.B

3.7 If condition was caused by a problem in step 3.5 or a problem other than an unplanned release, re-establish release.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0211 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** R. Fuller

TUOI: A1LP-AO-SW **Objective:** 6 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal

System Number: 076 **System Title:** Service Water System

Description: Knowledge of bus power supplies to the following: Service water.

K/A Number: K2.01 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.7	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 53 **SRO:**

Given:

- The unit is operating at 100% power.
- Service water pumps P-4A and P-4B are in service.
- A loss of P-4A occurs.

What is required of the operator to restore service water to the required Technical Specifications configuration?

a. Swap P-4B MOD (A6) to the A3 supply and start P-4C.

b. Start P-4C, stop P-4B, swap P-4B MOD (A6) to the A4 supply and start P-4B.

c. Stop P-4B, swap P-4B MOD (A6) to the A3 supply and start P-4B and P-4C.

d. Start P-4C, stop P-4B, swap P-4B MOD (A6) to the A3 supply and start P-4B.

Answer:

d. Start P-4C, stop P-4B, swap P-4B MOD (A6) to the A3 supply and start P-4B.

Notes:

(a.) is incorrect. MOD should not be swapped under load.
(b.) is incorrect. P-4B must be powered from A3 to comply with Tech Specs since P-4C is powered from A4.
(c.) is incorrect. Stopping P-4B first would cause a loss of all service water.
(d.) is correct.

References:

STM 1-42, Rev. 9

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO 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.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0227 **Rev:** 1 **Rev Date:** 5/2/05 **Source:** Modified **Originator:** J. Cork

TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

Section: 3.8 **Type:** Plant Services Systems

System Number: 078 **System Title:** Instrument Air System (IAS)

Description: Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems.

K/A Number: K4.02 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 54 **SRO:**

Instrument Air pressure has dropped to 58 psig.
Field operators can not find an Inst. Air leak on Unit One.

Which of the following is the appropriate response for the given plant conditions to restore or conserve Instrument Air pressure?

A. Verify Service Air to Instrument Air cross-connect automatically opens.

B. Close Unit 1 to Unit 2 Instrument Air cross-connect.

c. Trip Reactor, actuate EFW and MSLI on both SGs.

d. If ICW available, isolate Seal Injection by closing CV-1206.

Answer:

b. Close Unit 1 to Unit 2 Instrument Air cross-connect.

Notes:

Per 1203.024, the U1 to U2 cross connect should be closed first, so [b] is correct.
[a] is incorrect, this does not occur until pressure is at 50 psig.
[c] is incorrect, this would not be done until pressure was less than 35 psig.
[d] is incorrect, this would not be done unless necessary to maintain PZR level <290".

References:

1203.024, Chg. 010-08-0

History:

Developed for 1998 RO/SRO Exam QID 0102.
Modified for A. Morris 98 RO Re-exam
Modified for 2005 RO exam.

SECTION 1 -- LOW INSTRUMENT AIR PRESSURE (≤75 PSIG)

DISCUSSION

Low IA pressure can be caused by numerous conditions. This section assumes a gradual loss of air pressure with no major malfunction of air operated equipment. Expeditious action is required to minimize the impact on air operated systems and components. For additional discussion, see Attachment D.

1.0 SYMPTOMS

1.1 IA header pressure dropping.

1.2 INST AIR HEADER PRESS LO (K12-B3) alarm

1.3 INST AIR COMPRESSOR TROUBLE (K12-C3) alarm

1.4 M-1/F-8 ΔP (K21-5) alarm

2.0 IMMEDIATE ACTION

None.

3.0 FOLLOW-UP ACTIONS

NOTE
IA HDR Pressure can be monitored using PMS point P5409.

3.1 Verify standby Instrument Air Compressor(s) (C-28A/B, C-2A/B) running.

3.2 Dispatch an operator to determine specific compressor, air dryer, and filter condition.

3.3 IF IA is supplying respirable air,
THEN inform RP of loss of IA pressure, and that workers must back out of work in progress and isolate the IA supply.

3.4 IF low IA header pressure is due to loss of IA on Unit 2,
AND IA is crossconnected,
THEN perform the following:

3.4.1 IF IA header pressure drops below 60 psig,
THEN direct Unit 2 control room operators to terminate crossconnection.

3.4.2 **GO TO** step 3.7.

CAUTION
If either Unit 1 or 2 has a significant IA leak, crossconnecting Unit 1 and 2 IA systems can result in low IA pressure in both units.

3.5 Direct Unit 2 control room operators to crossconnect Unit 2 IA system to Unit 1 IA system.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0569 **Rev:** 0 **Rev Date:** 5/2/05 **Source:** Modified **Originator:** J.Cork

TUOI: A1LP-RO-AOP **Objective:** 2 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment System

Description: Knowledge of the effect that a loss or malfunction of the Containment System will have on the following: Loss of containment integrity under normal operations.

K/A Number: K3.02 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 55 **SRO:**

With the plant in Mode 2, what type of occurrence would make it necessary to use AOP 1203.005, Loss of Reactor Building Integrity?

A. Failure to perform a LLRT on personnel hatch within 5 days after opening.

B. Failure to perform a LLRT on emergency hatch within 7 days after opening.

C. "A" LPI RB sump suction is inoperable, closed, and de-energized.

D. The Ops Manager's permission is given to open both doors on personnel hatch to allow for ventilation motor entry.

Answer:

B. Failure to perform a LLRT on emergency hatch within 7 days after opening.

Notes:

Only "D" meets the conditions specified in 1203.005.

References:

1203.005, Chg. 011-01-0

History:

Direct from regular exam bank, QID #737.
#12 on 2002 RO exam, #13 on 2004 RO exam.
Modified due to NRC comments during exam review for 2005 RO exam, replacement question.

ENTRY CONDITIONS

NOTE

- Reactor building shall be operable whenever the unit is in Modes 1, 2, 3, or 4 (Tech Spec 3.6.1).
- This procedure assumes that reactor building integrity is required.
- During periods of frequent containment entries, the 7 day criterion for performance of airlock LLRT can be extended by Engineering Programs to 30 days.

- Failure to meet the requirements of SR 3.6.1.1.
  - Reactor building is not operable due to visual examination results.
  - Air leakage is in excess of Reactor Building Leakage Rate Testing Program acceptance criteria.
- Either reactor building air lock not operable per TS 3.6.2.
- Reactor building equipment hatch not closed and sealed (TS 3.6.1 Bases).
- Inoperable reactor building isolation manual or automatic power operated valve per TS 3.6.3.
- Failure to perform local leak rate testing (LLRT) on personnel lock or emergency lock within 7 days after opening (SR 3.6.2.1).

ES-401 PWR Examination Outline Form ES-401-2

Plant systems – Tier 2/Group 2 (RO / SRO)

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	#
001 Control Rod Drive									X			A3.07 Ability to monitor automatic operation of the CRDS, including: Boration/dilution.	4.1	56
002 Reactor Coolant								X				A2.01 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.	4.3	57
011 Pressurizer Level Control												Not selected.		
014 Rod Position Indication												Not selected.		
015 Nuclear Instrumentation			X									K3.01 Knowledge of the effect that a loss or malfunction of the NIS will have on the following: RPS.	3.9	58
016 Non-nuclear Instrumentation												Not selected.		
017 In-core Temperature Monitor											X	2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.	3.4	59
027 Containment Iodine Removal												Not selected.		
028 Hydrogen Recombiner and Purge Control												Not selected.		
029 Containment Purge												Not selected.		
033 Spent Fuel Pool Cooling	X											K1.02 Knowledge of the physical connections and/or cause-effect relationships between the Spent Fuel Cooling System and the following systems: RHRS.	2.5	60
034 Fuel Handling Equipment												Not selected.		
035 Steam Generator				X								K4.01 Knowledge of S/GS design feature(s) and/or interlock(s) which provide for the following: S/G level control.	3.6	61
041 Steam Dump/Turbine Bypass Control					X							K5.07 Knowledge of the operational implications of the following concepts as they apply to the SDS: Reactivity feedback effects.	3.1	62
045 Main Turbine Generator							X					A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G System controls including: Expected response of secondary plant parameters following a T/G trip.	3.3	63
055 Condenser Air Removal												Not selected.		
056 Condensate												Not selected.		
068 Liquid Radwaste												Not selected.		
071 Waste Gas Disposal										X		A4.25 Ability to manually operate and/or monitor in the control room: Setting of process radiation monitor alarms, automatic functions, and adjustment of setpoints.	3.2	64
072 Area Radiation Monitoring												Not selected.		
075 Circulating Water												Not selected.		
079 Station Air												Not selected.		
086 Fire Protection						X						K6.04 Knowledge of the effect of a loss or malfunction of the following will have on the Fire Protection System: Fire, smoke, and heat detectors.	2.6	65
K/A Category Totals:	1	0	1	1	1	1	1	1	1	1	1	Group Point Total:		10

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0262 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D Slusher
TUOI: ASLP-RO-RXT14 **Objective:** 20 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control
System Number: 001 **System Title:** Control Rod Drive System
Description: Ability to monitor automatic operation of the CRDS, including: Boration/dilution

K/A Number: A3.07 **CFR Reference:** 41.7/45.13

Tier:	2	**RO Imp:**	4.1	**RO Select:**	Yes	**Difficulty:**	3.5
Group:	2	**SRO Imp:**	3.7	**SRO Select:**	No	**Taxonomy:**	An

Question: **RO:** 56 **SRO:**

Given:

- The plant is at 100 %.
- CRDs are at the normal rod index.
- The EHC controller is in manual.
- RCS boron concentration is 812 ppm.
- 1 ppm RCS boration requires 7.8 gallons of Boric Acid.

The CBOR is making a RCS addition with no concentration change and adds 92 gallons of boric acid and 8 gallons of DI water.

What effect will this have, without any further operator action?

a. Rods go full out, Tave stays the same, power goes down.

b. Rods go in ~10%, Tave stays the same, power goes down.

c. Rods go in ~10%, Tave goes down, power stays the same.

d. Rods go full out, Tave goes down, power stays the same.

Answer:

d. Rods go full out, Tave goes down, power stays the same.

Notes:

(a) is incorrect because rods go full out and Tave decreases to maintain power the same
(b) & (c) are is incorrect because rods go out when boric acid is added.
(d) is correct. With reactor power is maintained by Tave going down and rods moving out.

References:

Generic Fundamentals Reactor Theory Chapter 14 Rev 2

History:

Developed for 1999 exam.
Selected for 2005 RO exam.

MAINTAINING ACCEPTABLE POWER DISTRIBUTION

To prevent large distortions in the axial power profile which could lead to peaking factors outside of design limits, there are strict technical specification (tech specs) requirements that limit the allowed axial flux difference (ΔI). Each facility has plant technical specification requirements in this regard. The following three sections present an overview of power distribution requirements for B&W, CE, and Westinghouse plants.

BABCOCK &WILCOX POWER DISTRIBUTION REQUIREMENT

To prevent large distortions in the axial power profile that could lead to peaking factors outside of design limits, there are strict technical specification requirements that limit the allowed axial power imbalance. Figure 8-20 compares four upper power range detector outputs to four lower range detector outputs, looking at the axial power imbalance.

$$\text{Axial Power Imbalance} = \frac{\phi_{top} - \phi_{bottom}}{\phi_{equiv.\ of\ 100\%\ power}}$$

Equation 8-9

The operating crew is required to operate the reactor within the "Permissible Region" shown under the curve. There are different curves derived based on of fuel burnup (core life). The curve in Figure 8-20 represents middle-of-life (MOL) conditions.

The allowed axial flux imbalance for a B&W reactor facility must be maintained according to tech specs as represented by this MOL axial power imbalance curve.

Normally, this is accomplished by maintaining control rods almost fully withdrawn from the core. Minimizing control rod movement during power operation minimizes axial flux shifts that can start xenon oscillations. However, some B&W facilities employ axial power shaping rods with 6 feet of active absorber positioned near the center of the core to also provide flux (power) shaping.

Because most nuclear facilities are "base loaded," the load dispatcher tries to ensure nuclear units on the grid do <u>not</u> change power.

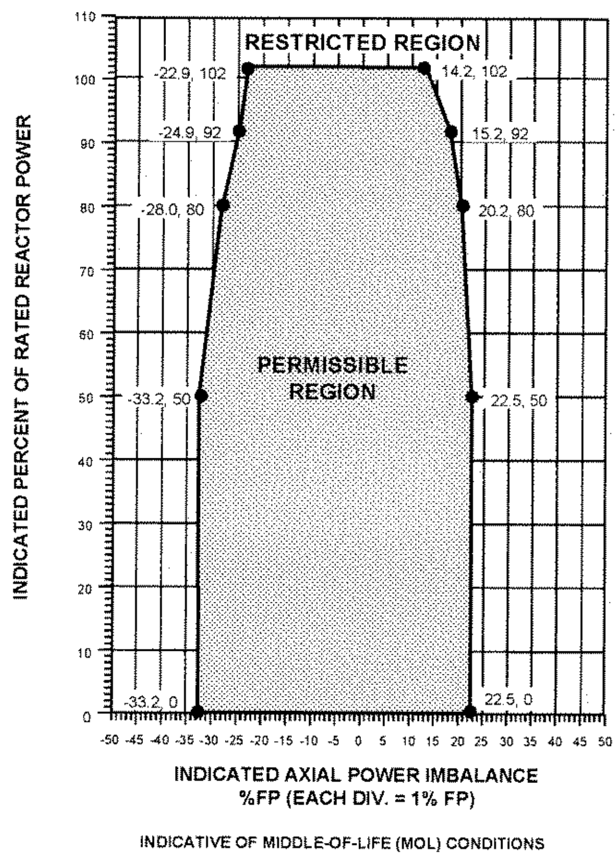


Figure 8-20 Indicated Axial Power Imbalance for a B&W Station

However, if load follow operations are warranted, then reactivity changes during load follow operations are made by changing the RCS boron concentration. This helps keep the control rods out of the core except where axial power shaping rods (APSRs) are used. As reactor power changes, however, the axial flux

tends to shift because of moderator temperature reactivity effects.

Changing reactor power upsets equilibrium conditions and induces transient xenon behavior. APSRs are most useful for handling these axial power imbalances. However, where no APSRs are used, boron concentration changes are necessary to move rods in or out.

The reactor operator must be aware of the direction of the resulting reactivity change and compensate with boron concentration adjustments to maintain RCS temperature. Rod movement in or out of the core for reactivity control is undesirable because it distorts the natural axial distribution of the flux and may initiate a xenon transient.

For negative power imbalances, boration is required to move control rod assemblies (CRAs) out. For positive power imbalances, dilution is required to move CRAs in.

Core imbalance must be monitored at a minimum of once every two hours during power operation above 40% of rated power. Except for physics tests, corrective measures (reduction of imbalance by APSR movement if allowed, and/or reduction in reactor power) must be taken to maintain operation within the envelope defined by Figure 8-20. If the imbalance is not within the envelope defined by this figure, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power must be reduced until imbalance limits are met according to Figure 8-20.

Control Rod and APSR Recommendations (B&W Procedural Examples)

Movement of the control rods and/or adjustment of the boron concentration control core power. Control rods should be maintained within the steady-state operating rod position range (Control Rod Group- 7 > 90% withdrawn) whenever steady load conditions exist for more than one-two hours or during load changes less than 0.5% per minute. This action will minimize imbalance changes. Greater rod insertion is used during all transients over 0.5% per minute and whenever power changes greater than 15% are expected.

The APSRs should be maintained within the position limits established by the plant operations director via the lead nuclear engineer (Control Rod Group 8 > 30% withdrawn) within eight hours after obtaining steady-state conditions. If these position limits cannot be maintained, the lead nuclear engineer is notified per station procedures.

COMBUSTION ENGINEERING POWER DISTRIBUTION REQUIREMENTS

To prevent large distortions in the axial power profiles that could lead to peaking factors outside CE plant design limits, Combustion Engineering power stations place strict requirements on the axial shape index in the plant technical specifications. The axial shape index (ASI) is calculated using the excore nuclear instruments and is an indication of core power distribution.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0604 **Rev:** 0 **Rev Date:** 6/30/05 **Source:** New **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:** No **Taxonomy:** C

Question: **RO:** 57 **SRO:**

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. 028-03-0

History:

New for 2005 RO exam, modified as a replacement question.

INSTRUCTIONS	CONTINGENCY ACTIONS
26. Check Pressurizer Level Control valve (CV-1235) maintains PZR level > 55".	26. Perform the following: A. **<u>IF</u>** CV-1235 fails to respond in AUTO, **<u>THEN</u>** operate CV-1235 in HAND to control PZR level 90 to 110". B. **<u>IF</u>** PZR level is < 55" with **<u>no</u>** indication of recovery, **<u>THEN</u>** isolate Letdown by closing either: Letdown Cooler Outlet (CV-1221), **<u>OR</u>** Letdown Cooler Outlets (CV-1214 and 1216). C. **<u>IF</u>** PZR level drops below 55", **<u>THEN</u>** verify Pressurizer Heaters off. D. **<u>IF</u>** PZR level drops below 30", **<u>THEN</u>** initiate HPI (RT 2).

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0571 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** Modified **Originator:** S.Pullin

TUOI: A1LP-RO-NI **Objective:** 7 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation System

Description: Knowledge of the effect that a loss or malfunction of the NIS will have on the following: RPS.

K/A Number: K3.01 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.3	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 58 **SRO:**

If at 90% power, NI channel 6 slowly fails to 103% power, which one of the following would occur?

A. ICS would insert control rods and runback feedwater.

B. RPS channel B would trip.

C. ICS would insert control rods and raise feedwater.

D. SASS would select NI channel 5.

Answer:

A. ICS would insert control rods and runback feedwater.

Notes:

"A" is correct, the highest of channels 5 & 6 input into ICS.
"B" is incorrect, the high flux trip setpoint is 104.9%.
"C" is incorrect, feedwater would be lowered, not raised.
"D" is incorrect, a slow failure will not cause SASS action.

References:

1105.004, Chg. 016-02-0

History:

Direct from regular exambank, QID#1793.
Modified for 2005 RO exam, as a replacement question.

6.13.3 If MFW Block Valve ΔP drops to ≤50 psid for >10 seconds, annunciator MFW A/B DELTA P LO (K07-D7) will alarm provided NONE of the following conditions exist:

- Main Block Valve Open.
- Aux FW Pump not running and both MFW Pumps tripped.
- Aux FW Pump not running and RFR Pump Runback in effect.

6.14 Neutron Error

6.14.1 Negative 1% neutron error causes rod withdrawal. Rod withdrawal terminates at -0.975% error.

6.14.2 Positive 1% neutron error causes rod insertion. Rod insertion terminates at +0.975% error.

6.14.3 Neutron error >±1% prevents transfer of Rod Controller (Diamond Panel) to automatic.

6.14.4 If neutron error is >±5%, a crosslimit will modify feedwater demand and cause a FEEDWATER IS REACTOR LIMITED (K07-C2) alarm.

6.15 If feedwater demand is more than 5% greater than feedwater flow, the reactor demand signal will be modified and cause a REACTOR IS FEEDWATER LIMITED (K07-C1) alarm.

6.16 Main Feedwater Block Valve (CV-2625, CV-2675) Interlocks

NOTE

50% loop demand is 50% of 5.56 MPPH which is ~46% POS indication.

6.16.1 If feedwater loop demand is >50% and Feedwater Pumps Disch Crosstie (CV-2827) is closed, MFW Block Valve opens automatically.

6.16.2 MFW Block closes automatically in fast speed upon reactor trip, even if ICS Control Override HS is in OVERRIDE.

NOTE

MFW Block Valve closure on MFP trip is blocked when power is >80%.

6.16.3 When power is <80%, unless ICS Control Override HS is in OVERRIDE, MFW Block closes automatically in fast speed upon trip of either Main Feedwater Pump (P-1A or P-1B).

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0572 **Rev:** 1 **Rev Date:** 8/1/05 **Source:** New **Originator:** J.Cork

TUOI: A1LP-RO-INCOR **Objective:** 12 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-core Temperature Monitor System

Description: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

K/A Number: 2.1.33 **CFR Reference:** 43.2 / 43.3 / 45.3

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 59 **SRO:**

Which of the following indications would cause entry into a Technical Specification?

A. Plant Monitoring System date and time is not updating.

B. Both Incore SPND Backup Recorders not printing or indicating point values.

C. Neither ICCMDS train's mimic display or local displays are updating.

D. Neither SPDS computer's date & time updating for greater than one hour.

Answer:

C. Neither ICCMDS train's mimic display or local displays are updating.

Notes:

"C" is the correct answer since the ICCMDS CETs are the qualified Reg. Guide 1.97 instruments and are required per 3.3.15.
"A" is incorrect, although PMS receives CET inputs, these are non-EQ CETS and thus not required.
"B" is incorrect, the Incore CET values are required by 3.3.15, not the SPND outputs.
"D" is incorrect, although this is a one hour reportable event and SPDS receives CET values, SPDS is not Tech Spec.

References:

T.S. 3.3.15, table 3.3.15-1

History:

New for 2005 RO exam, revised due to NRC comments, replacement question.

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 (VALIDATION)

QID: 0573 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-SFC **Objective:** 7 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 033 **System Title:** Spent Fuel Pool Cooling System (SFPCS)

Description: Knowledge of the physical connections and/or cause-effect relationships between the Spent Fuel Cooling System and the following systems: RHRS.

K/A Number: K1.02 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier:	2	**RO Imp:**	2.5	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	2.7	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 60 **SRO:**

What is the flowpath for the purification of the Fuel Transfer Canal and Reactor Vessel when using the Spent Fuel Cooling System?

A. Taking a suction on DHR, through the purification loop and then discharging to the Spent Fuel Pool.

B. Taking a suction on the Fuel Transfer Canal, through the purification loop and then discharging to the DHR system.

C. Taking a suction on the Spent Fuel Pool, through the purification loop and then discharging to the DHR system.

D. Taking a suction on DHR, through the purification loop and then discharging to the Fuel Transfer Canal.

Answer:

B. Taking a suction on the Fuel Transfer Canal, through the purification loop and then discharging to the DHR system.

Notes:

"B" is the correct alignment per 1104.006.
The rest are incorrect lineups per 1104.006.

References:

1104.006, Chg. 032-06-0

History:

Direct from regular exambank, QID#1977.
Selected for 2005 RO exam.

21.0 Fuel Transfer Canal and Reactor Vessel Purification **WITHOUT** Provisions for Emergency Boration

NOTE

During fuel handling, fuel transfer canal can be purified using the SF purification loop. The loop is aligned to take suction from the fuel transfer canal and discharge to the suction of the Decay Heat Pump (P-34A or P-34B). Both Spent Fuel Pool CLG Pumps (P-40A and P-40B) are available for SF Pool cooling as desired.

21.1 Initial Conditions

21.1.1 Reactor vessel head removed.

21.1.2 Fuel transfer canal filled to refueling level.

21.1.3 Fuel XFER Canal Fill/DRN ISOL Spectacle flange (M-307) rotated to OPEN position.

21.1.4 DH system in DH removal mode per Decay Heat Removal Operating Procedure (1104.004), "Decay Heat Removal During Refueling" section.

21.1.5 Spent Fuel Purification Loop secured.

CAUTION

Changes in system configuration or operational status with RCS flooded up and fuel transfer canal open to SFP could result in changes of RCS level and SFP level.

21.1.6 IF RCS is flooded up with fuel transfer canal open to SFP AND SFP level indication in the Control Room is NOT available,
THEN station an operator for local observation of SFP level (LI-2005) with direct communications to the Control Room.

NOTE

Aligning a flowpath through RB penetration P-19 is a breach of containment and requires tracking on Containment Closure Breach List 1015.002D.

21.2 Align P-66 for fuel transfer canal purification per Attachment J, "Fuel Transfer Canal Purification **WITHOUT** Emerg. Boration".

21.3 Make appropriate entry on Containment Closure Breach List 1015.002D.

21.4 Align SF purification per Attachment B, with the following exception:

21.4.1 SF to DH Suction Header (SF-20) open.

21.4.2 Close F-4A & B Discharge to SF Pool (SF-25).

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0201 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** J. Haynes

TUOI: A1LP-RO-EFIC **Objective:** 28 **Point Value:** 1

Section: 3.4 **Type:** RCS Heat Removal

System Number: 035 **System Title:** Steam Generator System

Description: Knowledge of S/GS design feature(s) and/or interlock(s) which provide for the following: S/G level control.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 61 **SRO:**

Startup Transformer #1 has locked out following a reactor trip from 100% power. Steam generator levels will be fed up to which of the following levels (assume no operator actions)?

a. 20-40 inches.

b. 100-150 inches.

c. 300-340 inches.

d. 370-410 inches.

Answer:

c. 300-340 inches.

Notes:

(a.) is incorrect. This is the Low Level EFIC control band.
(b.) is incorrect. EFIC does not control at this level in any conditions
(c.) is correct. A lockout of Startup Transformer #1 will result in the loss of all four RCPs. This causes EFIC to shift to the natural circulation setpoint of 312 inches.
(d.) is incorrect. This is the Reflux Boiling EFIC control band.

References:

1202.012, Chg. 004-03-0

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for use in 2005 RO exam as a replacement.

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)

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0574 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-STEAM **Objective:** 6 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core

System Number: 041 **System Title:** Steam Dump System and Turbine Bypass Control

Description: Knowledge of the operational implications of the following concepts as they apply to the SDS: Reactivity feedback effects.

K/A Number: K5.07 **CFR Reference:** 41.5 / 45.7

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 62 **SRO:**

Given:

- Normal startup with reactor power at 10%.
- Main Turbine is in Operator Auto.
- Both turbine bypass valves (TBVs) for the "A" SG fail open.

With NO operator action, which of the following will be the final reactor power?

A. ~14%

B. ~18%

C. ~22%

D. ~25%

Answer:

B. ~18%

Notes:

"B" is correct, 4 TBVs have a total capacity of 15%, so two TBVs is equal to 17.5% steam flow and reactor power will match steam flow.
The other distractors are simple answers for one, three, and four TBVs failed open.

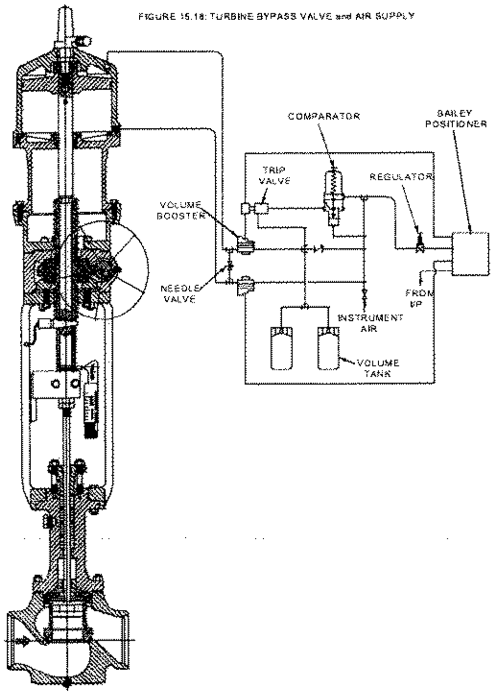
References:

STM 1-15, Rev. 7

History:

New for 2005 RO exam, modified as a replacement question.

FIGURE 15.18: TURBINE BYPASS VALVE and AIR SUPPLY



(Refer to Figure 15.18)

The TBV's are designed to provide a normal means of pressure control and heat removal from the steam generators during normal heatup, cooldown and prior to synchronizing the turbine generator to the grid. After the turbine generator is synchronized to the grid and load is greater than or equal to 15%, the turbine governor valves control header pressure. Turbine Bypass Valve capacity is 418,500 lbm/hr each. This allows all four of the TBV's to pass a total steam flow equivalent to 15% of main feedwater flow.

The inlet piping to the TBV's is a 6-inch line, while the outlet is a 10-inch line. The TBV's receive ICS signals in order to operate. The TBV's are electro-pneumatically-controlled globe valves. They have a stroke time of less than 3 seconds. These valves can also be manually operated.

If either of the main condensers' vacuum lowers below 21" Hg during normal operation, all four TBVs' will close. Both steam header ADVs are automatically unisolated and an alarm will sound--"Vacuum Low ADV Control Actuated".

Once condenser vacuum is restored above 24" Hg, the operator can reset the low vacuum signal block to the TBV's via 2 push-buttons (one for each condenser) on C02 (Low Vacuum Reset). Additionally, if it is desired to override the low vacuum signal block to the TBV's, a handswitch is provided on C02 (Turbine Bypass Low Vacuum Override). This handswitch allows the operator to select "Auto" for normal operation, or "Cond" to override the signal block. If this is done, excessive pressure can be developed in the condenser, which would be relieved by the condenser rupture discs.

TBV's have a sophisticated air supply system. IA supply is supplied to the valve positioner and provides supply air for valve operation. In addition to supplying motive air for operation it also maintains the two volume tanks pressurized to maintain the TBV closed on loss of instrument air.

The positioner is supplied with regulated air at 65 psig. With this supply of air, the positioner aligns to either the open volume booster or close volume booster through the trip valve. The instrument air supply through the volume booster then operates the valve. Operation of the TBV volume booster is identical to the ones used in the ADV's. Refer back to page 14 for explanation of volume booster operation.

The comparator is a spring-loaded valve that monitors air pressure continuously. If instrument air header pressure drops to 55 psig, the comparator's spring forces the internal valve down allowing the trip valve to realign the air accumulators to the turbine bypass valves closed volume booster thus maintaining the valve closed. Once air pressure in the two volume tanks is depleted, for whatever reason, system pressure can open the turbine bypass valve.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0575 **Rev:** 0 **Rev Date:** 5/4/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-ICS **Objective:** 17 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal from Reactor Core

System Number: 045 **System Title:** Main Turbine Generator System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G System controls including: Expected response of secondary plant parameters following a T/G trip.

K/A Number: A1.06 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.7 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 63 **SRO:**

Given:

- All ICS H/A stations are in automatic.
- Reactor power is 100%.
- Main Turbine trips.
- Normal post-trip response.

What is the expected OTSG pressure and RCS Tavg?

A. 895 psig and 532°F

B. 945 psig and 538°F

C. 995 psig and 547°F

D. 1020 psig and 550°F

Answer:

C. 995 psig and 547°F

Notes:

"C" is correct, a 100 psig bias is applied to the TBVs following a trip to limit RCS cooldown to 547°F.
"A" is incorrect, these parameters would be correct without the 100 psig bias.
"B" is incorrect, these parameters would be correct with the 50 psig bias applied when turbine above 15%.
"D" is incorrect, these parameters would be correct with the ADVs in control.

References:

STM 1-64, Rev. 9

History:

New for 2005 RO exam.

A bias of +100 psig will be added when the Reactor trips to limit the RCS cooldown. A pressure setpoint of 895 psig corresponds to a RCS Tave of 532^{0}F. Without the 100 psig bias the bypass valves would control pressure to 895 psig thus cooling the primary down from the normal operating Tave of 579^{0}F to 532^{0}F. With the 100 psig bias, the bypass valves would control to 995 psig which corresponds to ~547^{0}F Tave thus limiting the cooldown of the primary and the resultant drop in pressurizer level.

Each pair of turbine bypass valves can be manually controlled from a hand/automatic station. In position the meter indicates the hand demand signal and has a span of 0% open to 100% open. This signal is the demanded positions and not the actual valve positions. Actual valve position indications are provided above each hand/auto station by 0% to 100% indicators. Selection of measured variable on these stations indicates the biased turbine header pressure error signal. A 50% indication means that biased turbine header pressure error equals zero. If the 50 psi bias has been applied, and header pressure is at 895 psig, an indication of ~40% would be seen in measured variable.

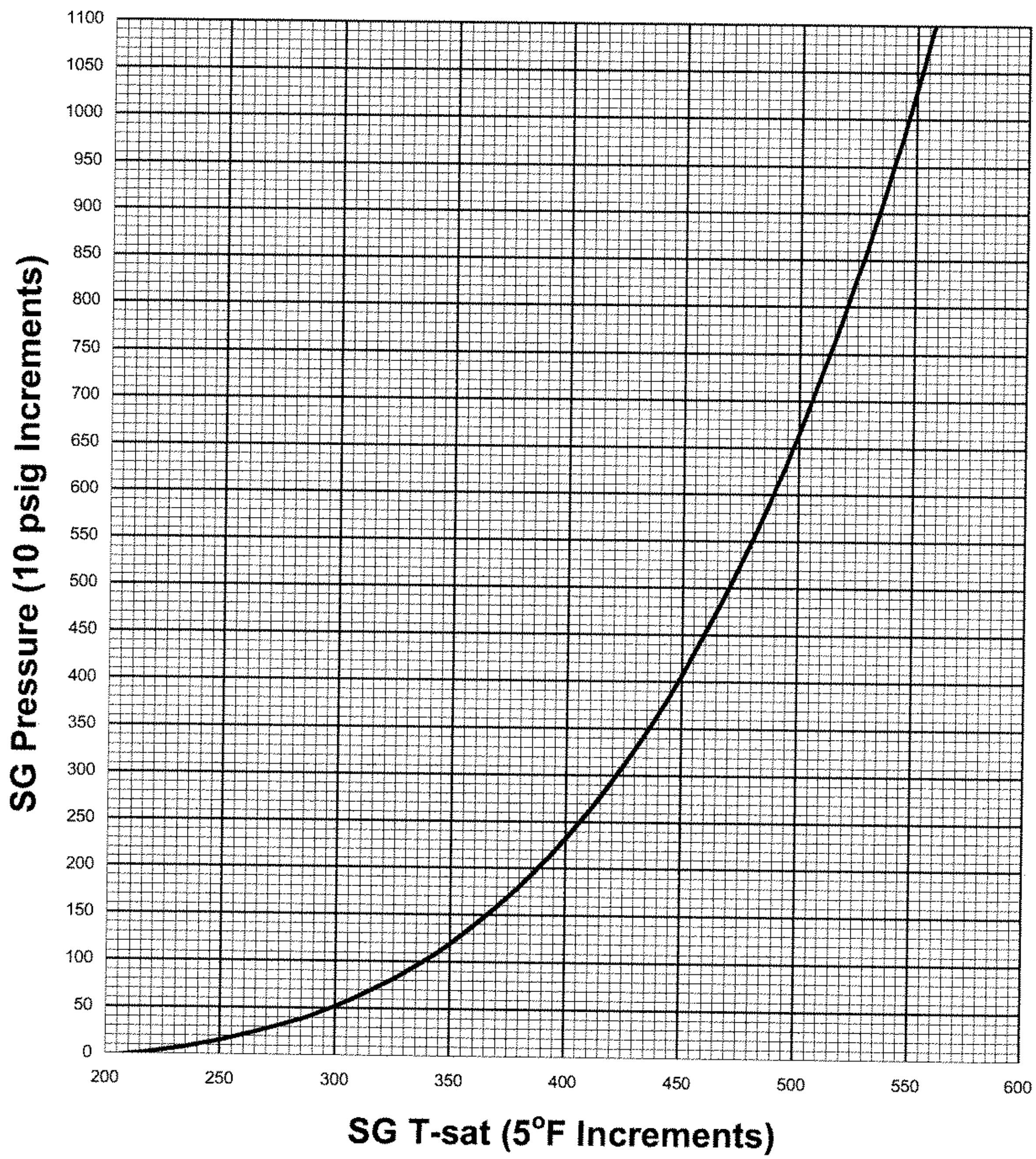
2.5.3.1 Turbine Bypass Valve Control Selector Switch

The signals to the turbine bypass valves may be affected by the position of a selector switch located on Panel C02. This switch has two positions, "Auto" and "Cond", which may be selected by the operator. The "Auto" position is normally used. In this position the control signal will always be passed to the turbine bypass valves unless a low condenser vacuum condition (<21") exists. In the event of low vacuum, the bypass valves receive a signal to keep them fully closed. If condenser vacuum increases to >23", the operator may return control to the valves by depressing two reset pushbuttons located on C02. If the operator selects the "Cond" position, the normal control signal will go to the bypass valves regardless of condenser vacuum value.

2.5.4 Calibrating Integral

During steady-state operation in the integrated control mode, the reactor and steam generator should be producing the proper amount of steam for the demanded load. If one of the following occurs, a megawatt error would develop: (1) the turbine generator efficiency changes, (2) the enthalpy of the steam to the turbine changes, or (3) an error occurs in the feedwater flow measurement. The megawatt error will, through a calibrating integral, be applied to alter the demand to both the steam generators (feedwater control) and reactor. (Refer to figure 64.20) This control action eliminates megawatt error by recalibrating the reactor and feedwater demand with respect to the unit load demand. With no megawatt error, the reactor and steam generator are producing the proper amount of steam for the demanded load.

FIGURE 2
SG Pressure vs T-sat



INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0196 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** R. Fuller

TUOI: A1LP-RO-RMS **Objective:** 11 **Point Value:** 1

Section: 3.9 **Type:** Radioactive Release

System Number: 071 **System Title:** Waste Gas Disposal System

Description: Ability to manually operate and/or monitor in the control room: Setting of process radiation monitor alarms, automatic functions, and adjustment of setpoints.

K/A Number: A4.25 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.2	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 64 **SRO:**

While performing a reactor building purge evolution, the operator notes all four RB Purge Isolation Valves go closed. What is the most likely cause?

a. An ESAS actuation of channels 1 and 2 has closed the valves.

b. A loss of load center B-5 has occurred causing the valves to fail closed.

c. A high radiation setpoint has been exceeded on SPING 1.

d. RB Purge Exhaust Fan (VEF-15) has tripped causing the valves to close.

Answer:

c. A high radiation setpoint has been exceeded on SPING 1.

Notes:

(a.) is incorrect. Channel 1 and 2 of ESAS does not cause the valves to close.
(b.) is incorrect. A loss of B-5 affects RB cooling and RB hydrogen recombiners but not RB purge.
(c.) is correct.
(d.) is incorrect. VEF-15 is interlocked with the RB supply fan but not the RB purge valves.

References:

STM 1-09 rev. 6

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2005 RO exam.

2.2.1.4 Purge Supply Containment Isolation

The fan discharges into the containment through an exterior isolation valve (CV-7402) and an interior isolation valve (CV-7404). CV-7402 is an air-to-open/spring-to-close butterfly valve supplied by instrument air and controlled by solenoid valve SV-7402 through a hand switch on panel C16. Position indication for CV-7402 is also on panel C16. On an ES actuation from Ch. 4, or high radiation signal from Sping 1, CV-7402 automatically closes to meet containment isolation requirements. SV-7402 is powered from breaker RA2-11. Interior isolation valve CV-7404 is a air-operated air-to-open/spring-to-close butterfly valve supplied by instrument air and controlled by solenoid valve SV-7404 through a hand switch on panel C18. Position indication for this valve is also on panel C18. Power to SV-7404 is RA1-9. On an ES actuation from Ch. 3, or high radiation signal from Sping 1, CV-7404 closes to meet containment isolation requirements. Purge Supply air is ducted to various areas in the lower half of the RB.

The Containment Purge Isolation Valves are expected to close within 5 seconds of receipt of an ES signal or an signal from Sping 1 by spring pressure. The interior isolation valves are air-operated valves sized at 24". The valves are 'Tricentric' which are designed to provide a positive seal by wedging the disk into the seat. An instrument airline goes through the penetration for the Purge lines to supply air to the two valves in the Reactor Building. Until a Tech. Spec. change is approved for the use of these valves above Cold Shutdown they will remain closed until shutdown with the conditions for Reactor Building Integrity no longer required.

CV-7401/CV-7402/CV-7403/CV-7404	
Valve MFG.	Atwood & Morrill
Type	Tricentric, Metal seated, Butterfly
Size	24"
Valvop MFG.	Bettis
Type	pneumatic, NT-520

2.2.2. RB Purge Exhaust Fan VEF-15

RB Purge Exhaust, VEF-15 is a centrifugal fan that takes suction on a common exhaust line from four ducts located in the upper portion of the Reactor Building. Feeding into the common exhaust line is one duct taking suction from the dome at 545' elevation and three other ducts taking suction from around the 445' elevation. VEF-15 is powered from B3164.

VEF-15	Clarage Fan
Flow	40KCFM
Type	Centrifugal, Backward Inclined
Motor	GE, 480VAC, 60 HP

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0619 **Rev:** 0 **Rev Date:** 8/10/05 **Source:** New **Originator:** J.Cork

TUOI: A1LP-RO-FPS **Objective:** 6 **Point Value:** 1

Section: 3.8 **Type:** Plant Service Systems

System Number: 086 **System Title:** Fire Protection System (FPS)

Description: Knowledge of the effect of a loss or malfunction of the Fire Protection System will have on the following:
Fire, smoke, and heat detectors.

K/A Number: K6.04 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 65 **SRO:**

The smoke detector string for the Cable Spreading Room has a trouble relay that is de-energized in it's respective ZIU.

Considering this, which of the following can actuate the deluge system for the Cable Spreading Room?

A. Take the Inhibit switch out of "Inhibit" on the Cable Spreading Room on C463.

B. Automatic actuation via smoke detector and protectowire detector for the Cable Spreading Room.

C. Automatic actuation of a protectowire detector for the Cable Spreading Room.

D. Manually operate the Operate switch for the Cable Spreading Room on C463.

Answer:

D. Manually operate the Operate switch for the Cable Spreading Room on C463.

Notes:

Answer "D" is correct, a de-energized Trouble Relay means a de-energized or open smoke detector string, therefore automatic operation is non-functional. Operating the Operate switch on C463 bypasses the automatic contacts.
Answer "B" and "C" are incorrect, automatic actuation is not available on a cross zoned system without both detection strings.
Answer "A", although the Inhibit switch is taken to Inhibit on cross zoned systems when a string is inoperable, taking it out of inhibit will do nothing in this case.

References:

1203.009, Chg. 022-05-0

History:

New for 2005 RO exam, replacement question.

ATTACHMENT A

B2-7 (U & L)
Zone 97-R Cable Spread Rm UAV-5638 Man Trip
Zone 97-R Cable Spread Rm UAV-5638 Inhibit

1.0 CAUSES

1.1 Zone 97-R Cable Spread Rm UAV-5638 Man Trip (B2-7U) LED: Manual Trip Switch (left-hand) in OPERATED position

NOTE
When automatic actuation inhibit is in effect, Cable Spread Rm Deluge Valve (UAV-5638) can only be opened using Man Trip Switch on this module or by using local pull station.

1.2 Zone 97-R Cable Spread Rm UAV-5638 Inhibit (B2-7L) LED: Inhibit switch (right-hand) is in OPERATED (down) position

2.0 ACTION REQUIRED

2.1 B2-7U manual trip LED: None.

2.2 B2-7L inhibit LED:

2.2.1 Declare suppression system inoperable.

2.2.2 Carry out required actions of Unit 1 Fire Protection Specifications, Attachment 1 of 1000.152 and report fire system impairment per "Instructions" section of 1000.152.

2.2.3 Review Unit 1 Fire Detection Database to determine if additional fire protection controls required per 1000.152.

3.0 TO CLEAR ALARM

3.1 B2-7U man trip LED:

3.1.1 WHEN Cable Spread Rm Deluge Valve (UAV-5638) is no longer required open,
THEN place Manual Trip Switch in NORMAL.

3.2 B2-7L inhibit LED:

3.2.1 WHEN inhibit is no longer needed,
THEN return Inhibit Switch to NORMAL.

Facility: Arkansas Nuclear One Unit 2 RO Written Outline Date of Exam: 01/21/2005

Category	K/A #	Topic	RO IR	RO #
1. Conduct of Operations	2.1	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	3.7	66
	2.1	2.1.20 Ability to execute procedure steps.	4.3	67
	2.1	2.1.22 Ability to determine mode of operation.	2.8	68
	2.1			
	2.1			
	2.1			
	Subtotal			3
2. Equipment Control	2.2	2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	3.7	69
	2.2	2.2.13 Knowledge of tagging and clearance procedures.	3.6	70
	2.2	2.2.33 Knowledge of control rod programming.	2.5	71
	2.2			
	2.2			
	Subtotal			3
3. Radiation Control	2.3	2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	2.5	72
	2.3	2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	2.9	73
	2.3			
	2.3			
	2.3			
	2.3			
	Subtotal			2
4. Emergency Procedures/ Plan	2.4	2.4.17 Knowledge of EOP terms and definitions.	3.1	74
	2.4	2.4.29 Knowledge of the emergency plan.	2.6	75
	2.4			
	2.4			
	2.4			
	Subtotal			2
Tier 3 Point Total				10

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0620 **Rev:** 0 **Rev Date:** 8/10/05 **Source:** Modified **Originator:** Cork/Pullin

TUOI: A1LP-RO-EOP06 **Objective:** 10 **Point Value:** 1

Section: 2.1 **Type:** Generic

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

K/A Number: 2.1.7 **CFR Reference:** 43.5 / 45.12 / 45.13

Tier:	3	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	3
Group:		**SRO Imp:**	4.4	**SRO Select:**	No	**Taxonomy:**	C

Question: **RO:** 66 **SRO:**

The plant is shutdown and a cooldown is in progress due to a tube leak in the "A" OTSG.

- RCS T-hot is 520 °F and lowering
- BWST level is at 35 ft and lowering
- A OTSG level is 405 inches and rising
- Dose rates at site boundary are at Alert level

Which of the following procedural RCS cooldown limits are in effect for the above conditions?

A. Less than or equal to 50 °F/hour

B. Less than or equal to 100 °F/hour

C. Less than or equal to 240 °F/hour

D. Less than or equal to 520 °F/hour

Answer:

C. Less than or equal to 240 °F/hour

Notes:

"C" is correct in accordance with 1202.006, Tube Rupture, due to "A" OTSG level approaching 410" and site boundary dose rates at the Alert level.
"A" is incorrect, it is the limit if RCS temp is between 300 and 170 °F.
"B" is incorrect, this is the normal cooldown limit above 300°F.
"D" is incorrect, it is the pressurizer cooldown limit.

References:

1202.006, Chg. 007-04-0

History:

Modified QID #38 for 2005 RO exam, replacement question.

INSTRUCTIONS	CONTINGENCY ACTIONS
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:	17. **GO TO step 18.**
A. For good SG, place TURB BYP valves in HAND **AND** adjust to maintain cooldown rate ≤240°F/hr.	A. **IF** TURB BYP valves are **not** available, **THEN** operate ATM Dump Control System for good SG in HAND to maintain cooldown rate ≤240°F/hr. SG A / — / SG B CV-2676 / ATM DUMP ISOL / CV-2619 CV-2668 / ATM DUMP CNTRL / CV-2618 1) **IF** both SGs are bad, **THEN** steam both SGs.
B. **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.	C. **IF** both SGs are bad, **THEN** steam both SGs.

Table in contingency action A:

SG A		SG B
CV-2676	ATM DUMP ISOL	CV-2619
CV-2668	ATM DUMP CNTRL	CV-2618

INSTRUCTIONS	CONTINGENCY ACTIONS

18. <u>IF</u> emergency cooldown rate is <u>not</u> required
<u>OR</u>
RCS T-hot is ≤500°F,
<u>THEN</u> establish RCS cooldown rate of ≤100°F/hr as follows:

A. For good SG, place TURB BYP valves in HAND
<u>AND</u>
adjust to maintain cooldown rate ≤100°F/hr.

Contingency: A. **<u>IF</u>** TURB BYP valves are **<u>not</u>** available, **<u>THEN</u>** 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) **<u>IF</u>** both SGs are bad, **<u>THEN</u>** steam both SGs.

B. **<u>When</u>** RCS press is <1700 psig, **<u>THEN</u>** bypass ESAS.

C. **<u>IF</u>** only one SG is bad, **<u>THEN</u>** 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: C. **<u>IF</u>** both SGs are bad, **<u>THEN</u>** steam both SGs.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0079 **Rev:** 1 **Rev Date:** 5/9/05 **Source:** Modified **Originator:** S.Pullin
TUOI: A1LP-WCO-PRMS **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As
System Number: 2.1 **System Title:** Conduct of Operations
Description: Ability to execute procedure steps.

K/A Number: 2.1.20 **CFR Reference:** 41.10 / 43.5 / 45.12
Tier: 3 **RO Imp:** 4.3 **RO Select:** Yes **Difficulty:** 2
Group: **SRO Imp:** 4.2 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 67 **SRO:**

Given:

- Unit is in Mode 5
- Reactor Building pressure is 15.9 psia and stable
- No other abnormal conditions exist

Which of the following procedural actions is required to establish RB purge with these conditions?

a. Open RB purge inlets first, then open outlets.

b. Vent RB via H2 sample lines.

c. Vent RB via RB leak detector.

d. Open RB purge outlets first, then open inlets.

Answer:

d. Open RB purge outlets first, then open inlets.

Notes:

"D" is correct due to positive pressure in RB to prevent reverse flow through filters.
"A" is incorrect since this would induce reverse flow through filters.
"B" and "C" are used at power to lower RB pressure when RB integrity is required.

References:

1104.033, Rev 060-10-0

History:

Developed for the 1998 RO/SRO Exam.
Used in 2001 RO/SRO Exam.
Modified for 2005 RO exam.

ATTACHMENT B

REACTOR BUILDING PURGE GASEOUS RELEASE PERMIT

CAUTION
The following sequence prevents reverse flow through RB purge exhaust filters. Reverse flow can cause damage to filters.

NOTE
SPING 1 is interlocked to close all four RB Purge Isolation Valves (CV-7401, CV-7402, CV-7403, CV-7404) on a high radiation signal.

5.7 Open RB purge dampers as follows:

5.7.1 IF RB pressure is negative,
THEN perform the following:

A. Open RB Purge Inlets (CV-7402 and CV-7404). ______

B. WHEN RB pressure equals atmospheric pressure,
THEN open RB Purge Outlets (CV-7401 and CV-7403). ______

5.7.2 IF RB pressure is positive,
THEN perform the following:

A. Open RB Purge Outlets (CV-7401 and CV-7403). ______

B. WHEN RB pressure equals atmospheric pressure,
THEN open RB Purge Inlets (CV-7402 and CV-7404). ______

5.7.3 IF RB pressure equals atmospheric pressure,
THEN open inlets and outlets. ______

- CV-7401
- CV-7402
- CV-7403
- CV-7404

5.7.4 Verify RB Purge Valves open. ______

- CV-7401
- CV-7402
- CV-7403
- CV-7404

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0576 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-SRO-TS **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to determine Mode of Operation.

K/A Number: 2.1.22 **CFR Reference:** 43.5 / 45.13

Tier: 3 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 68 **SRO:**

Which one of the following conditions would place the unit in Mode 4?

A. The reactor must be subcritical by at least 1.5% Delta k/k

B. RCS temperature is between 200 °F and 280 °F

C. K effective is >0.99

D. RCS temperature is 300 °F

Answer:

B. RCS temperature is between 200 °F and 280 °F

Notes:

Only "B" is correct per T.S. definition of mode 4.
The other choices are for other modes.

References:

T.S. table 1.1-1

History:

New for 2005 RO exam, modified for replacement question.

Table 1.1-1

MODES

MODE	TITLE	REACTIVITY CONDITION (K_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 280
4	Hot Shutdown[b]	< 0.99	NA	$280 > T_{avg} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

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:** No **Taxonomy:** K

Question: RO: 69 SRO:

During an INITIAL approach to criticality, if criticality is NOT achieved within ________ of the ECP, 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
   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
   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. 019-03-0

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 2005 RO 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 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.

9.9 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:

9.9.1 Inform Reactor Engineering.

9.9.2 Verify boron concentrations.

9.9.3 Verify ECC calculation.

9.9.4 Verify position of all control rods by comparing API to zone or limit position switches.

9.9.5 WHEN cause of ECC error is determined, AND cause corrected, THEN re-perform AND re-initial applicable steps of this procedure.

9.10 Record time reactor is made critical _________.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0621 **Rev:** 0 **Rev Date:** 8/10/05 **Source:** Direct **Originator:** Cork/Pullin

TUOI: ELP-OPS-CLR **Objective:** 2 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of tagging and clearance procedures.

K/A Number: 2.2.13 **CFR Reference:** 41.10 / 45.3

Tier: 3 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 70 **SRO:**

During the hanging of tagouts, the ________________ is responsible for the following:

-Reviews tagout detail of the tagout.

-Ensures components are positioned in accordance with the Tag Hang sheet.

-Ensures components are restored to their normal positions in accordance with Tagout Tags To Be Removed sheet.

-Ensures the tag(s) have been hung or removed.

A. Tagout Holder

B. Verifier

C. Preparer

D. Reviewer

Answer:

B. Verifier

Notes:

Answer "B" is correct. Per the responsibilities section of OP-102, the Verifier has the responsibilities listed. The other choices are other tagging positions.

References:

EN-OP-102, Rev. 2

History:

Direct from new exambank QID ANO-OpsCommon-02246
Selected for 2005 RO exam as a replacement question.

4.7 PREPARER

[1] Reviews each work order that will be covered by a tagout to determine the total work scope.

[2] Determines boundary isolation requirements for work being performed to ensure personnel and plant safety.

[3] Prepares the Tagout.

4.8 REVIEWER

[1] Reviews the tagout has been properly developed for the planned work.

[2] Shares equal responsibility with the preparer.

4.9 TAGGER

[1] Reviews tagout detail of the tagout.

[2] Ensures components are positioned in accordance with the Tag Hang sheet.

[3] Restores components to their normal positions in accordance with the Tags To Be Removed sheet.

[4] Hangs and removes Tags.

4.10 VERIFIER

[1] Reviews tagout detail of the tagout.

[2] Verifies components are positioned in accordance with the Tag Hang sheet.

[3] Verifies components are restored to their normal positions in accordance with Tagout Tags To Be Removed sheet.

[4] Verifies the tag(s) have been hung or removed.

4.11 TAGOUT HOLDER

[1] Overall job-associated lockout or tagout responsibility including coordination of affected work forces and ensuring continuity of protection.

[2] Reviews the tagout and verifies adequacy of the tagout for maintaining proper equipment protection and personnel safety.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0390 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Direct **Originator:** S. Pullin
TUOI: A1LP-RO-CRD **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic K & A's
System Number: 2.2 **System Title:** Control Rod Drive
Description: Knowledge of Control Rod programming.

K/A Number: 2.2.33 **CFR Reference:** 43.6
Tier: 3 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3
Group: **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** C

Question: **RO:** 71 **SRO:**

Given:

- A startup is in progress.
- All safety groups are 100% withdrawn.
- The group 6 rods' RPI were not re-zeroed prior to startup.
- The CBOR is continuing with the startup with group 5 rods.

Which of the following results?

a. Auto inhibit.

b. Asymmetric rod alarm.

c. Out inhibit.

d. Sequence inhibit.

Answer:

d. Sequence inhibit.

Notes:

Answer [d] is correct, input to sequence inhibit is from RPI.
Answer [a] is incorrect, input from safety group out limit.
Answer [b] is incorrect, input from API (reed switches).
Answer [c] is incorrect, input from High SUR and API.

References:

1105.009, Chg. 019-00-0

History:

Modified regular exambank QID # 3153 for use in 2001 RO exam.
Selected for 2005 RO exam.

the panel may best be described by identifying and explaining the function of each indicator and/or control.

2.8.1 Diamond Panel Alarms

2.8.1.1 Trip Confirmed Lamp

"Trip Confirmed" lamp (amber): When lighted, indicates that power to the CRDM's has been interrupted allowing rods to drop into the core. The lamp is turned on when the following conditions are met:

- "A" and "B" AC breakers open or
- "A" AC breaker and "D" DC breakers open with "F" electronic trips or
- "B" AC breaker open and "C" DC breakers open with "E" electronic trips or
- DC breakers "C" and "D" open with "E" and "F" electronic trips.

In addition to the Diamond Panel indication, the trip confirm circuit (Refer to figure 2.41) will trip the Control Rod Drive Control System (CRDCS) to manual, transfer the Sequence/Sequence Override circuit to the Sequence mode, and transfers the Group/Auxiliary Mode to Group.

A trip confirm will be annunciated "Reactor Trip" on control room annunciator panel K08. The electronic trip signal alone will be annunciated via the plant computer to K02 "Plant Computer Critical Alarm".

2.8.1.2 Asymmetric Rods Lamp

Asymmetric Rods Lamp (amber): When on, indicates that one or more rods within a group are more than 9 inches out of alignment with the group average position. More on Asymmetric Rods later.

2.8.1.3 Out Inhibit Lamp

Out Inhibit Lamp (amber): Indicates that control rods will not respond to any out command. Refer to figure 2.42. The following conditions will result in an "Out Inhibit":

- High startup rate signal from the RPS of 2 DPM in source range and 3 DPM in intermediate range. The high startup rate signals are bypassed in the RPS when greater than 10% power.
- If the Diamond is in automatic and greater than 40% reactor power, a loss of any safety group out limit, (refer to Safety Rods Out relay logic figure 2.43) <u>or</u> a 9" asymmetric rod fault. If either of these conditions occur they will "seal in" and the "Fault Reset" switch must be used to reset the circuit.

2.8.1.4 Sequence Inhibit Lamp

Sequence Inhibit Lamp (Amber): This lamp, when lighted, indicates excessive overlap between regulating groups (> 25%).

The signal for sequence inhibit (Sometimes referred to as sequence fault) is developed by one of two sequence monitor circuits. Refer to figure 2.44. Input to the sequence monitor circuits is group

average from relative position indication (RPI). RPI is utilized in order to provide the monitoring without incurring a sequence fault in the event an asymmetric rod condition alters the group average.

The same relay which causes the sequence inhibit will de-energize the sequence light in the "SEQ/SEQ OR" pushbutton.

Sequence inhibit will reject the Diamond Panel to manual

2.8.1.5 Auto Inhibit Lamp

Auto Inhibit Lamp (amber) indicates that the Diamond cannot be placed in automatic because:

- The safety groups are not at the out limit.
- The neutron error signal (demand versus actual) exceeds ± 1.00%.
- ICS power not available.

If the Diamond panel is in automatic, a loss of ICS Power will bring in this alarm and reject the Diamond to manual (refer to figure 2.45).

2.8.1.6 APSR Overlap Fault Lamp

APSR Overlap Fault Lamp (amber) Indicates that the group 6 lower poison section is less than three inches from group 8 upper poison section. This alarm is used for indication only.

2.8.2 Diamond Panel Indication Lights

2.8.2.1 Out Limit Lamps

Out Limit Lamps Groups 1 - 8 (red): Indicates that at least one rod out of its respective group is at the out limit. This will stop rod withdrawal for all rods within that group. On group 7, out limit occurs at 91.4% withdrawn unless the group 7 "91.4% key/bypass switch" is in bypass, then group 7 will actually be 100% withdrawn.

2.8.2.2 Control On Lamps

White lights that indicate a particular group has been selected (enabled for either automatic or manual command, or is selected for transfer. The corresponding control on lamps on the PIP must also be energized to allow insert/withdraw command.

Control on lamps for groups 1 - 4 indicate that the transfer logic is setup for that group to be transfer between DC Hold and the Auxiliary Power Supply. (If the white control on lamp on the Position Indication Panel is lit, the group is on the Auxiliary Power Supply.)

Control on Lamps Group 5 - 8 (refer to figure 2.46) indicate that a specific group has been selected (enabled) for either automatic or manual command or is selected for transfer.

2.8.2.3 In Limit Lamps

In Limit Lamp Group 1 - 8 (green): Indicates that at least one rod in that group is at the "in limit". This will stop rod movement in

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0605 **Rev:** 0 **Rev Date:** 6/30/05 **Source:** New **Originator:** Pullin/Cork

TUOI: ASLP-RO-RADPRO **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

K/A Number: 2.3.4 **CFR Reference:** 43.4 / 45.10 / 41.12

Tier: 3 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 72 **SRO:**

Which of the following describes the authorizations needed in order for an individual's exposure to reach 3300 mrem in a year?

A. Individual's supervisor approves.

B. Individual's supervisor recommends and RP supervision approves

C. Individual's supervisor recommends, RP Manager approves, and Plant General Manager concurs

D. Individual's supervisor recommends, Plant General Manager concurs, and Site Vice President approves

Answer:

C. Individual's supervisor recommends, RP Manager approves, and Plant General Manager concurs

Notes:

Answer "C" contains the correct approvals per ENS-RP-201.
Answer "A" is incorrect.
Answer "B" contains the correct approvals for > 2000 mrem but < 3000 mrem.
Answer "D" contains the correct approvals for > 4000 mrem.

References:

ENS-RP-201, Rev. 3

History:

New for 2005 RO exam, totally revised due to NRC comment, replacement question.

5.2.3.8 Unmonitored Individuals TEDE = 50 mrem/month, 100 mrem/year

5.2.3.9 Members of the General Public TEDE = 50 mrem

5.3 Instructions for Extending Routine Administrative Guidelines

5.3.1 Extend a Radiation Workers' administrative TEDE guidelines to the guidelines described in the following table, after obtaining the indicated approvals.

5.3.2 Document the authorization of the increased radiation exposure.

NOTE
Responsible individuals may be designated to authorize dose extensions.

Exposure Guideline	Requirements	Authorizations (Note)
Greater than 2000 mrem and less than or equal to 3000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends RP Supervision approves
Greater than 3000 mrem and less than or equal to 4000 mrem per year	No undocumented quarters in the current year	Individual's supervisor recommends Radiation Protection Manager approves Plant General Manager concurs
Greater than 4000 mrem per year for Radiation Workers.	No undocumented quarters in the current year	Radiation Protection Manager recommends Plant General Manager concurs Site Vice President approves
Greater than 1000 mrem and less than or equal to 2000 mrem for individuals whose lifetime exposure ≥ 1 n where n = age	No undocumented quarters in the current year	Individual's Supervisor recommends RP Supervision approves

5.4 Handling of Workers Receiving Radio-pharmaceutical Treatments

5.4.1 Individuals should notify Radiation Protection of any medical Radiopharmaceutical Treatments prior to entering the protected area of the plant.

5.4.2 Examples of procedures utilizing Radiopharmaceuticals include: cardiac stress tests and thyroid treatments. This list is not all inclusive and any other test requiring radiopharmaceuticals need to be mentioned as well.

5.4.3 Dosimetry evaluates any necessary radworker restrictions.

5.5 Program Reviews

5.5.1 Periodically, at least annually, document a review of plant isotopic composition (WBC library, passive monitoring basis, etc).

5.5.2 Periodically, at least annually, document a review of plant average beta energy.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0579 **Rev:** 0 **Rev Date:** 5/9/05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-EOP10 **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.3 **System Title:** Radiation Control

Description: Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

K/A Number: 2.3.10 **CFR Reference:** 43.4 / 45.10

Tier: 3 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

Group: **SRO Imp:** 3.3 **SRO Select:** No **Taxonomy:** C

Question: RO: 73 SRO:

During performance of the ESAS procedure with BWST level at 8 ft., which of the following actions is performed specifically to reduce plant personnel exposure?

A. Notifying RP to begin monitoring BWST suction line for back-leakage.

B. Throttling RB Spray flow 1050 to 1200 gpm per train.

C. Aligning HPI to provide PZR Aux Spray.

D. Removing all but C & D condensate polishers from service.

Answer:

C. Aligning HPI to provide PZR Aux Spray

Notes:

"C" is performed prior to BWST level reaching 6 ft. so the operator performing this alignment is not exposed to unknown dose from fluid being pumped from the RB through the AB.
"A" is performed following the swap to RB recirc but is done to prevent an unmonitored release to offsite personnel via the BWST vent.
"B" is performed just prior to the swap to RB recirc but is done for NPSH concerns for the RB and LPI pumps.
"D" is performed to prevent personnel exposure during a SGTR but not during the ESAS procedure.

References:

1203.010, Chg. 005-03-0

History:

New for 2005 RO exam.

INSTRUCTIONS | **CONTINGENCY ACTIONS**

NOTE

Aligning Pressurizer AUX Spray to LPI system before going on sump recirc reduces personnel exposure should the lineup be required for boron precipitation mitigation at a later time. Transfer to RB Sump suction must commence when BWST level reaches 6', even if this alignment is not complete.

13. **Dispatch an operator to align Pressurizer AUX Spray to LPI system using Decay Heat Removal Operating Procedure (1104.004), "DH System AUX Spray Alignment Prior to RB Sump Recirc" section.**

    A. IF BWST level reaches 6' before alignment is complete,
    THEN notify dispatched operator to exit the Aux Bldg, regardless of alignment status, until transfer to RB sump suction is complete and radiation levels can be determined.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0580 **Rev:** 0 **Rev Date:** 5-9-05 **Source:** New **Originator:** S.Pullin

TUOI: A1LP-RO-EOP **Objective:** 3 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of EOP terms and definitions.

K/A Number: 2.4.17 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** K

Question: **RO:** 74 **SRO:**

Given:

- Reactor trip from 100% power.
- Normal post-trip plant parameters.

The CRS asks for critical parameters.

Which of the following is NOT a critical parameter for the above conditions?

A. Subcooling margin

B. RCS pressure

C. Pressurizer level

D. RCS Tavg

Answer:

D. RCS Tavg

Notes:

Only "D" does not meet the definition of a critical parameter per 1015.043, 4.11. The RCS temperature which is a critical parameter is CET which is used to evaluate core cooling.

References:

1015.043, Chg. 000-03-0

History:

New for 2005 RO exam.

4.9 CONDITIONAL STATEMENTS AND LOGIC SEQUENCES

Used in EOPs/AOPs to describe a set of conditions or express complex combinations of conditions. Logic terms include IF, WHEN, THEN, AND, OR and NOT.

4.10 CONTINGENCY ACTION STEPS

Action steps performed if the instruction or expected response is not achieved.

4.11 CRITICAL PARAMETERS

Parameters which are closely monitored to provide early detection of upsets in heat transfer. Critical parameters include but are not limited to:

- Subcooling Margin
- CET temperature
- RCS pressure

- Pressurizer Level
- A/B SG levels
- A/B SG pressures

4.12 DUAL-COLUMN FORMAT

A procedure format using two parallel columns. The left column is designated "**INSTRUCTIONS**". This column contains the required operator action and the expected plant response. The right column is designated "**CONTINGENCY ACTIONS**". This column contains actions to be taken when the expected response is not obtained or required operator action cannot be performed.

4.13 EMERGENCY OPERATING PROCEDURE (EOP)

A plant specific document based on Emergency Procedure Guidelines which contains the steps needed to take the plant from a reactor trip to a safe, stable condition. Emergency Operating Procedures use a specific format for clarity of operator actions, Control Room personnel interactions, and compatibility with the design of the Control Room.

4.14 ENTRY CONDITIONS

Conditions that are written to explicitly identify those conditions that should exist for the user to enter an EOP/AOP.

4.15 FLOATING STEPS

Steps which apply at all times when the associated procedure is in use. They are located on foldout pages at the end of the procedure.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1

QID: 0128 **Rev:** 2 **Rev Date:** 8/10/05 **Source:** Modified **Originator:** JCork

TUOI: ASLP-RO-EPLAN **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of the emergency plan.

K/A Number: 2.4.29 **CFR Reference:** 43.5 / 45.11

Tier:	3	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	4.0	**SRO Select:**	No	**Taxonomy:**	K

Question: **RO:** 75 **SRO:**

Which of the following would be classified as a fission product barrier failure?

A. RCS leakage indicates greater than 50 gpm.

B. RCS pressure 2450 psig with ERV controlling pressure.

C. CNTMT radiation levels equal to Alert level from CNTMT Radiation EAL.

D. Engineering Assessment of core damage indicates 0.1% fuel cladding failure.

Answer:

d. EOF Director determines the Reactor Building is breached.

Notes:

"A" is correct, leakage is >50 gpm.
"B" is incorrect, this is a challenge but not a breach of the RCS pressure boundary.
"C" is incorrect, CNTMT radiation levels must reach the SAE level for this to be a breach.
"D" is incorrect, failed fuel must reach 1.0% to be a breach.

References:

1903.010, Chg. 037-03-0

History:

Developed for 1998 SRO exam.
Revised after 9/98 exam analysis review.
Used in 2001 SRO Exam.
Modified for 2005 RO exam as replacement question.

INITIAL RO/SRO EXAM BANK QUESTION DATA ARKANSAS NUCLEAR ONE - UNIT 1 (VALIDATION)

QID: 0128 **Rev:** 1 **Rev Date:** 11/4/98 **Source:** Direct **Originator:** JCork

TUOI: AA61002-006 **Objective:** 14 **Point Value:** 1

Section: 2.0 **Type:** Generic K/As

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of the emergency plan.

K/A Number: 2.4.29 **CFR Reference:** 43.5 / 45.11

Tier: 3 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 2

Group: G **SRO Imp:** 4.0 **SRO Select:** No **Taxonomy:** K

Question: **RO:** **SRO:**

PARENT Question

Which of the following would be classified as a fission product barrier failure?

a. RCS leakage indicates greater than 30 gpm.

b. The inability to monitor a Fission Product Barrier.

c. CNTMT radiation levels equal to Alert level from CNTMT Radiation EAL.

d. EOF Director determines the Reactor Building is breached.

Answer:

d. EOF Director determines the Reactor Building is breached.

Notes:

"d" is correct per EAL definition.
"a" is incorrect, leakage must be >50 gpm.
"c" is incorrect, rad levels must be equal to SAE level.
"b" is incorrect, the correct definition is two fission product barriers known to be breached with the inability to monitor the third.

References:

1903.010, Emergency Action Level Classification, Rev. 036-02-0, pages 4 and 5.

History:

Developed for 1998 SRO exam.
Revised after 9/98 exam analysis review.
Used in 2001 SRO Exam.

4.7 Emergency Response Organization (ERO) - The organization which is composed of the Initial Response Staff (IRS), the EOF staff, the TSC staff, the OSC staff, and the Emergency Team members. It has the capability to provide manpower and other resources necessary for immediate and long-term response to an emergency situation.

4.8 EPA Protective Action Guideline (PAG) Exposure Levels - The projected dose to reference man, or other defined individual, from an unplanned release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended (i.e., 1 Rem TEDE or 5 Rem Child Thyroid (CDE)).

4.9 Exclusion Area: That area surrounding ANO within a minimum radius of 0.65 miles of the reactor buildings, but outside the protected area and controlled to the extent necessary by ANO during periods of emergency.

4.10 FISSION PRODUCT BARRIER FAILURE

4.10.1 **Fuel Cladding Failure** - Condition where the fuel rod cladding becomes defective and cannot contain the fission gases that have accumulated between the fuel pellet and the fuel rod cladding (commonly referred to as the gap).

A. Unit 1 - Greater than 1% fuel cladding failure as indicated by ANY of the following:

1. Nuclear Chemistry analysis of RCS sample yields > 400 uCi/gm specific I-131.

2. Radiation levels that indicate >1% fuel cladding failure per Unit 1 Fuel Cladding Failure Radiation Plot (Att 7).

3. Failed Fuel Iodine process monitor (RE 1237S) indicates > 8.2×10^5 CPM.

4. Containment Radiation Levels correspond to a Site Area Emergency from Containment Radiation EAL Plot (Attachment 5).

5. Engineering assessment of core damage indicates >1% fuel cladding failure.

B. Unit 2 - Greater than 1% fuel cladding failure as indicated by ANY of the following:

1. Nuclear Chemistry analysis of RCS sample yields > 378 uCi/gm specific I-131.

2. Radiation levels that indicate >1% fuel cladding failure per Unit 2 Fuel Cladding Failure Radiation Plot (Att 8).]

3. Containment Radiation Levels correspond to a Site Area Emergency from Containment Radiation EAL Plot (Attachment 6).

(4.10.1 cont.)

4. Engineering assessment of core damage indicates > 1% fuel cladding failure.

4.10.2 **RCS Boundary Failure**

A. Unit 1 - RCS leakage greater than normal makeup capacity (50 gpm).

B. Unit 2 - RCS leakage greater than 44 gpm (capacity of a single Charging Pump).

4.10.3 **Containment Integrity Failure**

A. Abnormally high Containment High Range Radiation Monitor readings (RE-8060 or 8061 for Unit 1; 2RY-8925-1 or 2RY-8925-2 for Unit 2) and indications of radiological effluents outside of the Reactor Building that are not attributable to any other source.

B. In the judgement of the SM/TSC Director/EOF Director, a breach of the Reactor Building exists. The variety of possible Reactor Building integrity failure scenarios precludes the development of an all inclusive list. In the absence of the conditions described in 4.10.3.A above, the SM/TSC Director/EOF Director must judge the potential for an offsite release to occur based on a current status of Reactor Building isolation systems and structural integrity.

4.10.4 **Inability to Monitor a Fission Product Barrier**

A. Following the failure of two fission product barriers, the inability to monitor the third barrier is to be regarded as equivalent to a failure of that barrier.

4.11 FISSION PRODUCT BARRIER CHALLENGE

4.11.1 **Challenge to Fuel Cladding**: any event or condition which in the judgement of the SM/TSC Director/EOF Director presents the potential for greater than 1% fuel cladding failure; for example:

A. RCS temperature and pressure indicates superheated conditions.

B. Indications of the core being uncovered.

C. Exceeding safety limits (e.g. DNBR or Local Power Distribution)

4.11.2 **Challenge to RCS Boundary**: any event or condition which, in the judgement of the SM/TSC Director/EOF Director could result in RCS leakage in excess of normal makeup capacity (i.e., 50 gpm for Unit 1 or 44 gpm for Unit 2); for example:

A. RCS pressure > 2450 psig and not decreasing.

B. Two out of three seal stages failed on any RCP (U-1).