ANO Unit 1 Initial Exam given March 16, 2010

Written RO exam contains Questions 1-75
Written SRO exam contains Questions 76-100, located after the RO questions in this packet.
Reference files that were handouts are attached to back of exam.

The exam outlines are embedded within this file for each section of the exam.

Example, Tier 1, Group 1 topics for RO are the first several questions and therefore this section of the outline is located just in front of these questions in this package.

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

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18			6
	2	2	2	1				1	2			1	9			4
	Tier Totals	5	5	4				4	5			4	27			10
2. Plant Systems	1	3	3	2	3	3	2	2	2	2	3	3	28			5
	2	0	1	1	1	1	1	1	1	1	1	1	10			3
	Tier Totals	3	4	3	4	4	3	3	3	3	4	4	38			8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2 3 4		7
					3		2		1		4					

Note:

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

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

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)			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	#	QID	Type
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1			X				**EK3.01** – Actions contained in EOP for reactor trip	4.0	1	770	M
000008 Pressurizer Vapor Space Accident / 3						X	**2.2.37** – Ability to determine operability and/or availability of safety related equipment	3.6	2	771	N
000009 Small Break LOCA / 3				X			**EA1.02 –** RB Sump level	3.8	3	772	N
000011 Large Break LOCA / 3		X					**EK2.02** – Pumps	2.6*	4	198	D
000015/17 RCP Malfunctions / 4		X					**AK2.10** – RCP indicators and controls.	2.8*	5	609	D
000022 Loss of Rx Coolant Makeup / 2							**Not selected**	**N/A**			
000025 Loss of RHR System / 4	X						**AK1.01** - Loss of RHRS during all modes of operation.	3.9	6	773	N
000026 Loss of Component Cooling Water / 8							**Not selected**	**N/A**			
000027 Pressurizer Pressure Control System Malfunction / 3		X					AK2.03 – Controllers and positioners.	2.6	7	395	D
000029 ATWS / 1					X		**EA2.02** – Reactor trip alarm.	4.2	8	582	N
000038 Steam Gen. Tube Rupture / 3						X	**2.4.6** – Knowledge of EOP mitigation strategies.	3.7	9	364	D
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4	X						**AK1.01** – Consequence of PTS.	4.1	10	551	D
000054 (CE/E06) Loss of Main Feedwater / 4					X		**AA2.06** – AFW adjustments needed to maintain proper T-ave, and S/G level.	4.0	11	774	M
000055 Station Blackout / 6				X			**EA1.05** – Battery, when approaching fully discharged.	3.3	12	496	D
000056 Loss of Off-site Power / 6			X				**AK3.01** – Order and time to initiation of power for the load sequencer.	3.5	13	366	D
000057 Loss of Vital AC Inst. Bus / 6					X		**AA2.05** – S/G pressure and level meters.	3.5	14	624	D
000058 Loss of DC Power / 6	X						**AK1.01** – Battery charger equipment and instrumentation.	2.8	15	187	D
000062 Loss of Nuclear Svc Water / 4			X				**AK3.02 –** The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.	3.6	16	281	D
000065 Loss of Instrument Air / 8							**Not selected**	**N/A**			
W/E04 LOCA Outside Containment / 3							**Not selected**	**N/A**			

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

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

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#	QID	Type
W/E11 Loss of Emergency Coolant Recirc. / 4							**Not selected**	**N/A**			
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4				X			**EA1.3** – Desired operating results during abnormal and emergency situations.	3.6	17	335	D
000077 Generator Voltage and Electric Grid Disturbances / 6						X	**2.1.25** – Ability to interpret reference material, such as graphs, curves, tables, etc.	3.9	18	775	N
K/A Category Totals:	3	3	3	3	3	3	Group Point Total:		**18**		

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

QID: 770 **Rev:** 0 **Rev Date:** 9/2/2009 **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 the apply to a reactor trip: Actions contained in EOP for reactor trip.

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

Tier:	1	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.6	**SRO Select:**	Yes	**Taxonomy:**	C

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

The following conditions exist immediately after a reactor trip:

- Group 2, Rod 4 and Rod 5 failed to fully insert into the core
- RCS pressure is at 1730 psig
- Pressurizer level is at 50 inches
- A OTSG pressure is at 910 psig
- B OTSG pressure is at 905 psig
- CETs are 560°F and stable
- Turbine Trip Solenoid Power Available light is OFF

Which action is the operator required to perform FIRST in response to the given information as well as the reason for the action?

A. Manually actuate MSLI for affected SG(s) and EFW due to loss of D01.

B. Commence emergency boration per RT-12 due to stuck rods.

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

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

Answer:

B. Commence emergency boration per RT-12 due to stuck rods.

Notes:

(a) is incorrect since reactivity management is a higher priority and must be addressed first.
(b) is correct since two stuck rods require emergency boration.
(c) is incorrect because subcooling margin is adequate.
(d) is incorrect since pressurizer level is >30 inches and RCS pressure is >1700 psig.

References:

1202.001 Change 31

History:

This question modified from QID 0023
Selected for the 2010 RO/SRO Exam

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

Given:

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

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

- RCS pressure response abnormally slow with Pressurizer heaters energized.

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

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

A. Group 1 proportional heaters, Group 2 proportional heaters, Group 4 heaters, 124 KW output.

B. Group 1 proportional heaters, Group 2 proportional heaters, Group 5 heaters, 128 KW output

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

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

Answer:

B. Group 1 proportional heaters, Group 2 proportional heaters, Group 5 heaters, 128 KW output

Notes:

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

References:

1203.015 change 016
T.S. 3.4.9 amendment # 215

History:

New for the RO/SRO 2010 exam

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

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

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

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

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

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

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

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

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

B. RCS leak rate approximately 91 gpm, and the RB sump level can be used for 8 minutes.

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

D. RCS leak rate approximately 45 gpm, and the RB sump level can be used for 8 minutes.

Answer:

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

Notes:

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

References:

STM 1-08 Rev. 14

History:

New for the RO/SRO 2010 exam

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

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

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

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

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

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

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

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

A. To ensure adequate NPSH for ECCS pumps.

B. To prevent pump runout on the Spray pumps.

C. To prevent overloading EDGs on transfer.

D. To reduce radiation levels near RB Spray piping.

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. Flow rates will not effect radiation levels on sump recirc.

References:

1202.012, Chg. 008

History:

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

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

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

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

B. Seal bleedoff temperature 160°F

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

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

Answer:

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

Notes:

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

References:

1203.031, Chg. 018

History:

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

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

QID: 0773 **Rev:** 0 **Rev Date:** 9/3/2009 **Source:** New **Originator:** S. Pullin

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

Section: 4.2 **Type:** Generic APE

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

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

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

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.3	**SRO Select:**	Yes	**Taxonomy:**	a

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

Given:

- The RCS is drained to 374 feet for seal replacement.
- RCS Temperature 140 F.
- RCS pressure is 10 psig.
- RCS leakage measured at 50 gpm.
- "A" Decay Heat Pump has been stopped and CV-1050 Decay Heat Suction Valve has been closed per 1203.028, Loss of Decay Heat Removal AOP.

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

A. Gravity feed from the BWST.

B. Low Pressure Injection.

C. Spent Fuel Cooling Pump P-40A.

D. Borated Water Recirc Pump P-66.

Answer:

B. Low Pressure Injection.

Notes:

A. Gravity feed from the BWST is incorrect because the RCS is pressurized
B. Low Pressure Injection is correct.
C. Spent Fuel Cooling Pump P-40A is incorrect because it is the least preferred method allowed.
D. P-66 is incorrect because it is low on the perfered list.

References:

1203.028 Change 21

History:

New for the RO/SRO 2010 exam.

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

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

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

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

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

C. To maintain pressurizer heaters off during testing.

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

Answer:

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

Notes:

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

References:

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

History:

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

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

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

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

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

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

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

Subsequently the following indications are observed:
- Reactor power dropping rapidly,
- Turbine Generator Lockout alarm is in,
- EFW actuated on both trains.

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

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

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

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

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

Answer:

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

Notes:

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

References:

1102.002 Change 082
STM 1-59 Rev. 1

1102.002 Change 082
STM 1-59 Rev. 1

History:

New for the RO/SRO 2010 exam.

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of EOP mitigation strategies

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

Tier:	1	**RO Imp:**	3.7	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	An

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

After a reactor trip, the following indications are observed:

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

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

A. Initiate HPI per RT-2

B. Cooldown and isolate the "B" SG

C. Cooldown and isolate the "A" SG

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

Answer:

C. Cooldown and isolate the "A" SG

Notes:

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

References:

1202.006, Chg. 11

History:

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

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

A. HPI on with all RCPs off

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

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

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

Answer:

A. HPI on with all RCPs off

Notes:

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

References:

1202.012, Chg. 8

History:

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

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

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

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

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

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

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

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

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

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

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

The following conditions have existed for three minutes:

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

Which of the following operator actions will be performed?

A. Trip all running RCPs.

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

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

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

Answer:

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

Notes:

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

References:

1202.012 Change 008, RT-5

History:

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

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

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

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

Section: 4.1 **Type:** Generic EPEs

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

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

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

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

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

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

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

ELP-NLO-ELEC1

History:

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

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

1305.006, Chg. 030

History:

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

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

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

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

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

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

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

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

C. The NNI-X S1 and S2 switches would open and SASS would transfer to NNI-Y, SG pressure and level instruments would not be effected.

D. Instrument power would automatically be transferred to YO-1 by the ABT, SG pressure and level instruments would not be effected.

Answer:

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

Notes:

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

References:

STM 1-69, Rev. 13

History:

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

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

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

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

Section: 4.2 **Type:** Generic APE

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

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

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

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

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

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

Given the following indications at 100% power:

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

What are the actions required of the CBOT?

A. Trip the main generator output breakers.

B. Transfer D11 to emergency supply D01.

C. Trip all RCPs.

D. Transfer D21 to emergency supply D01.

Answer:

D. Transfer D21 to emergency supply D01.

Notes:

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

References:

1203.036, Chg. 08

History:

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

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

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

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

Section: 4.2 **Type:** Generic AOP

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

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

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

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

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

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

Service Water Pumps P-4A, P-4B (supplied from A-4), and P-4C are running.
An ES actuation channels (1-10) coincident with a loss of off-site power occurs.

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

A. P-4A, P-4B and P-4C, due to high service water loads with all 10 channels actuated

B. P-4A and P-4B, due to both being supplied from A-4 and #2 EDG tied on first

C. P-4B and P-4C, due to "B" service water pump is the swing pump and its perferred to be running

D. P-4A and P-4C, due to 3 service water pumps running prior to event to prevent EDG overloading

Answer:

D. P-4A and P-4C, due to 3 service water pumps running prior to event to prevent EDG overloading

Notes:

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

References:

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

History:

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

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

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

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

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

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

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

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

Tier: 1 **RO Imp:** 3.6 **RO Select:** Yes **Difficulty:** 2.5

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

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

Given:

- Loss of all Feedwater
- HPI core cooling started

What indications would you monitor to ensure adequate HPI core cooling?

A. CET temperatures stable at 100 minutes.

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

C. T-hot tracking CET temperatures.

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

Answer:

A. CET temperatures stable after 100 minutes.

Notes:

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

References:

1202.004 Change 6

History:

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

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

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

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

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

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

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

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

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

REFERENCE PROVIDED

Given:
Plant at 100% power
Generator output 880MWe
Electrical storm caused a grid disturbance
The Dispatcher calls Control Room and requests Unit 1 Generator be operated in the lagging mode at 180 Megavars.

What is the power factor for the above information?

A. 0.935 PF

B. 0.955 PF

C. 0.97 PF

D. 0.98 PF

Answer:

D. 0.98 PF

Notes:

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

References:

1102.004 Change 048

History:

Developed for the 2010 RO/SRO exam.

ES-401 PWR Examination Outline Form ES-401-2
Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#	QID	Type
000001 Continuous Rod Withdrawal / 1							**Not selected**	**N/A**			
000003 Dropped Control Rod / 1							**Not selected**	**N/A**			
000005 Inoperable/Stuck Control Rod / 1							**Not selected**	**N/A**			
000024 Emergency Boration / 1							**Not selected**	**N/A**			
000028 Pressurizer Level Malfunction / 2	X						**AK1.01** – PZR reference leak abnormalities.	2.8*	19	776	N
000032 Loss of Source Range NI / 7					X		**AA2.04** – Satisfactory source-range / intermediate-range overlap	3.1	20	777	N
000033 Loss of Intermediate Range NI / 7							**Not selected**	**N/A**			
000036 (BW/A08) Fuel Handling Accident / 8							**AK2.1 – Changed to randomly selected System 068 AK2.07**	**N/A**			
000037 Steam Generator Tube Leak / 3				X			**AA1.10** – CVCS makeup tank level indicator	2.9	21	778	N
000051 Loss of Condenser Vacuum / 4							**Not selected**	**N/A**			
000059 Accidental Liquid RadWaste Rel. / 9							**Not selected**	**N/A**			
000060 Accidental Gaseous Radwaste Rel. / 9							**AK1.04 – Changed to randomly selected System 028 AK1.01**	**N/A**			
000061 ARM System Alarms / 7			X				**AK3.02** – Guidance contained in alarm response for ARM system.	3.4	22	634	D
000067 Plant Fire On-site / 8	X						**AK1.02** – Fire Fighting	3.1	23	695	DR
000068 (BW/A06) Control Room Evac. / 8		X					**AK2.07** – ED/G	3.3	24	779	D
000069 (W/E14) Loss of CTMT Integrity / 5							**Not selected**	**N/A**			
000074 (W/E06&E07) Inad. Core Cooling / 4							**Not selected**	**N/A**			
000076 High Reactor Coolant Activity / 9							**Not selected**	**N/A**			
W/EO1 & E02 Rediagnosis & SI Termination / 3							**Not selected**	**N/A**			
W/E13 Steam Generator Over-pressure / 4							**Not selected**	**N/A**			
W/E15 Containment Flooding / 5							**Not selected**	**N/A**			
W/E16 High Containment Radiation / 9							**Not selected**	**N/A**			
BW/A01 Plant Runback / 1							**Not selected**	**N/A**			
BW/A02&A03 Loss of NNI-X/Y / 7							**Not selected**	**N/A**			
BW/A04 Turbine Trip / 4							**Not selected**	**N/A**			
BW/A05 Emergency Diesel Actuation / 6		X					**AK2.1** – Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	4.0	25	349	D
BW/A07 Flooding / 8					X		**AA2.2** - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	3.3	26	780	N
BW/E03 Inadequate Subcooling Margin / 4							**Not selected**	**N/A**			

BW/E08; W/E03 LOCA Cooldown - Depress. / 4							**Not selected**	**N/A**			
W/E09; CE/A13; W/E09&E10 Natural Circ. / 4							**Not selected**	**N/A**			
BW/E13&E14 EOP Rules and Enclosures						X	**2.2.22**- Knowledge of limiting conditions for operations and safety limits.	4.0	27	595	N
CE/A11; W/E08 RCS Overcooling - PTS / 4							**Not selected**	**N/A**			
CE/A16 Excess RCS Leakage / 2							**Not selected**	**N/A**			
CE/E09 Functional Recovery							**Not selected**	**N/A**			
K/A Category Point Totals:	**2**	**2**	**1**	**1**	**2**	**1**	Group Point Total:		**9**		

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

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

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

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

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

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

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

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

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

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

Given:

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

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

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

B. Indicated level decreases and pressurizer level control valve, CV-1235, fails as is.

C. Indicated level increases and pressurizer level control valve, CV-1235, fails as is.

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

Answer:

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

Notes:

D. is correct, a leak in the reference leg would cause indicated level to increase. As a result of the level rise CV-1235 will close in oreder to maintain level at setpoint.
A, B, and C are incorrect, using the different possible combinations. CV-1235 fails as is on a loss of Instrument Air not on a fialure of the reference leg.

References:

ASLP-RO-CMP02 Rev 2

History:

New selected for 2010 RO/SRO exam.

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

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

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

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

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

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

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

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

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

Given:

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

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

What is the required operator action for the given condition?

A. Immediately suspend operations involving positive reactivity changes..

B. Within 1 hour verify CRD trip breakers open.

C. Continue the startup..

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

Answer:

C. Continue the startup..

Notes:

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

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

References:

1203.021 Change 10

History:

New for the RO/SRO 2010 exam.

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

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

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

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

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

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

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

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

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

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

Given:

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

What is the A OTSG Tube Leak rate?

A. 10.2 gpm

B. 15.4 gpm

C. 20.4 gpm

D. 30.8 gpm

Answer:

B. 15.4 gpm

Notes:

B. 15.4 gpm is correct based on makeup tank level is 30.86 gallons per inch, at a rate of change of 1 inch per 2 minutes equals 15.4 gpm leak.
A, C and D are incorrect.

References:

1203.039 Change 011

History:

New for the RO/SRO 2010 exam.

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

QID: 0634 **Rev:** 0 **Rev Date:** 11/8/05 **Source:** Direct **Originator:** J.Cork

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

Section: 4.2 **Type:** Generic APEs

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

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

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

Tier: 1 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 4

Group: 2 **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** C

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

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

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

What reason(s) would cause both alarms above to come into alarm?

A. WARNING and POWER ON lights on

B. POWER ON light off

C. HIGH ALARM light on and POWER ON light off

D. FAILURE light on

Answer:

B. POWER ON light off

Notes:

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

References:

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

History:

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

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

QID: 0695 **Rev:** 1 **Rev Date:** 4/1/2008 **Source:** Repeat **Originator:** Steve Pullin

TUOI: ASLP-RO-FRHAZ **Objective:** 4B **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

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

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

Tier: 1 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

Group: 2 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** K

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

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

History:

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

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

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

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

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

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

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

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

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

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

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

Given:

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

What condition can automatically trip the Emergency Diesel Generators?

A. Positive crankcase pressure trip

B. Low lube oil pressure trip

C. Mechanical over speed trip

D. De-energized Governor Run Solenoid

Answer:

C. Mechanical over speed trip

Notes:

"C" will mechanically trip the fuel rack.
"A" and "B" require DC power to the emergency trip relay.
"D" is overriden on a No DC start

References:

1104.036, Emergency Diesel Generator Operation, Change 049

History:

Direct Selected for 2010 RO/SRO exam

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

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

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

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

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

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

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

Tier:	1	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	3	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

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

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

What will the ES Electrical response be?

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

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

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

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

Answer:

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

Notes:

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

References:

STM 1-32, Rev. 33

History:

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

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

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

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

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

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

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

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

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

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

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

Given:

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

What action is required per Natural Emergencies procedure 1203.025 section 4 Flood?

A. Perform Rapid Plant Shutdown and align "B" Decay Heat pump for Decay Heat

B. Perform Rapid Plant Shutdown and transfer plant auxiliaries to SU 2 transformer

C. Trip Reactor and perform a Forced flow Cool Down.

D. Trip Reactor and perform a Natural Circulation Cool Down.

Answer:

B. Perform Rapid Plant Shutdown and transfer plant auxiliaries to SU 2 transformer

Notes:

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

References:

Natural Emergencies 1203.025 change 028

History:

New for 2010 RO/SRO exam

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

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

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

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

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

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

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

Tier:	1	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.7	**SRO Select:**	Yes	**Taxonomy:**	K

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

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

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

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

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

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

Answer:

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

Notes:

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

References:

Technical Specifications bases B2.1.2 amendment # 215

History:

New for 2010 RO/SRO exam

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

Form ES-401-2

	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#	QID	Type
003 Reactor Coolant Pump					X							**K5.05** – The dependency of RCS flow rates upon the number of operating RCP's	2.8*	28	781	D
										X		**A4.08** – RCP cooling water supplies	3.2	29	782	M
004 Chemical and Volume Control											X	**2.1.34 changed to 2.2.38** – Knowledge of conditions and limitations in the facility license	3.6	30	796	N
				X								**K4.03** – Protection of ion exchangers (high letdown temperatures will isolate ion exchangers)	2.8*	31	259	D
005 Residual Heat Removal		X										**K2.01** – RHR Pumps	3.0*	32	786	M
006 Emergency Core Cooling						X						**K6.10** - Valves	2.6	33	783	N
007 Pressurizer Relief/Quench Tank					X							**K5.02** – Method of forming a steam bubble in the PZR	3.1	34	561	D
008 Component Cooling Water								X				**A2.08 changed to A2.01** - Loss of CCW Pump	3.3	35	787	N
010 Pressurizer Pressure Control			X									**K3.02** - RPS	4.0	36	788	N
012 Reactor Protection						X						**K6.10** – Permissive circuits	3.3	37	784	N
											X	**2.1.32** – Ability to explain and apply system limits and precautions	3.8	38	785	N
013 Engineered Safety Features Actuation				X								**K4.10** – Safeguards equipment control reset	3.3	39	142	D
022 Containment Cooling									X			**A3.01** – Initiation of safeguards mode of operation	4.1	40	135	D
025 Ice Condenser												**Not Selected**	**N/A**			
026 Containment Spray	X											**K1.01** - ECCS	4.2	41	78	D
039 Main and Reheat Steam								X				**A2.04** – Malfunctioning steam dump	3.4	42	202	D
059 Main Feedwater									X			**A3.03** – Feed water pump suction flow pressure	2.5	43	195	D
				X								**K4.16** – Automatic trips for MFW pumps	3.1	44	789	N
061 Auxiliary/Emergency Feedwater							X					**A1.04 changed to A1.01** – S/G level	3.9	45	270	D

PWR Examination Outline

Form ES-401-2

Plant Systems - Tier 2/Group 1 (RO)

	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#	QID	Type
062 AC Electrical Distribution											X	**2.4.35** – Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	3.8	46	790	N
		X										**K2.01** – Major system loads	3.3	47	316	D
063 DC Electrical Distribution			X									**K3.02** – Components using DC control power	3.5	48	86	D
064 Emergency Diesel Generator		X										**K2.01** – Air compressor	2.7	49	791	N
	X											**K1.05** – Starting air system	3.4	50	792	N
073 Process Radiation Monitoring					X							**K5.01** – Radiation theory, including sources, types, units, and effects	2.5	51	672	R
076 Service Water										X		**A4.02** – SWS valves	2.6	52	793	D
							X					**A1.02** – Reactor and turbine building closed cooling water temperatures	2.6	53	794	N
078 Instrument Air	X											**K1.03 changed to K1.02** – Service air	2.7	54	535	D
103 Containment										X		**A4.06** – Operation of the containment personnel airlock	2.7	55	795	D
K/A Category Point Totals:	3	3	2	3	3	2	2	2	2	3	3	Group Point Total:			28	

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

QID: 0107 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Direct **Originator:** JCork

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

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

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

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

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

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

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

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

The plant is operating at 60% power with Delta Tc and SG/RX Master stations in Hand.
All other ICS stations are in Auto.

If one RCP has to be tripped due to high vibration, how will the ICS respond?
(Assume no operator action other than tripping the RCP.)

A. The ICS will runback the plant to 45% load at 50%/min.

B. No change to FW will occur since the SG/RX Master is in Hand.

C. Demand is less than the RCP runback limit, no changes occur to FW.

D. The RC flow difference will re-ratio the FW flow demand.

Answer:

D. The RC flow difference will re-ratio the FW flow demand.

Notes:

Following an RCP trip Delta Tc will re-ratio feedwater demands, therefore answer (d) is correct. Answer (a) is incorrect since the plant is operating below the runback setpoint, while (b) and (c) are incorrect because they state that ICS will not re-ratio feedwater demands.

References:

1203.012F, Chg. 028
STM 1-64, Rev.11

History:

Modified QID 4408 for use on 1998 RO/SRO Exam.
Modified for use in 2001 RO Exam.
Selected for 2005 Jon Gray RO re-exam.
Selected for 2010 RO/SRO exam.

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

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

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

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

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

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

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

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

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

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

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

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

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

B. Seal injection flow is low.

C. RCP lift oil pressure is low.

D. RCP motor cooling flow is low.

Answer:

D. RCP motor cooling flow is low.

Notes:

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

References:

1103.006 change 032

History:

Modified from QID 559
Selected for 2010 RO/SRO exam

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

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

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

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

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

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

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

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

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

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

REFERENCE PROVIDED

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

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

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

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

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

Answer:

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

Notes:

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

REFERENCE PROVIDED FOR THIS QUESTION

References:

1104.003 change 046
TRM 3.5.1 rev 16

History:

New for 2010 RO/SRO exam

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

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

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

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

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

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

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

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

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

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

What is the function of the temperature interlock associated with RCS letdown?

A. Prevents letdown fluid from flashing to steam when pressure is reduced by closing CV-1221 (letdown isolation).

B. Prevents exceeding letdown piping thermal limits by shutting CV-1213 & 1215 (letdown cooler inlet MOV).

C. Prevents degrading T36A/B resin by shutting CV-1221 (letdown isolation).

D. Prevents exceeding letdown cooler capacity by shutting CV-1213 & 1215 (letdown cooler inlet MOV).

Answer:

C. Prevents degrading T36A/B resin by shutting CV-1221 (letdown isolation).

Notes:

"A" is incorrect, this is the function of the letdown coolers.
"B" is incorrect, interlock doesn't close the inlets and piping limits will not be exceeded before the resin is damaged.
"C" is correct
"D" although the letdown cooler capacity is exceeded when temperature is exceeded the interlock doesn't close the inlet valves.

References:

1104.002 Rev 051-02-0
STM1-04 Rev 5

History:

Used in 1999 exam.
Selected for 2010 RO/SRO exam

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

QID: 0786 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** Modified **Originator:** S. Pullin
TUOI: A1LP-RO-ELECD **Objective:** 11 **Point Value:** 1

Section: 3.4 **Type:** Heat Removal From Reactor Core
System Number: 005 **System Title:** Residual Heat Removal System
Description: Knowledge of bus power supplies to the following: RHR pumps.

K/A Number: K2.01 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.2	**SRO Select:**	Yes	**Taxonomy:**	K

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

Given:

- Plant is in Mode 6
- P-34B Decay Heat pump is running

Which of the following would cause a loss of Decay Heat Removal?

A. A-1 voltage of 2475 volts

B. A-2 voltage of 2475 volts

C. B-5 voltage of 428 volts

D. B-6 voltage of 428volts

Answer:

D. B-6 voltage of 428volts

Notes:

"B" Decay Heat Removal Pump is powered from A-4 via A-2. An undervoltage on the A buses or B buses will trip A-409 (A4 feeder breaker). The undervoltage setpoint for A-4 is 2450 volts. The undervoltage setpoint for B-6 is 429 volts. Therefore, "a","b", and "c" are incorrect.

References:

OP-1107.002 Change 025
STM 1-32 Figure 32.68 Rev 34

History:

Modified from QID 0293
Selected for 2010 RO/SRO Exam

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

QID: 0783 **Rev:** 0 **Rev Date:** 9/10/2009 **Source:** New **Originator:** Possage

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

Section: 3.3 **Type:** Reactor Pressure Control

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

Description: Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Valves

K/A Number: K6.10 **CFR Reference:** 41.7 / 45.7

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

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

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

Given

ESAS Channels 1-6 have actuated.
BWST Outlet Valve CV-1408 fails to open

What effect will this have on the ECCS with no operator action?

A. "A" High Pressure Injection Pump AND "A" Low Pressure Injection Pump will be damaged due to loss of suction.

B. "C" High Pressure Injection Pump AND "B" Low Pressure Injection Pump will be damaged due to loss of suction.

C. "A" High Pressure Injection Pump AND "A" Reactor Building Spray Pump will be damaged due to loss of suction.

D. "C" High Pressure Injection Pump AND "B" Reactor Building Spray Pump will be damaged due to loss of suction.

Answer:

B. "C" High Pressure Injection Pump AND "B" Low Pressure Injection Pump will be damaged due to loss of suction.

Notes:

B. Is the correct answer. CV-1408 is ES actuated open to provide suction to the Green Train ECCS components
A. Is incorrect, these are the RED Train ECCS Components and would not be effected by CV-1408.
C. Is incorrect, these are the RED Train ECCS Components and would not be effected by CV-1408.
D. Is incorrect, ESAS Channels 1-6 do not cause the Reactor Building Spray Pumps to start.

References:

STM 1-05 Rev. 16
STM 1-65 Rev. 5

History:

New for 2010 RO/SRO Exam

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

QID: 0561 **Rev:** 1 **Rev Date:** 8/10/05 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-RCS **Objective:** 21 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 007 **System Title:** Pressurizer Relief Tank/Quench Tank System

Description: Knowledge of the operational implications of the following concepts as they apply to the PRTS: Method of forming a steam bubble in the PZR.

K/A Number: K5.02 **CFR Reference:** 41.5 / 45.7

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

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

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

A plant startup is in progress with a steam bubble being drawn in the Pressurizer.
- Initial Quench Tank pressure is 3 psig.
- RCS pressure 75 psig.
- Pressurizer temperature 320°F.

Which of the following assures that venting and steam bubble formation is complete in the Pressurizer?

A. Quench Tank pressure 7.6 psig after a 3 minute blow of the ERV.

B. Quench Tank pressure 6.2 psig after a 3 minute blow of the ERV.

C. Quench Tank pressure 4.8 psig after a 3 minute blow of the ERV.

D. Quench Tank pressure 3.5 psig after a 3 minute blow of the ERV.

Answer:

D. Quench Tank pressure 3.5 psig after a 3 minute blow of the ERV.

Notes:

"D" is correct with Quench Tank pressure rise less than or equal to 1 psig.
All other choices contain greater than 1 psig pressure rise which indicates nitrogen is still being vented to the Quench Tank.

References:

1103.005, Chg. 036

History:

New for 2005 RO exam, later modified for replacement.
Selected for 2010 RO/SRO exam.

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

QID: 0787 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin

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

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

System Number: 008 **System Title:** Component Cooling Water System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of CCW Pump

K/A Number: A2.01 **CFR Reference:** CFR: 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	C

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

Given:

- 80% power,
- P33A and P33B ICW pumps in service.
- P33C (ICW Pump) out of service

- P33B (ICW Pump) trips

What impact would this have on plant operations, and what actions are required per 1104.028, ICW System Operating Procedure?

A. Loss of Non-Nuc ICW, open ICW cross connect valves CV-2238, CV-2239, CV-2240 and CV-2241

B. Loss of Non-Nuc ICW, close "A" to "B" cross connect valves CV-2238 and CV-2240

C. Loss of Nuc ICW, open ICW cross connect valves CV-2238, CV-2239, CV-2240 and CV-2241

D. Loss of Nuc ICW, close "A" to "B" cross connect valves CV-2238 and CV-2240

Answer:

C. Loss of Nuc ICW, open ICW cross connect valves CV-2238, CV-2239, CV-2240 and CV-2241

Notes:

"C" is correct P33C supplies the Nuc ICW loads, OP-1104.028 has the operator open the suction and discharge cross connect valves to supply both loops with one pump prior to reducing loads.
"A" is incorrect due to Non Nuc ICW loads were never lost
"B" is incorrect due to Non Nuc ICW loads were never lost
"D" is incorrect due to procedure has you open the valves and not close them

References:

OP-1104.028 Change 026

History:

New question, selected for 2010RO/SRO exam.

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

QID: 0788 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin
TUOI: A1LP-RO-RPS **Objective:** 5 **Point Value:** 1

Section: 3.3 **Type:** Reactor Pressure Control
System Number: 010 **System Title:** Pressurizer Pressure Control System (PZR PCS)
Description: Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RPS

K/A Number: k3.02 **CFR Reference:** 41.7 /45.6

Tier:	2	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	K

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

Given:

- 100% power,
- "A" MFW Pump trips

- PZR Spray valve (CV-1008) will not open.

What effect would this pressurizer control system malfunction have on the plant?

A Reactor trip due to AMSAC

B. Reactor trip due to anticipatory trip from RPS on loss of MFW pumps

C. Reactor trip due to High Power/Imbalance/Flow

D. Reactor trip due to High RCS Pressure

Answer:

D. Reactor trip due to High RCS Pressure

Notes:

A is incorrect because total feedwater flow will remain above trip setpoint
B is incorrect because only one MFW pump is tripped
C is incorrect because the flow in this coice refers to RCS flow
D is correct, without the spray valve opening RCS pressure will rise to the trip setpoint

References:

OP-1202.001 Change 31

History:

New selected for 2010 RO/SRO exam.

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

QID: 0784 **Rev:** 0 **Rev Date:** 9/10/2009 **Source:** New **Originator:** Possage

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

Section: 3.7 **Type:** Instrumentation

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

Description: Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Permissive circuits

K/A Number: K6.10 **CFR Reference:** 41.7 / 45.7

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 37 **SRO:** 37

Given:

The plant is at 100% power

I&C is troubleshooting RPS

"B" RPS is in Manual Bypass

The Power/Imbalance/Flow Trip bistable in Channel "A" has been pulled from the cabinet.

What would be the effect of a failure in the "B" RPS permissive circuitry that caused a short which de-energizes the "B" RPS Cabinet?

A. RPS would be in a 2 out of 3 coincidence trip logic

B. RPS would be in a 1 out of 2 coincidence trip logic

C. Reactor Trip would occur

D. Only RPS Channel "A" will be tripped.

Answer:

C. Reactor Trip would occur

Notes:

C. Is correct. The conditions given would result in the "A" Channel being tripped, when "B" is de-energized it would also be tripped and make up the logic to trip the reactor.
A and B are incorrect because the logic to trip the reactor has already been met.
D is incorrect, pulling the Power/Imbalance/Flow Trip bistable would trip the channel but due to de-energizing "B" RPS cabinet the reactor would trip.

References:

STM 1-63 Rev. 7

History:

Modified from Exam Bank ANO-OPS1-1670
Selected for 2010 RO/SRO exam

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

QID: 0785 **Rev:** 0 **Rev Date:** 9/10/2009 **Source:** New **Originator:** Possage

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

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

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

Description: Ability to explain and apply system limits and precautions.

K/A Number: 2.1.32 **CFR Reference:** 41.10 / 43.2 / 45.12

Tier: 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 38 **SRO:** 38

Given:

The plant is at 100% power

Which of the following is an applicable Limit & Precaution for the RPS System and why?

A. When testing an RPS protection channel any EFIC Channel can be placed in maintenance bypass simultaneously because RPS has no effect on an EFIC Channel.

B. Placing two RPS Channels in test simultaneously is allowed as long as Shift Manager permission is obtained because will have no effect on RPS operation but requries an LCO entry.

C. The key operated shutdown bypass switch associated with each RPS Channel can be used during power operation becuase this will have no effect on RPS operation but requries an LCO entry.

D. Only one RPS Channel Bypass Key shall be accessible for use in the control room because only one RPS Channel shall be key locked in the untripped state at any one time.

Answer:

D. Only one RPS Channel Bypass Keyshall be accessible for use in the control room because only one RPS Channel shall be key locked in the untripped state at any one time.

Notes:

D is correct, per limits and precautions of 1105.001
A, B and C are all incorrect versions of limits and precautions for 1105.001

References:

OP-1105.001 Change 024

History:

New Selected for 2010 RO/SRO exam

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

QID: 0142 **Rev:** 0 **Rev Date:** 10/28/97 **Source:** Direct **Originator:** G. Giles

TUOI: AA51002-012 **Objective:** 21 **Point Value:** 1

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

System Number: 013 **System Title:** Engineered Safety Features Actuation System(ESFAS)

Description: Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Safeguards equipment control reset.

K/A Number: K4.10 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	3.3	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 39 **SRO:** 39

Under what conditions can the Control Board Operator bypass or defeat a component automatically actuated by ESAS?

A. Bypassing or defeating a component automatically actuated by ESAS is not allowed.

B. The Control Board Operator, after careful consideration, determines that the component is no longer required.

C. ONLY when procedurally directed by the Emergency Operating or the Abnormal Operating procedures.

D. After it is determined that the component is no longer needed and approval is obtained from the SM/CRS.

Answer:

D. After it is determined that the component is no longer needed and approval is obtained from the SM/CRS.

Notes:

[A] is incorrect, provisions are made for this action.
[B] is partially correct, the component must not be needed but the CBO cannot make this decision on his own.
[C] is only one of the directions where a component can be bypassed/reset, CRS/SS permission is the other.
[D] contains all correct elements, lack of need and supervisory (SRO) permission.

References:

OP-1202..012 Change 008

History:

Taken from Exam Bank QID # 4791
Used in A. Morris 98 RO Re-exam
Previously used under K/A: 3.2 / Reactor Coolant System Inventory Control / 013 / Engineered Safety Features Actuation System / A4.02 / Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels. / CFR: 41.7 / 45.5 to 45.8 / RO: 4.3 / SRO: 4.4
Used on 2004 RO/SRO Exam (K/A T2 G1 013 K4.06)
Selected for the 2010 RO/SRO exam

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

QID: 0135 **Rev:** 1 **Rev Date:** 4/7/05 **Source:** Direct **Originator:** B. Short

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 022 **System Title:** Containment Cooling System

Description: Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation.

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

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

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

Question: **RO:** 40 **SRO:** 40

A LOCA has occurred.
Reactor Building (RB) pressure is 47 psia.

Which ESAS channels have actuated the RB cooling units and what is the correct RB cooling alignment?

A. ES channels 3 & 4, VSF-1A, 1B, 1C, & 1D running with service water aligned to the cooling coils.

B. ES channels 3 & 4, VSF-1A, 1B, 1C, 1D, & 1E running with chilled water aligned to the cooling coils.

C. ES channels 5 & 6, VSF-1A, 1B, 1C, & 1D running with service water aligned to the cooling coils.

D. ES channels 5 & 6, VSF-1A, 1B, 1C, 1D, & 1E running with chilled water aligned to the cooling coils.

Answer:

c. ES channels 5 & 6, VSF-1A, 1B, 1C, & 1D running with service water aligned to the cooling coils.

Notes:

ESAS channels 5 & 6 actuate RB cooling fans VSF-1A through 1D and also cause the bypass dampers to drop which allows air to bypass the 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. 9

History:

Developed for use in 98 RO Re-exam
Selected for 2005 RO exam.
Selected for 2010 RO/SRO exam

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

QID: 0078 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

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

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray System (CSS)

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

K/A Number: K1.01 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

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

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

Question: **RO:** 41 **SRO:** 41

If an ESAS occurs simultaneously with a Loss of Offsite Power, the start of RB Spray pumps is delayed by 35 sec. Why?

A. To allow the EDGs to come up to speed.

B. To allow SW pumps to start for spray pump cooling.

C. To prevent overload of the EDGs.

D. To prevent water hammer of the spray headers.

Answer:

C. To prevent overload of the EDGs.

Notes:

With an ES signal present, ES loads will sequence on to the EDG to prevent overload, therefore "C" is correct. (a), (b) and (d) are reasons for other aspects of RB spray operation but are not applicable to the basis for the time delay.

References:

1107.002, Chg. 025

History:

Developed for 1998 RO/SRO Exam.
Used in A. Morris 98 RO Re-exam
Selected for 2005 Jon Gray RO re-exam.
Selected for the 2010 RO/SRO exam.

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

QID: 0202 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** R. Walters

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

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

System Number: 039 **System Title:** Main and Reheat Steam System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump.

K/A Number: A2.04 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 42 **SRO:** 42

Given:

- A plant startup is in progress with the reactor critical below the point of adding heat.
- "B" OTSG Turbine Bypass Valve (CV-6688) fails full OPEN and is unable to be closed with the handjack.
- Tave 524 degrees and dropping
- Pressurizer level 205 inches and dropping
- RCS pressure 2120 psig and dropping

What is the proper course of action?

A. Close CV-6688 manual isolation valve MS-2A and maintain the reactor critical using 'A' OTSG Turbine Bypass Valve to control RCS temperature and pressure.

B. Continue the reactor startup maintaining startup rate <1 DPM while continuing to monitor primary and secondary plant parameters.

C. Go directly to 1203.003, OVERCOOLING for actions to mitigate the oversteaming of the 'B' OTSG.

D. Trip the reactor and follow the guidance of 1202.001 REACTOR TRIP.

Answer:

D. Trip the reactor and follow the guidance of 1202.001 REACTOR TRIP.

Notes:

(A.) is incorrect. You would not have time to take this action and the operator should take conservative action of tripping the reactor..
(B.) is incorrect. With the reactor below the point of adding heat with a stuck open TBV, this would not be possible.
(C.) is incorrect. This will be the ultimate tab that you will end up in, however, it is necessary to trip the reactor first and progress through the Reactor Trip EOP.
(D.) is correct. Taking the conservative action of tripping the reactor is appropriate due to being below the minimum temperature for criticality and the inability to maintain SUR below 1 DPM.

References:

1102.008 (Rev 023), Approach to Criticality, pages 4&5

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

QID: 0195 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** L. Kilby

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

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

System Number: 059 **System Title:** Main Feedwater System

Description: Ability to monitor automatic operation of the MFW, including: Feedwater pump suction flow pressure

K/A Number: A3.03 **CFR Reference:** 41.7 / 45.5

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 2.6 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 43 **SRO:** 43

Unit 1 is operating at 100% power with no abnormal conditions or alignments.
'B' MFP SUCT PRESS LO (K07-C8) annunciator is received.

Where can the Control Room Operators read the 'B' MFW pump suction pressure WITHOUT leaving the control room?

A. The 'B' MFP Lovejoy Operator Control Station (OCS).

B. 'B' MFP Suction Pressure (PI-2830) indicator.

C. 'B' MFP Suction Pressure computer point (P2830)

D. The Operator Information Touchscreen (OIT).

Answer:

C. 'B' MFP Suction Pressure computer point (P2830)

Notes:

(a.) & (d.) are incorrect. These panels are located in the control room, however, MFP suction pressure is not available on these panels.
(b.) is incorrect. This indicator is located outside the control room.
(c.) is correct. This computer point is found on the Plant Computer and the SPDS computer both of which are available in the control room.

References:

STM 1-19, Rev. 11

History:

Developed for use in 98 RO Re-exam
Selected for 2005 RO exam
Selected for 2010 RO/SRO exam

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

QID: 0789 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin

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

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

System Number: 059 **System Title:** Main Feedwater (MFW) System

Description: Knowledge of MFW design feature(s) and / or interlock(s) which provide for the following: automatic trips for MFW pumps.

K/A Number: K4.16 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 44 **SRO:** 44

Given:

- 100% power

Which of the following design features provide an automatic trip of the Main Feed Water Pump?

A. Main Feed Water Pump suction pressure reading 220 psig for 45 seconds

B. Main Feed Water Pump bearing oil pressure reading 15 psig

C. Main Feed Water Pump discharge pressure reading 1360 psig

D. Main Feed Water Pump vibration reading 14 mils

Answer:

C. Main Feed Water Pump discharge pressure reading 1360 psig

Notes:

A is incorrect, suction pressure would have to be less than 200 psig for 40 seconds.
B is incorrect, bearing oil pressure of 15 psig would cause an alarm but pressure must be less than 10 psig for a trip.
C is correct, pump discharge pressure of 1350 psig would result in a pump trip
D is incorrect, the high vibration trip is bypassed when the pump is in operation

References:

STM 1-24 Rev. 11

History:

New selected for 2010 RO/SRO exam

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

QID: 0270 **Rev:** 1 **Rev Date:** 11/8/05 **Source:** Direct **Originator:** D. Slusher

TUOI: A1LP-RO-EFIC **Objective:** 29 **Point Value:** 1

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

System Number: 061 **System Title:** Auxiliary/Emergency Feedwater System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: S/G level.

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

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 2.5

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

Question: **RO:** 45 **SRO:** 45

The EFIC automatic fill rate is designed to prevent overcooling.
With the plant in a degraded power condition and given a SG pressure of 885 psig, determine the proper OTSG fill rate by EFIC
for the EFW system:

A. ~3"/min

B. ~4"/min

C. ~5"/min

D. ~6"/min

Answer:

B. ~4"/min

Notes:

OTSG fill rate is adjusted so that OTSG levels raise at 2 inches/minute at OTSG pressure of 800 psig and 8 inches/minute at OTSG pressure of 1050 psig. This limits the overcooling effects of feeding OTSGs with EFW. At 885 psig OTSG fill rate will be 4 inches/minute. "b" is the correct answer.

References:

1105.005, Chg. 032

History:

Used in 1999 exam.
Direct from ExamBank, QID# 92 used in class exam
Modified for 2005 Jon Gray RO re-exam.
Selected for 2010 RO/SRO exam

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

QID: 0790 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S Pullin

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

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** A.C. Electrical Distribution

Description: Knowledge of local auxiliary operator tasks during emergency and the resultant operation effects.

K/A Number: 2.4.35 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 2 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** 46 **SRO:** 46

Given:

Unit 1 is in a Blackout condition.
Voltage has been recovered on SU#2 and is 155 kV

To restore power to A-3 and A-4, what action along with its purpose is required by the Auxiliary Operator?

A. Perform Attachment 1, Blackout Breaker Alignment and UV Relay Defeat, to defeat UV interlocks to allow for starting of equipment necessary to protect the core.

B. Perform Attachment 1, Blackout Breaker Alignment and UV Relay Defeat, to prevent excess current during starting of the motors.

C. Perform Attachment 2, Recovery from Blackout Breaker Alignment and UV Relay Defeat, to allow for starting of equipment necessary to protect the core.

D. Perform Attachment 2, Recovery from Blackout Breaker Alignment and UV Relay Defeat, to allow Unit 2 to tie on non-vital loads on SU#2.

Answer:

A. Perform Attachment 1, Blackout Breaker Alignment and UV Relay Defeat, to defeat UV Close Permissive interlocks to allow for starting of equipment necessary to protect the core.

Notes:

A is correct, with degraded voltage on SU#2, Att. 1 is required to defeat the UV interlocks.
B is incorrect, Att. 1 would have no effect on actual starting current for motors
C & D are incorrect, Att. 2 will only be performed when SU#2 voltage is greater than 158 kV.

References:

OP-1202.028 Change 010

History:

New selected for 2010 RO/SRO exam.

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

QID: 0316 **Rev:** 0 **Rev Date:** 9/5/99 **Source:** Direct **Originator:** J Haynes

TUOI: ANO-1-LP-RO-MU **Objective:** 3.5 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** A.C. Electrical Distribution

Description: Knowledge of bus power supplies to the following: Major system loads.

K/A Number: K2.01 **CFR Reference:** CFR: 41.7

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

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

Question: **RO:** 47 **SRO:** 47

Which of the following would explain why a loss of bus A1 will cause CV-1206 (RC Pump Seal Injection Block Valve) to close?

(Assume plant is at 100% power)

A. P36A (HPI) pump was the in-service pump.

B. Loss of instrument air to Seal Injection Control Valve, CV-1207.

C. P36C (HPI) pump was the in-service pump.

D. Loss of instrument air to Pressurizer Level Control valve CV-1235.

Answer:

A. P36A (HPI) pump was the in-service pump.

Notes:

"a" is correct, if P36A was the in-service pump, then a loss of A1 would cause a loss of A3, P-36A would cease to run, and CV-1206 would close when Seal Injection flow dropped to less than 22 gpm.
"b" is incorrect, CV-1207 fails open on a loss of Instrument Air.
"c" is incorrect, a loss of A1 would not affect P36C's power supply, bus A4.
"d" is incorrect, CV-1235 fails as-is on a loss of Instrument Air.

References:

1203.026, Change 11

History:

Used in 1999 exam.
Modified from ExamBank, QID# 3716.
Selected for 2010 RO/SRO exam.

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

QID: 0086 **Rev:** 0 **Rev Date:** 7/11/98 **Source:** Direct **Originator:** JCork

TUOI: A1LP-RO-ELECD **Objective:** 37 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** D.C. Electrical Distribution

Description: Knowledge of the effect that a loss or malfunction of the dc electrical system will have on the following: Components using dc control power.

K/A Number: K3.02 **CFR Reference:** 41.7 / 45.6

Tier:	2	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 48 **SRO:** 48

The plant is at 100% power.

Which of the following DC buses/panels, if de-energized, would cause a reactor trip?

A. Panel D41

B. Panel RA1

C. MCC D15

D. Panel D21

Answer:

B. Panel RA1

Notes:

Only "B" is capable of causing a reactor trip due to loss of two RCP contact monitors.
The others would cause a loss of vital equipment capability but as seen in Att. J of 1107.004, they would not cause a trip.

References:

1107.004, Chg. 016

History:

Developed for 1998 RO exam
Used in A. Morris 98 RO Re-exam
Selected for use in 2005 RO exam, but not used.
Selected for 2010 RO/SRO exam.

QID: 0791 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin

TUOI: A1LP-RO-EDG **Objective:** 19a **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators (ED/G)

Description: Knowledge of bus power supplies to the following: Air Compressor

K/A Number: K2.01 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.1	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 49 **SRO:** 49

What is the power supply to Emergency Diesel Generator Starting Air Compressors, C4A1 and C4B2?

A. B31 and B41

B. B32 and B42

C. B51 and B61.

D. B52 and B62

Answer:

A. B31 and B41

Notes:

A is correct, the other choices are alternate possibilities.

References:

OP-1107.001 Change 73

History:

New for 2010 RO/SRO exam.

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

QID: 0792 **Rev:** 0 **Rev Date:** 9/14/2009 **Source:** New **Originator:** S. Pullin

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

Section: 3.6 **Type:** Electrical

System Number: 064 **System Title:** Emergency Diesel Generators (ED/G)

Description: Knowledge of the physical connections and / or cause-effect relationships between the ED/G system and the following systems: Starting air system.

K/A Number: K1.05 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 50 **SRO:** 50

Given:

Plant at 100%
Performing #1 EDG monthly surveillance per 1104.036 Supplement 1

The CBOT presses the start pushbutton on C10
K01-B2, EDG 1 OVERCRANK, alarms

What is the cause of the alarm and how long did the starting air system attempt to start the engine?

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

B. #1 EDG did not exceed 300 rpm in 8 seconds and air start motors engaged for 45 seconds.

C. #1 EDG did not exceed 30 rpm in 45 seconds and air start motors engaged for 2.5 seconds.

D. #1 EDG did not exceed 30 rpm in 8 seconds and air start motors engaged for 8 seconds.

Answer:

A. #1 EDG did not exceed 300 rpm in 45 seconds and air start motors engaged for 8 seconds.

Notes:

A is correct, due to meeting the annunciator logic
B, C, and D are variations of the control logic for the starting air to the engine

References:

STM-1-31 rev 10
1203.012A change 038

History:

New 2010 RO/SRO exam

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

QID: 0672 **Rev:** 0 **Rev Date:** 12/16/06 **Source:** Repeat **Originator:** Possage

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

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radiation Monitoring System

Description: Knowledge of the operational implications of the following concepts as they apply to the PRM System: Radiation theory, including sources, types, units, and effects.

K/A Number: K5.01 **CFR Reference:** 41.5 / 45.7

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 51 **SRO:** 51

What type of detector is used by the Main Condenser Air Discharge Radiation Monitor to monitor for steam generator tube leaks?

A. Scintillation Detector

B. Geiger - Mueller Detector

C. Ion Chamber Detector

D. Beta Radiation Detector

Answer:

A. Scintillation Detector

Notes:

"A" is correct. The Main Condenser Air Dischagre Radiation Monitor is a scintillation detector.
"B" is incorrect. Area Monitors are G-M Detectors
"C" is incorrect. Ion chambers are used for RP surveys
"D" is incorrect. The Penentration Ventilation Monitors are beta sensitive monitors.

References:

STM 1-62 Rev. 11

History:

New for 2007 RO Exam.
Selected for 2010 RO/SRO exam

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

QID: 0793 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** Direct **Originator:** S Pullin

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

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

System Number: 076 **System Title:** Service Water System (SWS)

Description: Ability to manually operate and / or monitor in the control room SWS valves

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

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

Group: 1 **SRO Imp:** 2.6 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** 52 **SRO:** 52

When starting Service Water Pump P-4A after maintenance, you observe the following symptoms.

- Pump start is indicated by normal light indication above pump control HS on.
- Annunciator K10-B3 "SW DISCH PRESS HI" alarms.
- Valve position indication in the control room indicates proper valve alignment.
- SW Bay levels are 338 feet
- No change in SW flow or discharge pressure indications on the SPDS Diagnostics screen.
- No change in SW Loop pressure indications on control room panel C09.

Which of the following is the most likely cause of these symptoms?

A. The pump discharge valve was not opened when returned to service.

B. Warm weather conditions cause low demand from ACW/SW.

C. P-4A cannot pump into the system because of high system pressure from the other(running) pump.

D. P-4A is running without sufficient NPSH to pump water into the SW System.

Answer:

A. The pump discharge valve was not opened when returned to service.

Notes:

A is the correct answer. With the local discharge valve closed, the SW Pump would not be able to pump water to the loop, but since the discharge pressure switch is between the pump and discharge vlave, therefore a high discharge pressure would be seen.
B is incorrect, warm water would cause a high demand for flow not a low demand.
C is incorrect, if the maintenance performed had caused low discharge pressure such that the pump was unable to pump water to the loop, there would not be a high discharge pressure alarm.
D is incorrect, with a bay level of 338 feet, suction pressure would be (356.5-338)0.433= 8 psig which is adequate.

References:

OP-1203.012I Change 046

History:

Direct ANO Exam bank QID ANO-OPS1-3284
Selected for 2010 RO/SRO exam

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

QID: 0794 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** New **Originator:** S. Pullin

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

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

System Number: 076 **System Title:** Service Water System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures.

K/A Number: A1.02 **CFR Reference:** 41.5 / 45.5

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

Group: 1 **SRO Imp:** 2.6 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 53 **SRO:** 53

What would be the effect to service water pressure due to an inadvertent actuation of ES Channel 5 ?

A. Service Water Pressure would drop due to SW valves to the EDG Coolers opening.

B. Service Water Pressure would drop due to SW valves to the RB Coolers opening.

C. Service Water Pressure would rise due to ACW isolation valve closing.

D. Service Water Pressure would rise due to SW to ICW isolations closing.

Answer:

B. Service Water Pressure would drop due to SW valves to the RB Coolers opening.

Notes:

A is incorrect, SW to EDG Coolers open on diesel start. EDG starts on Channels 1 or 2
B is correct, ES Channel 5 will align SW to the RB Coolers
C is incorrect, ACW isolation valve would close on ES Channel 2
D is incorrect, SW to ICW isoaltion valve will close on ES Channels 1 and 2

References:

STM 1-65 Rev. 5

History:

New, Selected for 2010 RO/SRO exam

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

QID: 0535 **Rev:** 1 **Rev Date:** 10/13/2005 **Source:** Direct **Originator:** J.Cork
TUOI: A1LP-RO-AOP **Objective:** 3 **Point Value:** 1

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

System Number: 078 **System Title:** Instrument Air System

Description: Knowledge of the physical connections and / or cause-effect relationships between the IAS and the following systems: Service Air

K/A Number: K1.02 **CFR Reference:** 41.7 / 45.5

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.8	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 54 **SRO:** 54

Instrument Air pressure has dropped to 50 psig.

Which of the following manual or automatic actions should be performed or will occur in response to the low Instrument Air pressure?

Note: All actions for higher pressures have been completed at the required pressure and answer the question considering only the action for the current pressure.

A. Service Air to Instrument Air cross-connect automatically opens.

B. Unit 1 to Unit 2 Instrument Air cross-connect automatically opens.

C. Trip Reactor, actuate EFW and MSLI on both SGs.

D. Close Letdwon Cooler Outlet to isolate letdown

Answer:

A. Service Air to Instrument Air cross-connect automatically opens.

Notes:

"B" is incorrect, this valve is closed when either unit IA pressure reaches 60 psig.
"A" is correct, this automatically occurs when pressure drops to 50 psig.
"C" is incorrect, this would not be done until pressure was less than 35 psig.
"D" is incorrect, this would not be done until pressure was less than 35 psig

References:

1104.025, Chg. 014

History:

Developed for 1998 RO exam (similar to QID 102)
Modified question for A. Morris 98 RO Re-exam
Modified for J. Gray 2005 re-exam.
Selected for 2010 RO/SRO exam.

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

QID: 0795 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** Direct **Originator:** S. Pullin

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

Section: 3.5 **Type:** Containment Integrity

System Number: 103 **System Title:** Containment System

Description: Ability to manually operate and / or monitor in the control room: Operation of the containment personnel airlock door

K/A Number: A4.06 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 3

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

Question: **RO:** 55 **SRO:** 55

Given:

Plant refueling is in progress

The Reactor Building Coordinator calls the control room and reports the following:
The inner door of the reactor building personnel hatch will not close
The outer door is operable

In accordance with Technical Specifications for Refueling Operations, how does this affect fuel movement?

A. Irradiated fuel movement in the reactor building and auxiliary building must be suspended.

B. Irradiated fuel movement in the reactor building must be suspended.

C. Irradiated fuel movement in the auxiliary building must be suspended.

D. Irradiated fuel movement may continue without restriction.

Answer:

D. Irradiated fuel movement may continue without restriction.

Notes:

D is correct, fuel movement may continue in both the Reactor Building and Aux Building provided one of the air lock doors is capable of being closed.
A, B, and C are incorrect due to the outer door being operable.

References:

T.S. 3.9.3 Amendment No. 215

History:

Direct from ANO exam bank ANO-OPS1-6622
Selected for 2010 RO/SRO exam.

ES-401												PWR Examination Outline Plant Systems - Tier 2/Group 2 (RO)			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	#	QID	Type
001 Control Rod Drive		X										**K2.05 changed to K2.02 –** One-line diagram of power supply to trip breakers	3.6	56	429	D
002 Reactor Coolant								X				**A2.01-** Loss of coolant inventory	4.3	57	604	D
011 Pressurizer Level Control									X			**A3.03-** Charging and letdown	3.2	58	797	N
014 Rod Position Indication				X								**K4.05 –** Rod hold interlocks	3.1	59	308	D
015 Nuclear Instrumentation			X									**K3.04** - ICS	3.4	60	299	D
016 Non-nuclear Instrumentation	X											**K5.01- changed to K1.01** RCS	3.4	61	77	D
017 In-core Temperature Monitor						X						**K6.01-** Sensors and detectors.	2.7	62	240	D
027 Containment Iodine Removal												**Not selected**	**N/A**			
028 Hydrogen Recombiner and Purge Control												**Not selected**	**N/A**			
029 Containment Purge												**Not selected**	**N/A**			
033 Spent Fuel Pool Cooling												**Not selected**	**N/A**			
034 Fuel Handling Equipment												**Not selected**	**N/A**			
035 Steam Generator												**Not selected**	**N/A**			
041 Steam Dump/Turbine Bypass Control												**Not selected**	**N/A**			
045 Main Turbine Generator										X		**A4.06-** Turbine stop valves	2.8	63	138	D
055 Condenser Air Removal												**Not selected**	**N/A**			
056 Condensate												**Not selected**	**N/A**			
068 Liquid Radwaste												**K4.01-** Safety and environmental precautions for handling hot, acidic, and radioactive liquids **Rejected system to 014 Rod Position Indication**	**N/A**			
071 Waste Gas Disposal												**K3.05 –** ARM and PRM systems **Rejected system to 015 Nuclear Instrumentation**	**N/A**			
072 Area Radiation Monitoring												**Not selected**	**N/A**			
075 Circulating Water											X	**2.4.11-** Knowledge of abnormal condition procedures-	4.0	64	798	N
079 Station Air												**Not selected**	**N/A**			
086 Fire Protection							X					**A1.01-** Fire header pressure	2.9	65	542	D
K/A Category Point Totals:	1	1	1	1	0	1	1	1	1	1	1	Group Point Total:		10		

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

QID: 0429 **Rev:** 0 **Rev Date:** 4/30/2002 **Source:** Direct **Originator:** S.Pullin

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

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive System

Description: Knowledge of bus power supplies to the following: One-line diagram of power supply to trip breakers

K/A Number: K2.02 **CFR Reference:** 41.7

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

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

Question: **RO:** 56 **SRO:** 56

If breaker B631 opened while operating at 100% power, the response of the Control Rod Drive system would be:

A. A ratchet trip of all regulating rods since half of the power supply has been removed.

B. No effect on regulating rods, safety rods are held by a single phase (CC) energized.

C. A ratchet trip of the safety rods due to a single phase remaining energized.

D. A trip of all safety rods since the main power has been removed.

Answer:

B. No effect on regulating rods, safety rods are held by a single phase (CC) energized.

Notes:

B is correct. The one-line diagram shows the power supply configuration from A-501 providing power to the CC phase on the DC hold bus which will maintain the safety rods out. Regulating rods are not effected normal movement will be supplied by the Bus 2 power supplied by A-501.

References:

STM 1-02, Control Rod Drive System, page 9, step 2.4

History:

Direct from regular exambank QID 4208.
Selected for use in 2002 RO/SRO exam.
Selected for 2010 RO/SRO exam.

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

QID: 0604 **Rev:** 0 **Rev Date:** 6/30/05 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LPR-RO-RCS **Objective:** 5 **Point Value:** 1

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

System Number: 002 **System Title:** Reactor Coolant System (RCS)

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of coolant inventory.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

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

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** 57 **SRO:** 57

A reactor trip has occurred and the CRS is directing actions per 1202.001, Reactor Trip.

Assume all actions have been performed when required by system parameters.

5 minutes later the following is reported:

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 Repetitive Task (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 Repetitive Task (RT-2).

Notes:

Answer "A" is correct, this is done when level is < 30" per 1202.001.
Answer "B" is incorrect, this was done early in the procedure, shortly after immediate actions.
Answer "C" is incorrect, this was done earlier when level was < 50".
Answer "D" is incorrect, CV-1235 is operating properly in Auto, taking it to hand would not help.

References:

1202.001, Chg. 031

History:

New for 2005 RO exam, modified as a replacement question.
Selected for 2010 RO/SRO exam.

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

QID: 0797 **Rev:** 0 **Rev Date:** 9/15/2009 **Source:** New **Originator:** S. Pullin

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

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

System Number: 011 **System Title:** Pressurizer Level Control System (PZR LCS)

Description: Ability to monitor automatic operation of the PZR LCS, including: Charging and letdown.

K/A Number: A3.03 **CFR Reference:** 41.7 /45.5

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

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

Question: **RO:** 58 **SRO:** 58

Given:

Plant at 100%
Letdown flow 80 gpm indicated on FI-1236
Letdown pressure 50 psig on PI-1237

CV-1244 and CV-1245 Letdown DI Inlet Isolation valves lose power.

With no operator action what would be the expected automatic response of the pressurizer level control system ?

A. PI-1237 would read 50 psig and Pressurizer level control valve CV-1235 position would close.

B. PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would open.

C. PI-1237 would read 50 psig and Pressurizer level control valve CV-1235 position would open.

D. PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would close.

Answer:

B. PI-1237 would read 150 psig and Pressurizer level control valve CV-1235 position would open.

Notes:

B is correct, due to letdown DI Inlet Isolation Valves fail closed on a loss of power. Which would isolate letdown, letdown pressure would rise to the letdown relief setpoint of 150 psig, causing a LOCA. Pressurizer level would go down causing CV-1235 to open.
A,C, and D are variations of these possible combinations.

References:

STM 1-04 Rev. 9

History:

New for 2010 RO/SRO exam.

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

QID: 0308 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J. Cork

TUOI: ANO-1-LP-RO-CRD **Objective:** 16 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 014 **System Title:** Rod Position Indication System

Description: Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Rod hold interlocks.

K/A Number: K4.05 **CFR Reference:** CFR: 41.5 / 45.7

Tier:	2	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	2.5
Group:	2	**SRO Imp:**	3.3	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 59 **SRO:** 59

Given:

- Plant is at 100% power.
- ICS is in full automatic.

Subsequently, annunciator K07-B3 "ASYM ROD RUNBACK IN EFFECT" alarms.
A check of the PI panel shows that Rod 6 in Group 5 has dropped.

Which of the following alarms or indications would you expect to see on the diamond panel?

A. Sequence Inhibit lamp ON

B. Out Inhibit lamp ON

C. Auto Inhibit lamp ON

D. Manual lamp ON

Answer:

B. Out Inhibit lamp ON

Notes:

"A" is incorrect because the sequence inhibit is generated from relative position indications which do not use absolute position indications.
"B" is correct because the rods are interlocked so that they cannot move outward with an asymmetric rod fault with power greater than 40%.
"C" is incorrect because the rods are in auto and dropped rod is not a condition which will place the CRD system in manual.
"D" is incorrect because the diamond will not automatically revert to manual.

References:

1105.009 Change 32

History:

Developed for 1999 exam.
Selected for 2010 RO/SRO exam

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

QID: 0299 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** J Haynes

TUOI: ANO-1-LP-RO-NI **Objective:** 10 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation System

Description: Knowledge of the effect that loss or malfunction of the NIS will have on the following: ICS

K/A Number: K3.04 **CFR Reference:** 41.7 / 45.6

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** 60 **SRO:** 60

Given:

- The plant is at 80% power.
- The NI SASS mismatch alarm is bypassed due to a mismatch.

What would be the predicted plant response if NI-6 failed to 125%?

A. Control rods move inward, feedwater flows go up.

B. Control rods move inward, feedwater flows go down.

C. Control rods move outward, feedwater flows go up.

D. Control rods move outward, feedwater flow go down.

Answer:

A. Control rods move inward, feedwater flows go up.

Notes:

The mismatch alarm disables the SASS module automatic operation. When NI-6 fails to 125% power, ICS will see NI-6 as the input power. ICS will generate an error to drive rods in. At the same time a cross-limit is generated to keep feedwater balanced with reactor power. Feedwater will go up. Therefore, "B", "C", and "D" are incorrect.

References:

STM 1-64, Integrated Control System, rev 10, page 33, step 2.6.1, page 43, step 2.7

History:

Used in 1999 exam.
Direct from ExamBank, QID# 3723
Selected for 2002 RO exam.
Used on 2004 SRO/SRO Exam.
Selected for 2010 RO/SROexam

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

QID: 0077 **Rev:** 0 **Rev Date:** 9/29/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-1-LP-RO-NNI **Objective:** 5 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation System (NNIS)

Description: Knowledge of the physical connections and/or cause-efffect relationships between the NNIS and the following systems: RCS

K/A Number: K1.01 **CFR Reference:** 41.2 to 41.9/ 45.7 to 45.8

Tier:	2	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.4	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 61 **SRO:** 61

Given:

- Loop A RCS flow 70 E6 lbm/hr
- Loop B RCS flow 63 E6 lbm/hr
- Loop A Tave 578°F
- Loop B Tave 580°F
- Unit Tave 579°F

Which Tave will be selected by the SASS Auto/manual transfer switch and why?

a. Unit Tave due to Loop B flow

b. Loop A Tave due to Loop B flow

c. Loop B Tave due to Loop B flow

d. Unit Tave, flows are within tolerances

Answer:

b. Loop A Tave due to Loop B flow

Notes:

SASS will automatically select the Loop Tave for the Loop with the highest RCS flow should either flow drop below 95%. Normal RCS loop flow is ~70 E6 lbm/hr, therefore Loop B flow is <95% and SASS will select Loop A flow for Tave control, this control function protects the core from excessive heat transfer based upon flux to flow, therefore, (b) is the only correct response.

References:

STM 1-69 (Rev 13), Non-Nuclear Instrumentation System page 12 step 3.3.5

History:

Modified QID 2517 for 1998 RO/SRO Exam.
Used in A. Morris 98 RO Re-exam
Selected for 2002 RO/SRO exam.
Selected for 2010 RO/SRO exam

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

QID: 0240 **Rev:** 0 **Rev Date:** 8-17-99 **Source:** Direct **Originator:** Don Slusher

TUOI: ANO-1-LP-RO-NNI **Objective:** 25 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-Core Temperature Monitor (ITM) System

Description: Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors.

K/A Number: K6.01 **CFR Reference:** CFR: 41.7/45.7

Tier:	2	**RO Imp:**	2.7	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 62 **SRO:** 62

Given:

- Plant is at 100% power
- All CETs indicate 602 °F

ICC train "B" Core Exit Thermocouple TE-1152 fails to 900 °F.

What is the effect of this failure?

A. Core Exit Thermocouple TE-1152 will be removed from the average.

B. ICC Core Exit Thermocouple indication will go to ~627 °F.

C. "TRAIN B SUBCLG MARG LO" annunciator will alarm.

D. "B" SPDS will switch from ATOG to the ICC display.

Answer:

A. Core Exit Thermocouple TE-1152 will be removed from the average.

Notes:

CETs are averaged together to generate alarms, indication, or action. Therefore, "b", "c", and "d" are incorrect and "a" is correct since ICCMDS will determine that TE-1152 is unreliable and remove it from the average.

References:

1105.008 Rev 17

History:

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

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

QID: 0138 **Rev:** 0 **Rev Date:** 12/02/98 **Source:** Direct **Originator:** B. Short

TUOI: AA51002-013 **Objective:** 9 **Point Value:** 1

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

System Number: 045 **System Title:** Main Turbine Generator System

Description: Ability to manually operate and/or monitor in the control room: Turbine stop valves.

K/A Number: A4.06 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	2.7	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 63 **SRO:** 63

During the performance of Main Turbine Governor Valve testing, while governor valve #1 was closed in the test position governor valve #3 fails closed. What turbine problems does this impose?

A. Moisture impingement on the turbine blading.

B. Thermal shock to the turbine rotor.

C. Turbine will trip due to low load.

D. Turbine overspeed condition.

Answer:

B. Thermal shock to the turbine rotor.

Notes:

(A) is incorrect. The closure of both valves does not change the quality of the steam.
(B) is correct. Closure of GV1 and GV3 with GV2 & GV4 open or closure of GV2 & GV4 with GV1 & GV3 open causes thermal shock on the turbine rotor.
(C) is incorrect. The load shifts through the two valves that remain open.
(D) is incorrect. The load will stay essentially the same so that an overspeed condition should not occur.

References:

1106.009 (Change 37)

History:

Developed for use in A. Morris 98 RO Re-exam
Selected for 2010 RO/SRO exam

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

QID: 0798 **Rev:** 0 **Rev Date:** 9/16/2009 **Source:** New **Originator:** S. Pullin

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

Section: 3.8 **Type:** Plant Services System

System Number: 075 **System Title:** Circulating Water System

Description: Knowledge of abnormal condition procedures.

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

Tier:	2	**RO Imp:**	4.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	C

Question: **RO:** 64 **SRO:** 64

Given:

- Plant at 100% power
- Lake Temperature is stable at 65 **F**
- P-3A, P-3B, and P-3C Circulating Water Pumps are running
- P-3A Circulating Water Pump trips.
- P-3D Circulating Water Pump standby pump was started.
- It is noticed that the condenser waterbox discharge temperature is 10 degrees higher and condenser vacuum is dropping.
- AOP 1203.016, Loss of Condenser Vacuum, has been entered.

Which of the following is the cause for these conditions?

A. The stopping and starting of a circ pump caused fouling to be removed from the tube sheet promoting better heat transfer capabilities.

B. The discharge valve on the tripped pump did not go completely closed and circulating water is short cycling.

C. The debris on the bar grates of the circulating water bays was stirred up during the circ pump swap causing reduced flow.

D. Lake temperature is too high for 3 circulating water pump operation per 1104.008, Circulating Water and Water Box Vacuum System Operation.

Answer:

B. The discharge valve on the tripped pump did not go completely closed and circulating water is short cycling.

Notes:

(A.) is incorrect. Although some fouling can be removed during pump rotations, it should not result in a 10 degree change in waterbox discharge temperature.
(B.) is correct. The discharge valve on an idle pump can allow a significant amount of backflow from the operating pumps if it is not closed completely.
(C.) is incorrect. This condition is normal for a circ pump swap and may contribute to waterbox fouling, however, the service water system would be affected by this condition as well.
(D.) is incorrect. 1104.008 states that 4 CW Pumps are needed when lake temperature is above 67 F

References:

1104.008, Circulating Water System, change 27, page13, Caution

History:

New for 2010 RO/SRO exam

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

QID: 0542 **Rev:** 0 **Rev Date:** 12/8/2003 **Source:** Direct **Originator:** NRC

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

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

System Number: 086 **System Title:** Fire Protection System

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fire Protection System controls including: Fire header pressure.

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

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.3	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** 65 **SRO:** 65

You are on watch in the Control Room when the following annunciator alarms:

- K12-A1, "FIRE"

As Fire Water Header pressure drops from 110 psig to 80 psig select the order that fire pumps would start.

A. Jockey FWP P-11; Diesel Fire Pump P-6B starts second; Electric Fire Pump P-6A starts last.

B. Electric Fire Pump P-6A; Diesel Fire Pump P-6B starts second; Jockey FWP P-11 starts last.

C. Electric Fire Pump P-6A; Jockey FWP P-11 starts second; Diesel Fire Pump P-6B starts last.

D. Jockey FWP P-11; Electric Fire Pump P-6A starts second; Diesel Fire Pump P-6B starts last.

Answer:

D. Jockey FWP P-11; Electric Fire Pump P-6A starts second; Diesel Fire Pump P-6B starts last.

Notes:

"D" is correct, Jockey FWP P-11; Electric Fire Pump P-6A starts second; Diesel Fire Pump P-6B starts last. The other choices are incorrect based on pressure to start for each one.

References:

STM 1-60, Fire Protection System, rev 8, page 2.

History:

Developed by NRC.
Used on 2004 RO Exam.
Selected for 2010 RO/SRO exam

Facility: **Arkansas Nuclear One – Unit 1**		Date of Exam: **3/5/2010**				
Category	K/A #	Topic	RO			
			IR	#	QID	Type#
1. Conduct of Operations	**2.1.23**	Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.3	66	482	D
	2.1.31	Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	4.6	67	800	N
	2.1.32	Ability to explain and apply system limits and precautions.	3.8	68	799	N
	Subtotal			**3**		
2. Equipment Control	**2.2.1**	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could effect reactivity.	4.5	69	116	D
	2.2.37	Ability to determine operability and / or availability of safety related equipment	3.6	70	801	N
	Subtotal			**2**		
3. Radiation Control	**2.3.4**	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	71	802	N
	Subtotal			**1**		
4. Emergency Procedures / Plan	**2.4.6**	Knowledge of EOP mitigation strategies.	3.7	72	803	N
	2.4.11	Knowledge of abnormal condition procedures.	4.0	73	161	D
	2.4.14	Knowledge of general guidelines for EOP usage.	3.8	74	818	N
	2.4.3	Ability to identify post-accident instrumentation.	3.7	75	242	D
	Subtotal			**4**		
Tier 3 Point Total				**10**		

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

QID: 0482 **Rev:** 0 **Rev Date:** 10/7/2003 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-WCO-CZ **Objective:** 13 **Point Value:** 1

Section: 3.9 **Type:** Radioactivity Release

System Number: 068 **System Title:** Liquid Radwaste System (LRS)

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

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

Tier:	3	**RO Imp:**	4.3	**RO Select:**	Yes	**Difficulty:**	2
Group:		**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	K

Question: RO: 66 SRO: 66

Which of the following must be performed when you are releasing an Liquid radwaste tank and its Process Monitor is inoperable?

A. Estimate radiation level every four hours during the release.

B. Have an independent sample obtained and analyzed prior to release.

C. Estimate flow rate at least once every three hours during release.

D. The Tank can NOT be released iwhen its process monitor is inoperable.

Answer:

B. Have an independent sample obtained and analyzed prior to release.

Notes:

Answer "B" contains the requirement from Att. B1 of 1104.020. The other answers are incorrect.

2004 Exam Development Note: Randomly selected alternate K/A 2.1.23 to replace 2.1.31 due to lack of CR controls at ANO for the Liquid Radwaste system.

References:

1104.020, Change 49, Att. B1, section 2

History:

Modified regular exambank QID #2765.
Used on 2004 RO/SRO Exam.
Selected for 2010 RO/SRO

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

QID: 0800 **Rev:** 0 **Rev Date:** 9/16/2009 **Source:** New **Originator:** S Pullin

TUOI: A1LP-RO-ESAS **Objective:** 20 **Point Value:** 1

Section: 2 **Type:** Generic K&A

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to locate control room switches, controls, and indications, and to determine they correctly reflect the desired plant lineup.

K/A Number: 2.1.31 **CFR Reference:** 41.10 / 45.12

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

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

Question: **RO:** 67 **SRO:** 67

Given:

LOCA in progress has caused ESAS actuation of Channel 1-4

Which of the following combinations of indications and locations are correct for the given condition?

A. CV-3820, "SW TO ICW," green light, on C16;
CV-1270, "RCP SEAL BLEEDOFF FROM D RCP," red light, on C18;
CV-1053, "QUENCH TANK DRAIN," green light, on C16

B. CV-1233, "RCS MAKEUP," red light, on C16;
CV-1441, "BWST PURIF RECIRC ISOL," green light, on C13;
CV-5612 ,"FIRE WATER TO RB," green light, on C18.

C. CV-1285, "HIGH PRESSURE INJECTION," red light, on C16;
CV-1407, "BWST OUTLET," red light, on C18;
CV-3841, "LPI PUMP BRG CLR E-50 INLET," red light, on C16

D. CV-1408, "BWST OUTLET," red light, on C18;
CV-7402, "RB PURGE INLET," green light, on C18;
CV-4804, "RB VENT," red light, on C16

Answer:

C. CV-1285, "HIGH PRESSURE INJECTION," red light, on C16;
CV-1407, "BWST OUTLET," red light, on C18;
CV-3841, "LPI PUMP BRG CLR E-50 INLET," red light, on C16

Notes:

C is correct in that it has the correct indications and panel locations.
A is incorrect in that it has the incorrect indications and correct panel locations.
B is incorrect in that it has the correct indications and incorrect panel locations.
D is incorrect in that it has the incorrect indications and incorrect panel locations.

References:

STM 1-65 Rev 5 ESAS
STM 1-05 Rev 16 DHR

History:

New selected for 2010 RO/SRO exam

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

History:

New for 2010 RO/SRO exam

QID: 0799 **Rev:** 0 **Rev Date:** 9/16/2009 **Source:** New **Originator:** S Pullin

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

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

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to explain and apply system limits and precautions.

K/A Number: 2.1.32 **CFR Reference:** 41.10 / 43.2 / 45.12

Tier: 3 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 4

Group: **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** 68 **SRO:** 68

Procedure 1105.004, "Integrated Control system" limit and precaution states do not operate Reactor Demand H/A station in Auto with both S/Gs on low level limits.

What is the reason for this precaution and does any exception apply?

A. Due T-ave reduction as power lowers rods will pull to maintain T-ave at setpoint, you can operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits if you adjust T-ave setpoint to match reactor power

B. Due T-ave reduction as power lowers rods will not move due to T-ave error, you can not operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits

C. When S/Gs are on Low Level Limits, the Tave calibrating integral is blocked,. you can operate with Reactor Demand H/A station in Auto with both STGs on low level limits providing you verify calibrating integral is blocked on PDS.

D. When S/Gs are on Low Level Limits, the Tave calibrating integral is released, you can not operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits.

Answer:

A. Due T-ave reduction as power lowers rods will pull to maintain T-ave at setpoint, you can operate with Reactor Demand H/A station in Auto with both S/Gs on low level limits if you adjust T-ave setpoint to match reactor power

Notes:

A is correct, due to lowering power with S/G on LLL will cause Tave to ramp down. The Rx Demand station will try to pull rods to maintain 579 F. Limit & Precaution allows this mode of operation only if you reduce Tave setpoint to match Rx power.
B, C and D are incorrect

References:

OP-1105.004 Change 20

History:

New selected for 2010 RO/SRO exam.

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

QID: 0116 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork

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

Section: 2.0 **Type:** Generic K/As

System Number: 2.2 **System Title:** Equipment Control

Description: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

K/A Number: 2.2.1 **CFR Reference:** 45.1

Tier: 3 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 69 **SRO:** 69

During an INITIAL approach to criticality, if criticality is NOT achieved within _________ of the ECC, then insert ________ and ________ .

A. Plus or minus 1.0% delta k/k
control rods to achieve 1.5% SD margin
establish hot shutdown conditions

B. Plus or minus 1.0% delta k/k
regulating groups to achieve 1.0% SD margin
notify Reactor Engineering

C. Plus or minus 0.5% delta k/k
control rods to achieve 1.5% SD margin
verify calculation

D. plus or minus 0.5% delta k/k
regulating groups to achieve 1.0% SD margin
verify calculation

Answer:

C. plus or minus 0.5% delta k/k
control rods to achieve 1.5% SD margin
verify calculation

Notes:

Answer "C" is correct per 1102.008.

References:

1102.008, Chg. 023

History:

Used in 1998 RO exam
Used in NRC developed RO exam 8/24/92, no. 88
Used in A. Morris 98 RO Re-exam
Used in 2001 RO Exam
Selected for 2010 RO/SRO exam

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

QID: 0801 **Rev:** 0 **Rev Date:** 9/17/2009 **Source:** New **Originator:** S Pullin

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

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

System Number: 2.2 **System Title:** Equipment Control

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

K/A Number: 2.2.37 **CFR Reference:** 41.7 / 43.5 / 45.12

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

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

Question: **RO:** 70 **SRO:** 70

REFERENCE PROVIDED

Which of the following plant conditions would require entry into LCO 3.2.1 due to exceeding Regulation Rod Insertion Limits per the COLR?

A. 80% Power, 4 RCP's in service, 150 EFPD, Rod Index of 250 %

B. 70% Power, 4 RCP's in service, 300 EFPD, Rod Index of 220 %

C. 60% Power, 3 RCP's in service, 100 EFPD, Rod Index of 265 %

D. 50% Power, 3 RCP's in service, 350 EFPD, Rod Index of 255 %

Answer:

B. 70% Power, 4 RCP's in service, 300 EFPD, Rod Index of 220 %

Notes:

Per the graphs in the COLR answer (B) falls within the Operation Restricted area of the figure and would require entry into LCO 3.2.1.
A, C, and D do not require entry into LCO

References:

ANO-1 Cycle 22 COLR Figures 3-A through 4-B

History:

New for 2010 RO/SRO exam

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

QID: 0802 **Rev:** 0 **Rev Date:** 9/17/2009 **Source:** New **Originator:** S. Pullin

TUOI: ASLP-RO-RADP **Objective:** 15 **Point Value:** 1

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

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits under normal or emergency conditions.

K/A Number: 2.3.4 **CFR Reference:** 41.12 / 43.4 / 45.10

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

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

Question: **RO:** 71 **SRO:** 71

Given:

- A General Emergency has been declared on Unit 1.
- A Maintenance crew must enter a radiological area with a dose rate of 150 Rem/Hr to protect valuable property.

Which of the following is the MAXIMUM time an individual team member can stay in this area?

A. 4 minutes

B. 6 minutes

C. 8 minutes

D. 10 minutes

Answer:

A. 4 minutes

Notes:

A is correct, for protecting valuable property 10 Rem is the does limit.
B, C and D exceed 10 Rem limit.

References:

OP-1903.033 Change 019-01-0

History:

New for 2010 RO/SRO exam

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

QID: 0803 **Rev:** 0 **Rev Date:** 9/17/2009 **Source:** New **Originator:** S Pullin

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

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

System Number: 2.4. **System Title:** Emergency procedure / plan

Description: Knowledge of EOP mitigation strategies.

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

Tier: 3 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

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

Question: **RO:** 72 **SRO:** 72

General rules of the Generic Emergency Operating Guidelines are that symptoms are treated whenever they occur based on priorities.

Which of the following transients has top priority per the GEOG?

A. Overheating

B. Overcooling

C. Loss of Subcooling Margin

D. Steam Generator Tube Rupture

Answer:

C. Loss of Subcooling Margin

Notes:

C is correct per the GEOG LOSM has top priority.

References:

Volume 1 GEOG Part 1, Introduction

History:

New for 2010 RO/SRO exam.

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

QID: 0161 **Rev:** 1 **Rev Date:** 4/24/2002 **Source:** Direct **Originator:** J. Cork

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

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

System Number: 2.4 **System Title:** Emergency procedure / plan

Description: Knowledge of abnormal condition procedures.

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

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

Group: G **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** 73 **SRO:** 73

Given:

- Power escalation is in progress following a shutdown.
- Reactor power is 35%.
- Rod 6 of Group 7 drops.

Which of the following actions should be taken?

A. Insert all regulating rods in sequential mode.

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

C. Verify plant stabilizes at 320 MWe after ICS runback.

D. Verify SDM within COLR limit within one hour.

Answer:

D. Verify SDM within COLR limit within one hour.

Notes:

[a] would only be performed if power was <2%.
[b] would not be done because only one rod dropped.
[c] power is <360 MWe so there wouldn't be any runback, the value given would require a power increase.
[d] is the correct answer per TS.

References:

1203.003, Control Rod Drive Malfunction Action, change 023, page 12, step 4

History:

Developed for use in 98 RO Re-exam.
Used in 2001 RO/SRO Exam.
Selected for 2002 RO/SRO exam. Revised to agree with ITS.
Selected for 2010 RO/SRO exam

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

QID: 818 **Rev:** 0 **Rev Date:** 2/5/2010 **Source:** New **Originator:** S.Pullin

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

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

System Number: 2.4 **System Title:** Emergency procedure / plan

Description: Knowledge of general guidelines for EOP usage

K/A Number: 2.4.14 **CFR Reference:** 41.10/45.13

Tier: 3 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 4.5 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** 74 **SRO:** 74

The EOP/AOP user guide procedure states Reactor Trip (1202.001) is the entry point for all EOPs with one exception.

Which of the following is the exception?

A. Loss of SCM

B. Overcooling

C. Overheating

D. Tube Rupture

Answer:

D. Tube Rupture

Notes:

D. is correct this is the only EOP that would be entered in the absence of Reactor trip

A., B., and C, are incorrect these procedures are only entered after a Reactor trip

References:

1015.043 ANO-1 EOP/AOP user guide change 003

History:

New selected for 2010 RO exam

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

QID: 0242 **Rev:** 0 **Rev Date:** 9-1-99 **Source:** Direct **Originator:** D. Slusher

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

Section: 2 **Type:** Generic

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

Description: Ability to identify post-accident instrumentation.

K/A Number: 2.4.3 **CFR Reference:** CFR: 41.6/45.4

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

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

Question: **RO:** 75 **SRO:** 75

What instruments are marked with a green dot?

A. Instruments designated for use during an alternate shutdown.

B. Instruments that should be reliable during accident conditions.

C. Instruments the Shift Engineer uses after a reactor trip.

D. Instruments designated for use during a loss of NNI-Y power.

Answer:

B. Instruments that should be reliable during accident conditions.

Notes:

Instruments which are to be used for accident conditions are marked by a green dot as required by Reg Guide 1.97. Therefore, b is the correct answer.
(a) is incorrect because SPDS is designated for the alternate shutdown.
(c) is incorrect because SE instruments used after a reactor trip are designated by the 1015.037.
(d) is incorrect because NNI-X instruments are available in a loss of NNI-Y.

References:

1305.028 change 12 page 2

History:

Developed for 1999 exam.
Selected for the 2010 RO/SRO exam

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/2	036 Fuel Handling Accident – AK2.01 Fuel Handling Equipment.	Could not write a credible question since only RO action is to suspend fuel movement and exit. Randomly selected new system 068 Control Room Evacuation – AK2.07 ED/G.
1/2	060 Accidental Gaseous Radwaste Release – AK1.04 Calculation of Offsite Doses due to release from the power plant.	The RO would not perform this function. Randomly selected new system 028 Pressurizer Level Malfunction – AK1.01 Pressurizer reference leak abnormalities.
2/1	004 Chemical and Volume Control – 2.1.34 Knowledge of primary and secondary chemistry limits.	Could not write a credible question to match the K/A and tie to that system. Randomly selected – 2.2.38 Knowledge of conditions and limitations in the facility license.
2/1	008 Component Cooling Water – A2.08 Effects of shutting (automatically or otherwise) the isolation valves of the letdown cooler.	No credible tie for this K/A exists for the System. Randomly selected A2.01 – Loss of CCW Pump.
2/1	061 Emergency/Auxiliary Feedwater – A1.04 AFW source tank level	Not possible to prepare a psychometrically sound question related to the subject K/A. Randomly selected - A1.01 S/G level.
2/1	078 Instrument Air System (IAS) – K1.03 Containment Air.	Not possible to prepare a psychometrically sound question related to the subject K/A. Randomly selected – K1.02 Service Air.
2/2	001 Control Rod Drive – K2.05 M/G Sets	No credible tie for this K/A exists for the System. Randomly selected – K2.02 One-line diagram of power supply to trip breakers.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/2	068 Liquid Radwaste – K4.01 Safety and environmental precautions for handling hot, acid, and radioactive liquids.	The RO would not perform this function. Randomly selected new system 014 Rod position indication – K4.05 Rod hold interlocks.
2/2	071 Waste Gas Disposal – K3.05 ARM and PRM systems.	Not possible to prepare a psychometrically sound question related to the subject K/A. Randomly selected new system 015 Nuclear Instrumentation – K3.04 ICS.
3	2.3.11 Ability to control radiation releases.	NRC comment that the ODCM was too cueing, therefore rejected 2.3.11. Randomly selected new Generic K/A – 2.4.11.

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

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1													3	3	6
	2													3	1	4
	Tier Totals				N/A					N/A				6	4	10
2. Plant Systems	1													3	2	5
	2													0 1	2	3
	Tier Totals													4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4			1 2	3 4	7
														2 2	1 2	

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

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

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

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

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	065 Loss of Instrument Air – 2.4.18 Knowledge of the specific bases for EOPs	Could not write a credible SRO level question. Randomly selected new system 038 Steam Generator Tube Rupture 2.4.18
1/2	001 Continuous Rod Withdrawal – AA2.05 Uncontrolled Rod withdrawal from available indications	Could not write a credible SRO level question. Randomly selected new system 005 Inoperable stuck control rod – AA2.03 Required actions if more than one rod is stuck or inoperable.
2/2	028 Hydrogen Recombiner and Purge Control – 2.4.23 Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.	Could not write a credible SRO level question. Randomly selected new system 016 Non-Nuclear Instrumentation – 2.2.40 Ability to apply Technical Specifications for a system.

ES-401	PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (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	#	QID	Type
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1					X		**EA2.1**- Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.0	76	588	D
000008 Pressurizer Vapor Space Accident / 3							**Not selected**	**N/A**			
000009 Small Break LOCA / 3							**Not selected**	**N/A**			
000011 Large Break LOCA / 3							**Not selected**	**N/A**			
000015/17 RCP Malfunctions / 4							**Not selected**	**N/A**			
000022 Loss of Rx Coolant Makeup / 2					X		**AA2.04**- How long PZR level can be maintained within limits	3.8	77	805	N
000025 Loss of RHR System / 4						X	**2.4.31** Knowledge of annunciator alarms, indications, or response procedures.	4.1	78	806	N
000026 Loss of Component Cooling Water / 8					X		**AA2.01**- Location of a leak in the CCWS	3.5	79	807	N
000027 Pressurizer Pressure Control System Malfunction / 3							**Not selected**	**N/A**			
000029 ATWS / 1							**Not selected**	**N/A**			
000038 Steam Gen. Tube Rupture / 3						X	**2.4.18** – Knowledge of the specific bases for EOPs.	4.0	80	585	N
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4						X	**2.4.6**- Knowledge of symptom based EOP mitigation strategies	4.7	81	584	D
000054 (CE/E06) Loss of Main Feedwater / 4							**Not selected**	**N/A**			
000055 Station Blackout / 6							**Not selected**	**N/A**			
000056 Loss of Off-site Power / 6							**Not selected**	**N/A**			
000057 Loss of Vital AC Inst. Bus / 6							**Not selected**	**N/A**			
000058 Loss of DC Power / 6							**Not selected**	**N/A**			
000062 Loss of Nuclear Svc Water / 4							**Not selected**	**N/A**			
000065 Loss of Instrument Air / 8							**2.4.18** – Knowledge of the specific bases for EOPs **Rejected system to 038 Steam Gen Tube Rupture**	**N/A**			
W/E04 LOCA Outside Containment / 3							**Not selected**	**N/A**			

ES-401	PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (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	#	QID	Type
W/E11 Loss of Emergency Coolant Recirc. / 4							**Not selected**	**N/A**			
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4							**Not selected**	**N/A**			
000077 Generator Voltage and Electric Grid Disturbances / 6							**Not selected**	**N/A**			
K/A Category Totals:					3	3	Group Point Total:		**6**		

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

QID: 0588 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-EOP04 **Objective:** 11 **Point Value:** 1

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

System Number: E10 **System Title:** Post-Trip Stabilization

Description: Ability to determine and interpret the following as they apply to the (Post-Trip Stabilization): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A Number: EA2.1 **CFR Reference:** 43.5 /45.13

Tier:	1	**RO Imp:**	2.5	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question: RO: SRO: 76

Given:

- Reactor tripped due to a loss of both MFWPs approximately 15 minutes ago.
- Annunciator K02-B6 "A3 L.O. RELAY TRIP" is in alarm.
- AFW pump, P-75, is tagged out for maintenance.
- Steam Driven EFW Pump, P-7A, has tripped on overspeed.
- RCS pressure is 2000 psig.
- CETs are 612°F.
- Both OTSG levels are 30".

Which of the following procedures should be in use for the above conditions?

A. 1202.002, Loss of Subcooling Margin

B. 1202.004, Overheating

C. 1202.011, HPI Cooldown

D. 1203.037, Abnormal ES Bus Voltage

Answer:

B. 1202.004, Overheating

Notes:

Answer "B" is correct, the Overheating EOP should be entered with CETs > 610°F and all MFW and EFW lost during loss of adequate Subcooling Margin.
Answer "A" is incorrect, this procedure would have been in use up to the point where CETs became > 610°F.
Answer "C" is incorrect, this procedure is entered from Loss of Subcooling Margin.
Answer "D" is incorrect, this procedure is used when ES bus voltage is low but not de-energized.

References:

1202.004, Chg. 006

History:

New for 2005 SRO exam.
Selected for 2010 SRO exam

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

QID: 0805 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Ability to determine and interpret the following as they apply to Reactor Coolant Makeup: How long PZR level can be maintained within limits.

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

Tier: 1 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 4

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

Question: **RO:** **SRO:** 77

Given:

- RCS Cooldown in progress
- Tave is 295 F
- RCS Pressure is 440 psig.
- Pressurizer level is 65 inches
- All makeup has been lost
- Pressurizer level is dropping at 5 inches per minute
- Assuming pressurizer level rate of change remains the same

When will LCO 3.4.9 Pressurizer, be entered due to low Pressurizer level and what is the bases per Technical Specification for the low level?

A. 2 minutes and to maintain the minimum ES bus powered pressurizer heaters OPERABLE.

B. 2 minutes and to maintain on scale pressurizer level indication.

C. 4 minutes and to maintain the minimum ES bus powered pressurizer heaters OPERABLE.

D. 4 minutes and to maintain on scale pressurizer level indication.

Answer:

D. 4 minutes and to maintain on scale pressurizer level indication.

Notes:

D is correct, the limit per LCO 3.4.9 is less than or equal to 45 inches and the minimal water level limit has been established to ensure that water level is above the minimum detectable level.
A is incorrect, due to PZR level would be 55 inches which is below the administrative limit per OP-1102.010 for PZR level, but does not require entry into the LCO.
B is incorrect, due to PZR level would be 55 inches which is below the administrative limit per OP-1102.010 for PZR level, but does not require entry into the LCO.
C is incorrect, due to PZR level would be 45 inches which is at the pressurizer heater cutoff level which would deenergize the ES powered heaters.

References:

T.S. 3.4.9 Amendment 215

History:

New selected for 2010 SRO exam.

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

QID: 0806 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Knowledge of annunciator alarms, indications, or response procedures.

K/A Number: 2.4.31 **CFR Reference:** 41.10 / 45.3

Tier: 1 **RO Imp:** 4.2 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 78

Given:

- Mode 5
- RCS temperature 170 F
- CV-1050 and CV-1410 interlocks are not bypassed
- RCS pressure 0 psig
- "A" RCP seal removed for maintenance
- "A" Decay Heat in service
- Following alarms are received
  - DECAY HEAT FLOW HI/LO (K09-A8)
  - DECAY HEAT VORTEX WARNING (K09-D8)
  - ISOL VLV OPEN RC PRESS LO (K10-E5)

Which section of OP-1203.028, Loss of Decay Heat Removal, will be entered for the given conditions?

A. Section 6, Decay Heat Pump Trip

B. Section 7, Suction Valve Closure

C. Section 9, Loss of Both DH Systems - RCS Pressure Boundary Intact

D. Section 10, Loss of Both DH Systems - RCS Pressure Boundary Open

Answer:

B. Section 7, Suction Valve Closure

Notes:

B is correct, with the given alarms K10-E5 would automatically cause the DHR Suction valve to close.
A is incorrect, the DHR Pump would still be running for the given condition. The pump does not automatically stop on valve closure.
C is incorrect, although the RCS is still intact with an RCP seal removed, the transition to loss of both DHR Pumps does not occur until RCS temperature is greater than 280 F
D incorrect, the RCS is not open and the transition to loss of both DHR Pumps does not occur until RCS temperature is greater than 280 F

References:

OP-1203.028 Change 021
op-1203.012I Change 046

History:

New selected for 2010 SRO exam.

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

QID: 0807 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** S Pullin

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

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

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

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

K/A Number: AA2.01 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 79

Given:

- Plant at 100%
- The following alarms are received
- ICW COOLER OUTLET TEMP HI (K12-E4)
- RCP BEEDOFF TEMP HI (K08-C7)
- "A" RCP seal temperature rising
- Skewed RCP Seal Injection flows indicated on CO4
- RCS leak rate is 50 gpm

Which of the following procedures provide the actions necessary to mitigate the abnormal operating condition?

A. OP-1203.039, Excess RCS Leakage

B. OP-1203.026, Loss of Reactor Coolant Makeup

C. OP-1203.031, Reactor Coolant Pump and Motor Emergency

D. OP-1102.016, Power Reduction and Plant Shutdown

Answer:

A. OP-1203.039, Excess RCS Leakage

Notes:

A is correct, because Excess RCS Leakage procedure is the only procedure that combats an intersystem LOCA.
B is incorrect, OP-1203.026 has a section to address makeup & purification system leaks, but with the indications given this is not considered a makeup & purification system leak.
C is incorrect, with the given indications the student could misdiagnose this as a seal failure issue,
D is incorrect, with the given leak rate, a rapid plant shutdown would be necessary.

References:

OP-1203.039 Change 11

History:

New selected for 2010 SRO exam

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

QID: 0585 **Rev:** 0 **Rev Date:** 9/21/2009 **Source:** New **Originator:** B. Possage

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

Section: 4.1 **Type:** Generic EPE's

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

Description: Knowledge of the specific bases for EOPs.

K/A Number: 2.4.18 **CFR Reference:** 41.10 / 43.1 / 45.13

Tier: 1 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 80

Given:
- SGTR in progress
- Rx is tripped
- RCS pressure 1350 psig
- RCS Thot 540°F
- Projected dose rate at site boundary at NUE criteria
- "B" SG level at 395" and rising rapidly
- "A" SG level stable at 40"

Considering the above conditions, which of the following procedural actions will cause higher tube stresses than normal limitations but is acceptable during a SGTR per the EOP technical bases document?

A. Perform a cool down to less than 500°F at 100°F/hr and isolate bad SG.

B. Steam bad SG to maintain bad SG Tube-to-Shell DT <150°F (tubes colder).

C. Steam bad SG to maintain bad SG Tube-to-Shell DT <100°F (tubes hotter).

D. Establish a cool down rate of 250°F/hr to 500°F Thot.

Answer:

B. Steam bad SG to maintain bad SG Tube-to-Shell DT <150°F (tubes colder).

Notes:

B is correct, per Technical Bases during emergency cool downs the tube to shell delta T limits are relaxed. With the given information an emergency cool down is required at the rate of </= 240 F/hr.
A is incorrect, this rate is the normal cool down rate.
C is incorrect, this is the normal tube to shell delta T limit.
D is incorrect, this exceeds the allowed emergency cool down limit.

References:

OP-1202.006 Change 11
B&W EOP Technical Bases Document

History:

New selected for 2010 SRO exam

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

QID: 0584 **Rev:** 0 **Rev Date:** 5/20/05 **Source:** Direct **Originator:** J.Cork

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

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of symptom based EOP mitigation strategies.

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

Tier: 1 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 4

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

Question: **RO:** **SRO:** 81

A steam line rupture has occurred in the Reactor Building with the following conditions now present:
- ESAS actuated on channels 1 thru 6.
- All RCPs secured per RT-10.
- RB pressure 19 psig and dropping.
- HPI throttled due to existence of adequate SCM.
- RCS pressure is 1050 psig.
- T-hot is 490°F.
- EOP actions have terminated the overcooling.

The SE recommends to the CRS to restore normal operating pressure per RT-14 in order to reset ESAS and re-start RCPs.
As CRS, does this recommendation follow the EOP mitigation strategies?

A. Yes, overcooling event has been terminated.

B. No, this could overstress reactor vessel.

C. Yes, adequate SCM has been restored.

D. No, RB pressure is not within normal limits.

Answer:

B. No, this could overstress reactor vessel.

Notes:

"B" is correct, trainee must recognize that with RCPs secured and HPI having been initiated that PTS limits apply until an evaluation is performed prior to returning to normal pressure. PTS limits prevent overstressing reactor vessel.
"A" is incorrect, yes the overcooling has been terminated but normal operating pressure would violate procedure.
"C" is incorrect, subcooling margin was never lost but normal operating pressure would violate procedure.
"D" is incorrect, although RB pressure is a concern the overriding concern is with PTS concerns.

THIS QUESTION IS TIED to 43.1

References:

1202.012, chg. 004-03-0, RT-14

History:

New for 2005 SRO exam.
Selected for the 2010 SRO exam

ES-401	PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (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	#	QID	Type
000001 Continuous Rod Withdrawal / 1							**AA2.05-** Uncontrolled rod withdrawal from available indications **Rejected system to 005 Inoperable/Stuck Control Rod**	**N/A**			
000003 Dropped Control Rod / 1							**Not selected**	**N/A**			
000005 Inoperable/Stuck Control Rod / 1					X		**AA2.03** – Required actions if more than one rod is stuck or inoperable	4.4	82	589	D
000024 Emergency Boration / 1					X		**AA2.05** – Amount of boron to add to achieve the required SDM	3.9	83	808	M
000028 Pressurizer Level Malfunction / 2							**Not selected**	**N/A**			
000032 Loss of Source Range NI / 7							**Not selected**	**N/A**			
000033 Loss of Intermediate Range NI / 7							**Not selected**	**N/A**			
000036 (BW/A08) Fuel Handling Accident / 8							**Not selected**	**N/A**			
000037 Steam Generator Tube Leak / 3							**Not selected**	**N/A**			
000051 Loss of Condenser Vacuum / 4							**Not selected**	**N/A**			
000059 Accidental Liquid RadWaste Rel. / 9							**Not selected**	**N/A**			
000060 Accidental Gaseous Radwaste Rel. / 9							**Not selected**	**N/A**			
000061 ARM System Alarms / 7							**Not selected**	**N/A**			
000067 Plant Fire On-site / 8							**Not selected**	**N/A**			
000068 (BW/A06) Control Room Evac. / 8							**Not selected**	**N/A**			
000069 (W/E14) Loss of CTMT Integrity / 5							**Not selected**	**N/A**			
000074 (W/E06&E07) Inad. Core Cooling / 4							**Not selected**	**N/A**			
000076 High Reactor Coolant Activity / 9							**Not selected**	**N/A**			
W/EO1 & E02 Rediagnosis & SI Termination / 3							**Not selected**	**N/A**			
W/E13 Steam Generator Over-pressure / 4							**Not selected**	**N/A**			
W/E15 Containment Flooding / 5							**Not selected**	**N/A**			
W/E16 High Containment Radiation / 9							**Not selected**	**N/A**			
BW/A01 Plant Runback / 1							**Not selected**	**N/A**			
BW/A02&A03 Loss of NNI-X/Y / 7					X		**AA2.1** – Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.0	84	591	D
BW/A04 Turbine Trip / 4							**Not selected**	**N/A**			
BW/A05 Emergency Diesel Actuation / 6							**Not selected**	**N/A**			
BW/A07 Flooding / 8							**Not selected**	**N/A**			
BW/E03 Inadequate Subcooling Margin / 4							**Not selected**	**N/A**			

BW/E08; W/E03 LOCA Cooldown - Depress. / 4						X	**2.4.16**- Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe management guidelines	4.4	85	592	D
BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4							**Not selected**	**N/A**			
BW/E13&E14 EOP Rules and Enclosures							**Not selected**	**N/A**			
CE/A11; W/E08 RCS Overcooling - PTS / 4							**Not selected**	**N/A**			
CE/A16 Excess RCS Leakage / 2							**Not selected**	**N/A**			
CE/E09 Functional Recovery							**Not selected**	**N/A**			
K/A Category Point Totals:					**3**	**1**	Group Point Total:		**4**		

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

QID: 0589 **Rev:** 0 **Rev Date:** 6/1/05 **Source:** Direct **Originator:** S.Pullin
TUOI: A1LP-RO-TS **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs
System Number: 005 **System Title:** Inoperable/Stuck Control Rod
Description: Ability to determine and interpret the following as they apply to the Inoperable/Stuck Control Rod: Required actions if more than one rod is stuck or inoperable.

K/A Number: AA2.03 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	3.5	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.4	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 82

Given:
- Plant is at 40% power.
- Group 4, Rod 4 is stuck and is mis-aligned from the group by 7.5%.
- The rod can not be re-aligned with the group.

Subsequently Group 7 Rod 6 drops to 0% withdrawn.

What are the required action(s) per Technical Specifications for the above conditions?

A. Open Control Rod Drive breakers, within 1 hour.

B. Borate to restore SDM within 1 hour and perform Linear Heat Rate surveilllance, SR 3.2.5.1, within 6 hours.

C. Borate to restore SDM within 1 hour and verify the potential ejected rod worth is within the assumptions of the rod ejection analysis within 6 hours.

D. Borate to restore SDM within 1 hour and place the plant in Mode 3 within 6 hours.

Answer:

D. Borate to restore SDM within 1 hour and place the plant in Mode 3 within 6 hours.

Notes:

Answer "D" is correct per TS 3.1.4 action "C" for two inoperable rods.
Answer "A" is incorrect, this action is performed for two dropped rods.
Answer "B" is incorrect, this action is performed for one inoperable rod and the time given for the stated condition is incorrect.
Answer "C" is incorrect, this action is performed for one inoperable rod and the time given for the stated condition is incorrect.

References:

T.S. 3.1.4 amendment 215

Do not include this spec in the student handout!!!

History:

New for 2005 SRO exam.
Selected for 2010 SRO exam

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

QID: 0808 **Rev:** 0 **Rev Date:** 9/22/2009 **Source:** Modified **Originator:** S.Pullin

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

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

System Number: 0024 **System Title:** Emergency Boration

Description: Ability to determine and interpret the following as they apply to the Emergency Boration: Amount of boron to add to achieve the required SDM.

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

Tier: 1 **RO Imp:** 3.3 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 83

REFERENCE PROVIDED

- Rx has tripped with three CRDs stuck full out.
- Core lifetime = 150 EFPD
- RCS initial Boron concentration = 810 ppm
- Chemistry reports that the RCS boron concentration is 2200 ppm.

Which of the following contains guidance that must be used, for the above conditions?

A. No action required, SDM is adequate

B. 1202.012, RT-12 Emergency Boration

C. 1203.017, Moderator Dilution

D. 1103.015, Reactivity Balance Calculation

Answer:

B. 1202.012, RT-12 Emergency Boration

Notes:

Answer "B" is correct, using Att. B-16 from the plant data book, the examinee should determine that adequate SDM has not been established and Emergency Boration must be performed until adequate SDM is established.
Answer "A" is incorrect, SDM is not adequate.
Answer "C" is incorrect, although this might seem like a logical choice, this procedure should not be used for these conditions.
Answer "D" is incorrect, although this might seem like a logical choice, use of the Reactivity Balance Calculation procedure does not have any plant actions in it.

References:

1202.012 RT-12, Chg. 008
CALC-ANO1-NE-08-00007
NOTE: CALC Att. B-16 must be in SRO handout!!!!

History:

Modified from QID 678
Selected for 2010 SRO exam

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

QID: 0591 **Rev:** 0 **Rev Date:** 6/6/05 **Source:** Direct **Originator:** S.Pullin

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

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

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

Description: Ability to determine and interpret the following as they apply to the (NNI-X): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A Number: AA2.1 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** An

Question: **RO:** **SRO:** 84

Given:

- Pressurizer Level Control Valve CV-1235 indicates 50% open.
- RC Pump Seals Total Inj Flow valve CV-1207 indicates 50% open.
- Letdown flow indication is zero.
- Letdown pressure indication is zero.
- Letdown Orifice Bypass valve CV-1223 indicates 50% open.
- RCS pressure is 2210 psig and slowly rising.
- Pressurizer Spray valve CV-1008 indicates closed.

What procedure should be in use due to the above conditions?

A. 1203.015, Pressurizer Systems Failure

B. 1203.024, Loss of Instrument Air

C. 1203.047, Loss of NNI Power

D. 1203.012B, ACA for K10-A8 "LETDOWN TEMP HI"

Answer:

C. 1203.047, Loss of NNI Power

Notes:

Answer "C" is correct since the conditions given are representative of a loss of NNI X and Y power.
Answer "A" is incorrect, this would be in use if Spray valve was failed due to something other than a loss of NNI power.
Answer "B" is incorrect, this would be in use for failed valves due to loss of IA, but the positions given are different than for loss of air alone.
Answer "D" is incorrect, this is chosen for hi letdown temp but letdown flow would still be indicated while the question states there is none.

References:

1203.047, Chg. 000-01-0

History:

New for 2005 SRO exam.
Selected for 2010 SRO exam

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

QID: 0592 **Rev:** 0 **Rev Date:** 6/6/05 **Source:** Direct **Originator:** S.Pullin

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

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

System Number: E08 **System Title:** LOCA Cool down

Description: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe management guidelines

K/A Number: 2.4.16 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 3.5 **RO Select:** No **Difficulty:** 4

Group: 2 **SRO Imp:** 4.4 **SRO Select:** Yes **Taxonomy:** Ap

Question: RO: SRO: 85

Given:
- Rx was shutdown using 1203.045 Rapid Plant Shutdown,
- Due to a RCS leak
- RCS pressure 1720 psig and lowering slowly
- HPI flow 150 gpm
- A & B SG pressure 910 psig
- RCS cool down rate 35°F per hour
- All Turbine bypass valves closed

Which procedure should be in use?

A. 1202.001, Overcooling

B. 1203.041, Small Break LOCA cool down

C. 1203.040, Forced Flow cool down

D. 1202.010, ESAS

Answer:

B. 1203.041, Small Break LOCA cool down

Notes:

Answer "B" is correct with an uncontrolled cool down continuing due to break/HPI flow, regardless of SG status.
Answer "A" is incorrect, Overcooling entry conditions have not yet been met
Answer "C" is incorrect, although RCPs are running, there is no control of the cool down.
Answer "D" is incorrect, although parameters are close to ES actuation setpoints, the ESAS procedure would eventually transition to 1203.041.

References:

1203.039, Chg. 011

History:

New for 2005 SRO exam.
Selected for 2010 SRO exam

PWR Examination Outline
Plant Systems - Tier 2/Group 1 (SRO)
Form ES-401-2

	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#	QID	Type
003 Reactor Coolant Pump								X				**A2.02** – Conditions which exist for an abnormal shutdown of a RCP in comparison to a normal shutdown of RCP	3.9	86	809	N
004 Chemical and Volume Control												**Not Selected**	**N/A**			
005 Residual Heat Removal												**Not Selected**	**N/A**			
006 Emergency Core Cooling												**Not Selected**	**N/A**			
007 Pressurizer Relief/Quench Tank												**Not Selected**	**N/A**			
008 Component Cooling Water												**Not Selected**	**N/A**			
010 Pressurizer Pressure Control								X				**A2.02** – Spray failures	3.9	87	762	R
012 Reactor Protection												**Not Selected**	**N/A**			
013 Engineered Safety Features Actuation								X				**A2.06** – Inadvertent ESFAS actuation	4.0	88	812	N
022 Containment Cooling												**Not Selected**	**N/A**			
025 Ice Condenser												**Not Selected**	**N/A**			
026 Containment Spray												**Not Selected**	**N/A**			
039 Main and Reheat Steam												**Not Selected**	**N/A**			
059 Main Feedwater												**Not Selected**	**N/A**			
061 Auxiliary/Emergency Feedwater											X	**2.2.22** – Knowledge of limiting conditions for operations and safety limits	4.7	89	811	N
062 AC Electrical Distribution												**Not Selected**	**N/A**			
063 DC Electrical Distribution											X	**2.2.42** – Ability to recognize system parameters that are entry-level conditions for Technical Specifications	4.6	90	810	N
064 Emergency Diesel Generator												**Not Selected**	**N/A**			
073 Process Radiation Monitoring												**Not Selected**	**N/A**			
076 Service Water												**Not Selected**	**N/A**			
078 Instrument Air												**Not Selected**	**N/A**			
103 Containment												**Not Selected**	**N/A**			
K/A Category Point Totals:										**3**	**2**	Group Point Total:	**5**			

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

QID: 0809 **Rev:** 0 **Rev Date:** 9/23/2009 **Source:** New **Originator:** S Pullin

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

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPs; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP

K/A Number: A2.02 **CFR Reference:** 41.5/43.5/45.3/45/13

Tier: 2 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 86

Given:

- 100% Power,
- "C" RCP seal bleed off temperature 210 F.
- "C" RCP motor bearing temperature 185 F.
- "C" RCP motor inboard vibration alert alarm,
- "C" RCP seal cavity pressure oscillating from 650 to 1250 psig.

What is the appropriate section and action of 1203.031, "Reactor Coolant Pump and Motor Emergency" which will mitigate the consequences of these malfunctions?

A. Section 2, "Seal Failure", Reduce reactor power to within the capacity of unaffected RCP combination and stop the affected RCP per Reactor Coolant Pump Operation, OP1103.006.

B. Section 2, "Seal Failure", Trip the Reactor and trip the affected RCP.

C. Section 5, "Motor / Bearing Trouble", Reduce reactor power to within the capacity of unaffected RCP combination and stop the affected RCP per Reactor Coolant Pump Operation, OP1103.006.

D. Section 5, "Motor / Bearing Trouble", Trip the Reactor and trip the affected RCP.

Answer:

B. Section 2, "Seal Failure", Trip the Reactor and trip the affected RCP.

Notes:

B is correct, a seal bleedoff temperature of greater than 200 F with no change in cooling (seal injection or ICW flow) meets the requirements to trip the RCP due to seal failure section.
A is incorrect. The given conditions require an abnormal shutdown of an RCP instead of a normal shutdown of an RCP.
C is incorrect. The given conditions require an abnormal shutdown of an RCP instead of a normal shutdown of an RCP.
D is incorrect. The given conditions do not indicate a bearing problem that warrents stopping the RCP.

References:

OP-1203.031 Change 018

History:

New selected for 2010 SRO exam

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

QID: 0762 **Rev:** 0 **Rev Date:** 11/11/2000 **Source:** Repeat **Originator:** Steve Pullin

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

Section: 3.3 **Type:** Reactor Pressure Control

System Number: 010 **System Title:** Pressurizer Pressure Control System (PZR PCS)

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures

K/A Number: A2.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.9 **RO Select:** No **Difficulty:** 4

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

Question: **RO:** **SRO:** 87

Given:

-Unit 1 is operating at 40% power.
-The Unit is in three pump ops due to the failure of P-32B.
-The Pressurizer Spray Control valve (CV-1008) fails open.

The ATC attempts to close the Pressurizer Spray Isolation valve (CV-1009) and it will NOT close

-Reactor Coolant Pressure is at 2100 psig and slowly lowering with all Pzr Heaters on.

What is the correct procedure and correct action for this condition?

A. 1202.001 Reactor Trip, and trip the Reactor.

B. 1202.001 Reactor Trip, and stop P-32C.

C. 1203.015 PZR System Failure, and trip the Reactor.

D. 1203.015 PZR System Failure, and stop P-32C.

Answer:

D. 1203.015 PZR System Failure, and stop P-32C.

Notes:

A is incorrect. Since the Power to Pump trip entry conditions are not met.
B is incorret with the correct action but with the incorrect procedure since the Power to Pump trip entry conditions are not met.
C is incorrect with the correct procedure but incorrect action.
D is correct.

References:

1203.015 Pzr System Failure Chg 16

History:

New for the 2009 Retake SRO Exam
Selected for 2010 SRO exam REPEAT

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

QID: 0812 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S. Pullin

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

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

System Number: 013 **System Title:** Engineered Safety Features Actuation System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based ability on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertant ESFAS actuation.

K/A Number: A2.06 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.7 **RO Select:** No **Difficulty:** 3

Group: 1 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question: **RO:** **SRO:** 88

Given

- Plant at 100% power
- P-2B Condensate Pump OOS
- Inadvertent actuation of ES Channel #1
- S/U #1 OOS for maintenance LCO 3.8.1.A 72 hour Time Clock in effect

What would be the impact to the plant due to this malfunction and what procedure would be used to mitigate the effects?

A. #1 Emergency Diesel Generator would start and use OP-1105.003, Engineered Safeguards Actuation System to reset the tripped channel.

B. Red Train High Pressure Injection would occur and use 1202.010, ESAS EOP to override HPI

C. Loss of power to A-1 bus and use 1202.001, Reactor Trip EOP

D. All Seal Return isolates and use OP1203.031, Reactor Coolant Pump and Motor Emergencies to realign seal bleed off.

Answer:

C. Loss of power to A-1 bus and use 1202.001, Reactor Trip EOP

Notes:

C is correct, the Unit Aux supply breaker to A-1 would open on ES Channel #1 actuation and would result in a reactor trip due to a loss of all Condensate pumps resulting in a losss of Main Feedwater.
A is incorrect, although the EDG would start with a reactor trip the EOP would have priority over securing the EDG
B is incorrect, although HPI would occur the ESAS EOP would not be utilized to secure HPI for an inadvertant actuation.
D is incorrect, seal return would be realigned to the Quench Tank rather than isolate.

References:

STM 1-32 Rev 33
OP-1107.001 Change 073

History:

New selected for 2010 SRO exam

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

QID: 0811 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

TUOI: A1LP-RO-EFIC **Objective:** 43 **Point Value:** 1

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

System Number: 061 **System Title:** Auxiliary / Emergency Feewater System

Description: Knowledge of limiting conditions for operations and safety limits

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

Tier: 2 **RO Imp:** 4.0 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 89

Given

- 'A' SG Low level transmitter feeding the 'D' EFIC Channel failed Lo
- 'B' SG Pressure transmitter feeding the 'C' EFIC Channel failed Hi

What operator actions are required per Technical Specifications?

A. Place 'D' channel in bypass per 3.3.11.A

B. Place 'C' channel in bypass per 3.3.11.B

C. Trip 'D' channel per 3.3.11.B

D. Trip 'C' channel per 3.3.11.A

Answer:

B. Place 'C' channel in bypass per 3.3.11.B

Notes:

B is correct, the Low Level transmitter failing low will result in a trip of the D Channel, 3.3.11.B requirements for two inoperable channels requires one to be placed in bypass and the other one tripped.
A is incorrect, "D" Channel is already tripped and placing in bypass would have no effect. TS 3.3.11.A is only applicable to one inoperable channel. The question asks what to do for two inoperable channels
C is incorrect, because it is only half of the action required by 3.3.11.B
D is incorrect because tripping "C" Channel would result in an EFIC actuation.

References:

TS 3.3.11 Amendment 215

History:

New selected for 2010 SRO exam

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

QID: 0810 **Rev:** 0 **Rev Date:** 9/23/2009 **Source:** New **Originator:** S. Pullin

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

Section: 2 **Type:** Generic Knowledge and Abilities

System Number: 063 **System Title:** DC Electrical Distribution

Description: Ability to recognize system parameters that are entry-level conditions for Technical Specifications

K/A Number: 2.2.42 **CFR Reference:** 41.7/41.10/43.2/43.3/45.3

Tier: 2 **RO Imp:** 3.9 **RO Select:** No **Difficulty:** 3

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

Question: **RO:** **SRO:** 90

Which of the following conditions requires entry into Technical Specification 3.8.4, "DC Sources, Operating" and what is the bases for Technical Specification 3.8.4?

A. D04A, "Battery Charger" inoperable and D06, "Battery" operable.
Bases is to insure reactor coolant pressure boundary limits are not exceeded as a result of abnormalities

B. D04B, "Battery Charger" inoperable and D03B, "Battery Charger" inoperable.
Bases is to insure reactor coolant pressure boundary limits are not exceeded as a result of abnormalities

C. D04A, "Battery Charger" inoperable and D04B, "Battery Charger" inoperable.
Bases is to insure adequate core cooling is provided, and reactor building operability and other functions are maintained in the event of a postulated DBA

D. D03B, "Battery Charger" inoperable and D07, "Battery" operable.
Bases is to insure adequate core cooling is provided, and reactor building operability and other functions are maintained in the event of a postulated DBA

Answer:

C. D04A, "Battery Charger" inoperable and D04B, "Battery Charger" inoperable.
Bases is to insure adequate core cooling is provided, and reactor building operability and other functions are maintained in the event of a postulated DBA

Notes:

C is correct, with both battery chargers on the same train being inoperable, the subsystem is inoperable requiring entry into TS 3.8.4. The bases for TS 3.8.4 is to insure adequate core cooling is provided, and reactor building operability and other functions are maintained in the event of a postulated DBA
A is incorrect, Only one of the two charges being inoperable does not affect the operability of the subsystem. The bases used for this option is partially correct.
B is incorrect, two battery chargers are inoperable but since they are on different trains they do not affect the operability of either subsystem. The bases used for this option is partially correct.
D is incorrect, Only one of the two charges being inoperable does not affect the operability of the subsystem. The bases used for this option is partially correct.

References:

T.S. 3.8.4 Amendment 215

History:

New selected for 2010 SRO exam

ES-401 PWR Examination Outline Form ES-401-2
Plant Systems - Tier 2/Group 2 (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	#	QID	Type
001 Control Rod Drive												**Not selected**	**N/A**			
002 Reactor Coolant												**Not selected**	**N/A**			
011 Pressurizer Level Control												**Not selected**	**N/A**			
014 Rod Position Indication												**Not selected**	**N/A**			
015 Nuclear Instrumentation												**Not selected**	**N/A**			
016 Non-nuclear Instrumentation											X	**2.2.40 –** Ability to apply technical specifications for a system	4.7	91	599	D
017 In-core Temperature Monitor												**Not selected**	**N/A**			
027 Containment Iodine Removal												**Not selected**	**N/A**			
028 Hydrogen Recombiner and Purge Control												**2.4.23 –** Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations. **Rejected system replaced with 016 Non-Nuclear Instrumentation**	**N/A**			
029 Containment Purge												**Not selected**	**N/A**			
033 Spent Fuel Pool Cooling												**Not selected**	**N/A**			
034 Fuel Handling Equipment											X	**2.1.40 –**Knowledge of refueling administrative requirements.	3.9	92	600	D
035 Steam Generator								X				**A2.01 –** Faulted or ruptured S/Gs.	4.6	93	813	N
041 Steam Dump/Turbine Bypass Control												**Not selected**	**N/A**			
045 Main Turbine Generator												**Not selected**	**N/A**			
055 Condenser Air Removal												**Not selected**	**N/A**			
056 Condensate												**Not selected**	**N/A**			
068 Liquid Radwaste												**Not selected**	**N/A**			
071 Waste Gas Disposal												**Not selected**	**N/A**			
072 Area Radiation Monitoring												**Not selected**	**N/A**			
075 Circulating Water												**Not selected**	**N/A**			
079 Station Air												**Not selected**	**N/A**			
086 Fire Protection												**Not selected**	**N/A**			
K/A Category Point Totals:								**1**			**2**	Group Point Total:		**3**		

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

QID: 0599 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** J.Cork

TUOI: A1LP-RO-NNI **Objective:** 35 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation

Description: Ability to apply technical specifications for a system.

K/A Number: 2.2.40 **CFR Reference:** 41.10 / 43.2 / 43.5 / 45.3

Tier: 2 **RO Imp:** 3.4 **RO Select:** No **Difficulty:** 4

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

Question: **RO:** **SRO:** 91

REFERENCE PROVIDED

The plant is operating at 100% power.
Both PZR level transmitters LT-1001 and LT-1002 have failed LOW.

Which of the following actions are required by Technical Specification 3.3.15 and Table 3.3.15-1?

A. Be in Mode 3 within 6 hours.

B. Both channels must be restored within 7 days.

C. Restore one channel to operable status within 30 days or be in Mode 3 within 6 hours.

D. Restore one channel to operable status within 7 days or be in Mode 3 within 6 hours.

Answer:

D. Restore one channel to operable status within 7 days or be in Mode 3 within 6 hours.

Notes:

Answer "D" is correct in accordance with Table 3.3.15-1 and actions C and E.
Answer "A" is incorrect, there is still an allowance of 7 days per action C.
Answer "B" is incorrect, only one channel must be restored.
Answer "C" is inocrrect, this is a combination of A and E.

References:

T.S. 3.3.15 Amendment 232

Note: T.S. 3.3.15 must be in students' handout.

History:

Direct from regular exam bank QID#ANO-OPS1-6623
Selected for 2005 SRO exam.
Selected for 2010 SRO exam.

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

QID: 0600 **Rev:** 0 **Rev Date:** 6/27/05 **Source:** Direct **Originator:** J.Cork

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

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

System Number: 034 **System Title:** Fuel Handling Equipment

Description: Knowledge of refueling administrative requirements.

K/A Number: 2.1.40 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 2 **RO Imp:** 2.8 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** K

Question: **RO:** **SRO:** 92

Given:

- Plant is in a Refueling outage.
- Core re-load is in progress.
- Approximately 90% of the core is in the Reactor vessel.

The Main Fuel Handling Bridge has a once-burned fuel assembly and is in the process of indexing over the specified core location when NI-502 fails to 0.1 cps.

What action should be taken?

A. No action necessary because with NI-501 operating, Tech Spec NI requirements for operablility are met.

B. Contact the Main Fuel Bridge operator and place the assembly in a core location without any adjacent fuel assemblies.

C. Halt operations on the Main Fuel Bridge. Core geometry cannot be changed unless two neutron flux monitors are operable.

D. Verify boron concentration in the Refueling Canal is greater than 2326 ppm and then continue fuel load.

Answer:

C. Halt operations on the Main Fuel Bridge. Core geometry cannot be changed unless two neutron flux monitors are operable.

Notes:

Answer "C" is correct per 1502.004, 5.3, and T.S. 3.9.2
Answer "A" is incorrect, although only one is required in Mode 6, two NI's are required during core alterations.
Answer "B" is incorrect, this is still a core alteration.
Answer "D" is incorrect, this is simply a requirement for refueling.

References:

1502.004, Chg. 041
T.S. 3.9.2 Amendment 215

History:

Direct from regular exam bank QID#3178
Selected for 2005 SRO exam.
Selected for 2010 SRO exam

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

QID: 0813 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

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

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

System Number: 035 **System Title:** Steam Generator System (S/GS)

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the S/G and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulted or ruptured S/Gs.

K/A Number: A2.01 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier:	2	**RO Imp:**	4.5	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.6	**SRO Select:**	Yes	**Taxonomy:**	An

Question: **RO:** **SRO:** 93

Given:

- Plant at 100% power

Simultaneously the following occurs:

- Reactor trips on low RCS Pressure
- N-16 alarm on "A" Steam Generator
- Steam Line High Range Radiation monitor RI-2681 in alarm.
- RCS pressure drops to 1300 psig
- CET's indicate 550°F
- Reactor Building and Aux Building sump levels are stable.

Starting with 1202.001, Reactor Trip EOP, which of the following lists the order of EOP's to mitigate this event?

A. 1202.002 Loss of Subcooling Margin and 1202.006 Tube Rupture

B. 1202.002 Loss of Subcooling Margin and 1202.010 ESAS

C. 1202.006 Tube Rupture and 1202.010 ESAS

D. 1202.006 Tube Rupture and 1202.012 RT-10

Answer:

A. 1202.002 Loss of Subcooling Margin and 1202.006 Tube Rupture

Notes:

A is correct, The Reactor Trip EOP immediate actions will send the operator to Loss of Subcooling margin, with the only LOCA being a tube rupture the Loss of Subcooling Margin procedure will send the operator to Tube Rupture.
B is incorrect, ESAS would only be entered if RCS pressure dropped below 150 psig.
C and D are incorrect, Reactor Trip would send the operator to Loss of Subcooling Margin EOP first.

References:

OP-1202.001 Change 031
OP-1202.002 Change 006

History:

New selected for 2010 SRO exam

Facility: Arkansas Nuclear One – Unit 1		Date of Exam: 3/5/2010				
Category	K/A #	Topic	SRO			
			IR	#	QID	Type#
1. Conduct of Operations	2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.7	94	492	D
	2.1.35	Knowledge of the fuel-handling responsibilities of SROs	3.9	95	814	N
	Subtotal		2			
2. Equipment Control	2.2.25	Knowledge of bases and technical specifications for limiting conditions of operations and safety limits.	4.2	96	646	D
	2.2.19	Knowledge of maintenance work order requirements.	3.4	97	815	N
	Subtotal		2			
3. Radiation Control	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.8	98	816	N
	Subtotal		1			
4. Emergency Procedures / Plan	2.4.30	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.	4.1	99	411	D
	2.4.35	Knowledge of local auxiliary operator tasks during an emergency and the operational resultant effects.	4.0	100	750	D
	Subtotal		2			
Tier 3 Point Total			7			

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

QID: 0492 **Rev:** 1 **Rev Date:** 12/4/06 **Source:** Direct **Originator:** S.Pullin

TUOI: A1LP-RO-EOP08 **Objective:** 7 **Point Value:** 1

Section: 2.0 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

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

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

Tier: 3 **RO Imp:** 4.4 **RO Select:** No **Difficulty:** 4

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

Question: **RO:** **SRO:** 94

Given the plant conditions following a reactor trip:

- RCS temperature: 605 degrees stable
- RCS pressure: 2300 psig slowly dropping
- ERV: open in AUTO
- OTSG shell temperature: 558 degrees
- OTSG levels 20 inches, steady
- PZR level 180 inches, rising

Which of the following actions are required?

A. Trip the running RCP per 1202.002, Loss of Subcooling Margin.

B. Isolate the ERV per 1202.001, Reactor Trip.

C. Select the reflux boiling setpoint per RT-5.

D. Initiate Full HPI per RT 3.

Answer:

B. Isolate the ERV per 1202.001, Reactor Trip.

Notes:

Answer "B" is correct. A pressurizer steam space leak is indicated by PZR level rising with RCS pressure dropping and no rise in RCS temperature. ERV is open and should have closed at 2395 psig.
Answer "A" is incorrect, Tube to Shell delta T of 60 degrees tubes hotter would require this action however the delta T is only 47 degrees in the question.
Answer "C" is incorrect, although RCS temperature/pressure conditions are close to a loss of subcooling margin which would require selection of Reflux Boiling but SCM is still adequate.
Answer "D" is incorrect, Full HPI would be required if the ERV opened in Auto with the Overheating EOP in effect but the Overheating entry conditions are not met.

References:

1202.001, Chg. 031

History:

Modified from regular exambank QID#3314.
Used on 2004 SRO Exam.
Modified for use on 2007 SRO Exam.
Selected for 2010 SRO exam.

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

QID: 0814 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

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

Section: 2 **Type:** Generic Knowledge and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of the fuel-handling responsibilities of SROs

K/A Number: 2.1.35 **CFR Reference:** 41.10 / 43.7

Tier:	3	**RO Imp:**	2.2	**RO Select:**	No	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** **SRO:** 95

Which of the following conditions would require the SRO in charge of fuel handling to order a stop to fuel movement in the Reactor Building?

A. Outage Control Center reports that the reactor has been subcritical for 90 hours.

B. National Weather Service declares a Tornado Watch in effect for Conway County.

C. One Control Room Emergency Air Conditioning System (CREACS) inoperable for the past 5 days.

D. Reactor Building Radiation monitor RE-8017 inoperable, and portable survey instrument is being monitored on the fuel handling bridge.

Answer:

A. Outage Control Center reports that the reactor has been subcritical for 90 hours.

Notes:

A is correct, the reactor must be subcritical for greater than 100 hours prior to fuel movement.
B is incorrect, Pope, Johnson, Yell and Logan counties in a tornado watch would require stopping fuel movement. Conway county is immediately east of Pope county.
C is incorrect, with one CREACS channel inoperable we have 30 days to repair prior to stopping fuel movement.
D is incorrect, RE-8017 is desired to be operable for monitoring radiation levels on the bridge, however if it becomes inoperable any portable survey instrument is allowed for monitoring rad levels and continue fuel movement.

References:

OP-1502.004 Change 041

History:

New selected for 2010 SRO exam.

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

QID: 0646 **Rev:** 0 **Rev Date:** 10/23/2006 **Source:** Direct **Originator:** Cork/Possage

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

Section: 2 **Type:** Generic Knowledge and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

K/A Number: 2.2.25 **CFR Reference:** 41.5 / 41.7 / 43.2

Tier: 3 **RO Imp:** 3.2 **RO Select:** No **Difficulty:** 3

Group: G **SRO Imp:** 4.2 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 96

Given:

- #1 EDG has one Air Start Compressor and it's associated Air Receiver Tanks tagged out.
- The remaining Air Start Compressor on #1 EDG trips while EDG is running for a surveillance.
- The Air Receiver Tanks' pressure is 145 psig.

In accordance with Technical Specifications, what is the required action for the above conditions?

A. No actions are necessary since the EDG is running and an air start system is not needed.

B. Restore required starting air receiver pressure to within limits in 48 hours.

C. Declare #1EDG inoperable immediately.

D. Be in Mode 3 within 12 hours.

Answer:

C. Declare #1EDG inoperable immediately.

Notes:

Answer "C" is correct, with only one receiver bank and pressure <158 psig the EDG must be declared inoperable per 3.8.3.E.1.
Answer "A" is incorrrect, although the EDG is running, if it tripped there would not be enough air for a re-start.
Answer "B" is incorrect, this is the action from 3.8.3.D and would be applicable if pressure was between 158 and 175 psig.
Answer "D" is incorrect, this action is from 3.8.1.F and would be applicable if the EDG was not made operable within 7 days.

References:

3.8.3 and Bases Amendment 215

History:

Uses QID 447 stem with some modifications, all answers are different, therefore it is a new question.
New question for 2007 SRO exam.
Selected for the 2010 SRO exam

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

QID: 0815 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

TUOI: ASLP-SRO-MNTC **Objective:** 2 **Point Value:** 1

Section: 2 **Type:** Generic K&A

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of maintenance work order requirements

K/A Number: 2.2.19 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	3	**RO Imp:**	2.3	**RO Select:**	No	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.4	**SRO Select:**	Yes	**Taxonomy:**	K

Question: **RO:** **SRO:** 97

Given:
- Power 100%
- All equipment operable and there are no Tech Spec LCOs in effect

- Annunciator K12-B5, P-7A Turbine Trip alarm is received
- WCO reports that the linkage for the trip throttle valve has broken.

You are the Shift Manager,

Per EN-WM-100, "Work Request (WR) Generation, Screening and Classification," which work order process should be used to correct this condition.

A. Emergency Maintenance

B. Priority One Work Order

C. Priority Two Work Order

D. Priority Three Work Order

Answer:

B. Priority One Work Order

Notes:

B is correct, since P-7A inoperability is a 72 hour Time Clock, a Priority 1 work order would be initiated to begin maintenance and work around the clock to completion.
A is incorrect, Emergency Maintenance is only used to protect the public or prevent serious injury/death
C is incorrect, Priority 2 work orders are entered into the T-3 week schedule and would exceed TS time limit
D is incorrect, Priority 3 work orders timing would exceed TS time limit

References:

EN-WM-100 Rev 3

History:

New selected for 2010 SRO exam

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

QID: 0816 **Rev:** 0 **Rev Date:** 9/24/2009 **Source:** New **Originator:** S Pullin

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

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

K/A Number: 2.3.14 **CFR Reference:** 41.12 / 43.4 / 45.10

Tier: 3 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 3

Group: G **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 98

During a fuel handling accident Krypton-85 is the major source of gaseous activity released from a damaged Fuel assembly that has decayed for >190 days.

Which portion of the body will receive the highest dose after a fuel handling accident?

A. Skin dose from Beta

B. Whole body dose from Gamma

C. Extremities dose from Beta

D. Internal Organ dose from Gamma

Answer:

A. Skin dose from Beta

Notes:

A is correct, skin dose rates from K-85 are 100 times higher than the whole body , gamma dose rates.
B, C, and D are all incorrect.

References:

OP-1203.042 Change 005-03-0

History:

New selected for 2010 SRO exam

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

QID: 0411 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** Direct **Originator:** E-Plan

TUOI: ASLP-RO EPLAN **Objective:** 7 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

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

Description: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

K/A Number: 2.4.30 **CFR Reference:** 41.10 / 43.5 / 45.11

Tier: 3 **RO Imp:** 2.7 **RO Select:** No **Difficulty:** 2

Group: G **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 99

A fire was reported at 0844 in the vicinity of the Old Radwaste Building.
It is now 0920 and the fire is still burning.

What is the Emergency Plan time requirement for notification to the NRC?

A. Notification to the NRC is required within 15 minutes of the declaration of an emergency class.

B. Notification to the NRC is required immediately following notification of the ADH and within 1 hour of the declaration of an emergency class.

C. Notification to the NRC is required immediately following declaration of an emergency class and notify the ADH within 1 hour.

D. Notification to the NRC is required within 4 hours of the declaration of an emergency class.

Answer:

B. Notification to the NRC is required immediately following notification of the ADH and within 1 hour of the declaration of an emergency class.

Notes:

Answer [B] is correct since this is the procedural requirement.
Answer [A], [C], [D] are incorrect, these are not in accordance with 1903.011.

References:

1903.011Y, Emergency Initial Notification Message Change 036

History:

Modified E-Plan exam bank QID#61 for use in 2001 SRO Exam.
Selected for use in 2002 SRO exam.
Selected for 2010 SRO exam

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

[redacted]

QID: 0750 **Rev:** 2 **Rev Date:** 6/23/08 **Source:** Direct **Originator:** Spullin

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

Section: 2.0 **Type:** Generic K/As

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

Description: Knowledge of local auxiliary operator tasks during an emergency and the operational resultant effects

K/A Number: 2.4.35 **CFR Reference:** 41.10/43.5/45.13

Tier: 3 **RO Imp:** 3.8 **RO Select:** No **Difficulty:** 3

Group: **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

Question: **RO:** **SRO:** 100

Given:
- Severe Fire on 335 Auxiliary Building on Unit 1
- Reactor has been tripped

Which of the following actions would the CRS direct the Outside AO to perform and what procedural guidance would be used?

A. Fire fighting tasks per "Fire or Explosion" procedure 2203.034.

B. Securing Polishers per "Reactor Trip/Outage Recovery" procedure 1102.006.

C. Placing the Startup Boiler in service per "Startup Boiler Operation" procedure 1106.022.

D. Throttle CV-2627 EFW Supply to "A" SG per "Fires in Areas Affecting Safe Shutdown" procedure 1203.049

Answer:

D. Throttle CV-2627 EFW Supply to "A" SG per "Fires in Areas Affecting Safe Shutdown" procedure 1203.049

Notes:

"A." is incorrect; due to recent procedures changes have the opposite Unit AO's fighting the fire, but the WCO non licensed operator has fire fighting duties
"B." is incorrect; under normal Reactor trip conditions this would be an Outside AO action promptly following Rx trip, but it is not in the Reactor procedure
"C." is incorrect; under normal Reactor trip conditions this would be an Outside AO would perform following Rx trip
"D." is correct; this is a new procedure action for the non licensed operators

References:

1203.049 Fires in Areas affecting Safe Shutdown Change 005

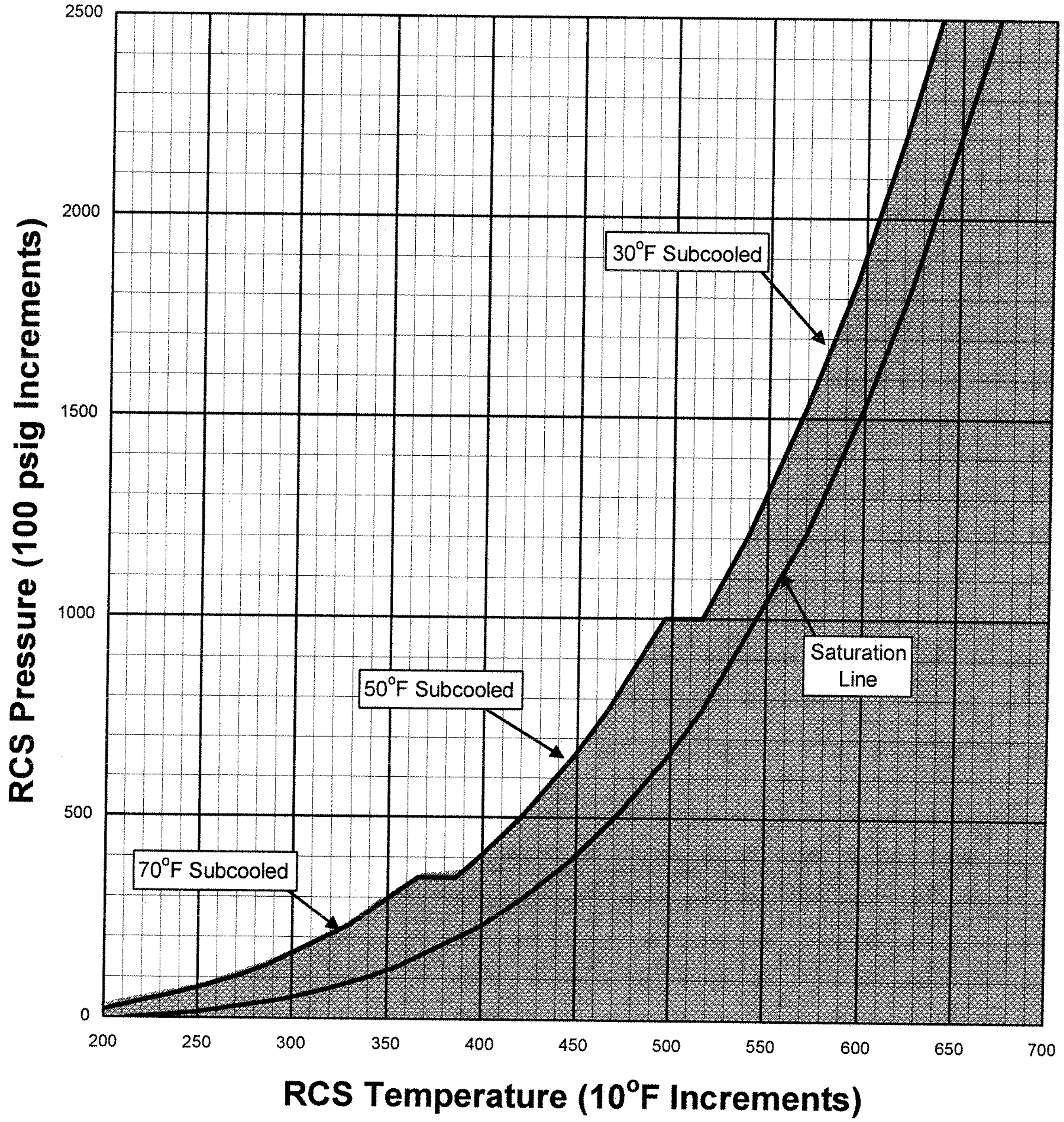
History:

Selected for 2010 SRO exam

RO NRC Exam

HANDOUT FOR 2010 EXAM

FIGURE 1
Saturation and Adequate SCM



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

FIGURE 2
SG Pressure vs T-sat

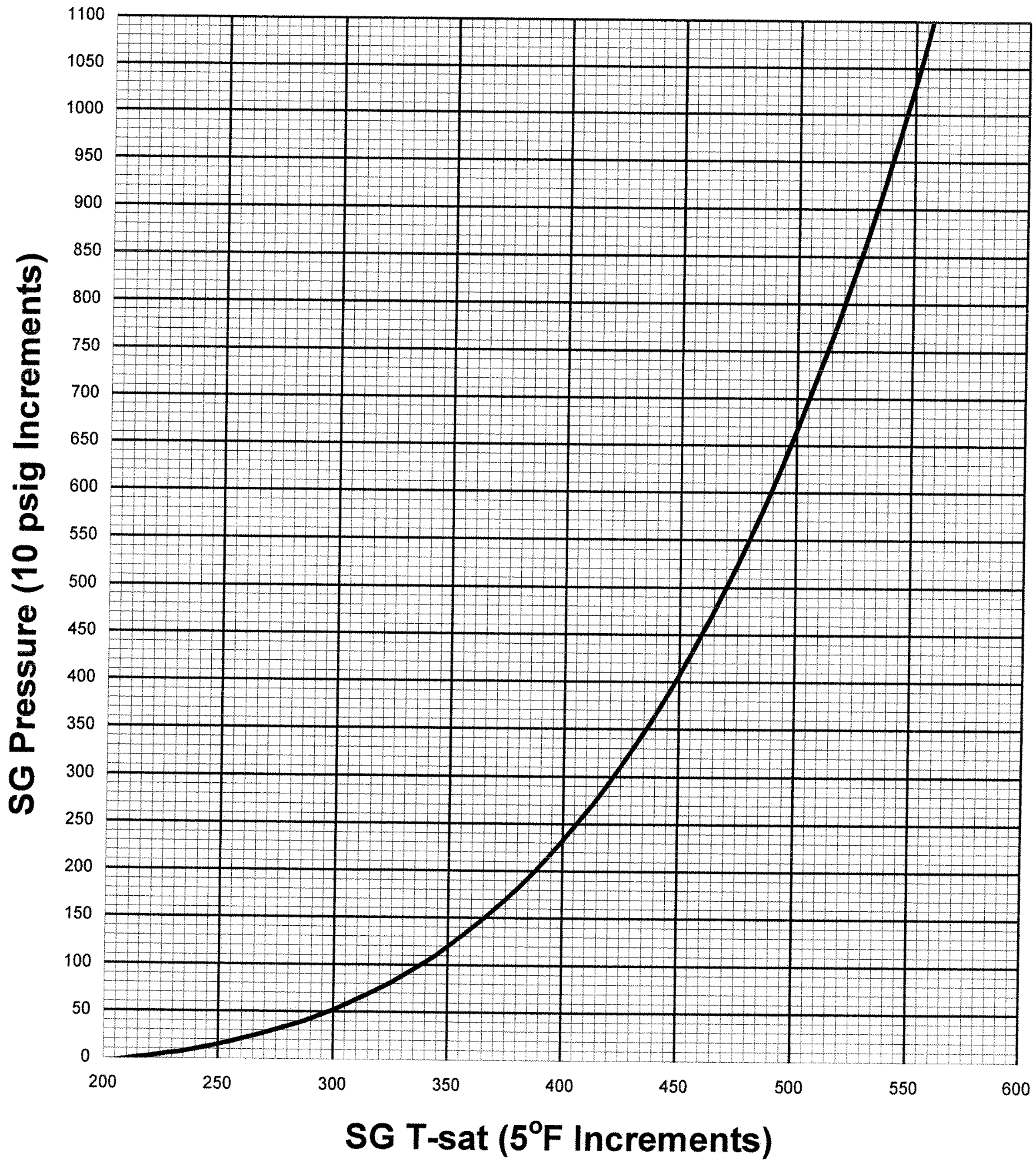


FIGURE 3
RCS Pressure vs Temperature Limits

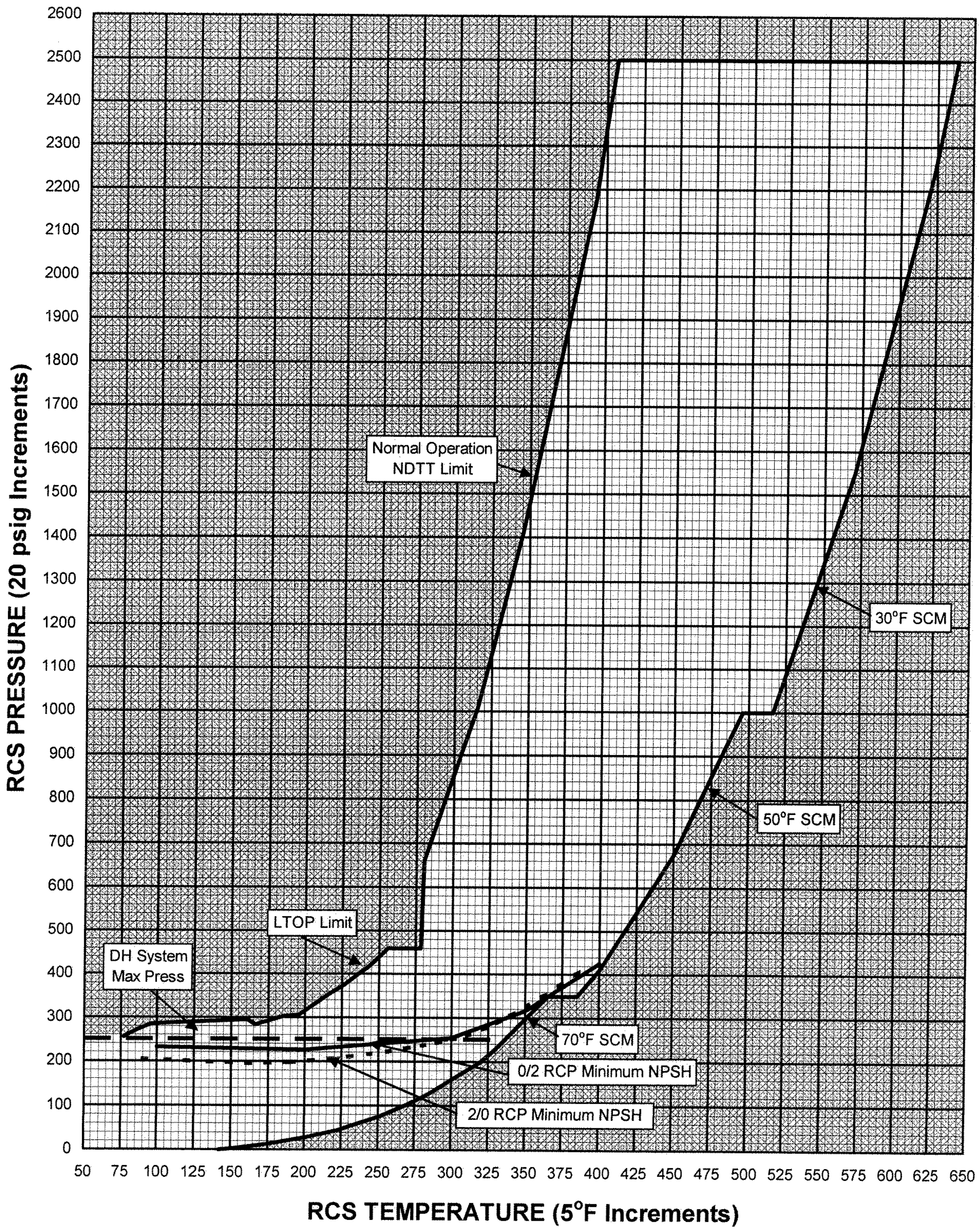


FIGURE 4
Core Exit Thermocouple for Inadequate Core Cooling

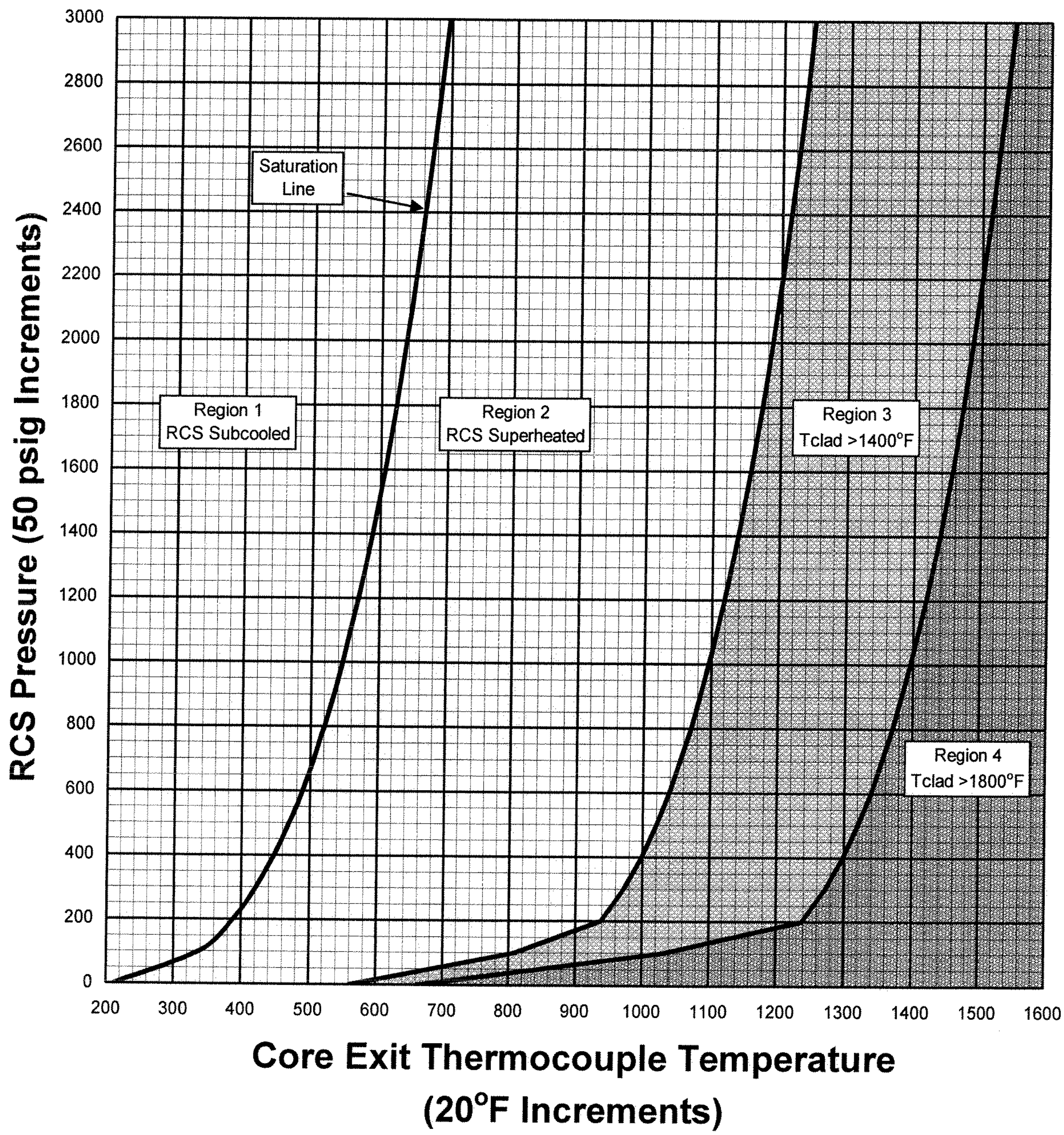


FIGURE 5

SG Pressure to Establish 40° to 60°F Primary to Secondary ΔT

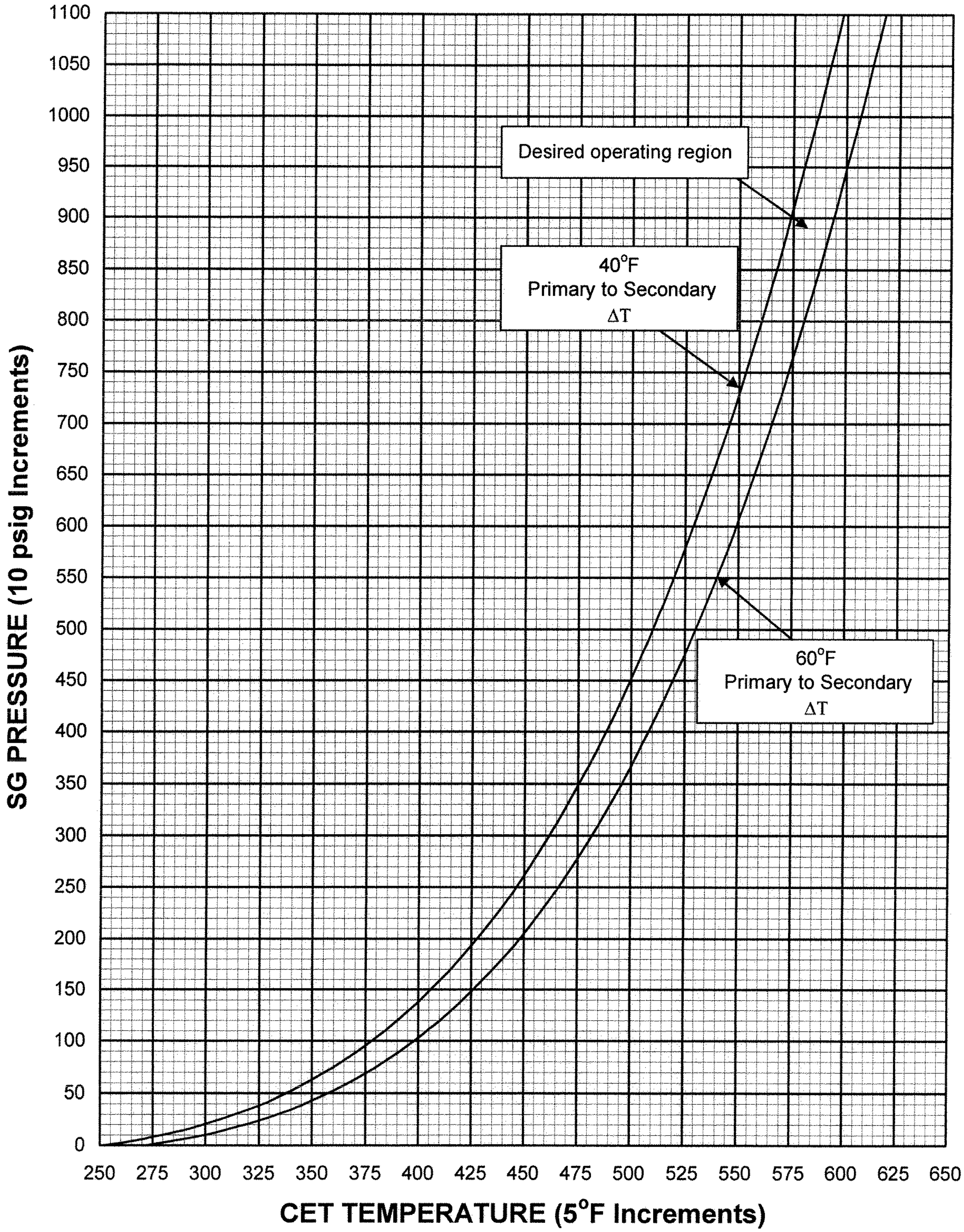
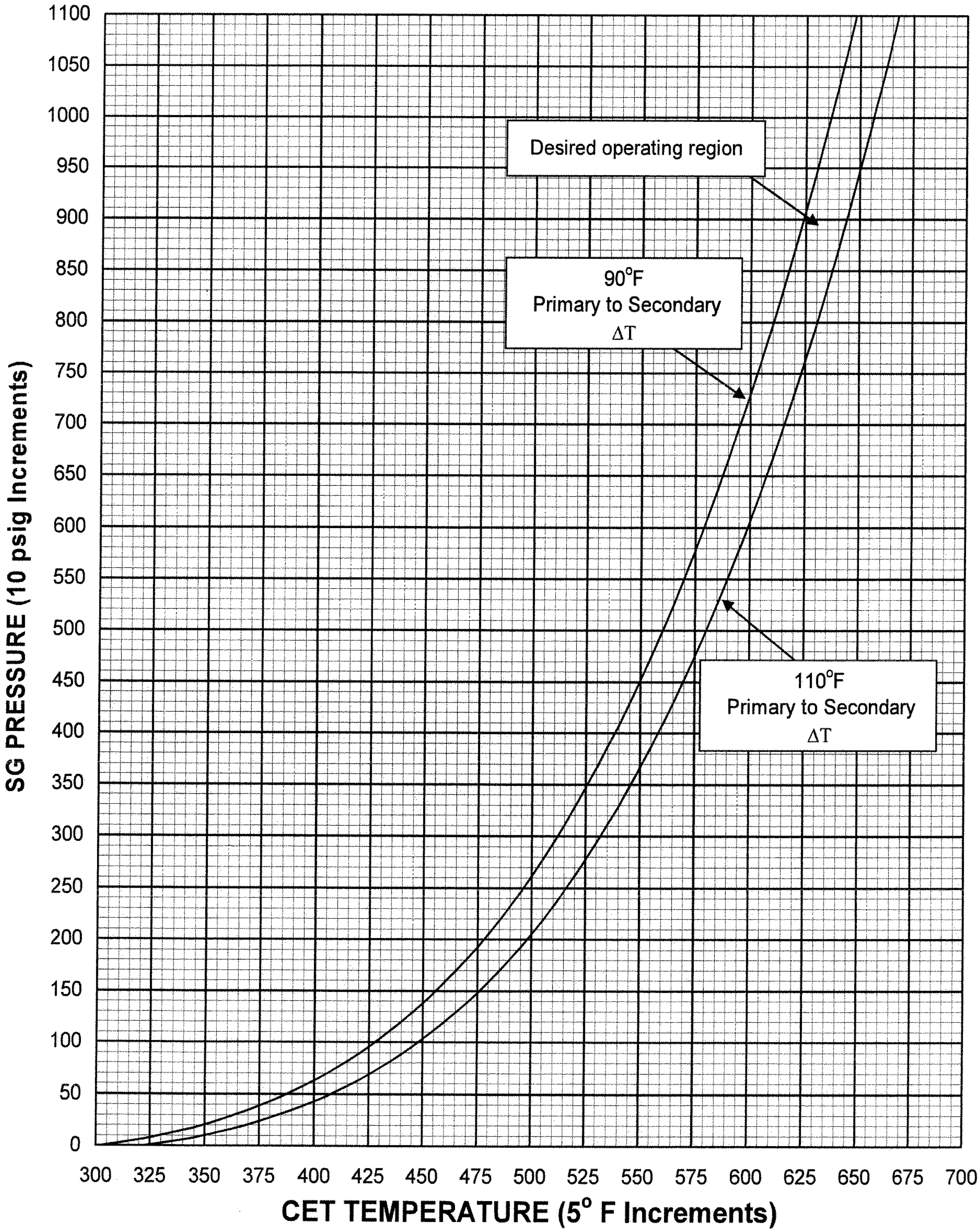


FIGURE 6

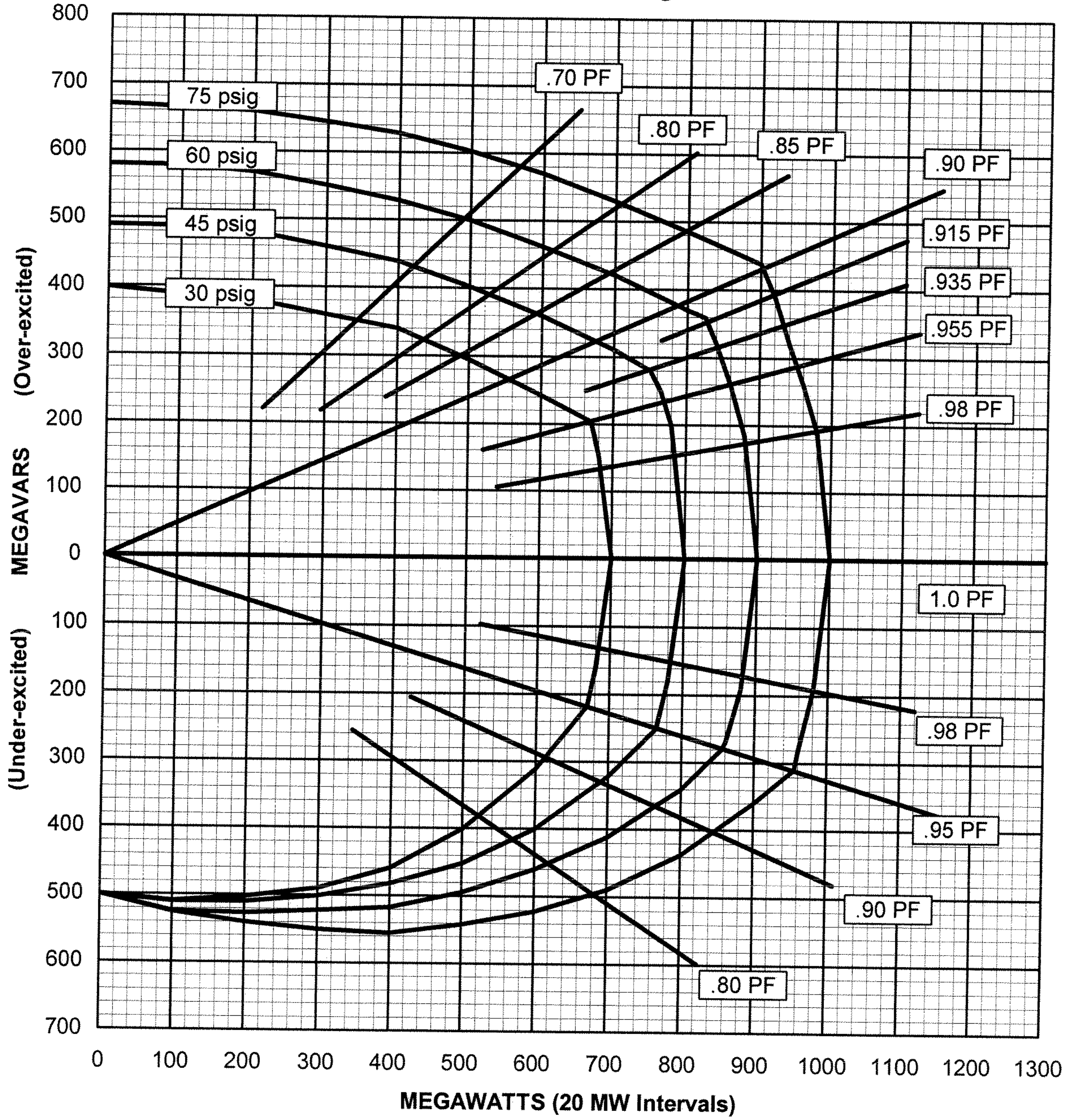
SG Pressure to Establish 90° to 110°F Primary to Secondary ΔT



ATTACHMENT N

page 1 of 1

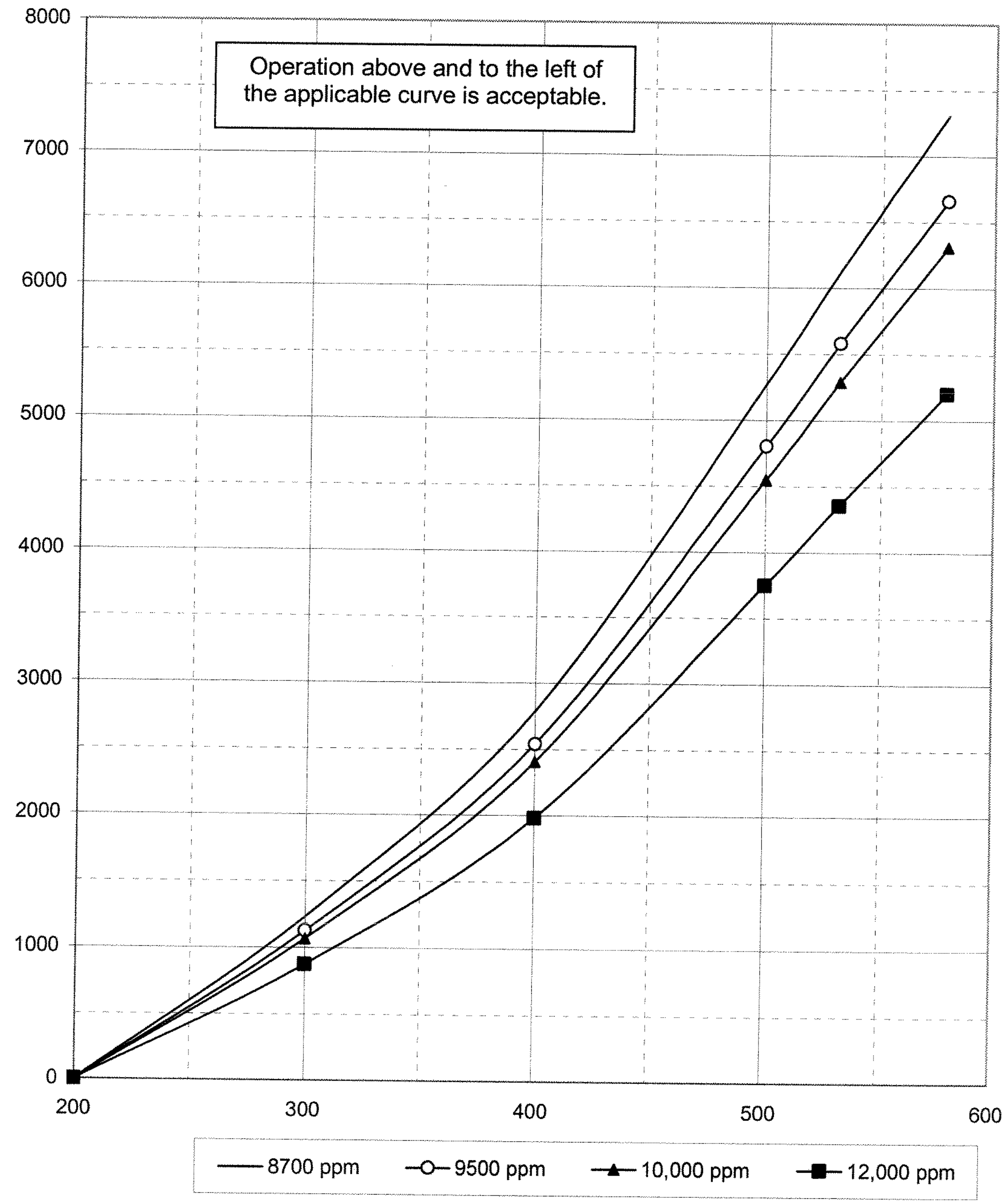
Hydrogen Inner-Cooled Turbine Generator Calculated Capability Curve at Rated Voltage



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

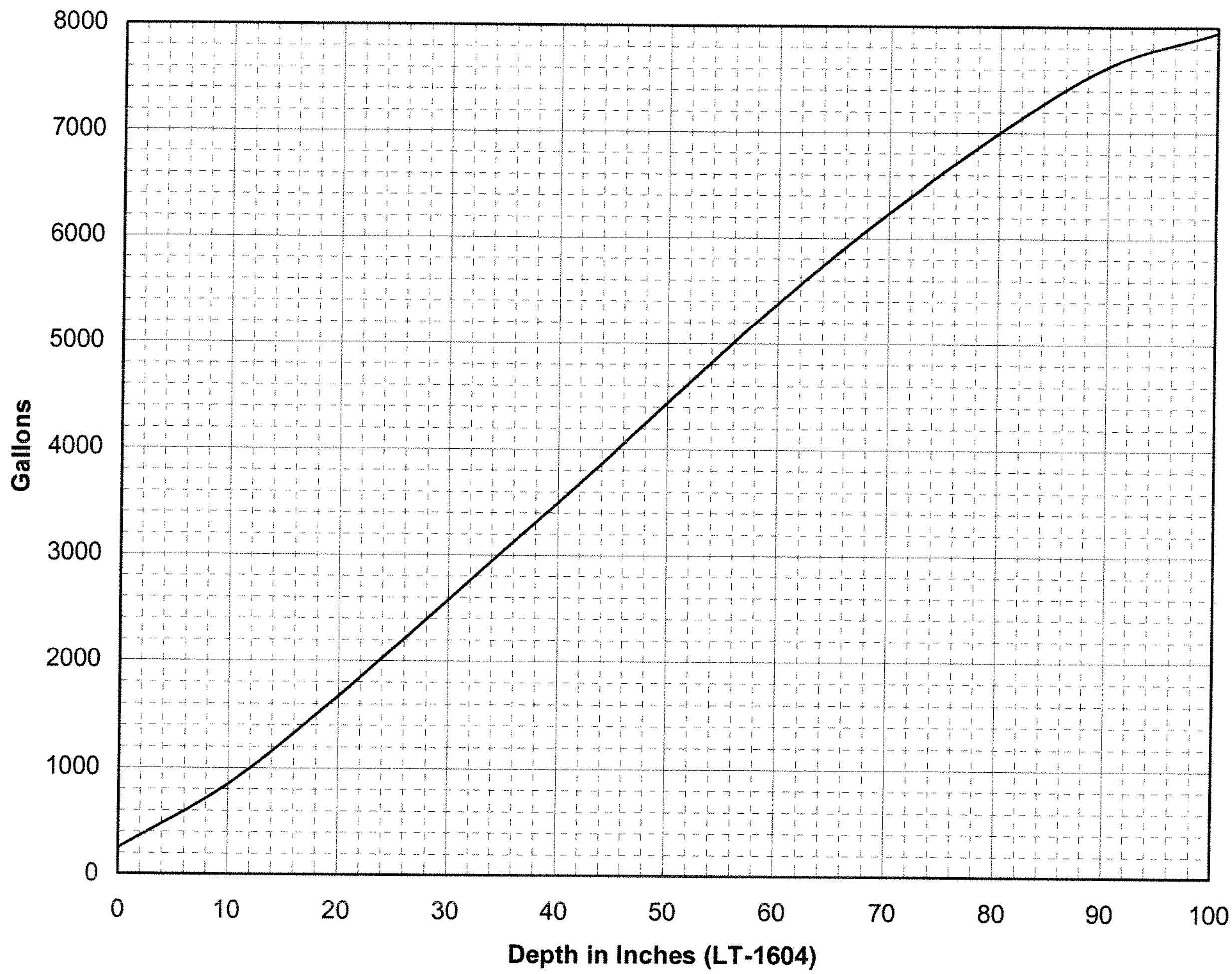
ATTACHMENT G

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



ATTACHMENT H

Volume of BAAT vs. Depth of Liquid



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

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

1.2 Subtract feed volume from initial tank level.

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

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

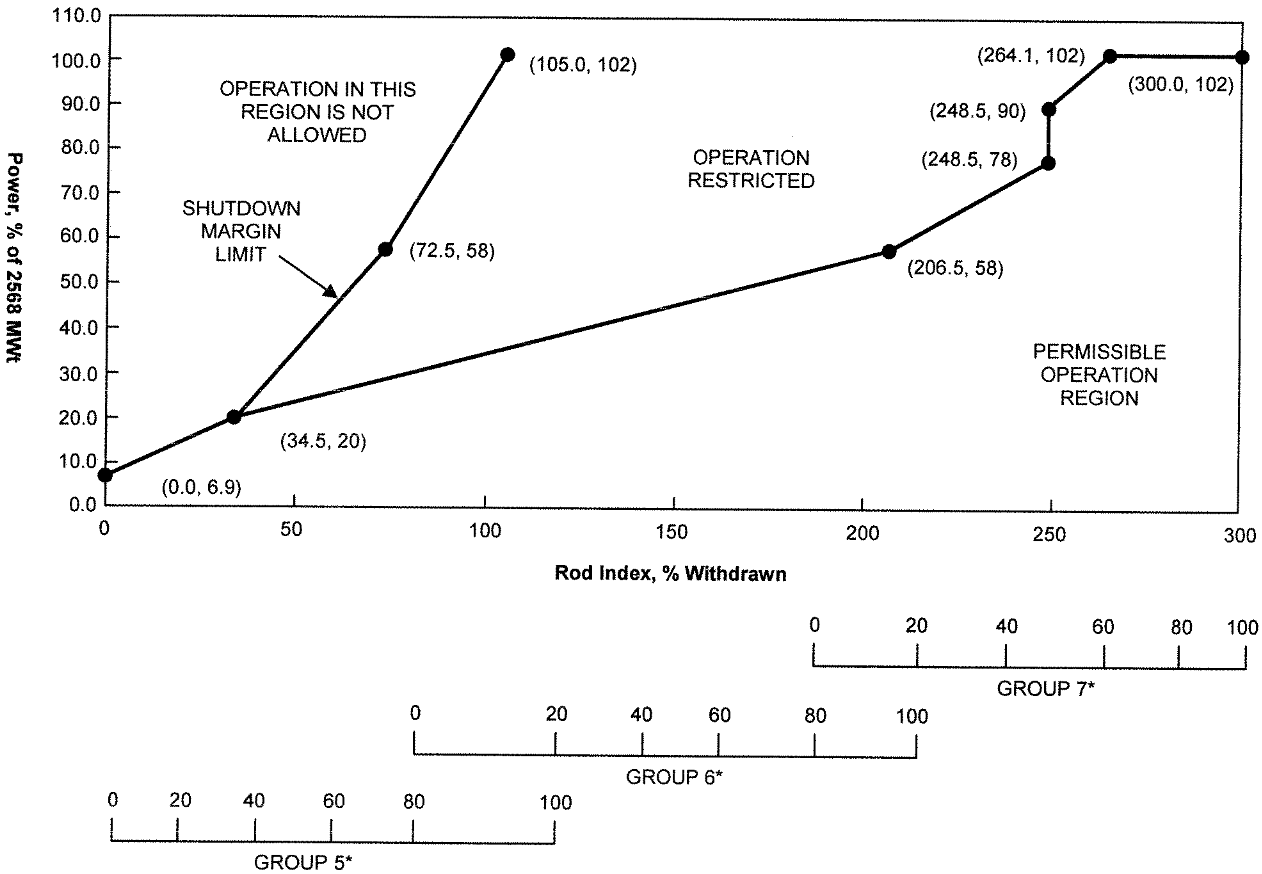
B. Initial volume - feed volume = 7100 - 530 = 6570 gal.

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

Figure 3-A

Regulating Rod Insertion Limits for Four-Pump Operation From 0 to 200 ± 10 EFPD

(Figure is referred to by Technical Specification 3.2.1)

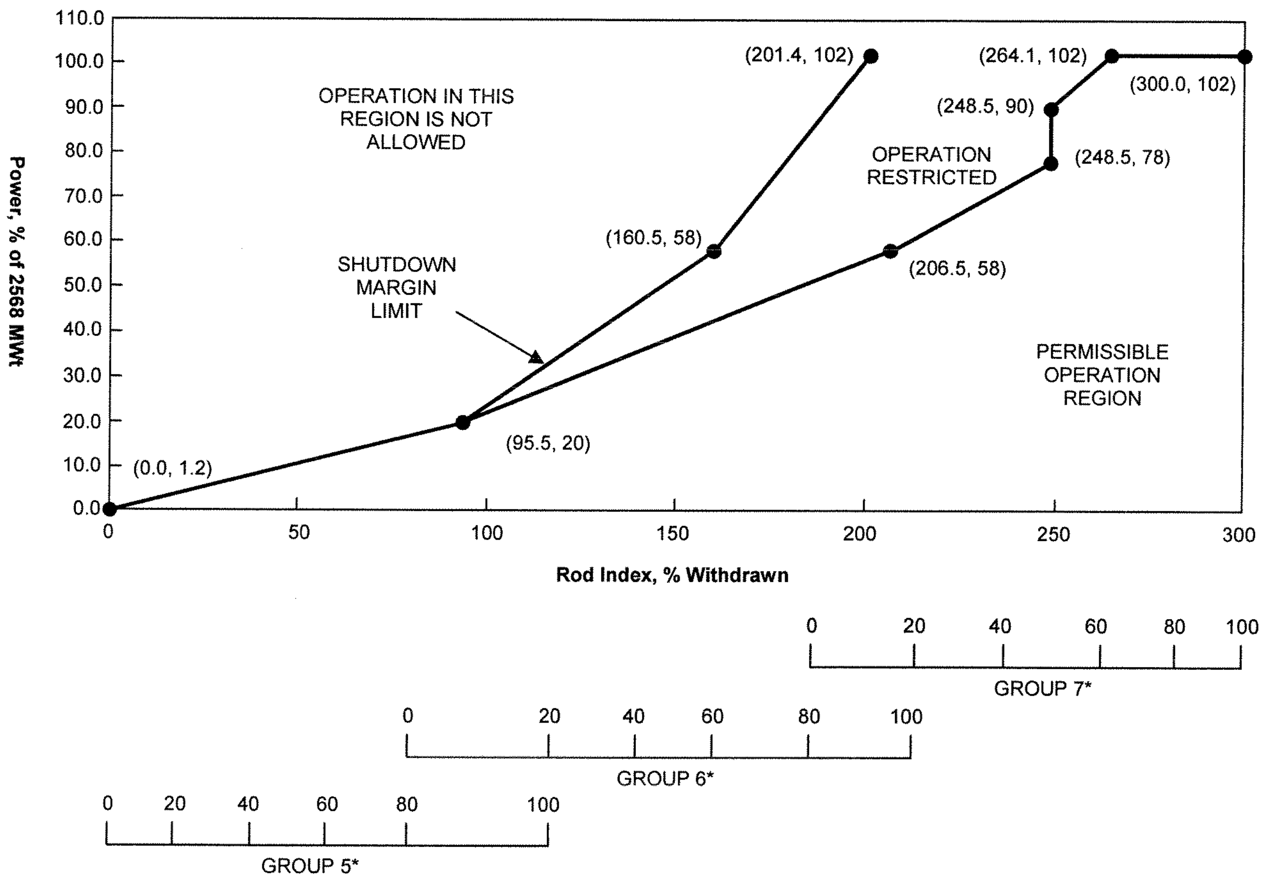


* Technical Specification 3.5.2.5(2) requires that operating rod group overlap be 20% ± 5% between two sequential groups, except for physics tests.

Figure 3-B

Regulating Rod Insertion Limits for Four-Pump Operation From 200 ± 10 EFPD to EOC

(Figure is referred to by Technical Specification 3.2.1)



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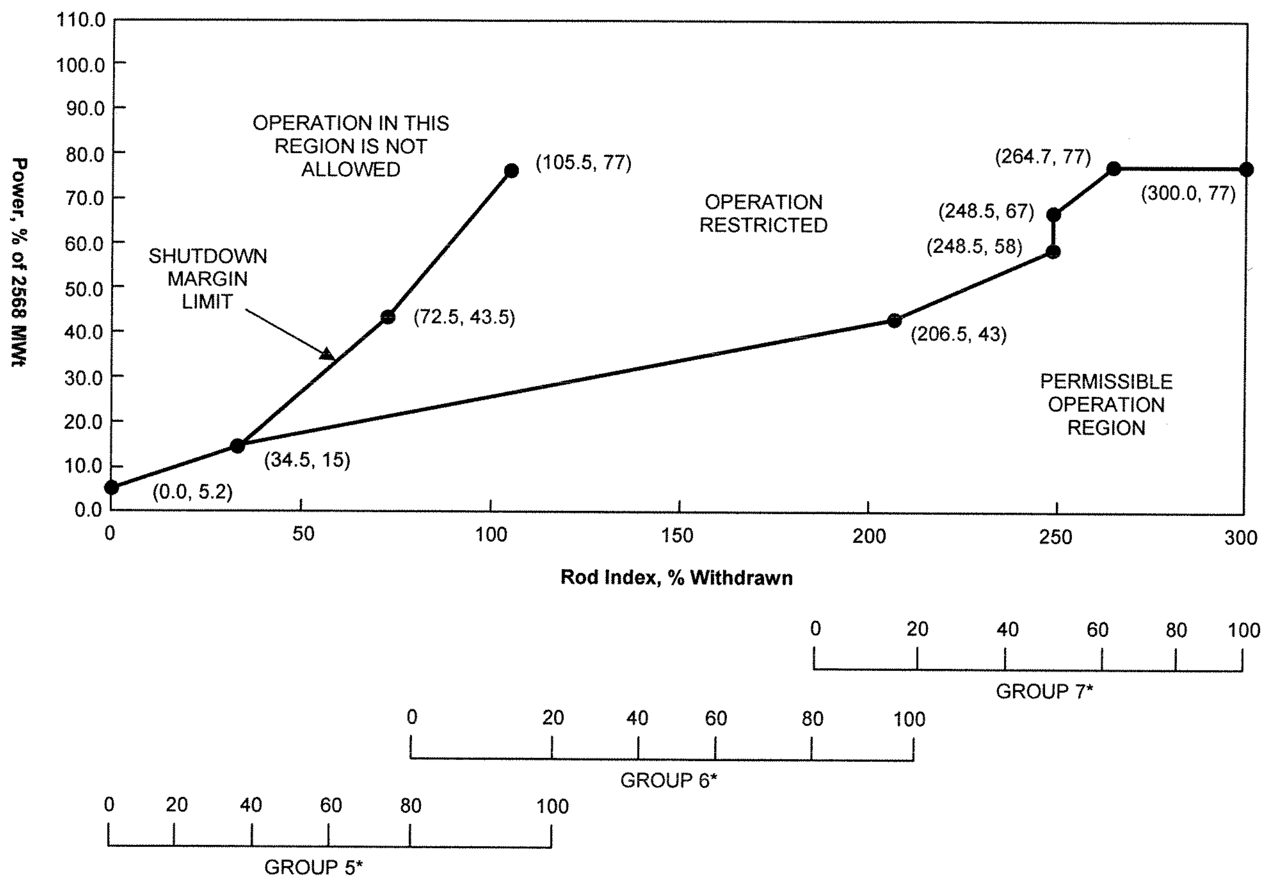

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SRO NRC Exam

HANDOUT FOR 2010 EXAM

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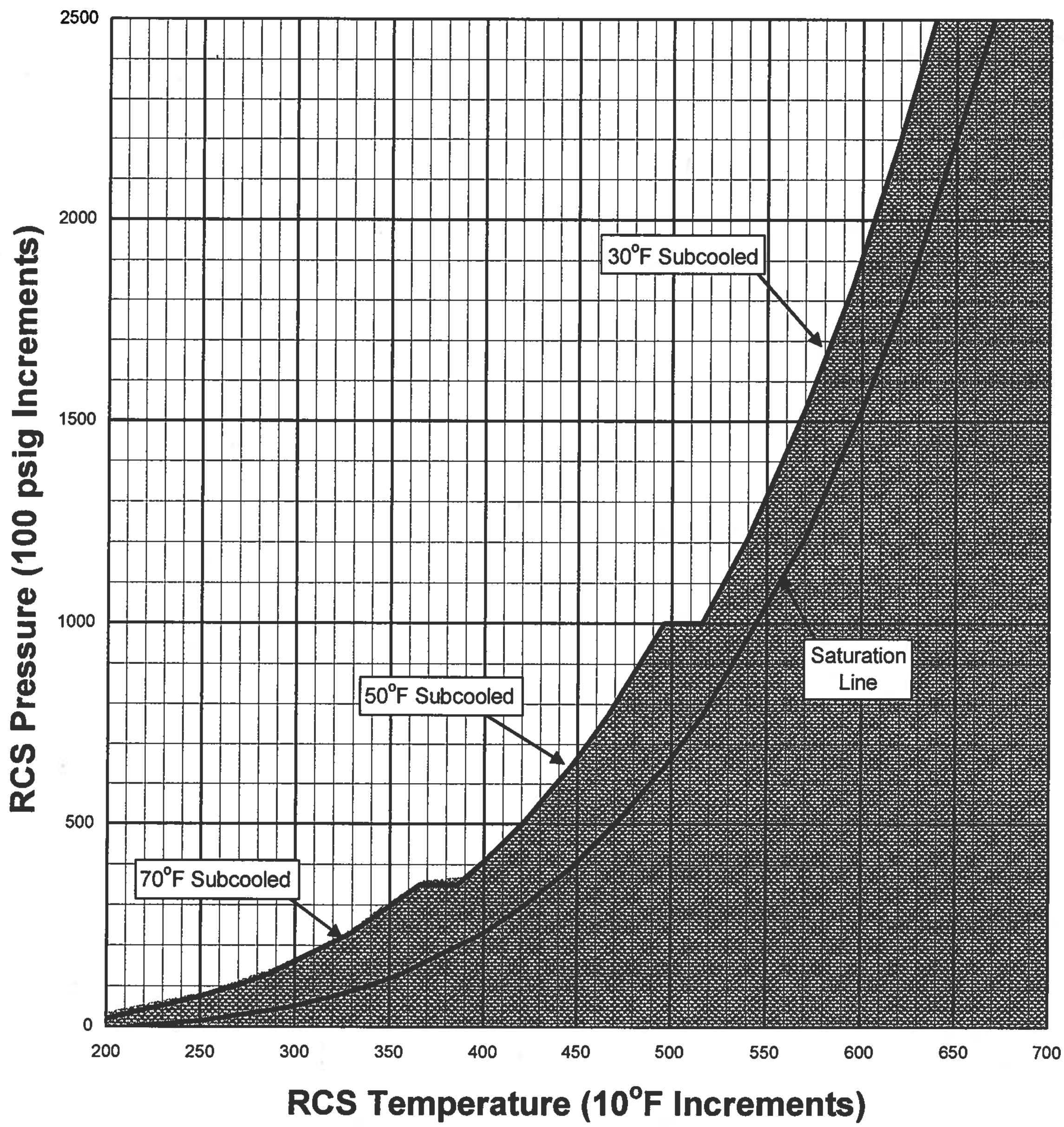

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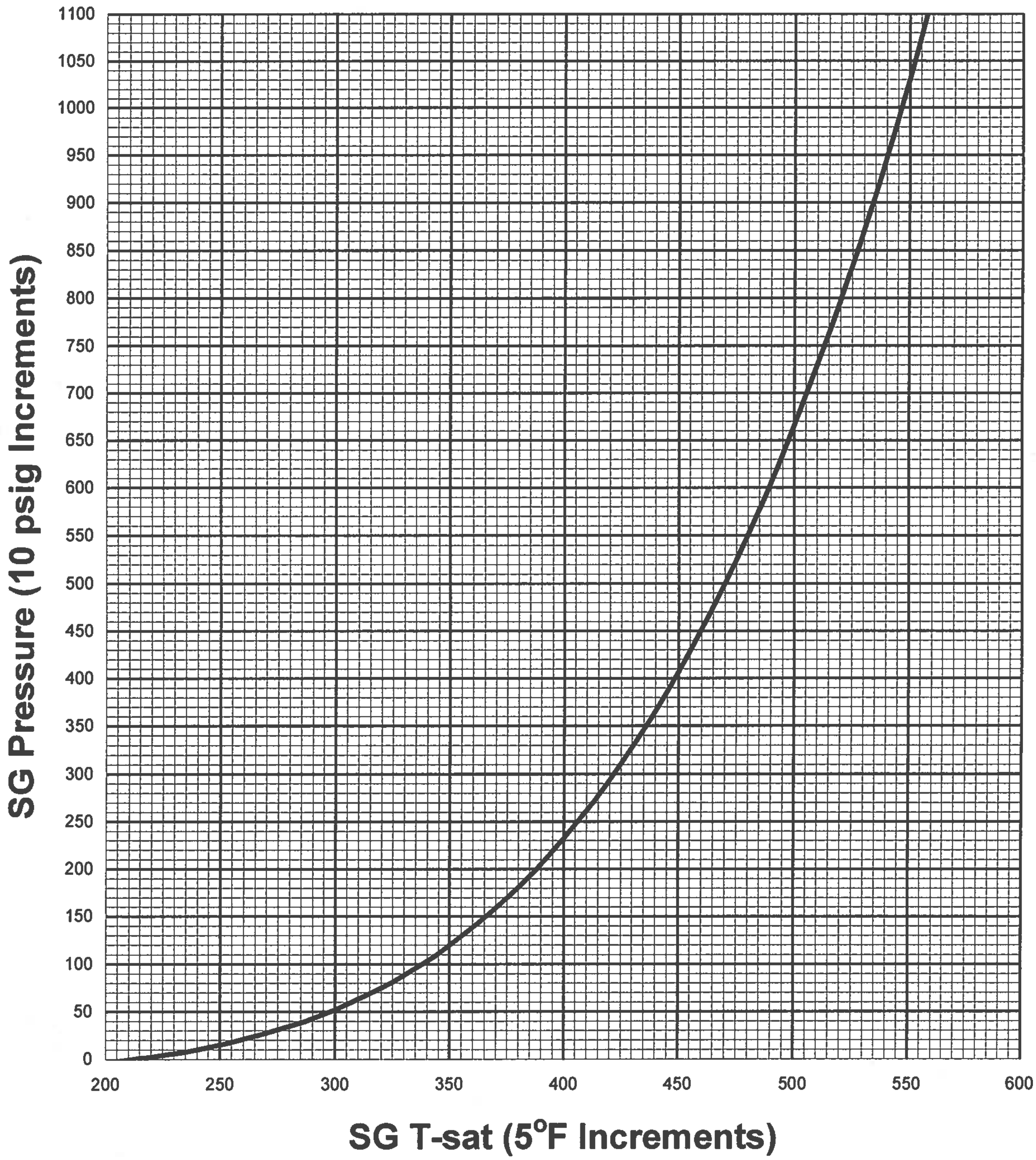
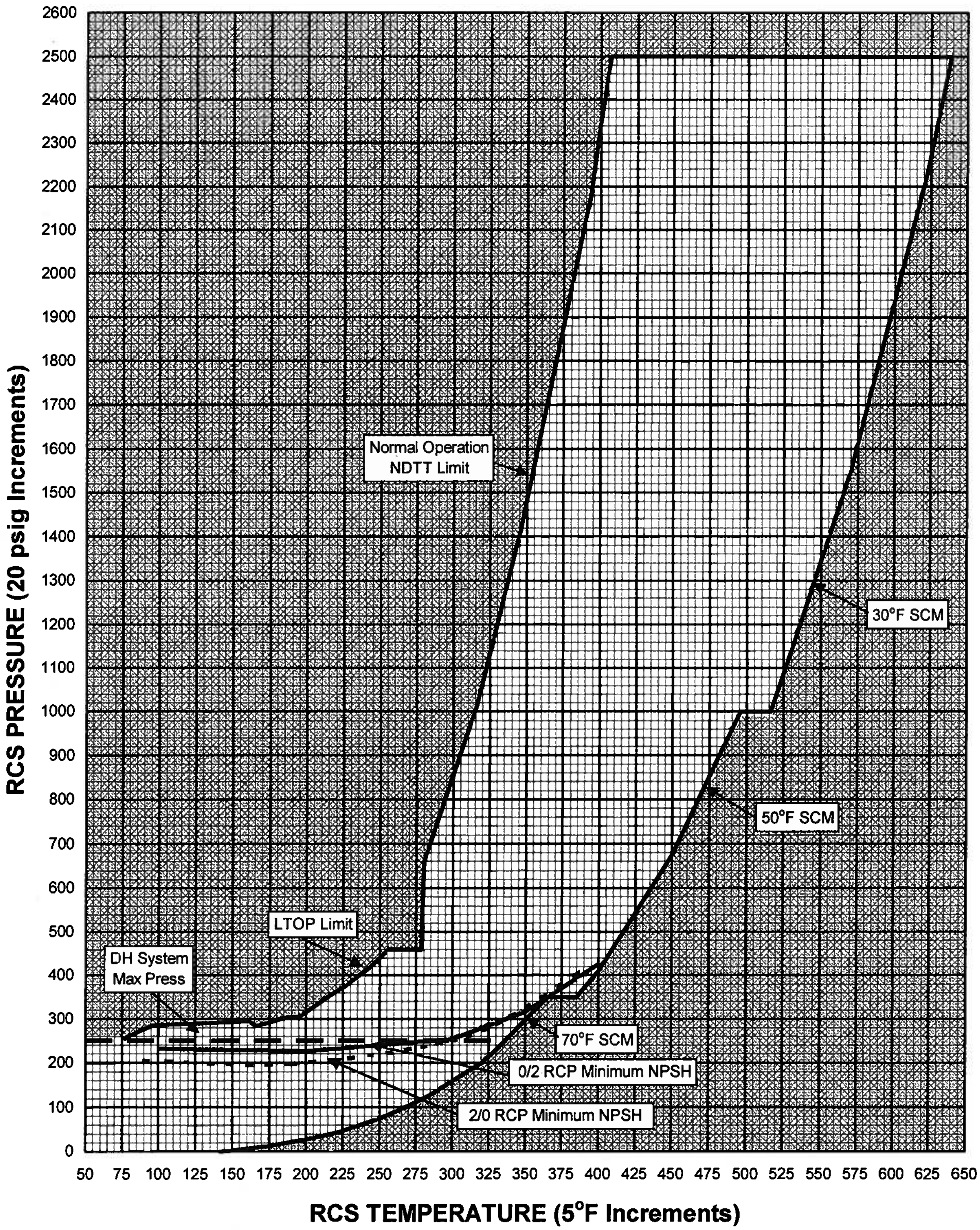


FIGURE 3
RCS Pressure vs Temperature Limits



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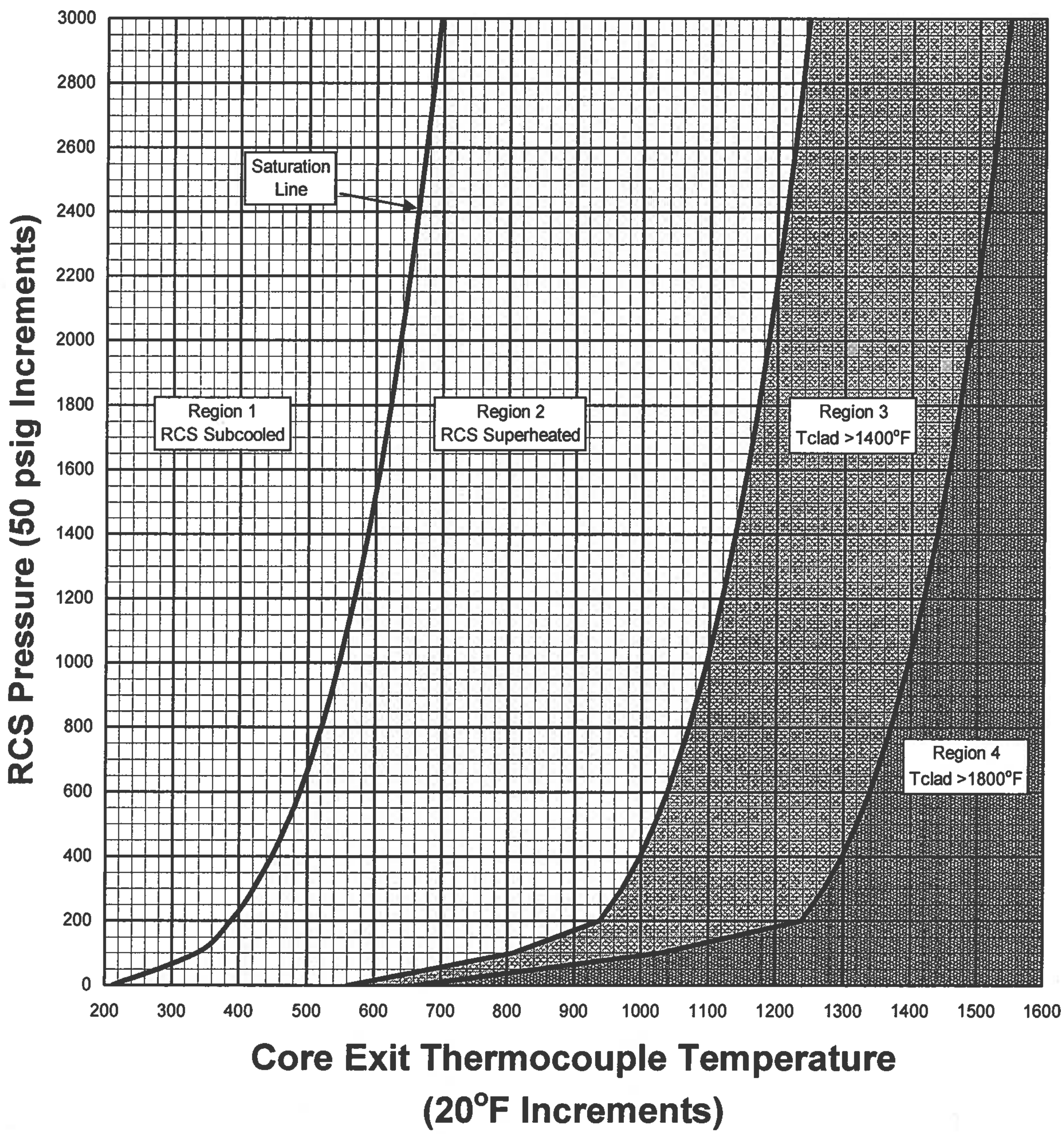

FIGURE 5

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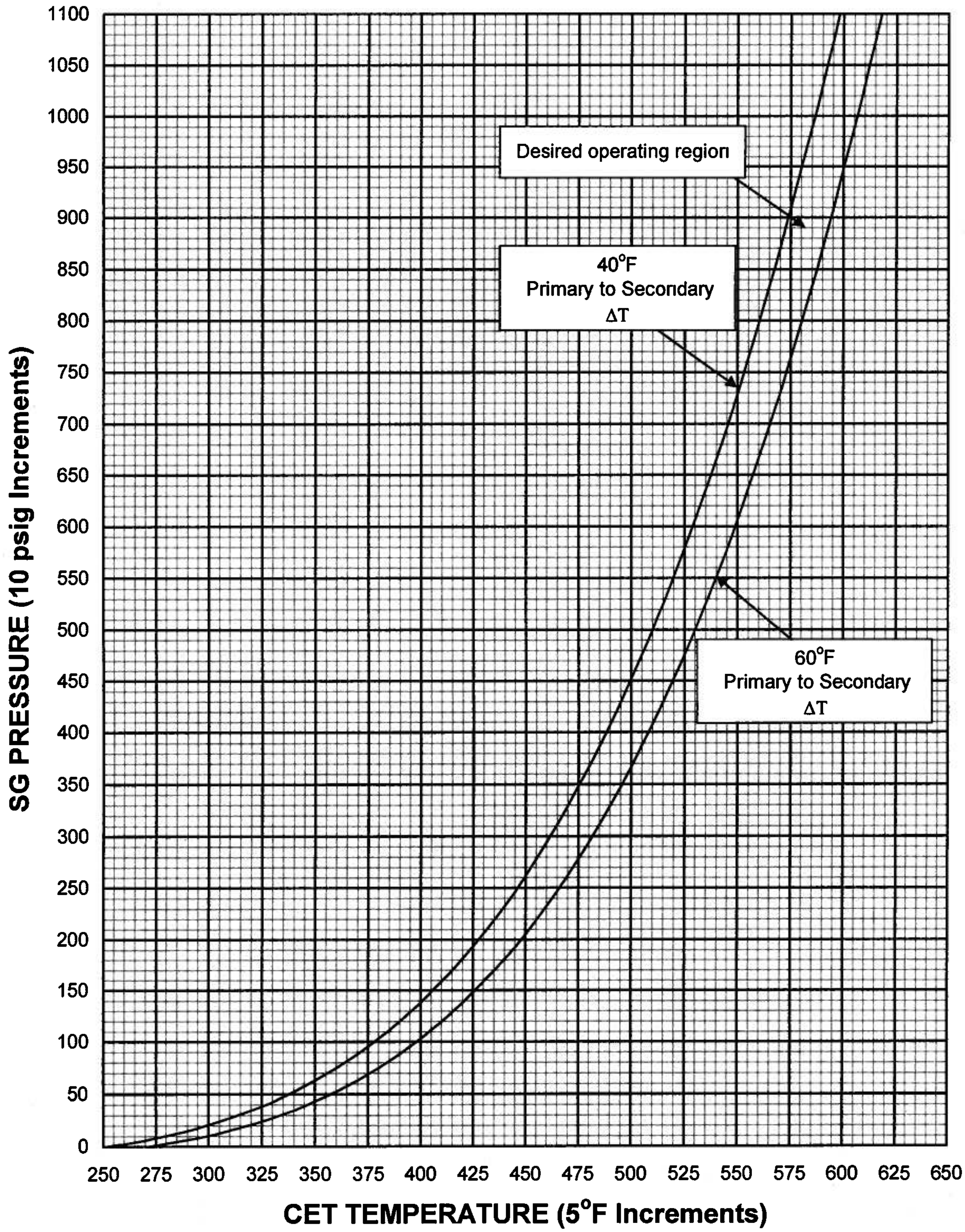
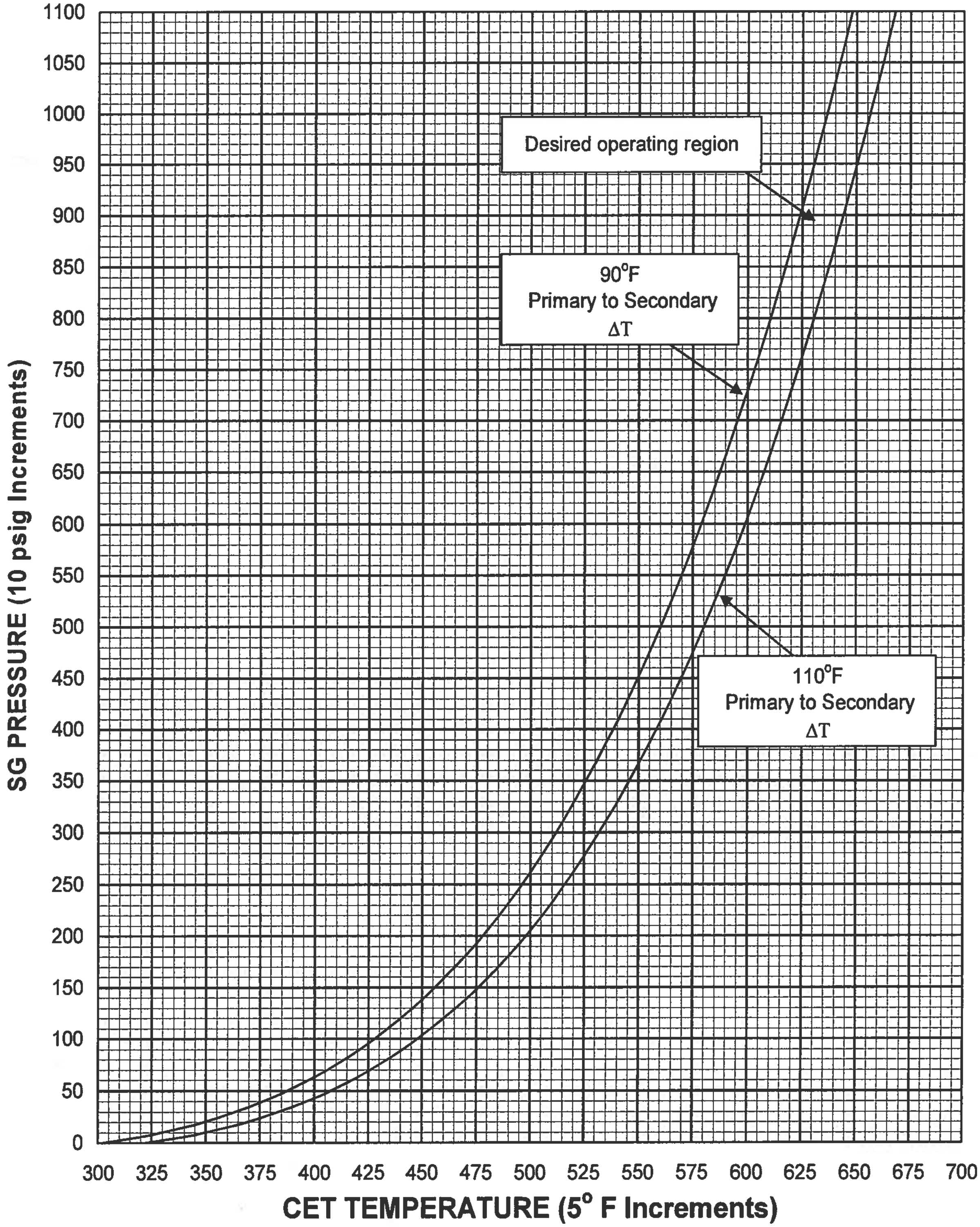


FIGURE 6

SG Pressure to Establish 90° to 110°F Primary to Secondary ΔT



ATTACHMENT N

page 1 of 1

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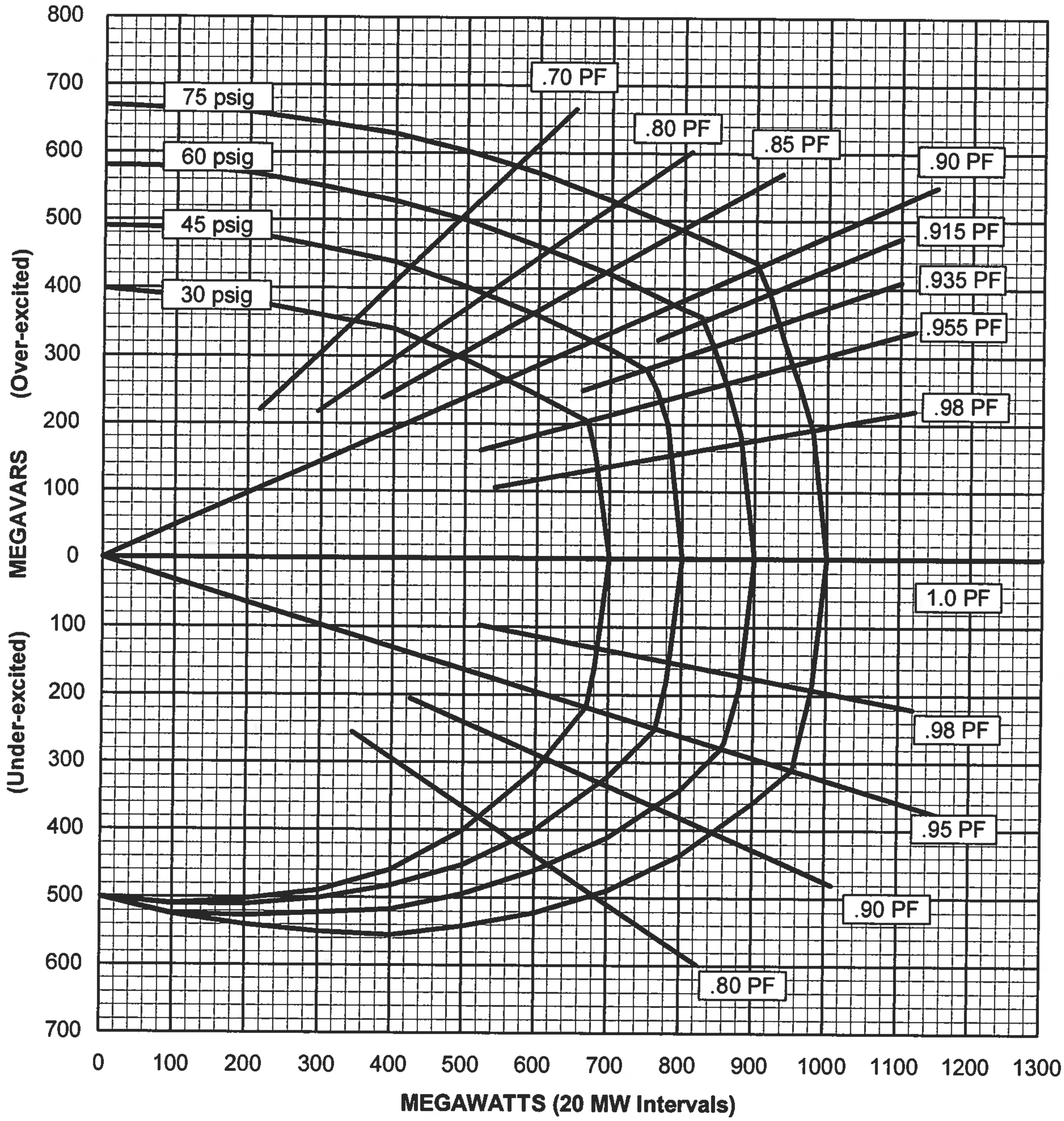

Basis Specifications:	1002.6 MVA	3 Phase
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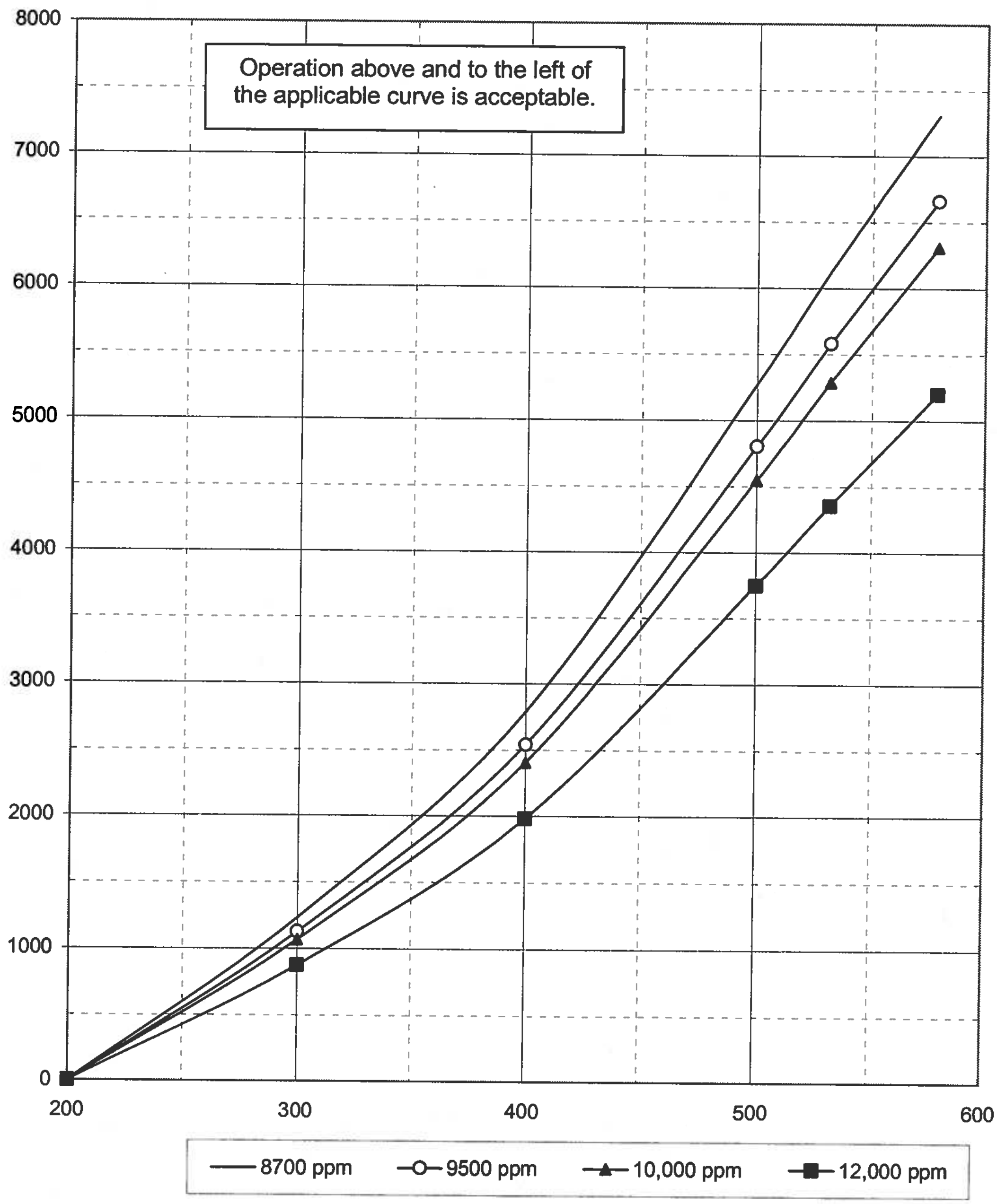
ATTACHMENT G

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



ATTACHMENT H

Volume of BAAT vs. Depth of Liquid

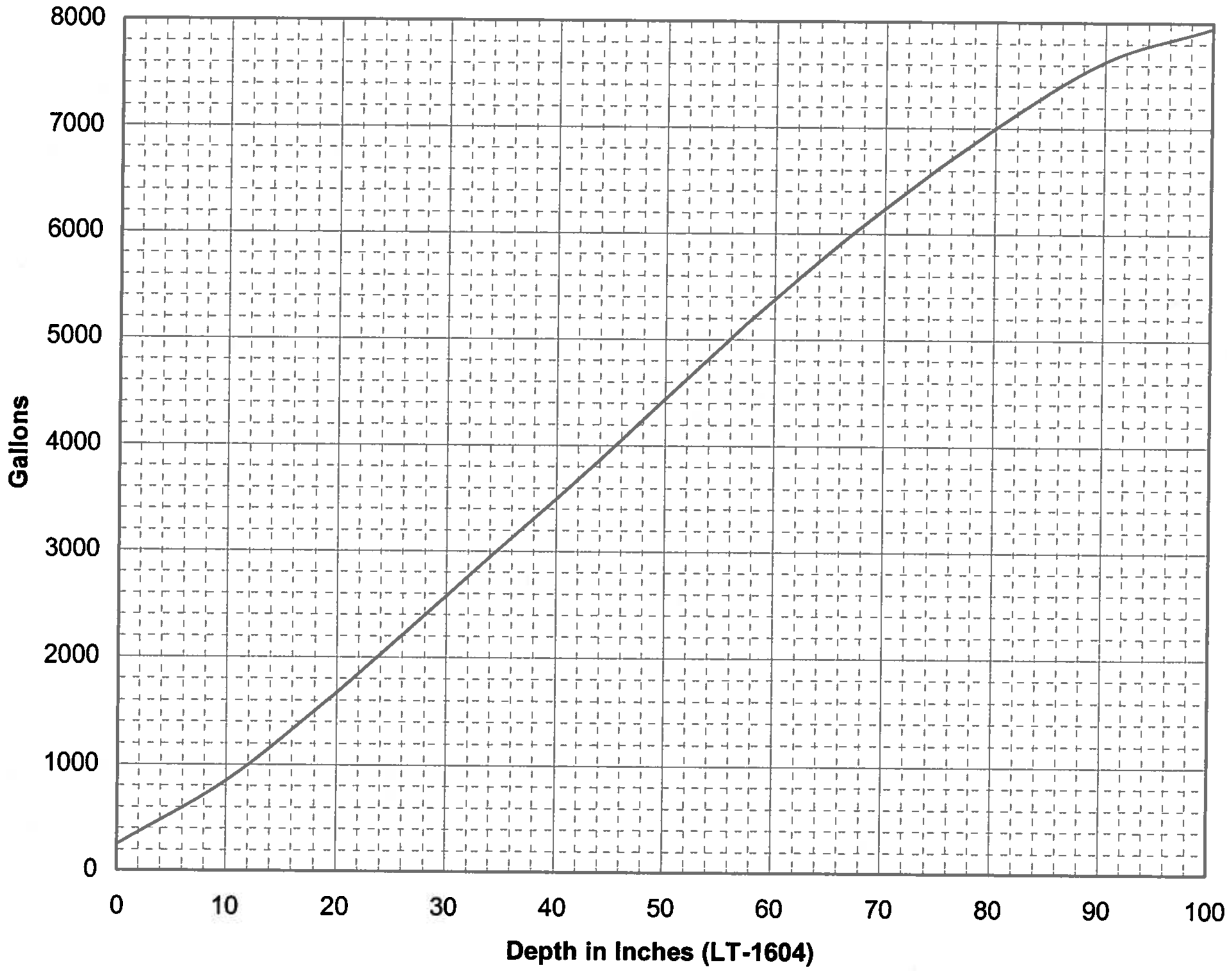


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Figure 3-A

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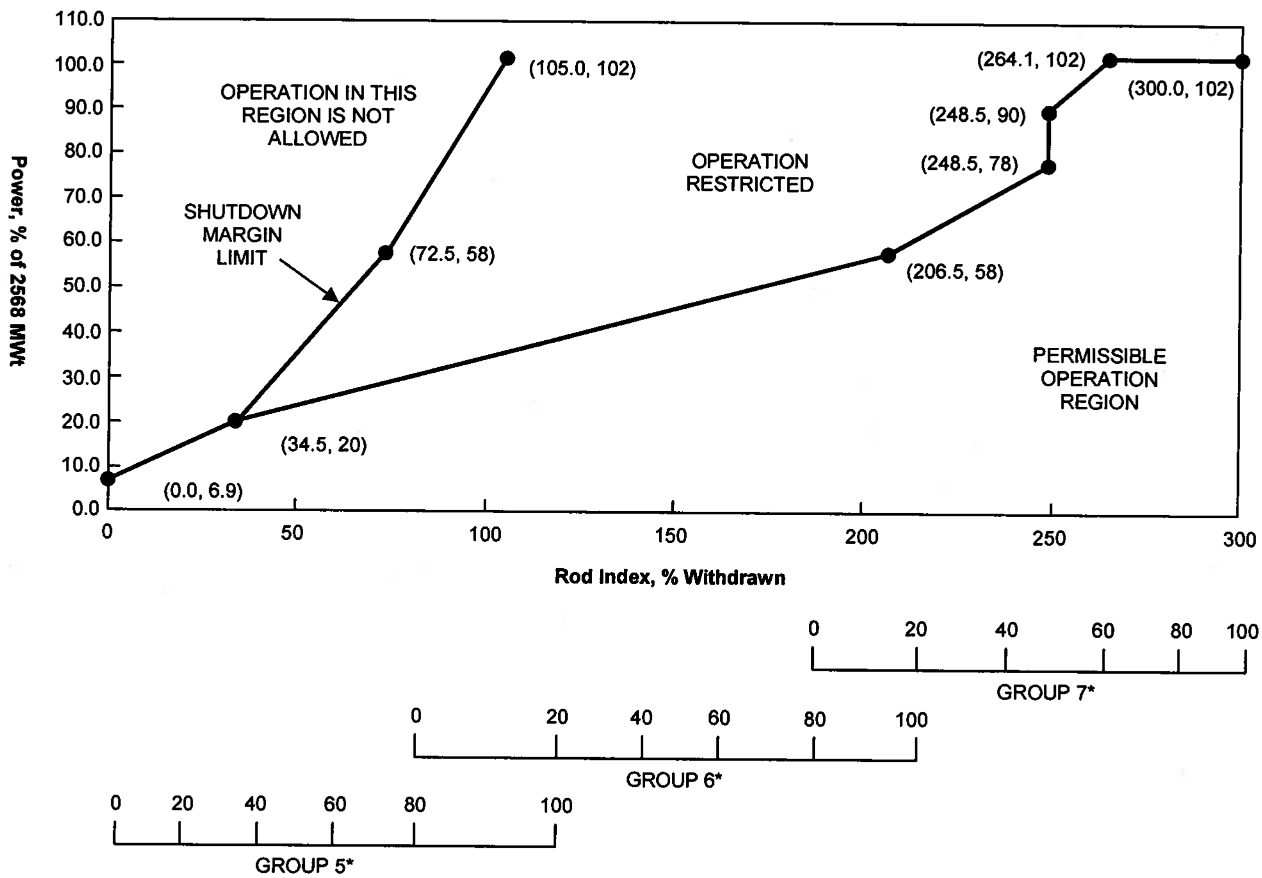
(Figure is referred to by Technical Specification 3.2.1)



* Technical Specification 3.5.2.5(2) requires that operating rod group overlap be 20% ± 5% between two sequential groups, except for physics tests.

Figure 3-B

Regulating Rod Insertion Limits for Four-Pump Operation From 200 ± 10 EFPD to EOC

(Figure is referred to by Technical Specification 3.2.1)

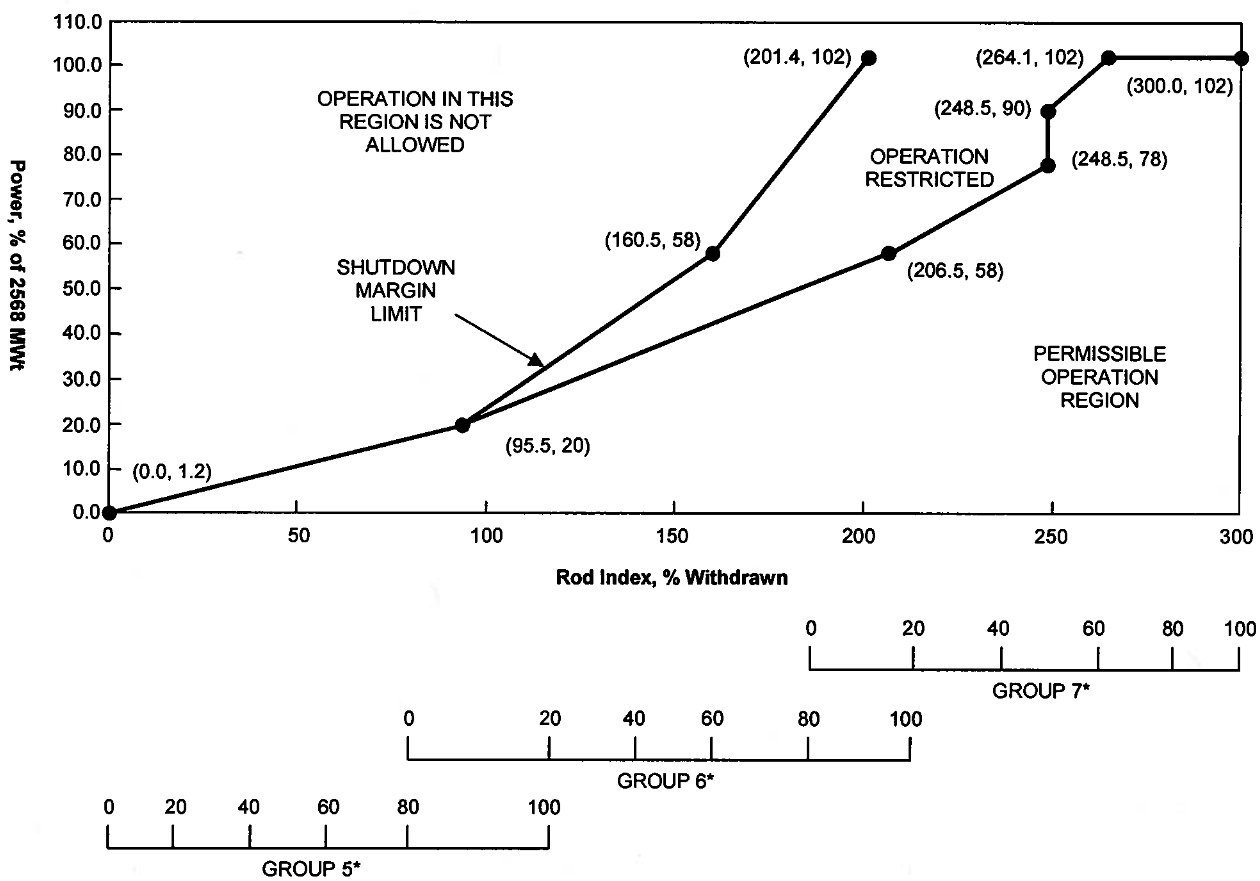


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Figure 4-A

Regulating Rod Insertion Limits for Three-Pump Operation From 0 to 200 ± 10 EFPD

(Figure is referred to by Technical Specification 3.2.1)

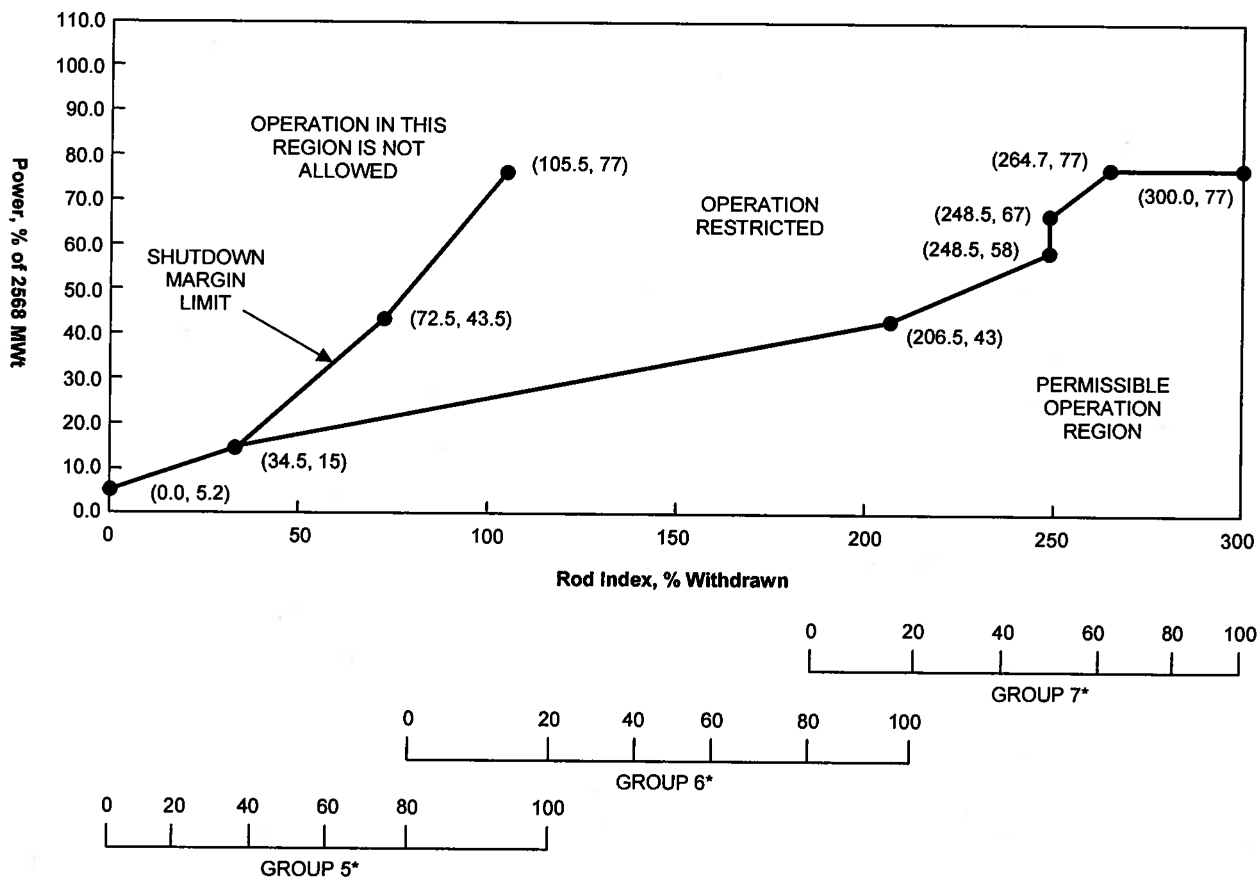

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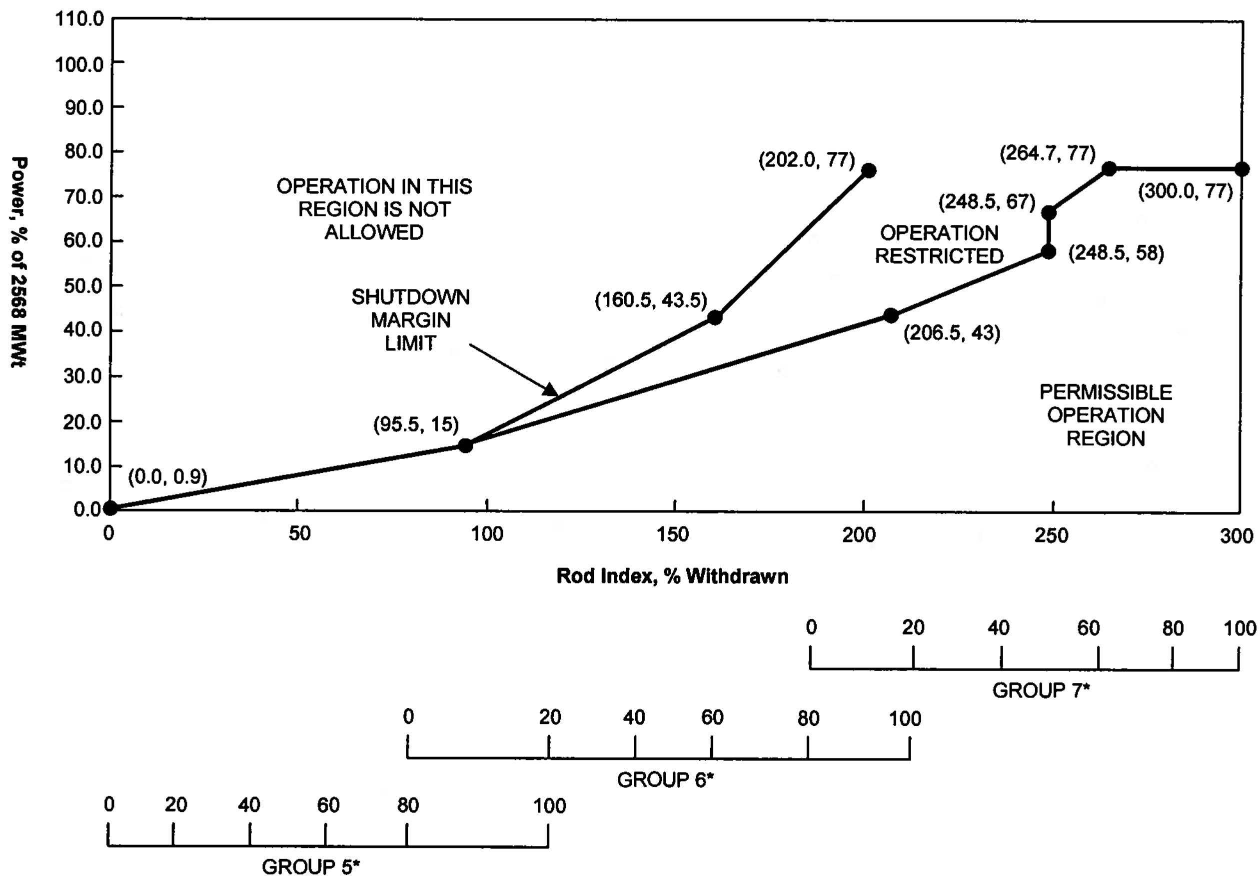
Figure 4-B

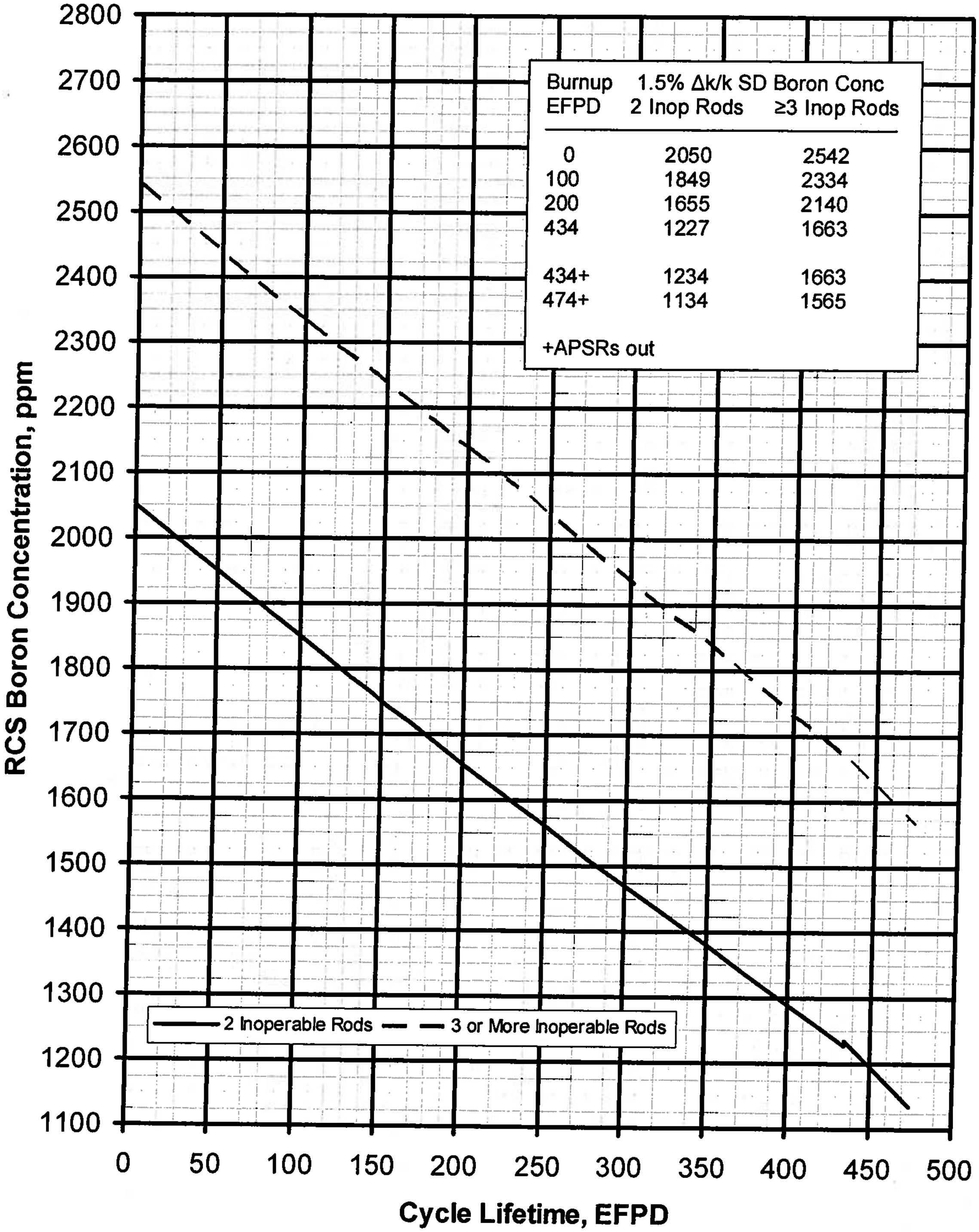
Regulating Rod Insertion Limits for Three-Pump Operation From 200 ± 10 EFPD to EOC

(Figure is referred to by Technical Specification 3.2.1)



* Technical Specification 3.5.2.5(2) requires that operating rod group overlap be 20% ± 5% between two sequential groups, except for physics tests.

Attachment B-16: Boron Concentration for 1.5% Shutdown Margin
During Emergency Boration



Burnup EFPD	1.5% Δk/k SD Boron Conc 2 Inop Rods	≥3 Inop Rods
0	2050	2542
100	1849	2334
200	1655	2140
434	1227	1663
434+	1234	1663
474+	1134	1565

+APSRs out

3.3 INSTRUMENTATION

3.3.15 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.15 The PAM instrumentation for each Function in Table 3.3.15-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to prepare and submit a Special Report.	Immediately
C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.15-1 for the channel.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.15.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	18 months

Table 3.3.15-1
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Wide Range Neutron Flux	2	E
2. RCS Hot Leg Temperature	2	E
3. RCS Hot Leg Level	2	F
4. RCS Pressure (Wide Range)	2	E
5. Reactor Vessel Water Level	2	F
6. Reactor Building Water Level (Wide Range)	2	E
7. Reactor Building Pressure (Wide Range)	2	E
8. Penetration Flow Path Automatic Reactor Building Isolation Valve Position	2 per penetration flow path(a)(b)	E
9. Reactor Building Area Radiation (High Range)	2	F
10. Deleted		
11. Pressurizer Level	2	E
12. a. SG "A" Water Level – Low Range	2	E
b. SG "B" Water Level – Low Range	2	E
c. SG "A" Water Level – High Range	2	E
d. SG "B" Water Level – High Range	2	E
13. a. SG "A" Pressure	2	E
b. SG "B" Pressure	2	E
14. Condensate Storage Tank Level	2	E
15. Borated Water Storage Tank Level	2	E
16. Core Exit Temperature (CETs per quadrant)	2	E
17. a. Emergency Feedwater Flow to SG "A"	2	E
b. Emergency Feedwater Flow to SG "B"	2	E
18. High Pressure Injection Flow	2	E
19. Low Pressure Injection Flow	2	E
20. Reactor Building Spray Flow	2	E

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.