ANO Unit One PWR RO Examination Outline

K/A Category Points

Tier	Group	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Point Total	Temp Total	Average	Std. Dev.
Tier 1	**1**	1	2	3				2	3			5	**16**	*16*		
Plant	**2**	2	3	5				3	3			1	**17**	*17*		
Evolutions	**3**	1	0	1				1	0			0	**3**	*3*		
	Tier Totals	4	5	9				6	6			6	**36**	*36*	*6.00*	*1.67*
Tier 2	**1**	2	1	1	3	1	2	3	0	2	6	2	**23**	*23*		
Plant	**2**	1	0	3	6	1	1	1	5	0	0	2	**20**	*20*		
Systems	**3**	0	1	0	1	2	0	0	2	0	1	1	**8**	*8*		
	Tier Totals	3	2	4	10	4	3	4	7	2	7	5	**51**	*51*	*4.64*	*2.46*
Tier 3		**Cat1**	**Cat2**	**Cat3**	**Cat4**											
Generic		5	3	3	2								**13**	*13*		
K/A/G/ Totals		*7*	*7*	*13*	*10*	*4*	*3*	*10*	*13*	*2*	*7*	*24*	100	*100*	*9.09*	*6.17*

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier1/Group1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
000005 Inoperable/Stuck Control Rod / 1						1	2.1.12 Ability to apply technical specifications for a system.	2.9	1
000015/17 RCP Malfunctions / 4				1			AA1.05 Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCS flow.	3.8	1
BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4			1				BW/E09 EK3.1 Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Cooldown): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.	3.2	1
000024 Emergency Boration / 1					1		AA2.03 Ability to determine and interpret the following as they apply to the Emergency Boration: Correlation between boric acid controller setpoint and boric acid flow.	2.9	1
000026 Loss of Component Cooling Water / 8				1			AA1.05 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: The CCWS surge tank, including level control and level alarms, and radiation alarm.	3.1	1
000027 Pressurizer Pressure Control System Malfunction / 3		1					AK2.03 Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners.	2.6	1
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4					1		BW/EA2.2 Ability to determine and interpret the following as they apply to the (Excessive Heat Transfer): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.6	1
CE/A11; W/E08 RCS Overcooling - PTS / 4							Not applicable.		0
000051 Loss of Condenser Vacuum / 4						1	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.	3.9	1
000055 Station Blackout / 6	1						EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling.	4.1	1
000057 Loss of Vital AC Elec. Inst. Bus / 6									0

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier1/Group1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
000062 Loss of Nuclear Service Water / 4			1				AK3.02 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.	3.6	1
000067 Plant Fire On-site / 9									0
000068 (BW/A06) Control Room Evac. / 8		1					BW/A06 AK2.1 Knowledge of the interrelations between the (Shutdown Outside Control Room) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.8	1
000069 (W/E14) Loss of CTMT Integrity / 5						1	2.4.11 Knowledge of abnormal condition procedures.	3.4	1
000074 (W/E06&E07) Inad. Core Cooling / 4						1	2.4.6 Knowledge symptom based EOP mitigation strategies.	3.1	1
BW/E03 Inadequate Subcooling Margin / 4			1				B/W EK3.4 Knowledge of the reasons for the following responses as they apply to the (Inadequate Subcooling Margin): RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.	3.2	1
000076 High Reactor Coolant Activity / 9						1	2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.0	1
BW/A02&A03 Loss of NNI-X/Y / 7					1		BW/A02 AA2.2 Ability to determine and interpret the following as they apply to the (Loss of NNI-X): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	4.0	1
K/A Category Totals:	1	2	3	2	3	5	Group Point Total = 16		16

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier1/Group2

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
000001 Continuous Rod Withdrawal / 1									0
000003 Dropped Control Rod / 1				1			AA1.02 Ability to operate and/or monitor the following as they apply to the Dropped Control Rod: Controls and components necessary to recover rod.	3.6	1
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1		1					BW/E10 EK2.1 Knowledge of the interrelations between the (Post-Trip Stabilization) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.5	1
BW/A01 Plant Runback / 1									0
BW/A04 Turbine Trip / 4			1				BW/AK3.2 Knowledge of the reasons for the following responses as they apply to the (Turbine Trip): Normal, abnormal and emergency operating procedures associated with (Turbine Trip).	3.4	1
000008 Pressurizer Vapor Space Accident / 3					1		AA2.27 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Effects on indicated PZR pressure and/or level of sensing line leakage	2.9	1
000009 Small Break LOCA / 3			1				EK3.21 Knowledge of the reasons for the following responses as they apply to the small break LOCA: Actions contained in EOP for small break LOCA/leak.	4.2	1
000011 Large Break LOCA / 3		1					EK2.02 Knowledge of the interrelations between the Large Break LOCA and the following: Pumps.	2.6	1
W/E04 LOCA Outside Containment / 3							Not applicable.		0
BW/E08; W/E03 LOCA Cooldown/Depress. / 4		1					BW/EK2.2 Knowledge of the interrelations between the (LOCA Cooldown) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	4.0	1
W/E11 Loss of Emergency Coolant Recirc. / 4							Not applicable.		0
W/E01 & E02 Rediagnosis & SI Termination / 3							Not applicable.		0
000022 Loss of Reactor Coolant Makeup / 2									0
000025 Loss of RHR System / 4									0
000029 Anticipated Transient w/o Scram / 1						1	2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	3.9	1

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier1/Group2

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
000032 Loss of Source Range NI / 7			1				AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Startup termination on source-range loss.	3.2	1
000033 Loss of Intermediate Range NI / 7			1				AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Guidance contained in EOP for loss of intermediate- range instrumentation	3.6	1
000037 Steam Generator Tube Leak / 3	1						AK1.02 Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Leak rate vs. pressure drop.	3.5	1
000038 Steam Generator Tube Rupture / 3					1		EA2.02 Ability to determine or interpret the following as they apply to a SGTR: Existence of an S/G tube rupture and its potential consequences.	4.5	1
000054 (CE/E06) Loss of Main Feedwater / 4				1			AA1.04 Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): HPI, under total feedwater loss conditions	4.4	1
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	1						BW EK1.2 Knowledge of the operational implications of the following concepts as they apply to the (Inadequate Heat Transfer): Normal, abnormal and emergency operating procedures associated with (Inadequate Heat Transfer).	4.0	1
000058 Loss of DC Power / 6									0
000059 Accidental Liquid RadWaste Rel. / 9				1			AA1.01 Ability to operate and / or monitor the following as they apply to the Accidental Liquid Radwaste Release: Radioactive-liquid monitor.	3.5	1
000060 Accidental Gaseous Radwaste Rel. / 9			1				AK3.01 Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste: Implementation of E-plan.	2.9	1
000061 ARM System Alarms / 7					1		AA2.05 Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Need for area evacuation; check against existing limits.	3.5	1
W/E16 High Containment Radiation / 9							Not applicable.		0
CE/E09 Functional Recovery							Not applicable.		0
K/A Category Totals:	2	3	5	3	3	1	Group Point Total = 17		17

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier1/Group3

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
000028 Pressurizer Level Malfunction / 2				1			AA1.02 Ability to operate and / or monitor the following as they apply to the Pressurizer Level Control Malfunctions: CVCS.	3.4	1
000036 (BW/A08) Fuel Handling Accident / 8	1						BW A08/ AK1.2 Knowledge of the operational implications of the following concepts as they apply to the (Refueling Canal Level Decrease): Normal, abnormal and emergency operating procedures associated with (Refueling Canal Level Decrease).	3.7	1
000056 Loss of Off-site Power / 6			1				AK3.01 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.	3.5	1
000065 Loss of Instrument Air / 8									0
BW/E13&E14 EOP Rules and Enclosures									0
BW/A05 Emergency Diesel Actuation / 6									0
BW/A07 Flooding / 8									0
CE/A16 Excess RCS Leakage / 2							Not applicable.		0
W/E13 Steam Generator Over-pressure / 4							Not applicable.		0
W/E15 Containment Flooding / 5							Not applicable.		0
K/A Category Totals:	1	0	1	1	0	0	Group Point Total = 3		3

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier2/Group1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic	Imp.	Points
001 Control Rod Drive										1	1	A4.10 Ability to manually operate and/or monitor in the control room: Determination of an ECP. 2.1.27 Knowledge of system purpose and or function.	3.5 2.8	2
003 Reactor Coolant Pump			1						1			K3.03 Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: Feedwater and emergency feedwater. A3.01 Ability to monitor automatic operation of the RCPS, including: Seal injection flow.	2.8 3.3	2
004 Chemical and Volume Control									1	1		A3.10 Ability to monitor automatic operation of the CVCS, including: PZR level and pressure. A4.12 Ability to manually operate and/or monitor in the control room: Boration/dilution batch control.	3.9 3.8	2
013 Engineered Safety Features Actuation						1	1					K6.01 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors. A1.02 Ability to predict and/or monitor changes in parameters (to Prevent exceeding design limits) associated with operating the ESFAS controls including: Containment pressure, temperature, and humidity.	2.7 3.9	2
015 Nuclear Instrumentation					1							K5.10 Knowledge of the operational implications of the following concepts as they apply to the NIS: Ex-core detector operation.	2.8	1
017 In-core Temperature Monitor	1						1					K1.02 Knowledge of the physical connections and/or cause-effect relationships between the ITM system and the following systems: RCS. A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including: Core exit temperature.	3.3 3.7	2
022 Containment Cooling		1								1		K2.01 Knowledge of power supplies to the following: Containment cooling fans. A4.03 Ability to manually operate and/or monitor in the control room: Dampers in the CCS.	3.0 3.2	2
025 Ice Condenser												Not applicable.		0

Emergency and Abnormal Plant Evolutions - Tier2/Group1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic	Imp.	Points
056 Condensate				1						1		K4.11 Knowledge of Condensate System design feature(s) and/or interlock(s) which provide for the following: Bypass of heater stream. **(Replaced by Plant Specific Priority in T2G3)** A4.08 Ability to manually operate and monitor in the control room: Condensate automatic makeup valve controller. *Justification for <2.5 Importance: The use of Condensate 056 A4.08 is appropriate due to the importance of maintaining condenser level which is a suction source of the Condensate Pumps and thus necessary to maintain the secondary system's ability to remove heat from the primary.	1.7 *1.7	2
059 Main Feedwater				1			1					interlock(s) which provide for the following: MFW and startup feedwater valve combination. **(Replaced by Plant Specific Priority in T2G3)** A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or	2.4 2.9	2
061 Auxiliary/Emergency Feedwater	1					1						K1.03 Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: Main steam system. K6.02 Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps.	3.5 2.6	2
068 Liquid Radwaste										1		A4.04 Ability to manually operate and/or monitor in the control room: Automatic isolation.	3.8	1
071 Waste Gas Disposal				1							1	K4.01 Knowledge of design features(s) and/or interlock(s) which provide for the following: Pressure capability of the waste gas decay tank. **(Replaced by Plant Specific Priority in T2G3)** 2.1.28 Knowledge of the purpose and function of major system components and controls.	2.6 3.2	2
072 Area Radiation Monitoring										1		A4.01 Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments.	3.0	1
K/A Category Totals:	2	1	1	3	1	2	3	0	2	6	2	Group Point Total =	23	23

23

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier2/Group2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic	Imp.	Points
002 Reactor Coolant					1							K5.10 Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship between reactor power and RCS differential temperature.	3.6	1
006 Emergency Core Cooling						1						K6.19 Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: HPI/LPI systems (mode change).	3.7	1
010 Pressurizer Pressure Control											1	2.1.28 Knowledge of the purpose and function of major system components and controls.	3.2	1
011 Pressurizer Level Control								1				A2.07 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Isolation of letdown.	3.0	1
012 Reactor Protection								1				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument power.	3.6	1
014 Rod Position Indication	1											K1.01 Knowledge of the physical connections and/or cause-effect relationships between the RPIS and the following systems: CRDS.	3.2	1
016 Non-nuclear Instrumentation								1				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.	3.0	1
026 Containment Spray			1									K3.01 Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS.	3.9	1
029 Containment Purge							1					A1.03 Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the Containment Purge System controls including: Containment pressure, temperature, and humidity.	3.0	1
033 Spent Fuel Pool Cooling			1									K3.01 Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Area ventilation systems.	2.6	1
035 Steam Generator											1	2.4.10 Knowledge of annunciator response procedures.	3.0	1

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier2/Group2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic	Imp.	Points
039 Main and Reheat Steam								1				A2.04 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.	3.4	1
055 Condenser Air Removal				1								K4.01 Knowledge of CARS design feature(s) and/or interlock(s) which provide for the following: Turbine startup. *Justification for <2.5 Importance: The use of CARS 058 K4.01 is appropriate due to this system's effects on the Main Turbine's ability to function as a Secondary Heat Removal device.	*1.9	1
062 AC Electrical Distribution				1								K4.01 Knowledge of ac distribution system design eature(s) and/or interlock(s) which provide for the following: Bus lockouts.	2.6	1
063 DC Electrical Distribution								1				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds.	2.5	1
064 Emergency Diesel Generator				1								K4.05 Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Incomplete-start relay.	2.8	1
073 Process Radiation Monitoring			1									K3.01 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases.	3.6	1
075 Circulating Water				1								K4.01 Knowledge of circulating water system design feature(s) and interlock(s) which provide for the following: Heat sink.	2.5	1
079 Station Air				1								K4.01 Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: Cross-connect with IAS.	2.9	1
086 Fire Protection				1								K4.03 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Detection and location of fires.	3.1	1
K/A Category Totals:	1	0	3	6	1	1	1	5	0	0	2	Group Point Total = 20		20

PWR RO Examination Outline
Emergency and Abnormal Plant Evolutions - Tier2/Group2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic	Imp.	Points

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier2/Group3

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic	Imp.	Points
005 Residual Heat Removal					1							K5.09 Knowledge of the operational implications of the following concepts as they apply the RHRS: Dilution and boration considerations.	3.2	1
007 Pressurizer Relief/Quench Tank								1				A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the PS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overpressurization of the PZR.	3.6	1
008 Component Cooling Water														0
027 Containment Iodine Removal												Not applicable.		0
028 Hydrogen Recombiner and Purge Control					1							K5.03 Knowledge of the operational implications of the following concepts as they apply to the HRPS: Sources of hydrogen within containment.	2.9	1
034 Fuel Handling Equipment											1	2.2.28 Knowledge of new and spent fuel movement procedures.	2.6	1
041 Steam Dump/Turbine Bypass Control														0
045 Main Turbine Generator										1		A4.01 Ability to manually operate and/or monitor in the control room: Turbine valve indicators (throttle, governor, control, stop, intercept), alarms, and annunciators.	3.1	1
076 Service Water				1								K4.01 Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Conditions initiating automatic closure of closed cooling water auxiliary building header supply and return valves.	2.5	1
078 Instrument Air		1										K2.01 Knowledge of bus power supplies to the following: Instrument air compressor.	2.7	1
103 Containment								1				A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry.**(Replaced by Plant Specific Priority in T2G3)**	2.9	1
														0
K/A Category Totals:	0	1	0	1	2	0	0	2	0	1	1	Group Point Total = 8		8

PWR RO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier2/Group3

Plant-Specific Priorities

System/Topic	Recommended Replacement for………	Reason	Points
041 K2.01 SDS: Knowledge of bus power supplies to the following: ICS, normal and alternate power supply. 2.8	071 K4.01 (T2G1)	This item has a higher importance rating and has more importance to overall plant operation than pressure ratings of Waste Gas Decay Tanks.	1
045 K4.12 MT/G: Knowledge of MT/G system design feature(s) and/or interlock(s) which provide for the following: Automatic turbine runback. 3.3	056 K4.11 (T2G1)	This item has a higher importance rating and has more importance to overall plant operation than bypass of FW heaters.	1
059 A3.07 Ability to monitor automatic operation of the MFW, including: ICS. 3.4	103 A2.05 (T2G3)	This item has a higher importance rating and has more importance to overall plant operation than Condensate automatic makeup valve controller.	1
059 K1.07 Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: ICS. 3.2	059 K4.01 (T2G1)	This item has a higher importance rating and has more importance than MFW and startup valve combination.	1
Plant-Specific Priority Total: (limit 10)			4

QID: 0003 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** GGiles

TUOI: ANO-1-LP-RO-EFW **Objective:** 10 **Point Value:** 1

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

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

Description: Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: Feedwater and Emergency Feedwater.

K/A Number: K3.03 **CFR Reference:** CFR: 41.7 / 45.6

Tier: 1 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 4

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

Question:

The following plant conditions exist:

- Reactor is tripped
- All 4 RCPs are OFF
- RCS pressure is 1800 psig and rising
- RCS CET average is 600 °F
- "A" and "B" OTSG pressures are ~980 psig

The CBOR reports that RT-5 (Verification of EFW Actuation and Control) is completed.

Which of the following describes the expected EFIC system response?

a. OTSG levels will be rising at a rate of ~4 inches per minute to 370 to 410 inches.

b. OTSG levels will be rising at a rate of ~6 inches per minute to 300 to 340 inches.

c. OTSG levels will be rising at a rate of ~6 inches per minute to 370 to 410 inches.

d. OTSG levels will be rising at a rate of ~4 inches per minute to 300 to 340 inches.

Answer:

c. OTSG levels will be rising at a rate of ~6 inches per minute to 370 to 410 inches.

Notes:

RT-5 directs the operator to select REFLUX BOILING setpoint (~378 inches) if subcooling margin is less than adequate. The given plant conditions indicate a loss of subcooling margin. EFIC will automatically control OTSG levels at the REFLUX BOILING setpoint of ~378 inches. The rate of OTSG level rise when RCPs are OFF is variable from 2 to 8 inches per minute, depending on OTSG pressure (2 inches per minute at 800 psig, 8 inches per minute at 1050 psig). Therefore, (c) is the only correct response. (a) is incorrect because the wrong rate is given. (b) is incorrect because the wrong setpoint is given. (d) is incorrect because the wrong rate and wrong setpoint are given.

References:

1202.012 (Rev 004-01-0), Repetitive Tasks, RT5

History:

Developed for 1998 SRO Exam.
Modifed for use in 2001 RO/SRO Exam.

QID: 0005 **Rev:** 0 **Rev Date:** 6/17/98 **Source:** Direct **Originator:** GGiles

TUOI: AA51003-006 **Objective:** 6.2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Emergency Boration: Correlation between boric acid controller setpoint and boric acid flow.

K/A Number: AA2.03 **CFR Reference:** 43.5 / 45.13
43.5 / 45.13

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

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

Question:

The reactor has tripped and 3 control rods failed to fully insert.
The CRS has instructed you to perform Emergency Boration in accordance with RT-12.

Which of the following best describes the initial setting on the batch controller?

a. Set the batch controller to the batch size determined by the plant computer boron program to compensate for the reactivity worth of the stuck rods.

b. Set the batch controller to the maximum batch size setting of 999999 gallons and commence adding boric acid to the make up tank.

c. Set the batch controller to the batch size required to obtain the boron concentration as determined by a reactivity balance calculation.

d. Set the batch controller to the batch size required to maintain make up tank level between 55 and 86 inches while maintaining pressurizer level >100 inches.

Answer:

b. Set the batch controller to the maximum batch size setting of 999999 gallons and commence adding boric acid to the make up tank.

Notes:

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

References:

1202.012 (Rev 004-01-0), Repetitive Tasks, RT-12, Emergency Boration

History:

Developed for 1998 RO Exam.
Used in 2001 RO/SRO Exam.

QID: 0008 **Rev:** 0 **Rev Date:** 7/9/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-1-LP-AO-ICW **Objective:** 9 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 026 **System Title:** Loss of Component Cooling Water (ICW at ANO)

Description: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: : The CCWS surge tank, including level control and level alarms, and radiation alarm.

K/A Number: AA1.05 **CFR Reference:** CFR: 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 4

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

Question:

Given:

- Process Radiation Monitor RI-2236, Nuclear ICW, is in alarm.
- Nuclear ICW flow rate is >3100 gpm
- Local reports of Nuclear ICW Surge Tank overflowing

A leak in which of the following components would be capable of causing these conditions?

a. RCP Seal Return Coolers

b. Spent Fuel Coolers

c. Letdown Coolers

d. Pressurizer Sample Cooler

Answer:

c. Letdown Coolers

Notes:

"C" is correct since it is the only component with the piping size and differential pressure to cause the indications given.
All of the other choices have either small piping size or relatively low differential pressures.

References:

STM 1-43, rev. 3 ch. 1, Intermediate Cooling Water System, page 27, 28

History:

Developed for 1998 SRO Exam.
Used in 2001 RO/SRO Exam.

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

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

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

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

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

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

Question:

Given:

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

Choose the appropriate operator actions:

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

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

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

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

Answer:

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

Notes:

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

References:

1203.016 (Rev 011-00-0), Loss of Condenser Vacuum, page 1

History:

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

QID: 0011 **Rev:** 0 **Rev Date:** 7/1/98 **Source:** Direct **Originator:** GGiles

TUOI: AA51003-013 **Objective:** 13.5 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

System Number: 055 **System Title:** Loss of Offsite & Onsite Power (Station Blackout)

Description: Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling.

K/A Number: EK1.02 **CFR Reference:** CFR: 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:

Due to severe weather the plant is in a station blackout condition.
The steam driven emergency feedwater pump (P-7A) is feeding both steam generators.

Under what conditions would an "Emergency RCS Cooldown" be performed?

a. Loss of subcooling margin, no HPI available, and reactor vessel head voids are indicated.

b. Reactor vessel head voids are indicated and Core Exit Thermocouple temperature > 610 °F.

c. Loss of subcooling margin and steam generator tube to shell delta-T >100 °F (tubes colder).

d. Core Exit Thermocouple temperature > 610 °F and steam generator tube to shell delta-T >100 °F (tubes colder).

Answer:

a. Loss of subcooling margin, no HPI available, and reactor vessel head voids are indicated.

Notes:

Answer (a) is correct. Blackout EOP floating step states: If adequate subcooling margin is lost and RV head voids are indicated, then perform step 3 contingency. This contingency directs operators to perform emergency cooldown per step 49. A note preceeding step 49 states: This section is used for emergency RCS cooldown if adequate SCM is lost and no HPI is available and RV head voids are indicated. Answers (b), (c) and (d) are incorrect for the following reasons: CET temperature > 610 °F is indicative of an overheating condition but does not meet the criteria for establishing an emergency cooldown. Steam generator tube to shell delta-T should be maintained <100 °F when stabilizing RCS temperature and does not apply to the emergency cooldown criteria.

References:

1202.008 (Rev6), Blackout, step 3 contingency

History:

Developed for 1998 RO/SRO Exam.
Used in 2001 RO/SRO Exam.

QID: 0015 **Rev:** 0 **Rev Date:** 6/30/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.3 **Point Value:** 1

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

System Number: A06 **System Title:** Shutdown Outside Control Room

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

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

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

Question:

A fire in the control room forced an immediate evacuation.
An alternate shutdown is in progress and the crew is attempting to stabilize the plant at hot shutdown natural circulation conditions.

It is desired to raise reactor coolant system pressure. How is this accomplished?

a. Energize all pressurizer heaters from their respective power supply breaker cubicles.

b. Raise steaming rate at the Atmospheric Dump Valves to raise RCS temperature and pressure.

c. Manually initiate High Pressure Injection to compress the pressurizer steam bubble.

d. Manually throttle open on the pressurizer makeup block valve to raise pressurizer level.

Answer:

c. Manually initiate High Pressure Injection to compress the pressurizer steam bubble.

Notes:

The CRS will manually initiate HPI by manually closing in the breaker for a HPI pump as directed by the TSC, therefore (c) is the correct response. (a) and (d) are not options covered in the alternate shutdown procedure. (b) is incorrect since raising the steaming rate would result in a lower RCS temperature and pressure.

References:

1203.002 (Rev 015-01-0), Alternate Shutdown, page 8

History:

Developed for 1998 RO Exam.
Used in 2001 RO/SRO Exam.

QID: 0019 **Rev:** 0 **Rev Date:** 7/6/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.3 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 076 **System Title:** High Reactor Coolant Activity

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

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

Tier: 1 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 4

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

Question:

Fuel pin leakage has caused higher than normal activity in the reactor coolant system.

Which of the following indications on the Failed Fuel Monitor (RI-1237) would be indicative of failed fuel and require power reduction?

a. A marked rise by 20% in the IODINE/GROSS ratio.

b. A marked rise by 40% in the GROSS/IODINE ratio.

c. A marked drop by 20% in the IODINE/GROSS ratio.

d. A marked drop by 40% in the GROSS/IODINE ratio.

Answer:

d. A marked drop by 40% in the GROSS/IODINE ratio.

Notes:

The plant computer provides indication of the Calculated Failed Fuel Gross/Iodine Ratio (R1237R). Step 3.1 of 1203.019 specifies that if the failed fuel ratio drops by 40% as indicated by WCO logs or the Plant Computer, reduce reactor power by 50% of present power level, therefore, (d) provides the only correct response. Answers (a), (b) and (c) provide the wrong magnitude and/or direction of the change in failed fuel ratio.

References:

1203.019 (Rev 010-02-0), High Activity in Reactor Coolant, page 4

History:

Developed for 1998 RO/SRO Exam.
Used in A. Morris 98 RO Re-exam
Used in 2001 RO/SRO Exam

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0020 **Rev:** 0 **Rev Date:** 7/6/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-NNI **Objective:** 6 **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 (Loss of NNI-X): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A Number: AA2.2 **CFR Reference:** CFR: 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 4.0 **RO Select:** Yes **Difficulty:** 4

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

Question:

Given the following indications/alarms:

- SASS Mismatch alarm (fast flash)
- SG BTU Limit alarm (slow flash)
- SG "B" FW Temp signal select switch selected to SASS Enable with the white indicating light out and the blue "Y" light on.

What operator action is procedurally required?

a. Place the SG "B" FW Temp signal select switch to the "Y" position.

b. Depress the Auto pushbutton for SG "B" FW Temp on the SASS panel in C47-2.

c. No action necessary, SASS has automatically transferred to "Y" NNI.

d. Place both FW loop demands in manual.

Answer:

a. Place the SG "B" FW Temp signal select switch to the (Y) position.

Notes:

"a" is the correct response per procedure 1203.012F and 1105.006.
"b" is incorrect, this action is performed when resetting a failed signal.
"c" is incorrect although SASS has transferred to [Y], the response in "a" the only procedurally required action.
"d" is not necessary, the SASS system has transferred to a good signal, no ICS upset should occur, and this action should include placing the SG/Rx master in manual as well.

References:

1203.012F, Rev. 026-00-0, Annunciator K07 Corrective Action, page 17, 20
1105.006 rev. 009-00-0 Reactor Coolant System NNI, page 13

History:

Developed for 1998 RO/SRO Exam.
Modified QID 3127
Used in 2001 RO/SRO Exam.

QID: 0030 **Rev:** 0 **Rev Date:** 7/8/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-EOP10 **Objective:** 5 **Point Value:** 1

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

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

Description: Knowledge of the interrelationships between the (LOCA Cooldown) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

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

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

Question:

During a small break LOCA cooldown, which of the following criteria must be met before High Pressure Injection may be throttled?

a. Restoration of adequate subcooling margin.

b. Total low pressure injection flow >2800 gpm.

c. Restoration of forced flow cooling.

d. At least one OTSG is available as a heat sink.

Answer:

a. Restoration of adequate subcooling margin.

Notes:

Answer (a) is correct in accordance with B&W guidelines for HPI throttling/termination. Answer (b) is incorrect because it states incorrect LPI flow criteria, answers (c) and (d) are incorrect because they state other non-applicable cooldown methods.

References:

B&W Technical Document, Part VI, Specific Rules

History:

Developed for 1998 RO/SRO Exam.
Used in RO/SRO Exam.

QID: 0031 **Rev:** 0 **Rev Date:** 7/9/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-ELECD **Objective:** 7.19 **Point Value:** 1

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

System Number: E10 **System Title:** Post Trip Stabilization

Description: Knowledge of the interrelationships between the (Post-Trip Stabilization) 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: EK2.1 **CFR Reference:** CFR: 41.7 / 45.7

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

Given:

- The unit is operating at 100% power.
- SU-1 transformer is INOPERABLE.
- The Startup Transformer Preferred Transfer Switches on C-10 for A1/H1 and A2/H2 are selected to SU-2 transformer.
- SU-2 feeder breakers to A1, A2, H1 and H2 are out of the pull-to-lock position.

Which of the following best describes the electrical system response to a reactor/turbine trip?

a. With SU-1 inoperable, the load shed protective circuitry will require a manual transfer of electrical buses to SU-2 due to the limited capacity of SU-2.

b. The reactor/turbine trip will actuate the generator lockout relay and result in a "slow transfer" of A1, A2, H1 and H2 electrical buses to SU-2 transformer.

c. The tie breakers between the vital and non-vital buses will open, the emergency diesel generators will start and supply the vital buses and SU-2 will power A1, A2, H1 and H2.

d. The generator lockout relay trip will cause a "fast transfer" to SU-2 transformer and applicable loads will be shed to limit the loading on SU-2 transformer.

Answer:

d. The generator lockout relay trip will cause a "fast transfer" to SU-2 transformer and applicable loads will be shed to limit the loading on SU-2 transformer.

Notes:

Answer (d) is correct. The reactor/turbine trip will actuate the generator lockout relay trip which will cause a fast transfer to SU-2. The load shed circuitry will automatically shed applicable loads to limit the loading on SU-2. Answer (a) is incorrect since an automatic transfer will occur. Answer (b) is incorrect since the transfer will be a "fast transfer". Answer (c) is incorrect because tie breakers will not open and the EDGs will not start.

References:

1107.001 (Rev 057-02-0), Electrical System Operation, page 4

History:

Developed for 1998 RO/SRO Exam.
Used in 2001 RO/SRO Exam.

QID: 0039 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-ICS **Objective:** 12 **Point Value:** 1

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

System Number: 035 **System Title:** Steam Generator System (S/GS)

Description: Knowledge of annunciator response procedures.

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

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

Group: 2 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** C

Question:

ICS is in full automatic and the CBOR is verifying proper plant response to a main feedwater pump trip from 100% power.

Which of the following should the CBOR expect to occur?

a. The operating main feedwater pump demand will have a bias of 30% added.

b. The pressurizer spray valve will come open when RCS pressure reaches 2205 psig.

c. The control rods will start inserting immediately due to a crosslimit from feedwater.

d. The main feedwater block valves will close in fast speed once power goes below 80%.

Answer:

d. The main feedwater block valves will close in slow speed once power goes below 80%.

Notes:

Answer (d) is correct since the main feedwater block valve closure is inhibited above 80%, once <80% they will close in fast speed. Answers (a) is incorrect, no bias is applied to demand although it will be maximized due to the high power level and the MFP trip.
Answer (b) is incorrect, a bias of 125 psig is subtracted from the normal setpoint of 2205, the spray valves should open at 2080 psig.
Answer (c) is incorrect, control rods will start to insert but not due to a crosslimit from feedwater, rather the reactor system will be crosslimiting feedwater. A bias subtraction from the reactor demand signal will cause the rods to insert.

References:

1203.012F, Rev. 026-00-0, Annunciator K07 Corrective Action, page 9

History:

Developed for 1998 RO Exam.
Used in 2001 RO Exam.

QID: 0040 **Rev:** 0 **Rev Date:** 12/06/00 **Source:** New **Originator:** S. Pullin

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

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

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

Description: Knowledge of the operational implications of the following concepts as they apply to the (Inadequate Heat Transfer): Normal, abnormal and emergency operating procedures associated with (Inadequate Heat Transfer).

K/A Number: EK1.2 **CFR Reference:** CFR: 41.8 / 41.10 / 45.3

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

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

Question:

A reactor trip due to a loss of electrical buses H1 and H2 has occurred.
While performing reactor trip follow-up actions, the CBOR reports that CET temperature is rising.

When would transition to 1202.004, Overheating be required?

a. CET temperature exceeds 580 degrees.

b. CET temperature exceeds 610 degrees.

c. Thot temperature exceeds 580 degrees.

d. Thot temperature exceeds 610 degrees.

Answer:

b) CET temperature exceeds 610 degrees.

Notes:

(b) is the correct answer per 1202.004, Overheating, entry conditions.
(a) is incorrect, entry is required at CET temperature of 610 degrees.
(c) and (d) are incorrect because Thot indication is not valid with no RCPs running.

References:

1202.004, Rev 4, Overheating, entry conditions.

History:

Developed for 2001 RO/SRO NRC Exam.

QID: 0041 **Rev:** 0 **Rev Date:** 7/10/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.2 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems.

K/A Number: AA2.03 **CFR Reference:** 43.5 / 45.13

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

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

Question:

A reactor trip has occurred from 100% power following a loss of D01.
Attempts to transfer 125V DC panel D11 to its emergency supply are unsuccessful.
Both OTSGs pressures are ~890 psig.

Which of the following is the appropriate action?

a. Crosstie D-11 and D-21 from panel C-10 in the control room to provide power to D11.

b. Manually actuate MSLI and EFW for both OTSGs and verify proper actuation and control.

c. Select main turbine control to TURBINE MANUAL and close the governor valves in fast speed.

d. Dispatch an operator to manually trip the main turbine from the main turbine front standard.

Answer:

b. Manually actuate MSLI and EFW for both OTSGs and verify proper actuation and control.

Notes:

Answer (b) is correct per reactor trip EOP contingency actions and Loss of D01 AOP follow up actions. (a) is not physically possible, (c) and (d) are not procedural options in the Loss of D01 procedure.

References:

1203.036, Rev. 005-00-0, Loss of 125V DC, page 2

History:

Developed for 1998 Exam.
Used in 2001 SRO Exam.

QID: 0044 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** GGiles

TUOI: AA61002-006 **Objective:** 6.8 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste Release: Implementation of the E-plan.

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

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

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

Question:

Events are in progress which have resulted in an unplanned gaseous radioactive offsite release which is expected to exceed the EPA Protective Action Guideline exposure levels.

What emergency classification would you recommend to the Shift Manager?

a. Notification of Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

Answer:

d. General Emergency

Notes:

Answer (d) is correct per the definition of a General Emergency. (a) is incorrect since for an NUE no releases requiring offsite response or monitoring are expected. (c) is incorrect since for an SAE releases are not expected to exceed limits except near the site boundary. (b) is incorrect because for an Alert, releases are not expected to be more than just a fraction of the EPA guidelines.

References:

1903.010 (Rev 036-02-0), Emergency Action Level Classification, page 3

History:

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

QID: 0052 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** GGiles
TUOI: ANO-1-LP-RO-RXBAL **Objective:** 3 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control
System Number: 001 **System Title:** Control Rod Drive System
Description: Ability to manually operate and/or monitor in the control room: Determination of an ECP.

K/A Number: A4.10 **CFR Reference:** CFR: 41.7/ 45.5 to 45.8
Tier: 2 **RO Imp:** 3.5 **RO Select:** Yes **Difficulty:** 4
Group: 1 **SRO Imp:** 3.9 **SRO Select:** No **Taxonomy:** Ap

Question:

The plant had been operating at 100% power for 200 days.
Following a plant trip, preparations for startup are in progress.
The CBOR is performing a calculation for Estimated Critical Position (ECP).

At a given boron concentration, which of the following times would result in the lowest rod index due to the effects of Xenon?

a. 4 to 6 hours

b. 8 to 12 hours

c. 40 to 60 hours

d. 70 to 90 hours

Answer:

d. 70 to 90 hours

Notes:

Answer (d) is correct since Xenon is essentially depleted at ~80 hours following a reactor trip from 100% power (thumb rule), thus adding essentially no reactivity for which rod withdrawal must compensate for. (a) is incorrect since xenon is still building in at this time, (b) is incorrect since xenon has peaked at it's highest concentration pre-trip and (c) is incorrect because at this time the core is still not xenon free.

References:

General Physics Corporation PWR / Reactor Theory, ch.6, p.25

History:

Developed for the 1998 Unit 1 RO/SRO Exam.
Modified for use in 2001 RO Exam.

QID: 0055 **Rev:** 0 **Rev Date:** 7/11/98 **Source:** Direct **Originator:** GGiles

TUOI: ANO-1-LP-RO-MUP **Objective:** 10 **Point Value:** 1

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

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

Description: Ability to manually operate and/or monitor in the control room: Boration/dilution batch control.

K/A Number: A4.12 **CFR Reference:** CFR: 41.7 / 45.5 to 45.8

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

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

Question:

Batch Controller Flow Control Valve (CV-1249) should normally be open no more than 20% when feeding boric acid. Why?

a. To prevent pump runout on the boric acid pumps.

b. To prevent exceeding design flow through the batch controller.

c. To prevent overheating of the boric acid pumps.

d. To prevent inaccurate readings on the batch totalizer.

Answer:

d. To prevent inaccurate readings on the batch totalizer.

Notes:

Answer (d) is correct per note in 1103.004, Soluble Poison Concentration Control.
Answer (a) will certainly not runout the pumps at only 20% open, [b] is incorrect for the same reason.
Answer [c] is a precaution against dead heading the pumps, 20% open on the batch controller meets minimum flow requirements.

References:

1103.004, (Rev. 016-00-0) Soluble Poison Concentration Control, page 12

History:

Developed for the 1998 RO exam
Used in 2001 RO Exam.

QID: 0059 **Rev:** 0 **Rev Date:** 7/12/98 **Source:** Direct **Originator:** GGiles

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 System (ITM)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including: Core exit temperature.

K/A Number: A1.01 **CFR Reference:** CFR: 41.5 / 45.7

Tier: 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

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

Question:

During a large break LOCA the value on the ICCMDS CET Subcooling Margin Display is negative and flashing. What does this indicate?

a. An ICCMDS communications error.

b. CET readings are invalid.

c. Core damage has occurred.

d. CET readings indicate superheat.

Answer:

d. CET readings indicate superheat.

Notes:

(d) is correct. (a) would cause a trouble alarm, (b) is incorrect since invalid readings are automatically removed from the ICCMDS calculations, (c) is incorrect since indications of superheat does not mean core damage has occurred.

References:

1105.008 (Rev 012-01-0), Inadequate Core Cooling Monitor, page 7

History:

Developed for the 1998 RO/SRO exam.
Modified existing QID 727.
Used in 2001 RO/SRO Exam.

QID: 0060 **Rev:** 0 **Rev Date:** 7/12/98 **Source:** Direct **Originator:** GGiles

TUOI: AA41002-010 **Objective:** 11 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 022 **System Title:** Containment Cooling System (CCS)

Description: Knowledge of power supplies to the following: Containment cooling fans

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

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

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

Question:

Which of the following load centers supply power to the five (5) Reactor Building Ventilation Fans?

a. B5, B6 and B7

b. B3, B4 and B2

c. B3, B4 and B7

d. B5, B6 and B2

Answer:

a. B5, B6 and B7

Notes:

(a) is correct per 1107 series procedures. (b), (c) and (d) list combination of wrong load centers.

TUOI ANO-1-LP-RO-VENT, refers to TUOI ANO-1-LP-WCO-RBPUR (AA41002-010), for review of Reactor Building Ventilation System. Objective 10 covers system interrelationships including electrical distribution.

References:

1107.002 (Rev 017-03-0), ES Electrical System Operation, p. 53-54
1107.001, Rev. 057-02-0, Electrical System Operation, p. 59

History:

Developed for the 1998 RO/SRO exam.
Used in 2001 RO/SRO Exam.

QID: 0062 **Rev:** 0 **Rev Date:** 7/12/98 **Source:** Direct **Originator:** GGiles
TUOI: ANO-1-LP-RO-ICS **Objective:** 12 **Point Value:** 1

Section: 4.3 **Type:** B&W EOP/AOP
System Number: A01 **System Title:** Plant Runback
Description: Knowledge of abnormal condition procedures.

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

Tier:	1	**RO Imp:**	3.4	**RO Select:**	No	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question:

Given the following plant conditions:

- 100% power
- Condensate Pump P-2A OOS
- K06-E7 "COND PUMP MTR WDG TEMP HI" is in alarm
- AO reports fire in P-2C motor

The CRS instructs the CBOT to trip P-2C, which of the following best describes the correct response?

a. Trip P-2C, perform immediate actions per 1203.027, Loss of Steam Generator Feed.

b. Trip P-2C, monitor ICS runback to 40% power and dispatch the fire brigade per 1203.034, Smoke, Fire or Explosion.

c. Trip P-2C and reduce power per 1203.045, Rapid Plant Shutdown, to maintain adequate main feed pump suction pressure.

d. Trip P-2C then trip the turbine and reactor and carry out immediate actions per 1202.001, Reactor Trip.

Answer:

b. Trip P-2C, monitor ICS runback to 40% power and dispatch the fire brigade per 1203.034, Smoke, Fire or Explosion.

Notes:

The plant is designed to survive a loss of 2 condensate pumps. ICS will run the plant back at 50%/min to 40% power (360 MWe). Immediate action for fire is to dispatch the fire brigade, therefore (b) is the correct response. (a) is actions for a loss of a main feedwater pump which should not occur. (c) main feed pump suction pressure will go down but recover as ICS runs plant back. (d) a reactor/turbine trip should not be required.

References:

1105.004 Rev 013-02-0, Integrated Control System, p.11
1203.034, Rev. 012-01-0, Smoke, Fire, or Explosion, p. 4

History:

Used in 2001 SRO Exam.

QID: 0065 **Rev:** 2 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** GGiles

TUOI: ANO-1-LP-RO-RMS **Objective:** 2 **Point Value:** 1

Section: 3.9 **Type:** Radioactivity Release

System Number: 068 **System Title:** Liquid Radwaste System (LRS)

Description: Ability to manually operate and/or monitor in the control room: Automatic isolation.

K/A Number: A4.04 **CFR Reference:** CFR: 41.7 / 45.5 to 45.8

Tier:	2	**RO Imp:**	3.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

A radioactive liquid release is in progress.
RE-4642, Liquid Radwaste Process Monitor detects high radiation.

What effect will this have on the release?

a. This will cause a process monitor trouble alarm to alert the control room staff to manually terminate the release.

b. An interlock will cause instrument air to be vented from radwaste flow control valve, CV-4642, terminating the release.

c. An interlock will align instrument air to the radwaste flow control valve, CV-4642, and automatically terminate the release.

d. An interlock will trip the running radwaste transfer pump to terminate the release.

Answer:

b. An interlock will cause instrument air to be vented from radwaste flow control valve, CV-4642, terminating the release.

Notes:

A loss of power to RI-4642 will result in a process monitor trouble alarm and a process monitor high radiation alarm. In addition, the loss of power will cause a high radiation signal trip of CV-4642. CV-4642 is an air to open valve. Answer (b) is correct. Answer (a) is incorrect because manual termination of the release is not required. Answer (c) is incorrect because the loss of power will vent instument air off of CV-4642 vice align it to the valve. Answer (d) is incorrect since CV-4642 will fail closed and manual termination is not required.

References:

1203.012I, Rev 038-02-0, Annunciator K10 Corrective Actions, p.9
1203.007, Rev. 8, Liquid Waste Discharge Line High Radiation Alarm, p.1,2

History:

Developed for 1998 RO/SRO Exam.
Revised after 9/98 exam analysis review.
Modified for use in 2001 RO/SRO Exam.

QID: 0079 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-1-LP-WCO-PRMS **Objective:** 2 **Point Value:** 1

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

System Number: 029 **System Title:** Containment Purge System (CPS)

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment Purge System controls including: Containment pressure, temperature, and humidity.

K/A Number: A1.03 **CFR Reference:** CFR: 41.5 / 45.5

Tier:	2	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.3	**SRO Select:**	Yes	**Taxonomy:**	K

Question:

Given:

Unit operating at 100%
Reactor Building pressure is 15.9 psia and stable
No other abnormal conditions exist

What action should be taken to lower RB pressure?

a. Open RB purge inlets first, then open outlets.

b. Vent RB via H2 sample lines.

c. Use vent flowpath via RB leak detector.

d. Open RB purge outlets first, then open inlets.

Answer:

c. Use vent flowpath via RB leak detector.

Notes:

The only procedural guidance (1104.033) for depressurizing the RB at power is via the RB Leak Detector, therefore (c) is correct. (a) & (d) are distractors that imply use of the RB Purge system which would not be done at power. (b) is incorrect because RB depressurization is not a function of the Hydrogen Sampling system.

References:

1104.033, Rev 058-00-0, Reactor Building Ventilation, p.10, 11

History:

Developed for the 1998 RO/SRO Exam.
Used in 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0088 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

TUOI: AA51002-016 **Objective:** 18 **Point Value:** 1

Section: 3.6 **Type:** Electrical

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

Description: Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Incomplete-start relay.

K/A Number: K4.05 **CFR Reference:** CFR: 41.7

Tier: 2 **RO Imp:** 2.8 **RO Select:** Yes **Difficulty:** 4

Group: 2 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** An

Question:

The CRS directs you to perform Supplement 1 of 1104.036, #1 EDG Monthly Test. You depress the start pushbutton on C10, nothing happens, then the "EDG 1 OVERCRANK" annunciator K01-B2 alarms. The CRS directs the inside AO to check the EDG out and then depress the local RESET pushbutton.

Which of the following would occur after the AO depresses the RESET pushbutton?

a. EDG would be ready for another manual start.

b. EDG will not manually or automatically start.

c. EDG output breaker will be locked out.

d. EDG will immediately start cranking.

Answer:

d. EDG will immediately start cranking.

Notes:

Answer (d) is correct since the EDG will start after reset pushbutton is depressed if stop pushbutton is not depressed to reset the logic. Answer (a) is incorrect because the EDG start logic must be reset after an overcrank, (b) is incorrect since it will automatically start and (c) is incorrect since the breaker is not affected.

References:

STM-1-3,1 Rev.6, Emergency Diesel Generators, p.8

History:

Developed for 1998 RO/SRO Exam.
Used in A. Morris 98 RO Re-exam
Used in 2001 RO Exam.

QID: 0090 **Rev:** 1 **Rev Date:** 11/4/98 **Source:** Direct **Originator:** JCork

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

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

System Number: 086 **System Title:** Fire Protection System (FPS)

Description: Knowledge of design feature(s) and/or interlocks which for the following: Detection and location of fires.

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

Tier: 2 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 4

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

Question:

Given:

- The yellow trouble LED is ON on #1 EDG flame detector module on C463 and has been acknowledged on C463.
- Subsequently, a #1 EDG smoke detector spuriously actuates.

Besides the "FIRE PROT SYSTEM TROUBLE", K12-D1, which of the following annunciators would also alarm?

a. "FIRE WATER FLOW", K12-A2

b. "FIRE WATER PRESSURE LOW", K12-B1

c. "FIRE", K12-A1

d. "FIRE PUMP AUTO START", K12-B2

Answer:

c. "FIRE", K12-A1

Notes:

The fire detection system for the EDGs require a smoke detector and a flame detector to actuate and trip the deluge, which will cause a FIRE alarm. However, because there is no fire, the system's fuseable heads will prevent firewater flow and the fire pump will not autostart. Therefore the only expected annunciator is (is (c).

References:

1203.009, Rev. 020-04-0, Fire Protection System Annunciator Corrective Action, p.2, 20, 86-90

History:

Developed for the 1998 RO/SRO Exam.
Revised after 9/98 exam analysis review.
Used in 2001 RO/SRO Exam.

QID: 0102 **Rev:** 1 **Rev Date:** 11/17/00 **Source:** Modified **Originator:** JCork

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

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

System Number: 079 **System Title:** Station Air System (SAS)

Description: Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: Cross-connect with IAS.

K/A Number: K4.01 **CFR Reference:** 41.7

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

Question:

Given:

- Instrument Air pressure has fallen to 45 psig,
- Unit Two is in a refueling outage with Breathing Air in use.

Which of the following will be in use to restore or conserve Instrument Air pressure?

a. Instrument Air to Service Air X-over valve, SV-5400

b. Cross-connect with Unit Two Instrument Air

c. Breathing Air to Instrument Air X-connection, HS-5503

d. If ICW available, isolate Seal Injection by closing CV-1206

Answer:

a. Instrument Air to Service Air X-over valve, SV-5400

Notes:

Answer [a] is correct, SV-5400 opens when IA <50 psig.
Answer [b] is incorrect, the Unit Two x-connect is isolated when IA pressure <60 psig.
Answer [c] is incorrect, Breathing Air to Inst Air X-Connect is used early in a loss of I.A. transient and is isolated if BA header pressure drops to <80 psig with personnel using BA.
Answer [d] is incorrect, CV-1206 is placed in Override and isolated only when ICW is not available.

References:

1104.024, Rev. 026-01-0, Instrument Air System, page 8

History:

Developed for 1998 RO exam
Used in A. Morris 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.

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

TUOI: ANO-1-LP-RO-ICS **Objective:** 22 **Point Value:** 1

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

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

Description: Ability to monitor automatic operation of the MFW, including: ICS.

K/A Number: A3.07 **CFR Reference:** CFR: 41.7 / 45.5

Tier: 2 **RO Imp:** 3.4 **RO Select:** Yes **Difficulty:** 4

Group: 3 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** Ap

Question:

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, Rev. 026-00-0 Annunciator K07 Corrective Action, p.7
STM 1-64, Rev.6, Intergrated Control System page 42

History:

Modified QID 4408 for use on 1998 RO/SRO Exam.
Modified for use in 2001 RO Exam.

QID: 0110 **Rev:** 1 **Rev Date:** 12/1/00 **Source:** Modified **Originator:** JCork

TUOI: ANO-S-LP-SRO-ADMIN **Objective:** 4 **Point Value:** 1

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

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to apply technical specifications for a system.

K/A Number: 2.1.12 **CFR Reference:** CFR: 43.2 / 43.5 / 45.3

Tier: 3 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 4

Group: G **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** Ap

Question:

The ONLY action statement in Tech Specs regarding hydrogen recombiner operability (3.14.2) states:
"With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or the reactor
shall be placed in the HOT SHUTDOWN condition within the next 6 hours."

Which of the following actions would be applicable if BOTH hydrogen recombiner were inoperable?

a. Place the plant in Hot Shutdown within one hour.

b. Within 1 hour commence a plant shutdown to place the plant in Hot Standby within the next 6 hours.

c. Restore at least one recombiner to operable status in 15 days or be in Hot Standby within 6 hrs.

d. Submit a report outlining the plan for restoring the system to OPERABLE status to the NRC within 7 days.

Answer:

b. Within 1 hour commence a plant shutdown to place the plant in Hot Standby within the next 6 hours.

Notes:

Answer [b] is correct. Without an action statement covering two inoperable analyzers, this condition falls under the jurisidiction of T.S. 3.0.3.
Answer [a] is incorrect, although a shutdown must commence in one hour, the plant does not have to be in Hot SDN in one hour.
Answer [c] is incorrect, this is similar to LCO statements where only one train of two redundant trains is inoperable.
Asnwer [d] is incorrect, the H2 analyzers may not be considered to be essential to plant operations and this statement is similar to action statements for non-ESF equipment.

References:

Technical Specifications, 3.03 and 3.14

History:

Modified for use in 2001 SRO exam.

QID: 0111 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** JCork

TUOI: ANO-1-LP-RO-PROCS **Objective:** 1 **Point Value:** 1

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

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of conduct of operations requirements.

K/A Number: 2.1.1 **CFR Reference:** CFR: 41.10 / 45.13

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

Group: G **SRO Imp:** 3.8 **SRO Select:** No **Taxonomy:** C

Question:

A LOCA has occurred.

- RCS pressure 1650 psig and falling SLOWLY
- RB pressure is 2.0 psig and rising SLOWLY
- RT-2, Initiate HPI, is complete

Which of the following operator actions is appropriate for these conditions?

a. Immediately actuate ESAS, then perform RT-3, Initiate Full HPI.

b. Obtain concurrence from CRS, then manually actuate ESAS and make announcement.

c. Announce parameter and imminent ESAS actuation, then verify proper automatic actuation.

d. Obtain concurrence from CRS and then bypass ESAS prior to automatic actuation.

Answer:

b. Obtain concurrence from CRS, then manually actuate ESAS and make announcement.

Notes:

Since the parameters and trend given indicate that time is available to obtain CRS/SS concurrence answer (b) is the most correct per 1015.001, Conduct of Ops. Answer (a) is applicable if there is not enough time but the wrong RT is given, (c) is wrong because it waits for automatic actuation and (d) performs the wrong action, bypass vs. actuate.

References:

1015.001, Rev 052-05-0, Conduct of Operations, p. 18,19

History:

Developed for 1998 RO/SRO exam.
Used in A. Morris 98 RO Re-exam
Modified for use in 2001 RO Exam.

QID: 0113 **Rev:** 0 **Rev Date:** 7/13/98 **Source:** Direct **Originator:** JCork

TUOI: AA61002-009 **Objective:** 5 **Point Value:** 1

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

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of less than one hour technical specification action statements for systems.

K/A Number: 2.1.11 **CFR Reference:** CFR: 43.2 / 45.13

Tier:	3	**RO Imp:**	3.0	**RO Select:**	Yes	**Difficulty:**	4
Group:	G	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	An

Question:

Given:

- The unit is operating at 100% power.
- A system engineer enters the control room with a condition report stating the PZR code safety valve (PSV-1002), replaced during the last outage, was set by the vendor using out of calibration equipment.
- The condition report estimates the setpoint for PSV-1002 could be as high as 2790 psig.
- The system engineer recommends declaring PSV-1002 inoperable.

What action would you initiate?

a. Restore PSV-1002 to operable status within 15 minutes or be in Hot Shutdown within 12 hours.

b. Within one hour initiate a shutdown to be in Hot Standby within 6 hrs and be in Hot Shutdown within another 6 hrs.

c. Restore the safety to operable status within 6 hrs or place the unit in Hot Shutdown within the following 12 hrs.

d. Restore safety to operable status within 12 hrs or place the unit in Hot Shutdown within the following 12 hrs.

Answer:

a. Restore PSV-1002 to operable status within 15 minutes or be in Hot Shutdown within 12 hours.

Notes:

(a) is the correct response per Tech Spec 3.1.1.3.A. (b), (c) and (d) provide erroneous time clocks.

References:

Technical Specification 3.1.3

History:

Developed for 1998 SRO exam
Used in 2001 RO/SRO Exam.

QID: 0116 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-1-LP-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:** CFR: 45.1

Tier: 3 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

Group: G **SRO Imp:** 3.6 **SRO Select:** No **Taxonomy:** K

Question:

During an INITIAL approach to criticality, if criticality is NOT achieved within _________ of the ECP, insert ________ and ________ .

a. plus or minus 1.0% delta k/k
control rods to achieve 1.5% SD margin
establish hot shutdown conditions

b. plus or minus 1.0% delta k/k
regulating groups
notify Reactor Engineering

c. plus or minus 0.5% delta k/k
control rods to achieve 1.5% SD margin
verify calculation

d. plus or minus 0.5% delta k/k
regulating groups
verify calculation

Answer:

c. plus or minus 0.5% delta k/k
control rods to achieve 1.5% SD margin
verify calculation

Notes:

Answer "C" is correct per 1102.008.

References:

1102.008, Approach to Criticality, Rev. 018-0-0, page 12

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

QID: 0117 **Rev:** 0 **Rev Date:** 6/29/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-S-LP-SRO-ADMIN **Objective:** 4 **Point Value:** 1

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

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for controlling temporary changes.

K/A Number: 2.2.11 **CFR Reference:** CFR: 41.10 / 43.3 / 45.13

Tier: 3 **RO Imp:** 2.5 **RO Select:** No **Difficulty:** 4

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

Question:

Which of the following would NOT require the use of a Temporary Alteration Package?

a. Installation of a test gauge for a surveillance.

b. Temporary power cables supplying temporary equipment.

c. Installing a jumper to prevent a malfunctioning instrument loop from actuating equipment.

d. Replacement of a blank flange with a vent flange.

Answer:

a. Installation of a test gauge for a surveillance.

Notes:

Answer (a) is correct in accordance with 1000.028, Control of Temporary Alterations. 4.15.2.h shows that activities using test gauges are NOT considered to be temporary alterations. The remaining answers are all covered by 4.15.1 as activities requiring a temporary alteration package.

References:

1000.028 (Rev 023-00-0), Control of Temporary Alterations, page 37

History:

Developed for the 1998 SRO exam.
Used in 2001 SRO Exam.

QID: 0120 **Rev:** 0 **Rev Date:** 7/14/98 **Source:** Direct **Originator:** JCork

TUOI: AA52001-009 **Objective:** 1 **Point Value:** 1

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

System Number: 2.3 **System Title:** Radiation Control

Description: Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

K/A Number: 2.3.10 **CFR Reference:** CFR: 43.4 / 45.10

Tier: 3 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 4

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

Question:

Given:

- A Site Area Emergency has been declared on Unit 1.
- An Emergency Medical Team member must enter a 50 REM/hr area to rescue a critically injured employee.

Which of the following is the MAXIMUM time an individual team member can stay in this area?

a. 15 minutes

b. 30 minutes

c. 45 minutes

d. 60 minutes

Answer:

b. 30 minutes

Notes:

The limit for life saving is 25 rem TEDE. 50R/hr means a 30 minute stay time, therefore "B" is correct.

References:

1903.033, Protective Action Guidelines for Rescue, Rev. 017-01-0, p.5

History:

Modifiied for use in 1998 SRO exam
Modified question from NRC developed SRO exam 2/6/95, no. 94
Used in A. Morris 98 RO Re-exam
Used in 2001 RO/SRO Exam.

QID: 0121 **Rev:** 0 **Rev Date:** 12/06/00 **Source:** New **Originator:** S. Pullin

TUOI: ANO-S-LP-RO-RADP **Objective:** 15 **Point Value:** 1

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

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of 10CFR20 and related facility radiation control requirements.

K/A Number: 2.3.1 **CFR Reference:** CFR: 41.12 / 43.4 / 45.9, 45.10

Tier:	3	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	G	**SRO Imp:**	3.0	**SRO Select:**	Yes	**Taxonomy:**	K

Question:

What is the federal occupational exposure limit to the skin of the whole body in accordance with 10CFR20?

a. 5.0 rems/calendar year

b. 15.0 rems/calendar year

c. 25.0 rems/calendar year

d. 50.0 rems/calendar year

Answer:

d) 50.0 rems/calendar year

Notes:

(d) is the correct answer.
(a), (b), and (c) are incorrect values.

References:

1012.021, Exposure Limits and Controls, Rev. 004-01-0, page 5.

History:

New question developed for 2001 RO/SRO NRC Exam.

QID: 0125 **Rev:** 0 **Rev Date:** 7/13/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-1-LP-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 RO tasks performed outside the main control room during emergency operations including system geography and system implications.

K/A Number: 2.4.34 **CFR Reference:** CFR: 43.5 / 45.13

Tier:	3	**RO Imp:**	3.8	**RO Select:**	No	**Difficulty:**	2
Group:	G	**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	K

Question:

During an Alternate Shutdown requiring an immediate control room evacuation,
which of following is performed by a control board operator (RO#1 or #2)?

a. Start and load EDGs to power vital components

b. Start and stop HPI pump to maintain PZR level

c. Throttle EFW to OTSGs to maintain heat sink

d. Strip 6900v H1 and H2 buses

Answer:

c. Throttle EFW to SG's to maintain heat sink

Notes:

Answer [c] is correct. The CRS performs all of the tasks with the exception of "c" which is shared between the RO's.
Answers [a], [b], [d] are performed during Alternate Shutdown by the CRS.

References:

1203.002, Rev. 015-01-0, Alternate Shutdown p. 23

History:

Used in 2001 SRO Exam.

QID: 0128 **Rev:** 1 **Rev Date:** 11/4/98 **Source:** Direct **Originator:** JCork

TUOI: AA61002-006 **Objective:** 14 **Point Value:** 1

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

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of the emergency plan.

K/A Number: 2.4.29 **CFR Reference:** CFR: 43.5 / 45.11

Tier: 3 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 2

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

Question:

Which of the following would be classified as a fission product barrier failure?

a. RCS leakage indicates greater than 30 gpm.

b. EOF Director determines the Reactor Building is breached.

c. CNTMT radiation levels equal to Alert level from CNTMT Radiation EAL.

d. The inability to monitor a Fission Product Barrier.

Answer:

b. EOF Director determines the RB is breached.

Notes:

"B" is correct per EAL definition.
"A" is incorrect, leakage must be >50 gpm.
"C" is incorrect, rad levels must be equal to SAE level.
"D" is incorrect, the correct definition is two fission product barriers known to be breached with the inability to monitor the third.

References:

1903.010, Emergency Action Level Classification, Rev. 036-02-0, pages 4 and 5.

History:

Developed for 1998 SRO exam.
Revised after 9/98 exam analysis review.
Used in 2001 SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0133 **Rev:** 0 **Rev Date:** 08/16/95 **Source:** Direct **Originator:** D. Walls

TUOI: AA51002-001 **Objective:** 23 **Point Value:** 1

Section: 3.3 **Type:** RX Pressure Control

System Number: 010 **System Title:** Pressurizer Pressure Control System

Description: Knowledge of the purpose and function of major system components and controls.

K/A Number: 2.1.28 **CFR Reference:** 41.7

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

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

Question:

Which of the following are the two purposes for maintaining a one gpm continuous spray flow bypassing the pressurizer spray valve?

a. Assists in pressurizer level control and maintains pressurizer heaters at minimal firing rate.

b. Minimizes surge and spray line temperature differentials and maintains pressurizer boron concentration near that in the RCS.

c. Assists in maintaining pressurizer level and maintains pressurizer boron concentration near that in the RCS.

d. Minimizes surge and spray line temperature differentials and raises the differential pressure across the spray valve.

Answer:

b. Minimizes surge and spray line temperature differentials and maintains pressurizer boron concentration near that in the RCS.

Notes:

[b] is correct since the bypass flow will keep the spray line warm and the extra flow will cause some circulation through the surge line. The continous flow will also help to maintain the PZR boron close to the RCS boron concentration.
[a] and [c] are incorrect since the spray flow comes from the RCS, it will do nothing to maintain PZR leve.
[d] is incorrect, although the first portion is correct, the DP across the valve will be minimal.

References:

STM 1-03, rev. 8 ch.1 Reactor Coolant System, page 14

History:

Taken from Exam Bank QID # 2182
Used in A. Morris 98 RO Re-exam
Used in 2001 RO/SRO Exam.

QID: 0134 **Rev:** 0 **Rev Date:** 11/30/98 **Source:** Direct **Originator:** B. Short

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

Section: 3.7 **Type:** Instrumentation

System Number: 015 **System Title:** Nuclear Instrumentation System

Description: Knowledge of the operational implications of the following concepts as they apply to the NIS: Excore detector operation.

K/A Number: K5.10 **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:**	K

Question:

The Gamma Metrics system uses discrimination circuitry to provide accurate indication of source range power levels.

Why is discrimination necessary in source range nuclear instrumentation and what is the result of incorrect discrimination?

a. Discrimination separates out alpha and neutron pulses to provide a true gamma pulse. With the discrimination set too high power will indicate higher than actual power.

b. Discrimination separates out gamma and neutron pulses to provide a true alpha pulse. With the discrimination set too low, power will indicate higher than actual power.

c. Discrimination separates out gamma and alpha pulses to provide a true neutron pulse. With the discrimination set too high, power will indicate lower than actual power.

d. Discrimination separates out beta and gamma pulses to provide a true neutron pulse. With the discriminator set too low, power will indicate lower than actual power.

Answer:

c. Discrimination separates out gamma and alpha pulses to provide a true neutron pulse. With the discrimination set too high, power will indicate lower than actual power.

Notes:

The discriminator module in Gamma Metrics passes only pulses produced from neutrons. Gamma and Alpha pulses are not above the minimum discrimination voltage. If the discrimination voltage was set too high, some of the neutron pulses would not be counted and thus indicated power would be less than actual power. (c) is correct.
(a) (b) and (d) have various forms of the answer in incorrect applications.

References:

STM 1-67, Rev. 6, ch. 1, Nuclear Instrumentation, page 16, 17

History:

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

QID: 0137 **Rev:** 1 **Rev Date:** 04/15/93 **Source:** Direct **Originator:** G. Alden

TUOI: ANO-1-LP-RO-ICS **Objective:** 28 **Point Value:** 1

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

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

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

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

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

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

Question:

Which one of the following is NOT a function of the Rapid Feedwater Reduction feature of ICS?

a. Low Load and Startup Control Valve demands are reduced to zero.

b. Main Feedwater Pump speed goes to minimum.

c. Both Main Feedwater Block Valves close in slow speed.

d. Both Loop Feedwater demands are reduced to zero.

Answer:

c. Both Main Feedwater Block Valves close in slow speed.

Notes:

[a], [b], & [d] are part of the RFR circuit and while [c] appears to be a logical component of this, the [c] function is independent of RFR.

References:

STM 1-64 Rev 6, Integrated Control System page 40

History:

Taken from Exam Bank QID # 3262 (modified answers slightly)
Used in A. Morris 98 RO Re-exam
Used in 2001 RO/SRO Exam.

QID: 0143 **Rev:** 2 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** D.Walls

TUOI: ANO-1-LP-SRO-ADMIN **Objective:** 4 **Point Value:** 1

Section: 2 **Type:** Generic K&As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of how to conduct and verify valve lineups.

K/A Number: 2.1.29 **CFR Reference:** 41.10 / 45.1 / 45.12

Tier:	3	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	2
Group:	G	**SRO Imp:**	3.3	**SRO Select:**	No	**Taxonomy:**	Ap

Question:

The plant is shut down for Refueling.
A Core Flood system valve alignment is in progress inside Controlled Access.
The primary sample room has become a high radiation area due to hydorgen peroxide cleanup.
The first check was made on CF-2, Core Flood Combined Sample Isolation,
but the Shift Manager decided to waive the second check to reduce the exposure to high radiation.

Which one of the following statements most accurately describes why the Shift Manager's decison is acceptable or unacceptable?

a. Acceptable, independent verifications are always waived for valve alignments inside High Radiation Areas.

b. Unacceptable, independent verification cannot be waived if remote valve position indication is provided.

c. Acceptable, independent verification can be waived for any valve with the Shift Manager's approval.

d. Unacceptable, independent verifications cannot be waived for valve alignments without the approval of the Manager of Plant Operations.

Answer:

b. Unacceptable, independent verification cannot be waived if remote valve position indication is provided.

Notes:

[b] meets the guidance of 1015.035.
[a] untrue, independent verifications are not always waived, they are only waived on a case by case basis.
[c] is untrue, the Shift Manager can only waive verification in specific situations.
[d] lists the wrong approval authority.

References:

1015.035, Rev 010-00-0, Valve Operations, p. 11

History:

Taken from Exam Bank QID # 3273
Used in A. Morris 98 RO Re-exam
Modified for use in 2001 RO Exam.

QID: 0145 **Rev:** 1 **Rev Date:** 11/16/94 **Source:** Direct **Originator:** G. Alden

TUOI: ANO-1-LP-RO-CRD **Objective:** 16 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive System

Description: Knowledge of system purpose and/or function.

K/A Number: 2.1.27 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	K

Question:

The purpose of the IN-LIMIT (LATCH) BYPASS switch on the Diamond panel is to:

a. Apply power to the CRD motor which will engage the latching mechanism.

b. Reset a fault condition provided the fault has cleared.

c. Reset the AC breakers, DC breakers, and programmer controls.

d. Allow driving in Groups 1 thru 7 to engage roller nuts with lead screws.

Answer:

d. Allow driving in Groups 1 thru 7 to engage roller nuts with lead screws.

Notes:

(a) is incorrect. The CRD motor has power applied at all times.
(b) is incorrect. This is accomplished with the fault reset pushbutton.
(c) is incorrect. This is accomplished with the Trip/reset pushbutton.
(d) is correct. The Latch pushbutton must be depressed to allow the in-limits to be bypassed when engaging lead scews on groups 1-7 after they have been deenergized.

References:

STM 1-02, Rev. 5, Control Rod Drive System, p. 24

History:

Taken from Exam Bank QID # 2684
Used in A. Morris 98 RO Re-exam
Used in 2001 RO Exam.

QID: 0156 **Rev:** 3 **Rev Date:** 06/28/97 **Source:** Direct **Originator:** M. Goad

TUOI: ANO-1-LP-RO-EOP03 **Objective:** 7 **Point Value:** 1

Section: 4.2 **Type:** B&W EOP/AOP

System Number: E05 **System Title:** Excessive Heat Transfer

Description: Ability to determine and interpret the following as they apply to the (Excessive Heat Transfer): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A Number: EA2.2 **CFR Reference:** 43.5 / 45.13

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

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

Question:

Given:
* A reactor trip has occurred.
* RCS pressure is 1800 psig,
* RCS T-cold is 532 degrees F,
* "A" OTSG pressure is 650 psig,
* "B" OTSG pressure is 970 psig,
* Reactor Building pressure is 6 psig.

Which emergency operating procedure contains the specific steps to mitigate the consequences of this event?

a. ESAS 1202.010

b. Overcooling 1202.003

c. HPI Cooldown 1202.011

d. Loss of Subcooling Margin 1202.002

Answer:

b. Overcooling 1202.003

Notes:

The key to this question is in realizing that T-cold is lower than normal and that one OTSG is <900 psig. Also, RCS pressure is above ESAS actuation pressure but RB pressure is greater than ESAS actuation setpoint. These are three of the five possible entry conditions for 1202.003, the Overcooling EOP, and all of these conditions are indicative of a steam line rupture inside the RB. Also, the floating steps for the Reactor Trip EOP send the user to the Overcooling EOP.

References:

1202.010, Rev. 005-00-0, ESAS page 2

History:

Taken from Exam Bank QID # 556
Used in A. Morris 98 RO Re-exam
Used in 2001 RO/SRO Exam

QID: 0158 **Rev:** 0 **Rev Date:** 11/11/98 **Source:** Direct **Originator:** JCork

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.3 **Point Value:** 1

Section: 4.2 **Type:** Generic AOPs

System Number: 069 **System Title:** Loss of Containment Integrity

Description: Knowledge of abnormal condition procedures.

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

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

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

Question:

The plant is at 100% power.
The outside door of the personnel lock was opened to replace a seal gasket 24 hours ago.

How long does operations have to perform an LLRT on the personnel lock before a loss of containment integrity will exist?

a. 1 hour

b. 12 hours

c. 6 days

d. 13 days

Answer:

c. 6 days

Notes:

[c] is the correct answer per 1203.005.
[a] is incorrect, this can occur but only one hour is allowed to perform the repairs.
[c] is incorrect, this is applicable to automatic containment isolation valves, but not the air lock.
[d] is an incorrect action since it states "within one hour" and is only necessary if the LLRT is unsuccessful.

References:

1203.005 [Rev 010-00-0], Loss of Reactor Building Integrity, page 1

History:

Developed for 1998 RO Re-exam
Used in 1999 exam.
Used in 2001 RO/SRO Exam.

QID: 0160 **Rev:** 1 **Rev Date:** 08/05/94 **Source:** Direct **Originator:** E. Wentz

TUOI: AA51003-014 **Objective:** 14.3 **Point Value:** 1

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

System Number: 041 **System Title:** Steam Dump System (SDS) and Turbine Bypass Control

Description: Knowledge of bus power supplies to the following: ICS, normal and alternate power supply.

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

Tier:	2	**RO Imp:**	2.8	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	K

Question:

During a Rx trip transient, all nine of the + or - 24v DC NNI/ICS power supply status lights go out.

The most likely cause is:

a. Loss of Offsite Power

b. Loss of DC Bus D11

c. Loss of Y02

d. Loss of DC Bus D21

Answer:

b. Loss of DC Bus D11

Notes:

The first step of EOP 1202.009 checks these indicating lights and a note gives the power supply as breaker 25 on D-11.
Therefore, [b] is the correct answer as long as no other abnormal indications are present.
[a] is incorrect, D11 is battery backed.
[c] is incorrect, although Y02 supplies power to ICS and NNI, neither of these power the indicating lights.
[d] is incorrect, the lights are only powered from D11, although the two are very similar in other functions.

References:

1203.047, Rev. 0, Loss of NNI Power, p.1

History:

Taken from Exam Bank QID # 3202
Used in A. Morris 98 RO Re-exam
Used in 2001 RO Exam.

QID: 0161 **Rev:** 0 **Rev Date:** 10/22/98 **Source:** Direct **Originator:** J. Cork

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.3 **Point Value:** 1

Section: 2 **Type:** Generic K&As

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to recognize indications for system operating parameters which are entry-level conditions for Technical Specifications.

K/A Number: 2.1.33 **CFR Reference:** 43.2, 43.3 / 45.3

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

Group: 2 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** C

Question:

Given:
- Power escalation is in progress following a shutdown.
- Reactor power is 30%.
- Rod 4 of Group 5 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. Refer to Tech Specs and recover the dropped rod.

Answer:

d. Refer to Tech Specs and recover the dropped rod.

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 1203.003, Tech Specs would be referenced, actions taken to ensure compliance, and the dropped rod recovered.

References:

1203.003, Rev. 19 pc1, Control Rod Drive Malfunction Action, page 3

History:

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

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0167 **Rev:** 0 **Rev Date:** 10/24/91 **Source:** Direct **Originator:** M. Cooper

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.2 **Point Value:** 1

Section: 4.2 **Type:** Generic AOP

System Number: 059 **System Title:** Accidental Liquid Radioactive-Waste Release

Description: Ability to operate and/or monitor the following as they apply to the Accidental Liquid Radwaste Release: Radioactive-liquid monitor.

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

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

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

Question:

What action is required upon receipt of Liquid Radwaste Process Monitor (RI-4642) high alarm?

a. Start another circ water pump to increase dilution flow.

b. Verify no release in progress at Disch Flow to Flume (FI-4642) on C19.

c. Verify with Unit 2 no other release is in progress.

d. Have chemistry sample discharge flume for radionuclides.

Answer:

b. Verify no release in progress at Disch Flow to Flume (FI-4642) on C19.

Notes:

Per 1203.007 immediate action for a liquid radwaste process monitor alarm is to verify that no release in progress at FI-4642.(answer b) All other actions are associated with radwaste discharges but are not the immediate action for the alarm.

References:

1203.007 (Rev 8), Liquid Waste Discharge Line High Radiation Alarm, p.1,2

History:

Modified from Exam Bank QID # 1725
Used in A. Morris 98 RO Re-exam
Used in 2001 RO/SRO Exam.

QID: 0169 **Rev:** 0 **Rev Date:** 11/19/98 **Source:** Direct **Originator:** J. Cork

TUOI: ANO-1-LP-RO-NNI **Objective:** 19 **Point Value:** 1

Section: 4.2 **Type:** Generic AOP

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

Description: Ability to operate and/or monitor the following as the apply to the Pressurizer Level Control Malfunction: CVCS.

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

Tier:	1	**RO Imp:**	3.4	**RO Select:**	Yes	**Difficulty:**	3
Group:	3	**SRO Imp:**	3.4	**SRO Select:**	Yes	**Taxonomy:**	An

Question:

Given:

- Plant is at 100% power.
- PZR level transmitter LT-1001 selected via HS-1002 on C04.
- PZR temperature element TE-1001A selected via HS-1000 on C04.

The PZR temperature indicator, TI-1000, on C04 drops suddenly to 50°F (bottom of scale).

Without operator action, what will be the effect on the PZR Level Control System?

a. PZR Level Control Valve, CV-1235, will open to establish a higher steady-state PZR level.

b. PZR Level Control Valve, CV-1235, will maintain the same steady-state PZR level.

c. PZR Level Control Valve, CV-1235, will close to establish a lower steady-state PZR level.

d. PZR Level Control Valve, CV-1235, will fail open to continuously raise PZR level.

Answer:

a. PZR Level Control Valve, CV-1235, will open to establish a higher steady-state PZR level.

Notes:

[a] is correct. A loss of temperature compensation will result which will appear as a low PZR level. This is the same reason which makes [b] & [c] incorrect.
(d) is incorrect. The loss of temperature compensation does not produce an indication that is similar to a high off scale indication.

References:

STM 1-69, rev. 4, Non-Nuclear Instrumentation System, p. 20 thru 22

History:

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

QID: 0170 **Rev:** 0 **Rev Date:** 08/10/95 **Source:** Direct **Originator:** J. Haynes

TUOI: AA51003-012 **Objective:** 12.2 **Point Value:** 1

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

System Number: 078 **System Title:** Instrument Air

Description: Knowledge of bus power supplies to the following: Instrument air compressor.

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

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

Group: 3 **SRO Imp:** 2.9 **SRO Select:** Yes **Taxonomy:** C

Question:

During a loss of offsite power with a SG tube leak, the A2 bus is re-energized from the A4 bus. The A4 bus is supplied by #2 EDG.

What is the key reason for this action?

a. To start P-7B EFW pump and secure P-7A.

b. To restart circ. water and re-establish condenser vacuum.

c. To allow operation of the Aux Feedwater pump (P-75).

d. To re-establish Instrument Air and ICW cooling.

Answer:

d. To re-establish instrument air and ICW cooling.

Notes:

The strategy here, regardless of the tube leak, is to re-establish Instrument Air and ICW and ease complications of this transient by restoring RCP seal cooling and Letdown. Thus, answer [d] is correct.
[a] is a good idea during a tube leak but P-7B is powered from A3, making it unnecessary to energize A2.
[b] is also a good idea but procedural actions eliminate the need to re-establish condenser vacuum.
[c] incorrect, although the Aux Feedwater Pump is powered from A2, it is not the basis for performing this action.

References:

1202.007 [Rev 005-01-0], Degraded Power, pages 53 thru 55

History:

Taken from Exam Bank QID # 2791
Used in A. Morris 98 RO Re-exam
Used in 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0178 **Rev:** 0 **Rev Date:** 11/21/98 **Source:** Direct **Originator:** J. Haynes

TUOI: ANO-1-LP-RO-EOP05 **Objective:** 3 **Point Value:** 1

Section: 4.1 **Type:** Generic EOP

System Number: 074 **System Title:** Inadequate Core Cooling

Description: Knowledge of symptom based EOP mitigation strategies.

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

Tier:	1	**RO Imp:**	3.1	**RO Select:**	Yes	**Difficulty:**	4
Group:	1	**SRO Imp:**	4.0	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

After entering Inadequate Core Cooling EOP (1202.005), RCS temperature and pressure indicate entry into Region 4 of Figure 4.

What RCP action is appropriate to restore core cooling?

a. Start all RCPs even if RCP services are not available.

b. Restore RCP services and start one RCP/loop.

c. Restore RCP services and bump RCPs until primary to secondary heat transfer is established.

d. Bump all RCPs even if RCP services are not available.

Answer:

a. Start all RCPs even if RCP services are not available.

Notes:

(a.) is correct. These are appropriate actions for Region 4.
(b.) is incorrect. These are apporpriate actions for Region 3.
(c.) is incorrect. RCPs are bumped in Region 2 only if RCP services are available.
(d) is incorrect. RCPs are never bumped without services restored.

References:

1202.005 Rev. 004-00-0, Inadequate Core Cooling, page 9

History:

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

QID: 0184 **Rev:** 0 **Rev Date:** 11/21/98 **Source:** Direct **Originator:** R. Fuller

TUOI: ANO-1-LP-RO-AOP **Objective:** 4.2 **Point Value:** 1

Section: 4.2 **Type:** Generic AOP

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

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Startup termination on source-range loss.

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

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.6	**SRO Select:**	Yes	**Taxonomy:**	An

Question:

Given:

- During a reactor startup with source range NI-2 and reactor power wide range recorder NR-502 inoperable, source range NI-1 fails to 10 E5.
- Intermediate range NI-3 indicates 5 E-11 amps
- Intermediate range NI-4 is off scale low.

What is required of the CBOR?

a. Continue the startup utilizing NI-3 until NI-4 comes on scale.

b. Perform a plant shutdown in accordance with normal operating procedures due to lack of proper overlap.

c. Trip the reactor due to no on-scale indication of neutron flux available.

d. Hold power constant and perform an NI calibration.

Answer:

c. Trip the reactor due to no on-scale indication of neutron flux available.

Notes:

[c] is correct per guidance in 1203.021, if the recorder NR.502 is inoperable AND no SR channel is >10 E5 cps AND no IR channel is > 1 E-10 amps AND 3/4 PR instruments are <10% power, then no on-scale flux indication exists and the reactor must be tripped.
[a] is incorrect, although NI-4 might come on scale, the startup should not be continued without valid neutron flux indication.
[b] is incorrect, although shutting down is conservative, per procedure the reactor must be tripped immediately.
[d] is incorrect, this action sounds like it could rectify this situation, however, it would be impossible to calibrate the NI's at this point and would be contrary to procedural guidance.

References:

1203.021 (Rev 007-01-0), Loss of Neutron Flux Indication, page 7

History:

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

QID: 0192 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** J. Haynes

TUOI: 51002-012 **Objective:** 4 **Point Value:** 1

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

System Number: 013 **System Title:** Engineered Safety Features Actuation System

Description: Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors.

K/A Number: K6.01 **CFR Reference:** 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 2.7 **RO Select:** Yes **Difficulty:** 4

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

Question:

Reactor Building Pressure Transmitter (PT-2407) has failed high causing an ES CH3 Trip (analog 3) and the ESAS Partial Trip annunciator (K11-F6) to come into alarm.

With the above conditions, a power loss to which of the following would cause an ESAS actuation?

a. RS1

b. B11

c. RS3

d. B21

Answer:

a. RS1

Notes:

(a.) is correct. A loss of power to RS1 will trip Analog Channel 1 which would then complete a 2 out of 3 analog trip causing an ESAS actuation.
(b.) & (d.) are incorrect. Loss of power to B11 or B21 does not result in a loss of any ESAS functions.
(c.) is incorrect. Analog 3 would be tripped as a result of a power loss to B11. Analog 3 is already tripped.

References:

STM 1-65, ESAS, 3,4 and 8

History:

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

QID: 0193 **Rev:** 0 **Rev Date:** 11/23/98 **Source:** Direct **Originator:** R. Fuller

TUOI: ANO-1-LP-RO-NI **Objective:** 8 **Point Value:** 1

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

System Number: 002 **System Title:** Reactor Coolant

Description: Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship between reactor power and RCS differential temperature.

K/A Number: K5.10 **CFR Reference:** 41.5 / 45.7

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

Group: 2 **SRO Imp:** 4.1 **SRO Select:** Yes **Taxonomy:** An

Question:

Several plant parameters can be monitored to ensure accurate indications of reactor power are available. Which of the following sets of parameters would be indicative of 60% reactor power?

a. Tave 579 degrees, Thot 593 degrees, Tcold 564 degrees, total FW flow 6.5 million lbm/hr.

b. Tave 580 degrees, Thot 599 degrees, Tcold 560 degrees, total FW flow 8.4 million lbm/hr.

c. Tave 579 degrees, Thot 588 degrees, Tcold 570 degrees, total FW flow 6.5 million lbm/hr.

d. Tave 581 degrees, Thot 590 degrees, Tcold 565 degrees, total FW flow 9.8 million lbm/hr.

Answer:

a. Tave 579 degrees, Thot 593 degrees, Tcold 564 degrees, total FW flow 6.5 million lbm/hr.

Notes:

(a.) is correct.
(b.) is incorrect. Parameters are indicative of >70% power.
(c.) is incorrect. Delta T is indicative of ~40% power.
(d.) is incorrect. Delta T is indicative of ~56% power, however, FW flow is indicative of ~90% power.

References:

1102.004 (Rev 039-03-0), Power Operations, pages 26, 31

History:

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

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

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

TUOI: AA51002-008 **Objective:** 8.9 **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:** C

Question:

A plant startup is in progress with the reactor critical below the point of adding heat. 'B' SG Turbine Bypass Valve (CV-6688) fails full open and is unable to be closed with the handjack.

Given the following plant conditions:

- Tave 526 degrees and dropping
- Pressurizer level 205 inches and dropping
- RCS pressure 2120 psig and dropping

What is the proper course of action?

a. Initiate MSLI for the 'B' SG and maintain the reactor critical using 'A' SG TBV 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 the Overcooling tab (1202.003) of the EOP for actions to mitgate the oversteaming of the 'B' SG.

d. Trip the reactor and go to Reactor Trip tab (1202.001) of the EOP.

Answer:

d. Trip the reactor and go to Reactor Trip tab (1202.001) of the EOP.

Notes:

(a.) is incorrect. You would not want to isolate a SG and maintain the reactor critical.
(b.) is incorrect. With the reactor below the point of adding heat with a stuck open TBV, this would not be possible.
(c.) is incorrect. This will be the ultimate tab that you will end up in, however, it is necessary to trip the reactor first and progress through the Reactor Trip tab.

References:

1102.008 (Rev 018-00-0), Approach to Criticality, page 4

History:

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

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0205 **Rev:** 0 **Rev Date:** 11/24/98 **Source:** Direct **Originator:** B. Short

TUOI: ANO-1-LP-RO-MSSS **Objective:** 1.4 **Point Value:** 1

Section: 3.8 **Type:** Plant Services System

System Number: 075 **System Title:** Circulating Water System

Description: Knowledge of circulating water system design feature(s) and interlock(s) which provide for the following: Heat Sink.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3

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

Question:

During 3 circulating water pump operation, the 'A' circ water pump trips.
The standby circ pump was started and plant conditions have been stabilized.
It is noticed that the condenser waterbox discharge temperature is 10 degrees higher and plant efficency has dropped. Which of the following is the most likely cause of this condition?

a. The stopping and starting of a circ pump caused fouling to be removed from the tube sheet promoting better heat transfer capabilities.

b. The discharge valve on the tripped pump did not go completely closed and circulating water is short cycling.

c. The debris on the bar grates of the circulating water bays was stirred up during the circ pump swap causing reduced flow.

d. These are normal conditions following rotation of circulating pumps and temperatures will return to normal within 30 minutes.

Answer:

b. The discharge valve on the tripped pump did not go completely closed and circulating water is short cycling.

Notes:

(a.) is incorrect. Although some fouling can be removed during pump rotations, it should not result in a 10 degree change in waterbox discharge temperature.
(b.) is correct. The discharge valve on an idle pump can allow a significant amount of backflow from the operating pumps if it is not closed completely.
(c.) is incorrect. This condition is normal for a circ pump swap and may contribute to waterbox fouling, however, the service water system would be affected by this condition as well.
(d.) is incorrect. There should not be such a large temperature difference even if only 3 CW pumps are in service.

References:

1104.008 (Rev 02-00-0), Circulating Water System, page11

History:

Developed for use in A. Morris 98 RO Re-exam
Used in 2001 RO Exam.

QID: 0213 **Rev:** 1 **Rev Date:** 11/20/00 **Source:** Modified **Originator:** S.Pullin

TUOI: ANO-1-LP-WCO-VENT **Objective:** 8 **Point Value:** 1

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

System Number: 034 **System Title:** Fuel Handling

Description: Knowledge of new and spent fuel movement procedures.

K/A Number: 2.2.28 **CFR Reference:** 43.7 / 45.13

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

Group: 3 **SRO Imp:** 3.5 **SRO Select:** Yes **Taxonomy:** C

Question:

Given:

- Refueling outage in progress.
- Fuel handlers are moving fuel from core to Spent Fuel Pool.
- You are assigned to perform Att. B, Refueling Boron, Temperature, and Level Check from 1502.004, Control of Unit 1 Refueling.

While performing Att. B you discover the Spent Fuel Ventilation Exhaust Fan VEF-14A flow rate is high outside of the allowable flow band.

What action should be promptly taken?

a. Notify fuel handlers to stop fuel movement in the Spent Fuel Pool area.

b. Stop Spent Fuel Ventilation Exhaust Fan VEF-14A.

c. Adjust ventilation dampers to restore flow rate below maximum.

d. No action is required when flow rate is high.

Answer:

a. Notify fuel handlers to stop fuel movement in the Spent Fuel Pool area.

Notes:

Answer [a] is correct, per Att. B and 1502.004 precaution and limitation, ventilation must be within the normal flow band during fuel movements in Spent Fuel Pool.
Answer [b] is incorrect, stopping the fan will decrease flow but then there will be no ventilation during fuel movement.
Answer [c] is incorrect, this will possibly correct high flow, but this will take local manual action again, the primary response should be to stop fuel movement.
Answer [d] is incorrect, although high flow might not seem like a problem, action is required as stated above.

References:

1502.004, Rev 031-04-0, Control of Unit 1 Refueling, Att B page 25

History:

Developed for use in A. Morris 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

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

TUOI: AA51003-011 **Objective:** 11.3 **Point Value:** 1

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

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

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

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

Tier:	1	**RO Imp:**	3.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

Given:

- A 15 gpm Steam Generator tube leak cooldown is in progress
- Normal cooldown limits are being used with the good OTSG.
- RCS pressure is 1000 psig, Tave is 405°F.
- The CBOR is maintaining the RCS at about 140°F subcooled.

What is the primary reason the CBOR's actions are incorrect for this accident?

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

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

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

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

Answer:

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

Notes:

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

References:

1202.006 [Rev 007-02-0) Tube Rupture page 11

History:

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

QID: 0221 **Rev:** 1 **Rev Date:** 11/16/00 **Source:** Modified **Originator:** B. Short

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

Section: 3.7 **Type:** Instrumentation

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

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

K/A Number: A2.02 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	An

Question:

Given:

- "A" RPS is in Channel Bypass due to RCS pressure transmitter failure.
- Plant is at 100% power and stable.
- Subsequently, a lightning strike on the Reactor Building has resulted in the trip of 120V Vital AC distribution panel RS-3.

What should your action be in response to this event?

a. Go to 1202.001, Reactor Trip EOP in response to automatic reactor trip.

b. Immediately remove "A" RPS channel from Channel Bypass.

c. Perform 1203.045, Rapid Plant Shutdown, to comply with Tech Specs.

d. Make a station log entry and take action to restore power to RS-3.

Answer:

d. Make a station log entry and take action to restore power to RS-3.

Notes:

Answer [d] is correct, RPS is in 2/3 logic and only channel "C" is tripped with a loss of power to RS-3.
Answer (a) is incorrect, only one channel is tripped and it still takes two channels to trip the reactor in this condition.
Answer (b) is incorrect, removing "A" channel from bypass would result in an automatic reactor trip.
Answer (c) is incorrect, given conditions do not meet entry conditions for 1203.045 or TS LCO.

References:

STM 1- 63, Rev. 4, Reactor Protection System, page 13,14, & 17
1015.001, rev. 052-05-0, Conduct of Operations, page 27, 28, 29

History:

Developed for A. Morris 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.

QID: 0225 **Rev:** 1 **Rev Date:** 12/1/00 **Source:** Modified **Originator:** B. Short

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

Section: 3.6 **Type:** Electrical

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the EDG System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Synchronization of the ED/G with other electrical power supplies.

K/A Number: A2.09 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	3.1	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.3	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

During the performance of the normal DG1 monthly surveillance test, while the CBOT is paralleling the diesel to the grid, he accidentally goes to closed on the output breaker with the synchroscope at the 15 minute before 12:00 position.

What would be the consequences of the CBOT's action?

a. A protective feature will prevent the output breaker from closing in.

b. The output breaker will close in and the diesel will pick up load.

c. The output breaker will close in and immediately trip back open.

d. Breaker will remain open due to a lockout relay trip.

Answer:

a. A protective feature will prevent the output breaker from closing in.

Notes:

Answer (a) is correct. The sync check protective relay protects the diesel from tying on out of phase.
Answers (b) & (c) are incorrect. Although there are other features associated with the diesel circuitry that could cause these two conditions to occur, the sync check protective relay will prevent the breaker from closing in at all.
Answer (d) is incorrect. Paralleling the diesel to the grid out of phase does not directly result in a lockout relay trip.

References:

1104.036, Rev. 039-03-0, Emergency Diesel Generator Operation

History:

Modified for use on 2001 SRO Exam.

QID: 0231 **Rev:** 1 **Rev Date:** 11/20/00 **Source:** Modified **Originator:** J.Cork

TUOI: AA30001-005 **Objective:** 3 **Point Value:** 1

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

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of tagging and clearance procedures.

K/A Number: 2.2.13 **CFR Reference:** 41.10 / 45.13

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

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

Question:

Which of the following conditions is correct with regard to preparation and installation authorization of a common unit tagout?

a. Installation may be authorized by either the Unit 1 or the Unit 2 Operations Supervisor.

b. Preparers and reviewers from both units must be licensed operators.

c. Preparer and reviewer may be non-licensed if authorized by both Unit Operations Supervisors.

d. Preparer and reviewer may be non-licensed if the opposite unit reviewer is licensed.

Answer:

b. Preparers and reviewers from both units must be licensed operators.

Notes:

Answer [b] is correct, procedure requires both the preparer and the reviewer on the unit preparing the tagout have to be licensed.
Answer [a] is incorrect, a common unit tagout requires both Unit's Operations Supervisors to approve it.
Answer (c) is incorrect, both Unit Ops Supervisors must approve but the preparation & review must be done by licensed operators.
Answer [d] is incorrect, the preparation & review must be done by licensed operators on their respective units.

References:

1000.027, Rev. 026-01-0, Protective Tagging Control, Att. A

History:

Developed for use on A. Morris 98 RO Re-exam
Modified for use in 2001 RO/SRO Exam.

QID: 0239 **Rev:** 0 **Rev Date:** 11/21/98 **Source:** Direct **Originator:** J. Haynes

TUOI: ANO-1-LP-RO-EOP05 **Objective:** 3 **Point Value:** 1

Section: 4.1 **Type:** Generic EOP

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

Description: Ability to determine the following as they apply to a Large Break LOCA: Actions to be taken based on RCS temperature and pressure - saturated and superheated.

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

Tier: 1 **RO Imp:** 4.2 **RO Select:** No **Difficulty:** 4

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

Question:

After entering Inadequate Core Cooling EOP (1202.005),
RCS CET's indicate 900°F and RCS pressure is 1400 psig.

What is the appropriate action to be taken?

a. Bypass start interlocks and start all RCPs.

b. Restore RCP services and start one RCP/loop.

c. Bump RCPs until primary to secondary heat transfer is established even if RCP services are not available.

d. Depressurize the RCS by opening ERV and bypass ESAS if it has not already activated as RCS pressure drops below 1700 psig.

Answer:

b. Restore RCP services and start one RCP/loop.

Notes:

(a.) & (d.) are incorrect. These are entry actions for Region 4 not Region 3

(b.) is correct. 1202.005 step 8

(c.) is incorrect. RCPs are bumped in Region 2 only if RCP services are available.

References:

1202.005, Rev 004-00-0, Inadequate Core Cooling

History:

Developed for use in previous exam.
Used in 2001 SRO Exam.

QID: 0254 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

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

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

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

Description: Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps.

K/A Number: K6.02 **CFR Reference:** CFR: 41.7 / 45.7

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

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

Question:

Given:

- EFW started 10 minutes ago
- EFW pump P-7A speed is 900 RPM

Which of the following would cause these indications?

a. EFW Pump P-7A governor valve has lost power.

b. EFW Pump P-7A trip/throttle valve does not indicate full open.

c. EFW steam admission valve CV-2613 is closed.

d. EFW Pump P-7A governor valve has an oil leak.

Answer:

b. EFW Pump P-7A trip/throttle valve does not indicate full open.

Notes:

The trainee should conclude that EFW pump speed is below normal.
"a" is incorrect since the governor fails full open on a loss of power and this does not support the low speed condition.
"c" is incorrect since either steam admission valve will bring the pump up to full speed.
"d" is incorrect since a leak in the governor will only serve to reduce hydraulic pressure and increase speed.
"b" is the only correct answer since ramp initiate requires the trip/throttle valve to be full open and at leas on steam admission valve greater than 90% open.

References:

1106.006 Rev 060-00-0
STM1-27 Rev 2

History:

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

QID: 0267 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-MU **Objective:** 10 **Point Value:** 1

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

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

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

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

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2.5

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

Question:

The CBOR observes a change in seal injection flow rates and notes the following values:

"A" RCP	6.5 gpm
"B" RCP	15.0 gpm
"C" RCP	5.0 gpm
"D" RCP	6.0 gpm

Which of the following explains the seal injection flow indications?

a. Reactor Coolant Pump P-32B trip due to a motor fault.

b. Seal injection line break in the Upper North Piping Penetration Room .

c. "B" Reactor Coolant Pump seal cooler leak.

d. "B" seal injection flow transmitter failure.

Answer:

c. "B" Reactor Coolant Pump seal cooler leak.

Notes:

"a" is incorrect because flow from B RCP would drop.
"b" is incorrect because all seal injection flows would drop to zero.
"c" is correct, a seal cooler leak lowers the seal pressure and "B" seal injection flow raises. The others lower because total seal injection flow is maintained at setpoint.
"d" is incorrect because a transmitter failure will not lower the other seal injection flows.

References:

1203.039, Rev. 005-01-0, Excess RCS Leakage, pages 3,4
STM 1-03 rev. 8 ch.1 Reactor Coolant System, pages 34, 35, 36

History:

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

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0271 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-RMS **Objective:** 2 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 073 **System Title:** Process Radation Monitoring System (PRM)

Description: Knowledge of the effect of a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases.

K/A Number: K3.01 **CFR Reference:** CFR: 41.7 / 45.6

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	2.5
Group:	2	**SRO Imp:**	4.2	**SRO Select:**	Yes	**Taxonomy:**	K

Question:

Which of the following must be performed to release T-16A contents with the Liquid Radwaste Process Monitor (RI-4642) inoperable?

a. Chemistry personnel must estimate radiation level every four hours during the release.

b. A Waste Control Operator must independently verify release path alignment prior to release.

c. The release flow rate must be estimated at least once every three hours during the release.

d. Discharge Flume process monitor RI-3618 must be checked for operability.

Answer:

b. A Waste Control Operator must independently verify release path alignment prior to release.

Notes:

The requirements for release when the Liquid Radwaste Process Monitor is inoperable are
a. An independent verification of the release path by a person qualified as Waste Control Operator.
b. An independent sample and analysis of the tank contents
c. Computer input data independently verified.

"b" is the correct answer.

References:

1104.020 Rev 039-03-0, Clean Waste System Operation, pages 3

History:

Used in 1999 exam.
Direct from ExamBank, QID# 2765
Used in 2001 RO/SRO Exam.

QID: 0272 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

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

Section: 3.9 **Type:** Radioactivity Release

System Number: 071 **System Title:** Waste Gas Disposal System (WGDS)

Description: Knowledge of the purpose and function of major system components and controls.

K/A Number: 2.1.28 **CFR Reference:** CFR: 41.7

Tier:	2	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	2
Group:	1	**SRO Imp:**	3.3	**SRO Select:**	Yes	**Taxonomy:**	K

Question:

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

1. Nitrogen is added for dilution.
2. The Aux. Building Vent Header diverts to the Waste Gas Surge Tank.
3. The Waste Gas Decay Tank effluent control valve (CV-4820) shuts.
4. The Aux. Building Vent Header diverts to the Waste Gas Decay Tank in service.

a. 1 and 2

b. 2 and 3

c. 3 and 4

d. 1 and 4

Answer:

b. 2 and 3

Notes:

"b" is correct
"a", "c", and "d" are incorrect because adding nitrogen is a manual operation and the ABVH is diverted to the Waste Gas Surge Tank.

References:

1203.006, Rev. 007-02-0, Waste Gas Discharge Line Radiation page 1

History:

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

QID: 0274 **Rev:** 0 **Rev Date:** 9-2-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-RBS **Objective:** 5 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

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

Description: Ability to manually operate and/or monitor in the control room: Dampers in the CCS

K/A Number: A4.03 **CFR Reference:** CFR: 41.7 / 45.5 to 45.8

Tier: 2 **RO Imp:** 3.2 **RO Select:** Yes **Difficulty:** 1.5

Group: 1 **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** K

Question:

What would be the consequences if the Reactor Building Cooler Chilled Water Bypass Dampers remained latched after an ESAS actuation?

a. Damage to RB ventilation plenum from excessive pressure

b. Excessive heat load on the Chilled Water System

c. Inadequate air flow through the Service Water Cooling Coils

d. Excessive current on the cooling fan motors

Answer:

c. Inadequate air flow through the Service Water Cooling Coils

Notes:

"a" is incorrect, RB ventilation plenum has been analyzed for these conditions and it will withstand the pressures after an ESAS.
"b" is incorrect, the Chilled Water System is isolated on ESAS and therefore no additional heat load will be placed on it.
"d" is incorrect, the current on the motors is not a concern in this situation.
"c" is the correct answer, the bypass dampers drop to allow more flow through the Service Water coils by bypassing the Chilled Water coils and thus more cooling to the RB atmosphere.

References:

1104.033, Rev. 058-00-0, Reactor Building Ventilation, pages 3, 4

History:

Developed for 1999 exam.
Used in 2001 RO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

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:** CFR: 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:

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

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

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

b. P-4A and P-4B

c. P-4B and P-4C

d. P-4A and P-4C

Answer:

d. P-4A and P-4C

Notes:

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

References:

1203.012J, Rev. 034-00-0, Annunicator K11 Corrective Action, page 27
STM 1-42, Rev. 4, Ch. 3, Service and Auxiliary Cooling Water, page 13, 14

History:

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

QID: 0297 **Rev:** 0 **Rev Date:** 11/20/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-RO-TURB **Objective:** 9E **Point Value:** 1

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

System Number: 045 **System Title:** Main Turbine Generator (MT/G) System

Description: Ability to manually operate and/or monitor in the control room: Turbine valve indicators (throttle, governor, control stop, intercept) alarms, and annunciators.

K/A Number: A4.01 **CFR Reference:** 41.7 / 45.5 to 45.8

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

Group: 3 **SRO Imp:** 2.9 **SRO Select:** Yes **Taxonomy:** K

Question:

While performing the Overspeed Test during Turbine startup, the Operator mistakenly places the OPC switch in the OPC Test position instead of the Overspeed Test position.

What will this mistake cause?

a. Block the Overspeed Protection Controller's overspeed protection.

b. Close the Throttle and Governor Valves only.

c. Close the Governor Valves and the Reheat Intercept Valves.

d. Close all Throttle, Governor, Reheat Intercept, and Reheat Stop Valves.

Answer:

c. Close the Governor Valves and the Reheat Intercept Valves.

Notes:

Answer [c] is correct, the OPC Test switch simulates an OPC condition.
Answer [a] is incorrect since the Overspeed Protection Controller actuation is blocked when the the OPC test switch is placed in the Overspeed Test position.
Answer [b] is incorrect, these are the wrong valves.
Answer [d] is incorrect, this is essentially a Turbine Trip.

References:

1106.009, Rev. 029-03-0, Turbine Startup, Warmup, and Roll, page 28

History:

New question for use in 2001 RO/SRO Exam.

QID: 0309 **Rev:** 0 **Rev Date:** 9-5-99 **Source:** Direct **Originator:** R Cool

TUOI: ANO-1-LP-RO-ICS **Objective:** 40 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-Nuclear Instrumentation System

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

K/A Number: A2.01 **CFR Reference:** CFR: 41.5 / 43.5 / 45.3 / 45.5

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.1 **SRO Select:** No **Taxonomy:** An

Question:

Given:

- The plant is operating at 100% power.
- Loop "A" T-cold Narrow Range Temperature instrument fails HIGH.

If this instrument was hard selected by the SASS selector switch, what ICS HAND/AUTO stations should be placed in HAND?

a. Reactor Demand and both Feedwater Loop Demands.

b. SG/Rx Master and Reactor Demand.

c. SG/Rx Master and both Feedwater Loop Demands.

d. Both MFW Pumps and Turbine (EHC).

Answer:

a. Reactor Demand and both Feedwater Loop Demands.

Notes:

A cold leg temperature instrument failure causes the reactor demand signal to drive rods inward due to a high indicated Tave. Feedwater flows are changed to balance loop cold leg temperatures. Therefore reactor demand and feedwater loop demand stations must be taken to manual. "a" is the correct answer.
"b" is incorrect because feedwater is affected downstream of the SG/Rx Master.
"c" is incorrect because reactor demand is affected downstream of the SG/Rx Master.
"d" is incorrect because the turbine is not affected.

References:

1105.006 rev. 009-00-0, Reactor Coolant System NNI, page 5, 6,
1105.004 rev. 013-02-0, Intergrated Control System, page 8

History:

Used in 1999 exam.
Direct from ExamBank, QID# 2871
Used in 2001 RO Exam.

QID: 0312 **Rev:** 1 **Rev Date:** 11/16/00 **Source:** Modified **Originator:** J Cork

TUOI: ANO-1-LP-RO-SFC **Objective:** 2 **Point Value:** 1

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

System Number: 033 **System Title:** Spent Fuel Pool Cooling System

Description: Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling system will have on the following: Area ventilation systems.

K/A Number: K3.01 **CFR Reference:** CFR: 41.5 / 45.5

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

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

Question:

The WCO reports the Spent Fuel Pool level is +1.5 ft.

What problem could this level pose for Spent Fuel Pool operations or fuel handling in the SFP?

a. SFP minmum water temperature limit will be exceeded.

b. SFP ventilation ducts will be flooded.

c. Area dose rates will rise.

d. SFP must be sampled within 5 hours.

Answer:

b. SFP ventilation ducts will be flooded.

Notes:

Answer [b] is correct since normal level is 0 ft with a maximum allowable level of +1.0 ft which prevents water carryover into the ventilation ducts.
Answer [a] is incorrect because this answer is associated with SF cooling capacity which is largely unaffected by pool level.
Answer [c] is incorrect since this problem is associated with a low water level.
Answer [d] is incorrect but plausible since the time for sampling is correct but level is greater than maximum allowed.

References:

STM 1-7, Rev. 2 Ch. 1, Spent Fuel Cooling System, page 2

History:

Developed for 1999 exam.
Modified for 2001 RO/SRO Exam.

QID: 0315 **Rev:** 2 **Rev Date:** 11/16/00 **Source:** Modified **Originator:** R Soukup

TUOI: ANO-1-LP-AO-VAC **Objective:** 17 **Point Value:** 1

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

System Number: 055 **System Title:** Condenser Air Removal System

Description: Knowledge of CARS design feature(s) and/or interlock(s) which provide for the following: Turbine Startup.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	1.9	**RO Select:**	Yes	**Difficulty:**	2.5
Group:	2	**SRO Imp:**	2.3	**SRO Select:**	No	**Taxonomy:**	K

Question:

Given:

Plant is stablized at ~17% power in preps for placing the Main Turbine on line.
Condenser vacuum is 27.5 in Hg and slowly trending down.
CBOT is in the process of swapping from TV control to GV control.

What action would be taken when condenser vacuum reaches ~26.5 in Hg?

a. Continue with TV/GV transfer. No action required.

b. Trip the operating MFW Pump Turbine.

c. Trip the Main Turbine and lower power.

d. Adjust programmable alarm setpoint to ~25 in Hg.

Answer:

c. Trip the Main Turbine and lower power.

Notes:

Answer [c} is correct, a manual trip of the turbine is required when vacuum reaches 26.5 in Hg when turbine load is < 270 Mwe.
Answer [a] is incorrect, continued operation in this condition can lead to turbine blading damage.
Answer [b] is incorrect, this action is not taken until vacuum drops to ~5" Hg.
Answer [d] is incorrect, this action would be taken if operating >270 Mwe to alert operator of approaching trip criteria of 24.5 in Hg.

References:

1203.016 rev. 011-00-0, Loss of Condenser Vacuum, page 1

History:

Developed for 1999 exam.
Modified for 2001 RO exam.

QID: 0322 **Rev:** 0 **Rev Date:** 9-6-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-ICS **Objective:** 14 **Point Value:** 1

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

System Number: 045 **System Title:** Main Turbine Generator

Description: Knowledge of the MT/G system design feature(s) and/or interlock(s) which provide for the following: Automatic turbine runback.

K/A Number: K4.12 **CFR Reference:** CFR: 41.7

Tier: 2 **RO Imp:** 3.3 **RO Select:** Yes **Difficulty:** 2.5

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

Question:

A reactor coolant pump trip has caused a plant runback.

What ensures ICS maintains power steady (does not return to its previous load demand) when the runback is complete?

a. The Unit Master H/A station input tracks the Rate and Load Limited Megawatt demand signal.

b. The ICS runback demand signal is fed directly into the input of the Unit Master H/A station.

c. The input to the Unit Master H/A station is driven by cross limits to match the runback back demand signal.

d. The ICS runback signal will clear only when the Unit Master H/A station output equals actual generated megawatts.

Answer:

a. The Unit Master H/A station input tracks the Rate and Load Limited Megawatt demand signal.

Notes:

(a) is correct. The runback demand signal from the rate and load limit circuit is fed back into the input of the Unit Master to force the Unit Master to track the runback load demand signal.
(b) is incorrect because the runback limit is fed into the high load limit.
(c) is incorrect. While cross limits may come into effect during a runback, they do not drive the input of the Unit Master.
(d) is incorrect since the runback signal will clear when demand is less than the runback limit.

References:

STM1-64 Rev 6, Intergrated Control System, page 23, 24

History:

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

QID: 0325 **Rev:** 1 **Rev Date:** 11/16/00 **Source:** Modified **Originator:** J. Cork

TUOI: ANO-1-LP-RO-EOP01 **Objective:** 2 **Point Value:** 1

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

System Number: 011 **System Title:** Pressurizer Level Control

Description: ?Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Isolation of letdown.

K/A Number: A2.07 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier: 2 **RO Imp:** 3.0 **RO Select:** Yes **Difficulty:** 2.5

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

Question:

Following a reactor trip, what pressurizer level value (dropping) requires isolation of letdown per 1202.001?

a. 110 inches

b. 90 inches

c. 55 inches

d. 30 inches

Answer:

c. 55 inches

Notes:

Answer [c] is correct, this is the heater cutoff value and the value for isolation of letdown.
Answr [a] is incorrect, this is the upper limit of the normal control band prescribed in the Rx Trip EOP.
Answer [b] is incorrect, this is the lower limit of the normal control band prescribed in the Rx Trip EOP.
Answer [d] is incorrect, this setpoint is the off-scale low value and requires initiation of HPI to offset the loss of inventory and to attempt to sustain an adequate subcooling margin.

References:

1202.001, Rev 27, Reactor Trip, page 14, and 19

History:

Used in 1999 exam.
Modified from ExamBank, QID# 7742.
Modified for use in 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0328 **Rev:** 0 **Rev Date:** 9-6-99 **Source:** Direct **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-EOP01 **Objective:** 2 **Point Value:** 1

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

System Number: 029 **System Title:** Anticipated Transient Without Scram

Description: Knowledge of system setpoints/interlocks and automatic actions associated with EOP entry conditions.

K/A Number: 2.4.2 **CFR Reference:** 41.7 / 45.7 / 45.8

Tier:	1	**RO Imp:**	3.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.1	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

You are the CBOR and you observe the following indications:

"A" and "B" Main Feedwater Pumps are tripped
CRD groups 1, 2, 3, and 4 are at the out limit.
CRD groups 5, 6, and 7 are at the in limit.
NI-3 indicates 1 E-8 and lowering.

What action should be performed FIRST?

a. Depress the CRD Power Supply Breaker Trip Pushbuttons.

b. Dispatch an operator to open the CRD AC Power Supply Breakers.

c. Commence Emergency Boration per RT-12.

d. Manually insert CRD groups 1, 2, 3, and 4.

Answer:

a. Depress the CRD Power Supply Breaker Trip Pushbuttons.

Notes:

The conditions given are indicative of an ATWS since an anticipatory Rx Trip should have occurred (both MFW pumps tripped at 100% power) and a partial trip did occur (5,6,7 at in limit and Intermediate Range NI power decreasing), however all of the safety groups are at the out limit.

All of the answers given are correct contingency actions to a failure of the Reactor to trip, however, "a" is the preferred and most expedient action to drop the safety groups. Answers "b" and "d" are not performed unless "a" was unsuccessful.

References:

1202.001 Rev 27, Reactor Trip, page 2

History:

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

QID: 0329 **Rev:** 1 **Rev Date:** 11/7/00 **Source:** Modified **Originator:** D. Slusher

TUOI: ANO-1-LP-RO-NI **Objective:** 10 **Point Value:** 1

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

System Number: 033 **System Title:** Loss of Intermediate Range Nuclear Instrumentation

Description: Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Guidance contained in EOP for loss of intermediate range instrumentation.

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

Tier:	1	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	3.9	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question:

Given:

Plant startup in progress
NI501 at 9 x E4 cps
NI502 at 1 x E5 cps
NR502 is operable and at 5 x E-2% power
NI3 at 2 x E-10 amps
NI4 at 5 x E-10 amps
NI5 thru 8 at 0%

Subsequently NI3 fails low.

What action should be taken by control room operators?

a. Maintain flux level in the source range

b. Trip the reactor

c. Continue with startup

d. Stabilize power at 1 x E-8 amps

Answer:

c. Continue with startup

Notes:

The Intermediate Range NI values are on scale and represent one decade of overlap with the Source Range indications. Therefore with IR NI3 failed, the startup may continue as in answer (c) in accordance with 1203.021.
(a) is incorrect, this action would only be taken if if less than one decade of overlap existed.
(b) is incorrect, although this action would be taken IAW 1203.021 if no on scale flux indication existed.
(d) is incorrect, power can continue up to 10-8 amps.

References:

1203.021, Rev. 007-01-0, Loss of Neutron Flux Indication, page 5, 6

History:

Used in 1999 exam
Direct from ExamBank, QID# 3099 used in class exam
Modified for use in 2001 RO/SRO Exam.

QID: 0337 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** D Slusher

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

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

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

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

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

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

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

Question:

ESAS has actuated. LPI/HPI flows for the past ten minutes have been as follows:

"A" LPI flow--2900 gpm
"B" LPI flow--2850 gpm
"A" HPI pump flow throttled to 100 gpm through CV-1220
"C" HPI pump flow throttled to 100 gpm through CV-1285

An overcurrent has resulted in an A-3 bus lockout and A-1 to A-3 tie breaker A-309 trip. The operator should:

a. restore full HPI flow on "C" HPI pump.

b. close A-308 to power A-3 from #1 EDG.

c. energize bus B-5 from bus B-6

d. start P-36B to supply 100 gpm train through CV-1220.

Answer:

a. restore full HPI flow on "C" HPI pump.

Notes:

"a" is the only correct action with the loss of the A-3 bus and the "A" LPI pump. The criteria for throttling HPI is contingent upon both LPI pumps flow >2800 gpm OR one LPI pump >3200 gpm, therefore full HPI flow must be restored on the only running HPI pump. "b" and "c" are actions to restore proper electrical alignment but they do not address the immediate need of maintaining proper core cooling. "d" is incorrect since flow is only supplied at the 100 gpm rate.

References:

1202.010 Rev. 005-00-0, ESAS, page 7, 8

History:

Used in 1999 exam
Direct from ExamBank, QID# 4566 used in class exam
Used in 2001 RO/SRO Exam.

QID: 0347 **Rev:** 0 **Rev Date:** 9-7-99 **Source:** Direct **Originator:** G. Alden

TUOI: ANO-1-LP-RO-FH **Objective:** 19 **Point Value:** 1

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

System Number: A08 **System Title:** Refueling Canal Level Decrease

Description: Knowledge of the operational implications as they apply to the (Refueling Canal Level Decrease): Normal, abnormal, and emergency operating procedures associated with (Refueling Canal Level Decrease).

K/A Number: AK1.2 **CFR Reference:** 41.8 / 41.10 / 45.3

Tier: 1 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 2

Group: 3 **SRO Imp:** 4.0 **SRO Select:** Yes **Taxonomy:** K

Question:

The main fuel bridge has a spent fuel assembly in route to the RB upender when a seal plate NI cover failure occurs.
Water level in the canal is falling at two inches per minute.

The main fuel bridge operator should:

a. Continue to the upender and insert the assembly for transport to the SFP.

b. Leave the fuel assembly in the mast and evacuate the area.

c. Place the assembly in the fuel rack in the deep end of the canal.

d. Return the assembly to any available location in the reactor vessel.

Answer:

d. Return the assembly to any available location in the reactor vessel.

Notes:

In this scenario the fuel transfer canal level is decreasing rapidly and thus shielding for the spent fuel assembly will be decreasing rapidly and the fuel assembly must be placed in an area that will remain covered with water after the canal is drained.
Therefore, "d" is the only correct answer. "a" is incorrect as this is a time consuming maneuver and the transfer tube should be isolated anyway to prevent loosing level in the SFP. "b" is very incorrect since this will expose the assembly to atmosphere and dose rates will be lethal for quite some time. "c" is incorrect since the deep end will not contain enough water to keep the assembly covered.

References:

1203.042 Rev 005-00-0, Refueling Abnormal Operations, page 7

History:

Used in 1999 exam.
Direct from ExamBank, QID# 4282 used in class exam
Used in 2001 RO/SRO Exam.

QID: 0350 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** New **Originator:** S.PULLIN

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

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

System Number: 013 **System Title:** Engineered Safety Features Actuation System

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based ability on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument bus.

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

Tier: 2 **RO Imp:** 3.6 **RO Select:** No **Difficulty:** 3.5

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

Question:

Given, the plant is operating at 100% power.

ESAS Analog 2 RC pressure transmitter fails LOW due to loss of instrument power.

What operator action will allow continued plant operation at 100% power?

a. Initiate administrative controls to document and correct the failure.

b. Continued power operation is not allowed, plant shutdown is required.

c. Immediately trip one of the two remaining operable channels.

d. Test ES components associated with Analog Channel 2 within 24 hours.

Answer:

a. Initiate administrative controls to document and correct the failure.

Notes:

"a" is correct per Tech Spec table 3.5.1-1, Note 6.
"b" is incorrect, with only one inoperable channel, plant shutdown is not required.
"c" is incorrect, this action would result in ES actuation.
"d" is incorrect, testing of ES components is not required.

References:

Tech. Spec 3.5.1-1 table note 6

History:

Developed for 2001 SRO exam

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0363 **Rev:** 0 **Rev Date:** 11/6/00 **Source:** New **Originator:** S.Pullin

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

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

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

Description: Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Cooldown): Facility operating characteristics during transient conditions, including coolant chemistry and the effects
of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

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

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.4	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

Given:

- Natural circulation cooldown in progress
- CETs at 550°F
- Reactor vessel head temperatures at 614°F
- Pressurizer level = 150 inches, then makes step change to 180 inches
- RCS pressure at 1700 psig and slowly dropping

The required operator action is to pressurize the RCS slightly and reduce cooldown rate.

What is the reason for this action?

a. Reduce the tensile stress on the Reactor Vessel.

b. Prevent violation of Pressurizer cooldown rates.

c. Collapse a steam void in the Rx Vessel head.

d. Comply with SG tube to shell delta-T limits.

Answer:

c. Collapse a steam void in the Rx Vessel head.

Notes:

Answer (c) is correct since this action is due to a steam void in the upper head as evidenced by the sudden change in Pzr level with no RCS pressure increase.
Answer (a) would be the proper response for PTS concerns.
Answer (b) is not applicable, PZR cooldown rates have not been exceeded.
Answer (d) only applies when the OTSGs are not being used to cool the RCS.

References:

1203.013 (Rev. 016-02-0), Natural Circulation Cooldown, page 8

History:

Developed for 2001 RO/SRO Exam

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

TUOI: ANO-1-LP-RO-EOP06 **Objective:** 1 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Ability to determine and interpret the following as they apply to the SGTR: Existence of an S/G tube rupture and its potential consequences.

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

Tier: 1 **RO Imp:** 4.5 **RO Select:** Yes **Difficulty:** 4

Group: 2 **SRO Imp:** 4.8 **SRO Select:** Yes **Taxonomy:** An

Question:

After a reactor trip, the following indications are observed:

Makeup Tank level has lost 5 inches in the last 5 minutes
RB and Aux. Bldg. Sump levels are stable
"A" EFIC level is 35 rising and "A" MFW Flow is .1 mlb/hr
"B" EFIC level is 31 stable and "B" MFW Flow is .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 30 gpm, 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, Rev. 007-02-0, Tube Rupture, page 3

History:

Created for 2001 RO/SRO Exam.

QID: 0365 **Rev:** 0 **Rev Date:** 11/8/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-1-LP-RO-EOP04 **Objective:** 3 **Point Value:** 1

Section: **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Loss of Main Feedwater: HPI, under total feedwater loss conditions.

K/A Number: AA1.04 **CFR Reference:** 41.7 / 45.5 / 45.6

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

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

Question:

Unit One is operating normally with the following equipment OOS for maintenance:
P7A, Steam Driven EFW Pump
AACDG, Blackout Diesel Generator

A tornado has touched down in the switchyard causing a Degraded Power event.
EDG #1 trips on low lube oil pressure.
4160v bus A3 lockout occurs and cannot be reset.
RCS pressure has risen to 2450 psig and the ERV is open.

Which of the following actions is required for these conditions?

a. Close CV-1000, ERV Isolation valve.

b. Initiate HPI Cooling per RT-4.

c. Crosstie A3 and A4 buses to restore EFW.

d. Depressurize both SGs and feed with SW.

Answer:

B. Initiate HPI Cooling per RT-4.

Notes:

Answer [b] is correct. Without any source of feedwater to the SGs and RCS pressure at 2450 psig, HPI cooling is required to ensure adequate core cooling.
Answer [a] is incorrect since an open ERV is an essential component of HPI cooling.
Answer [c] is incorrect, A3 bus is locked out and cannot be restored quickly.
Answer [d] is incorrect as this is done only if HPI is not available or adequate to cool the core.

References:

1102.007, Rev. 005-01-0, Degraded Power, page 29, 30

History:

Created for 2001 RO/SRO Exam.

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

TUOI: ANO-1-LP-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: 3 **SRO Imp:** 3.9 **SRO Select:** Yes **Taxonomy:** K

Question:

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, Rev. 018-02-0, Integrated ES System Test, page 61

History:

Created for 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0367 **Rev:** 0 **Rev Date:** 11/8/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-1-LP-RO-AOP **Objective:** 10 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Need for area evacuation; check against existing limits.

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

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

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

Question:

You are assigned to move spent fuel for the Dry Fuel Storage Project.
A SFP bridge interlock fails and an assembly is damaged while you are moving it toward the cask loading pit.
The SFP area radiation monitor, RE-8009, begins alarming.

Which of the following is your required action for this event?

a. Place the spent fuel assembly back in its original location.

b. Notify the Radiation Protection Dept. immediately.

c. Initiate a local evacuation of the Spent Fuel Pool area.

d. Place the assembly in the cask loading pit.

Answer:

c. Initiate a local evacuation of the Spent Fuel Pool area.

Notes:

Answer [c] is correct per AOP 1203.042.
Answers [a] and [d] are incorrect, an ARM in alarm means the area should be evacuated until damage and radiation levels can be assessed.
Answer [b] is incorrect, this would take too much time.

References:

1203.042, Rev. 005-00-0, Refueling Abnormal Operations, page 3, 4

History:

Created for 2001 RO/SRO Exam.

QID: 0368 **Rev:** 0 **Rev Date:** 11/13/00 **Source:** Direct **Originator:** J.Cork

TUOI: ANO-1-LP-RO-EOP02 **Objective:** 5 **Point Value:** 1

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

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

Description: Knowledge of the reason for the following responses as they apply to the (Inadequate Subcooling Margin): RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and admendments are not violated.

K/A Number: EK3.4 **CFR Reference:** 41.5 / 41.10, 45.6, 45.13

Tier:	1	**RO Imp:**	3.2	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.5	**SRO Select:**	Yes	**Taxonomy:**	An

Question:

A reactor trip has occurred from 100% power.
One minute later the following conditions exist:

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

Which of the following operator actions will be performed?

a. Trip one (1) RCP in each loop.

b. Verify EFW flow to each Steam Generator is ~430 gpm.

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

d. Initiate 1202.001, Reactor Trip, and go to Overheating EOP.

Answer:

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

Notes:

Answer [c] is correct since subcooling margin is lost and the Reflux Boiling setpoint is required to be selected in this situation.
Answer [a] is incorrect, this would be done for loss of subcooling margin but only if >2 minutes had expired without tripping the RCPs.
Answer [b] is incorrect, this is done for loss of subcooling margin but only if one SG is available.
Answer [d] is incorrect, this would not be done since the entry condition for Overheating have not been met.

References:

1202.012, Rev. 004-01-0, Repetitive Tasks, RT-5

History:

Used in 2001 RO/SRO Exam, regular exambank QID 3030.

QID: 0369 **Rev:** 0 **Rev Date:** 11/13/00 **Source:** New **Originator:** R.Soukup

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

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to operate and/or monitor the following as they apply to the Dropped Control Rod: Controls and components necessary to recover rod.

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

Tier:	1	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	2	**SRO Imp:**	3.4	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question:

Given:

- Plant is at 38% power.
- ICS is in full automatic.
- Rod 7 in Group 6 has dropped.
- Annunciator K08-C2 "CONTROL ROD ASYMMETRIC" is in alarm.
- All actions in response to the dropped rod have been completed.

Which of the following actions must be performed FIRST to recover the dropped rod?

a. Depress FAULT RESET on the Diamond panel.

b. Transfer dropped rod to its normal power supply.

c. Latch the dropped rod using auxiliary power supply.

d. Pull the dropped rod's motor power fuses.

Answer:

c] Latch the dropped rod using auxiliary power supply.

Notes:

Answer [c] is correct since an individual rod must be recovered using the auxiliary power supply and relatched.
Answer [a] is incorrect, this is not done until the rod is realigned with its group.
Answer [b] is incorrect, the rod's group is left on the normal supply.
Answer [d] is incorrect, this is only done for a rod with a high stator temperature condition.

References:

1105.009. Rev. 016-02-0, CRD System Operating Procedure, page 30, 31

History:

Created for 2001 RO/SRO Exam.

QID: 0370 **Rev:** 0 **Rev Date:** 11/13/00 **Source:** Modified **Originator:** J.Cork

TUOI: ANO-1-LP-RO-AOP **Objective:** 2 **Point Value:** 1

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

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

Description: Knowledge of the reasons for the following responses as they apply to the Turbine Trip: Normal, abnormal and emergency operating procedures associated with Turbine Trip.

K/A Number: AK3.2 **CFR Reference:** 41.5 / 41.10 / 45.6 / 45.13

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

Question:

A plant power escalation is in progress at 41% power.
The following conditions are observed:

- Rapid rise in RCS temperature
- Rapid rise in RCS pressure
- Rapid rise in PZR level
- Megawatt output = zero (0)
- MSSV open alarm

No other annunciators in alarm except for those expected for the above conditions.

What procedure contains the required mitigating operator actions?

a. 1203.001, "ICS Abnormal Operating"

b. 1203.018, "Turbine Trip below 43% Power"

c. 1203.027, "Loss of Steam Generator Feed"

d. 1202.001, "Reactor Trip Procedure"

Answer:

b. 1203.018, "Turbine Trip below 43% Power"

Notes:

Answer [b] is correct, with MW output at zero and power at 41%, then 1203.018 should be used.
Answer [a] is incorrect, this would be consulted if there was an indication of an instrument failure.
Answer [c] is incorrect, a loss of SG feed could produce these indications but other annunciators such as "RX IS FW LIMITED", MFP trip, etc., would be in.
Answer [d] is incorrect, Reactor would not have tripped unless power was greater than or equal to 43%.

References:

1203.018, Rev. 012-03-0, Turbine Trip Below 43% Power, page 1

History:

Modified regular exambank QID #2786 for 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0371 **Rev:** 0 **Rev Date:** 11/13/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-RO-NNI **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 008 **System Title:** Pressurize Vapor Space Accident

Description: Ability to determine and interpret the following as they apply to the Pressurize Vapor Space Accident: Effects on indicated Pressurizer pressure and/or level of sensing line leakage.

K/A Number: AA2.27 **CFR Reference:** 43.5 / 45.13

Tier: 1 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 3

Group: 2 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** Ap

Question:

If a leak exists on the upper tap of a Pressurizer level transmitter sensing line,
causing a PZR steam space leak. Indicated PZR level will ___________
and actual PZR level will _________.

a. drop, drop

b. drop, rise

c. rise, drop

d. rise, rise

Answer:

d] rise, rise

Notes:

Answer [d] is correct since a leak on the upper tap will cause the differential pressure to decrease on the affected transmitter, thus causing indicated level to rise. Likewise a steam space leak will cause actual level to increase. Answers [a] thru [c] are combinations of the correct answer, and could be correct if the leak was elsewhere.

References:

1304.022, Rev. 023-00-0, Unit 1 Pressurizer Level & Temperature Channel Calibration, page 6, 7

History:

Created for 2001 RO/SRO Exam.
Regular exambank QID #5470 used as inspiration.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0372 **Rev:** 0 **Rev Date:** 11/14/00 **Source:** Modified **Originator:** J.Cork

TUOI: ANO-1-LP-RO-EOP02 **Objective:** 3 **Point Value:** 1

Section: 4.1 **Type:** Generic EPEs

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

Description: Knowledge of the reasons for the following responses as they apply to the Small Break LOCA: Actions contained in EOP for Small Break LOCA/leak.

K/A Number: EK3.21 **CFR Reference:** 41.5 / 41.10 / 45.6 /45.13

Tier:	1	**RO Imp:**	4.2	**RO Select:**	Yes	**Difficulty:**	4
Group:	2	**SRO Imp:**	4.5	**SRO Select:**	Yes	**Taxonomy:**	An

Question:

A LOCA is in progress concurrently with a Degraded Power condition.

- RCS pressure: 1000 psig
- RCS temperature: 535 degrees F
- HPI flows on C16: 120 gpm, 125 gpm, 115 gpm, 180 gpm
- P36B is inoperable due to maintenance.
- EDG #1 tripped and will not run.

Select the most appropriate action below for this situation:

a. Close the HPI valve with 180 gpm flow to isolate break.

b. Stop HPI pump P-36C to reduce break flow.

c. Throttle HPI valve with 180 gpm flow until flow is 145 gpm.

d. Close HPI recirc valve, CV-1300 or CV-1301.

Answer:

c. Throttle HPI valve with 180 gpm flow until flow is 145 gpm.

Notes:

Answer [c] is correct since only one HPI pump is in service and although SCM has not been established, HPI flow must be throttled due to possiblity of an HPI line break robbing flow.
Answer [a] is incorrect, although high HPI flow thru one nozzle would be indicative of a break in this line, the line should not be isolated.
Answer [b] is incorrect although stopping HPI pump will stop possible LOCA pathway on that train, it is incorrect since only one pump is running.
Answer [d] would be correct if ESAS had not actuated but it has.

References:

1202.012, Rev. 004-01-0, Repetitive Tasks, RT 3

History:

Modified regular exambank QID #3116 for use in 2001 RO/SRO Exam..

QID: 0373 **Rev:** 0 **Rev Date:** 11/14/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-RO-RCS **Objective:** 5 **Point Value:** 1

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

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

Description: Ability to monitor automatic operation of the CVCS: PZR level and pressure.

K/A Number: A3.10 **CFR Reference:** 41.7 / 45.5

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

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

Question:

Unit One is operating at 100% power.
Chemistry requests that Letdown flow be raised to maximum due to increasing RCS activity.

What is the effect on the Pressurizer level control system?

a. Makeup flow will rise to a higher steady state value.

b. Makeup flow will drop to a lower steady state value.

c. PZR level will drop to a lower steady state level.

d. PZR level will rise to a higher steady state level.

Answer:

a) Makeup flow will rise to a higher steady state value.

Notes:

Answer [a] is correct since an increase in letdown flow will reduce PZR level and CV-1235 will open to maintain PZR level at setpoint, therefore Makeup flow will increase.
Answer [b], [c], [d] are options that the candidate might choose if he cannot recall proper system response.

References:

1104.002, Rev. 053-02-0, Makeup and Purification System Operation, page 6

History:

Created for 2001 RO/SRO Exam.

QID: 0374 **Rev:** 0 **Rev Date:** 11/14/00 **Source:** New **Originator:** R.Soukup

TUOI: ANO-1-LP-RO-RBS **Objective:** 7 **Point Value:** 1

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

System Number: 013 **System Title:** ESFAS

Description: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ESFAS controls including: Containment pressure, temperature, and humidity.

K/A Number: A1.02 **CFR Reference:** 41.5 / 45.5

Tier: 2 **RO Imp:** 3.9 **RO Select:** Yes **Difficulty:** 3

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

Question:

A large break LOCA has ocurred causing the BWST to be emptied.
The shift to RB sump suction has been completed.

What short term effect will this have on RB pressure and temperature?

a. RB pressure will trend down and RB temperature will remain constant.

b. RB pressure will trend down and RB temperature will trend down.

c RB pressure will trend up and RB temperature will trend up.

d. RB pressure will remain constant and RB temperature will remain constant.

Answer:

c RB Pressure will trend up and RB Temperature will trend up.

Notes:

Answer [c] is correct due to the RB spray pumps now pumping hot RB sump water vs. the cooler BWST water.
Answer [a] is incorrect, RB sump temperature is hotter and initially cause pressure and temperature to rise.
Answer [b] is incorrect, this would be the trend if the RB Spray pumps continued to pump cool BWST water.
Answer [d] is incorrect, RB pressure and temperature will not remain constant in any case.

References:

1202.012, Rev. 004-01-0, Repetitive Tasks, RT15

History:

Created for 2001 RO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0375 **Rev:** 0 **Rev Date:** 11/14/00 **Source:** Modified **Originator:** J.Cork

TUOI: ANO-1-LP-RO-NNI **Objective:** 4 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 017 **System Title:** In-Core Temperature Monitor System

Description: Knowledge of the physical connections and/or cause- effect relationships between the ITM system and the following systems: RCS.

K/A Number: K1.02 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

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

Group: 1 **SRO Imp:** 3.5 **SRO Select:** No **Taxonomy:** C

Question:

With the plant operating at 100%, which condition would cause an "ICC Event Train A" or "B" alarm?

a. Core exit thermocouple (one) fails high.

b. Wide range RCS pressure instrument fails high.

c. Hot leg level transmitter fails high.

d. CET subcooling margin < 25 degrees F.

Answer:

d. CET subcooling margin < 25 degrees F.

Notes:

Answer [d] is correct, if CET subcooling margin was less than the minimum, then an ICC alarm would be generated.
Answer [a] is incorrect, while this seemingly would generate the alarm, the alarm uses an average CET vs. any CET.
Answer [c] is incorrect, while the hot leg level transmitters have an input to this alarm, a level transmitter failing high would be indicative of no voids and thus the ICC alarm would not be in alarm.
Answer [b] is incorrect, while wide range RCS pressure is compared to avg. CET temp to calculate SCM, a pressure transmitter failing high would provide more SCM, not less.

References:

1203.012J, Rev. 034-00-0, Annunciator K11 Corrective Action, p.2,3

History:

Modified regular exambank QID #3685 for use in 2001 RO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0376 **Rev:** 0 **Rev Date:** 11/15/00 **Source:** New **Originator:** J.Cork

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

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

System Number: 056 **System Title:** Condensate System

Description: Ability to manually operate and monitor in the control room: Condensate automatic makup valve controller.

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

Tier: 2 **RO Imp:** 1.7 **RO Select:** Yes **Difficulty:** 2

Group: 1 **SRO Imp:** 1.5 **SRO Select:** No **Taxonomy:** C

Question:

Why would the Condensate Makeup valve, CV-2873, require manual operation during condenser tube cleaning in a condenser E-11B waterbox?

a. Differential pressure between the condensers could cause CV-2873 to open.

b. The level switch that controls CV-2873 is on E-11B and could be isolated.

c. Differential pressure between the condenser could prevent CV-2873 from opening.

d. Differential pressure could cause reverse flow of condensate into Condensate tank, T-41.

Answer:

a) Differential pressure between the condensers could cause CV-2873 to "fail" open.

Notes:

Answer [a] is correct since an isolation of an E-11B waterbox could cause E-11B to have a higher pressure than E-11A causing erroneous low level indication on E-11B and thus CV-2873 would be open continuously.
Answer [b] is incorrect, although CV-2873's level switch is on E-11B, it would not be isolated.
Answer [c] is incorrect, the differential pressure would cause a high level indication only if a E-11A waterbox was isolated.
Answer [d] is incorrect, even with a waterbox isolated the condenser pressure would never drop low enough for reverse flow into T-41 to occur.

References:

1104.008, Rev. 020-00-0, Circulating Water & Water Box Vacuum System, p.15
STM 1-20, Rev. 3, Condensate System, p.9

History:

Created for 2001 RO Exam.

QID: 0377 **Rev:** 0 **Rev Date:** 11/15/00 **Source:** Modified **Originator:** J.Cork

TUOI: ANO-1-LP-RO-EOP04 **Objective:** 3 **Point Value:** 1

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

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

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use
procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feeding a dry S/G.

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

Tier:	2	**RO Imp:**	2.9	**RO Select:**	Yes	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.4	**SRO Select:**	No	**Taxonomy:**	Ap

Question:

Given:

- RCS pressure 2350 decreasing
- RCS temperature 589 °F and slowly increasing
- ERV opened by CBOR
- "A" EFIC low range level = 18" and dropping slowly
- "B" EFIC low range level = 15" and steady
- HPI is in service
- EFW is NOT available

Select the most correct statement from the following:

a. Neither OTSG may be fed using MFW.

b. "A" OTSG may be fed using MFW.

c. "B" OTSG may be fed using MFW.

d. Both OTSGs may be fed using MFW.

Answer:

b. "A" OTSG may be fed using MFW.

Notes:

Answer [b] is correct since A OTSG level is 18" (<20" dry criteria) but is still dropping and is therefore NOT "dry." Answer [a], [c], and [d] are incorrect and any could be chosen if candidate cannot determine what event is occurring and/or cannot recall "dry" OTSG criteria.

References:

1202.004, Rev. 4, Overheating, p.3

History:

Modified regular exambank QID #1323 for use in 2001 RO Exam.

QID: 0378 **Rev:** 0 **Rev Date:** 11/15/00 **Source:** New **Originator:** J.Cork

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

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

System Number: 061 **System Title:** Auxiliary/Emergency Feedwater (AFW) System

Description: Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: Main steam system.

K/A Number: K1.03 **CFR Reference:** 41.2 to 41.9 / 45.7 to 45.8

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

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

Question:

Both Steam Admission valves, CV-2613 and CV-2663, to the Turbine Driven EFW Pump, P-7A, have solenoid operated bypass valves, SV-2613 and SV-2663.

What is the purpose of these solenoid bypass valves?

a. They open first to drain moisture from the steam lines thereby preventing an overspeed trip of P-7A.

b. They open first to allow governor oil pressure to build up prior to opening the larger valves.

c. They open to decrease the differential pressure across the steam admission valves.

d. They open to ramp turbine speed up to 3700 rpm prior to opening of steam admission valves.

Answer:

b. They open first to allow governor oil pressure to build up prior to opening the larger valves.

Notes:

Answer [b] is correct, the solenoids bypass the steam admission valves to spin the turbine via small diameter steam lines. The governor will then build up oil pressure and start to close prior to opening the larger diameter lines isolated by the steam admission valves.
Answer [a] is incorrect, although moisture in the steam lines definitely could cause an overspeed trip, there is a trap installed for this purpose.
Answer [c] is incorrect, this is a purpose of small bypass valves around larger valves but not in this case.
Answer [d] is incorrect, the turbine is ramped up to 3700 rpm but this is done by the governor.

References:

1106.006, Rev. 060-00-0, Emergency Feedwater Pump Operation, p.72
STM 1-27, Rev. 3, Emergency Feedwater System, p.14&22

History:

Created for use in 2001 RO/SRO Exam.

QID: 0379 **Rev:** 0 **Rev Date:** 11/15/00 **Source:** Modified **Originator:** J.Cork

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

Section: 3.7 **Type:** Instrumentation

System Number: 072 **System Title:** Area Radiation Monitoring (ARM) System

Description: Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments.

K/A Number: A4.01 **CFR Reference:** 41.7 / 45.5 to 45.8

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

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

Question:

During your performance of 1305.001, Supplement 6, Area Radiation Monitor Monthly Alarm Check, you discover Relay Room Area Monitor, RI-8002, high alarm setpoint is greater than the maximum allowable value.

What are the required actions?

a. Record the value found, notify the CRS, and initiate a Condition Report.

b. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

c. Record the value found, then have I&C make the required adjustment under a "blanket" MAI.

d. Record the value found and continue, nothing else need be done since RI-8002 is not a Tech Spec required monitor.

Answer:

b. Adjust the setpoint to less than or equal to max high alarm setpoint before recording the As-Left Setpoint.

Notes:

Answer [b] is correct per the procedure.
Answer [a] would be correct for discrepancies not governed by a procedural response.
Answer [c] is how this was handled in the past.
Answer [d] is how an incompetent operator might proceed.

References:

1305.001, Rev. 014-02-0, Radiation Monitoring System Check and Test, p.37, 40-43

History:

Modified regular exambank QID #2645 for use in 2001 RO Exam.

QID: 0380 **Rev:** 0 **Rev Date:** 11/15/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-RO-TS **Objective:** 4 **Point Value:** 1

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

System Number: 006 **System Title:** Emergency Core Cooling System (ECCS)

Description: Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: HPI/LPI systems (mode change).

K/A Number: K6.19 **CFR Reference:** 41.7 / 45.7

Tier: 2 **RO Imp:** 3.7 **RO Select:** Yes **Difficulty:** 4

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

Question:

Following a refueling outage, a plant "Heatup to 800 psig" is in progress.
RCS temperature is 290°F.
P-36B is out of service for maintenance.
P-36A is in ES standby and P-36C is operable.

After reviewing the surveillance data for P-36C, System Engineering determines that P-36C is inoperable.

Which of the following is allowable in accordance with Unit One Tech Specs?

a. Heatup may continue, two HPI pumps are required and two are operable.

b. Heatup should be stopped and RCS cooled down to < 200°F.

c. Heatup may continue up to, but not to exceed, an RCS temp of 350°F.

d. Heatup should be stopped and RCS placed in a condition NOT requiring CNTMT integrity.

Answer:

c. Heatup may continue up to, but not to exceed, an RCS temp of 350°F.

Notes:

Answer [c] is correct, two HPI pumps are required to be operable with RCS temp > 350°F.
Answer [a] is incorrect, although two HPI pumps are indeed operable, two HPI pumps must be operable from independent buses.
Answer [b] is incorrect, the RCS does not have to be cooled down, although most ECCS equipment must be operable prior to exceeding 200°F.
Answer [d] is incorrect, HPI is not required for CNTMT integrity.

References:

Unit One Technical Specifications 3.3.2

History:

Created for 2001 RO Exam.

QID: 0381 **Rev:** 0 **Rev Date:** 11/16/00 **Source:** Direct **Originator:** J.Cork

TUOI: ANO-1-LP-RO-CRD **Objective:** 22 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 014 **System Title:** Rod Position Indication System (RPIS)

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

K/A Number: K1.01 **CFR Reference:** 41.3 to 41.9 / 45.7 to 45.8

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

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

Question:

The CONTROL ROD ASYMMETRIC annunciator (K08-C2) in alarm indicates:

a. a rod is greater than 7 inches from its group average as measured by Relative Position Indication (RPI).

b. a rod is greater than 7 inches from its group average as measured by Absolute Position Indication (API).

c. a rod is greater than 9 inches from its group average as measured by Relative Position Indication (RPI).

d. a rod is greater than 9 inches from its group average as measured by Absolute Position Indication (API).

Answer:

b. a rod is greater than 7 inches from its group average as measured by Absolute Position Indication (API).

Notes:

Answer [b] is correct since API is used to generate this alarm and this alarm does NOT indicate a runback.
Answer [a] is incorrect, the setpoint is correct but RPI is not utilized to generate this alarm.
Answer [c] is incorrect, this setpoint is for a plant runback and RPI is not utilized to generate this alarm.
Answer [d] is incorrect, although API is the system that generates this alarm, the >9" setpoint is for a plant runback.

References:

1203.012G, Rev. 032-01-0, Annunciator K08 Corrective Action, p.12
1203.003, Rev. 019, pc-1, Control Rod Drive Malfunction, p.2

History:

Direct from regular exambank QID #5339 for use in 2001 RO/SRO Exam.

QID: 0382 **Rev:** 0 **Rev Date:** 11/16/00 **Source:** New **Originator:** R.Soukup

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

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray System

Description: Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS.

K/A Number: K3.01 **CFR Reference:** 41.7 / 45.6

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

Group: 2 **SRO Imp:** 4.1 **SRO Select:** No **Taxonomy:** C

Question:

During post-LOCA conditions in the Reactor Building, what percentage of containment cooling is provided with a failure of one RB Spray pump with no other failures?

a. 100% Cooling

b. 150% Cooling

c. 200% Cooling

d. 250% Cooling

Answer:

b. 150% Cooling

Notes:

Answer [b] is correct, both trains of CNTMT coolers will provide 100% cooling and one train of RB Spray provides 50% cooling capacity.
Answer [a] is incorrect, the combination of CNTMT coolers provides 100% cooling but this answer doesn't consider the cooling capability of the RB Spray pump.
Answer [c] is incorrect, this percentage is based on BOTH RB Spray pumps and both trains of CNTMT coolers.
Answer [d] is incorrect, this percentage is not achievable.

References:

STM 1-8, Rev. 5, Reactor Building Spray, p.10

History:

Created for 2001 RO/SRO Exam.

QID: 0384 **Rev:** 0 **Rev Date:** 11/17/00 **Source:** New **Originator:** R.Soukup

TUOI: ANO-1-LP-RO-ELECD **Objective:** 14.f **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 063 **System Title:** DC Electrical Distribution

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the DC Electrical Systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds.

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

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

Group: 2 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** K

Question:

Plant operating at 100% power.
Annunciator K01-D7, DO1 TROUBLE, is in alarm.
The AO reports the local trouble annunciator indicates a DC ground.

How would you determine which component in the DC system is grounded?

a. Electrical maintenance support is required to determine ground location.

b. Check ground indication lamps on the DC distribution panels.

c. At local panels, push TEST pushbutton, to determine grounded component .

d. Transfer D11 to D21 and see if ground is still present on D01.

Answer:

a. Electrical maintenance support is required to determine ground location.

Notes:

Answer [a] is correct in accordance with 1107.004.
Answer [b] is incorrect, individual ground detection lights are not available on DC distribution panels.
Answer [c] is incorrect, ground detection lights at local panel only indicate positive or negative ground, not the component.
Answer [d] is incorrect, transferring D11 to D21 is not allowed at power.

References:

1107.004, Rev. 011-03-0, Battery And 125V DC Distribution, p.17
1203.012A, Rev. 033-02-0, Annunciator K01 Corrective Action, p.49, 163, 166

History:

New for 2001 RO/SRO Exam.

QID: 0385 **Rev:** 0 **Rev Date:** 11/17/00 **Source:** New **Originator:** R Soukup

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

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

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

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

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

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

Group: 3 **SRO Imp:** 3.4 **SRO Select:** Yes **Taxonomy:** K

Question:

Per Technical Specifications boron concentration can not be reduced unless what MINIMUM conditions exist?

a. At least 1 RCP and its associated generator are available for heat removal.

b. All four RCP's available with at least 2 RCP's in service.

c. Either a RCP or DHR pump in service circulating reactor coolant.

d. Both RBS pumps available for emergency makeup.

Answer:

c. Either a RCP or DHR pump in service circulating reactor coolant.

Notes:

Answer [c] is correct, must have either a RCP or DHR pump in service for mixing during boron concentration changes.
Answer [a] is incorrect, does not satisfy the spec when only available.
Answer [b] is incorrect, only 1 RCP required to be in service not 2 and all available not required.
Answer [d] is incorrect, RBS pumps not used to reciculate the RCS.

References:

Technical Specification 3.1.1.1.B

History:

New for 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0386 **Rev:** 0 **Rev Date:** 11/20/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-1-LP-RO-AOP **Objective:** 4 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

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

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

K/A Number: A2.03 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.13

Tier:	2	**RO Imp:**	3.6	**RO Select:**	Yes	**Difficulty:**	3
Group:	3	**SRO Imp:**	3.9	**SRO Select:**	No	**Taxonomy:**	C

Question:

Immediately following an "A" MFP trip at 100% power, the following indications are observed:

- Both MFW Block valves open
- RCS pressure 2155 and rising
- PZR Spray valve, CV-1008, closed
- MFW Cross-Over valve, CV-2827, going open
- "REACTOR IS FW LIMITED" K07-C1 in alarm
- Reactor power is 90% and dropping

What operator actions should be taken in addition to the automatic actions occurring?

a. Manually close MFW Block valves.

b. Initiate EFW on both trains.

c. Open PZR spray valve, CV-1008, in manual.

d. Take manual control of Main Turbine.

Answer:

c. Open PZR spray valve, CV-1008, in manual.

Notes:

Answer [c] is correct, with a MFP trip >80% power, the PZR spray valve is biased open with a 100 reduction in setpoint, therefore it should be open from 2080 and closes at 2030.
Answer [a] is incorrect, MFW Block valves will not start going closed until power is <80%.
Answer [b] is incorrect, no EFW setpoints have been met.
Answer [d] is incorrect, this is not necessary for this runback.

References:

1203.015, Rev. 010-01-0, Pressurizer Systems Failure, p. 11, 12
1203.027, Rev. 010-00-0, Loss of Steam Generator Feed, p.1,2

History:

New for 2001 RO Exam.

QID: 0387 **Rev:** 0 **Rev Date:** 11/20/00 **Source:** New **Originator:** J.Cork

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

Section: 3.5 **Type:** Containment Integrity

System Number: 028 **System Title:** Hydrogen Recombiner and Purge Control System HRPS

Description: Knowledge of the operations implications of the following concepts as they apply to HRPS: sources of hydrogen within containment.

K/A Number: K5.03 **CFR Reference:** 41.5 / 45.7

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

Group: 3 **SRO Imp:** 3.6 **SRO Select:** Yes **Taxonomy:** K

Question:

Which one of the following is NOT a source of hydrogen production in the Reactor Building after a LOCA?

a. Metal-to-water reaction (zirconium steam/water)

b. Corrosion of aluminum

c. Sodium hydroxide/boric acid reaction with carbon steel

d. RCS water (hydrogen coming out of solution)

Answer:

c. Sodium hydroxide/boric acid reaction with carbon steel

Notes:

Answer [c] is correct, the boric acid/carbon steel reaction is a corrosion concern not a major source of hydrogen. Answers [a], [b], [d] are incorrect, because they are all sources of hydrogen production.

References:

STM 1-9, Rev. 3, ch. 2, Reactor Building Ventilation, p.11

History:

Direct from regular exambank QID#1067, used in 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0388 **Rev:** 0 **Rev Date:** 11/20/00 **Source:** New **Originator:** S. Pullin

TUOI: ANO-1-LP-RO-ESAS **Objective:** 5 **Point Value:** 1

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

System Number: 076 **System Title:** Service Water System

Description: Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Conditions initiating automatic closure of closed cooling water auxiliary building header supply and return valvels.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier: 2 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3

Group: 3 **SRO Imp:** 2.9 **SRO Select:** No **Taxonomy:** C

Question:

ESAS has actuated on RB pressure alone.
The CRS is in the Overcooling EOP.
Subcooling Margin is adequate.

You are directed to trip all running RCP's.

Why is this action being performed?

a. No cooling to the RCP motor and seal coolers.

b. Loss of subcooling margin is imminent.

c. RCP seal bleedoff path is isolated.

d. RCP seal injection is isolated.

Answer:

a. No cooling to the RCP motor and seal coolers.

Notes:

Answer [a] is correct since ESAS channels1-6 will actuate on RB pressure only and will isolate SW to ICW coolers and ICW to RCPs.
Answer [b] is incorrect, this is an overcooling event and SCM will be substantial.
Answer [c] is incorrect, RCP seal bleedoff is on alternate path and not isolated.
Answer [d] is incorrect, RCP seal injection does not close until flow <26 gpm and isn't closed by ESAS.

References:

1202.012, Rev. 004-01-0, Repetitive Tasks, RT-10

History:

New for 2001 RO Exam.

QID: 0389 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** S.Pullin

TUOI: ANO-S-LP-RO-PRCON **Objective:** 8 **Point Value:** 1

Section: 2 **Type:** Generic

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to obtain and verify controlled procedure copy.

K/A Number: 2.1.21 **CFR Reference:** 45.10 / 45.13

Tier: 3 **RO Imp:** 3.1 **RO Select:** Yes **Difficulty:** 2

Group: **SRO Imp:** 3.2 **SRO Select:** No **Taxonomy:** K

Question:

A job is in progress that will last for several weeks.
The procedure has been verified at the start of the job.
A pre-job brief has been completed for all participants.

How often should the procedure for this job be verified to be current?

a. Every 7 days.

b. Once every 24 hours.

c. Only prior to the start of the job.

d. Every 14 days.

Answer:

a. Every 7 days.

Notes:

Answer [a] is correct IAW 1000.006, all other choices are familiar frequencies of tasks.

References:

1000.006, Rev. 048-00-0, Procedure Control, step 12.8, p.32

History:

Modified regular exambank QID# 6054 for use in 2001 RO Exam.

QID: 0390 **Rev:** 1 **Rev Date:** 12/7/00 **Source:** Modified **Originator:** S. Pullin

TUOI: ANO-1-LP-RO-CRD **Objective:** 5 **Point Value:** 1

Section: 2.0 **Type:** Generic K & A's

System Number: 2.2 **System Title:** Control Rod Drive

Description: Knowledge of Control Rod programming.

K/A Number: 2.2.33 **CFR Reference:** 43.6

Tier:	3	**RO Imp:**	2.5	**RO Select:**	Yes	**Difficulty:**	3
Group:	G	**SRO Imp:**	2.9	**SRO Select:**	No	**Taxonomy:**	C

Question:

A startup is in progress.
All safety groups are 100% withdrawn.
The group 6 rods' RPI were not re-zeroed prior to startup.
The CBOR is continuing with the startup with group 5 rods.

Which of the following results?

a. Auto inhibit.

b. Assymetric rod alarm.

c. Out inhibit.

d. Sequence inhibit.

Answer:

d. Sequence inhibit.

Notes:

Answer [d] is correct, input to sequence inhibit is from RPI.
Answer [a] is incorrect, input from safety group out limit.
Answer [b] is incorrect, input from API (reed switches).
Answer [c] is incorrect, input from High SUR and API.

References:

1105.009 Rev 016-02-0, CRD Operating Procedure, p.2,8,9,19

History:

Modified regular exambank QID # 3153 for use in 2001 RO exam.

QID: 0391 **Rev:** 0 **Rev Date:** 11/20/00 **Source:** New **Originator:** R. Soukup

TUOI: ANO-1-LP-SRO-RAD **Objective:** 4 **Point Value:** 1

Section: 2.0 **Type:** Generic K & A's

System Number: 2.3 **System Title:** Radiation Control

Description: Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

K/A Number: 2.3.4 **CFR Reference:** 43.4 / 45.10

Tier: 3 **RO Imp:** 2.5 **RO Select:** Yes **Difficulty:** 3.5

Group: G **SRO Imp:** 3.1 **SRO Select:** Yes **Taxonomy:** Ap

Question:

A worker arrives on site with 2.5 Rem accumulative dose for the calendar year.
The worker's NRC form 4 is on file.
The worker's expected exposure will be 1.8 Rem for his assigned job.

Whose authorization is required to extend the worker's TEDE exposure limit?

a. The worker's Supervisor, Radiation Protection Manager, and General Manager, Plant Operations.

b. The worker's Supervisor and Radiation Protection Manager.

c. The worker's Supervisor, Radiation Protection Manager, General Manager, Plant Operations and Vice President, Operations.

d. This exposure limit can not be authorized per ANO Admin Exposure Limits.

Answer:

c. The worker's Supervisor, Radiation Protection Manager, General Manager, Plant Operations and Vice President, Operations.

Notes:

Answer [c] is correct IAW 1012.021 for doses >4 R but <4.5 R.
Answer [a] is incorrect, this is the authorization required for doses >3 R but <4 R.
Answer [b] is incorrect, this is the authorization required for doses >2 R but <3 R.
Answer [d] is incorrect, this is the authorization required for doses >4.5 R.

References:

1012.021, Rev. 004-01-0, Exposure Limits and Controls, p.7,9

History:

New question created for 2001 RO/SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0392 **Rev:** 0 **Rev Date:** 11/20/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-RO-FPS **Objective:** 10 **Point Value:** 1

Section: 2 **Type:** Generic K & A's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of fire protection procedures.

K/A Number: 2.4.25 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 2.9 **RO Select:** Yes **Difficulty:** 2

Group: G **SRO Imp:** 3.4 **SRO Select:** No **Taxonomy:** K

Question:

Who is the final authority on evaluating the fire loading of a safety related area for transient and insitu combustibles?

a. Safety Coordinator

b. Licensing Safety Engineer

c. Fire Barrier Watch Supervisor

d. Fire Protection Engineer

Answer:

d) Fire Protection Engineer

Notes:

Answer [d] is correct per 1000.152, all others have assigned responsibilities with fire protection or safety but not this one.

References:

1000.152, Rev. 003-01-0, Unit One and Two Fire Protection System Specifications, p.6

History:

New question created for 2001 RO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0393 **Rev:** 0 **Rev Date:** 11/20/0 **Source:** New **Originator:** R.Soukup

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

Section: 2 **Type:** Generic K & A's

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of operator response to loss of all annunciators.

K/A Number: 2.4.32 **CFR Reference:** 41.10 / 43.5 / 45.13

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

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

Question:

Both AC and DC "Power Available" lamps have gone out for all Control Room annunciator panels.

Which of the following actions should be taken?

a. Trip the reactor and enter 1202.001, Reactor Trip.

b. Commence power reduction per 1203.045, Rapid Plant Shutdown.

c. Commence normal plant shutdown per 1102.016, Power Reduction and Plant Shutdown.

d. Notify the Shift Manager to implement 1903.010, Emergency Action Level Classification.

Answer:

d. Notify the Shift Manager to implement 1903.010, Emergency Action Level Classification.

Notes:

Answer [d] is correct, power should be maintained steady and SM should consult 1903.010.
Answer [a] is incorrect, annunciators are vital when verifying plant conditions following a Rx trip.
Answers [b] and [c] are incorrect, steady state power should be maintained while annunciators are inoperable.

References:

1203.043, Rev. 001-00-0, Loss of Control Room Annunciators, p.1

History:

New question created for 2001 RO/SRO Exam.

QID: 0394 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Modified **Originator:** S.PULLIN

TUOI: AA-51001-001 **Objective:** 9.6 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to apply Technical Specifications for a system.

K/A Number: 2.1.12 **CFR Reference:** 43.2/43.5/45.3

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

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

Question:

Given:

- Rod 4 in Group 5 is inoperable, it did not pass rod drop time.
- Subsequently Rod 7 in Group 7 is found >9" from its group averrage.

What actions are required by Technical Specifications?

a. Boron must be added to equal the worth of both rods.

b. Power must be reduced to 60% of the maximum allowable power level for the number of RCP's running.

c. The reactor must be brought to Hot Standby within one hour unless 1% availble shutdown margin can be verified.

d. Plant shutdown is required since operation with more than one inoperable rod is not allowed.

Answer:

d. Plant shutdown is required since operation with more than one inoperable rod is not allowed.

Notes:

Answer [d] is correct, operation is limited to only 1 inoperable rod per Technical Specifications 3.0.3.
Answer [a] is incorrect, this action does not comply with T.S. for inoperable rods.
Answer [b] is incorrect, this action does not comply with T.S. for inoperable rods.
Answer [c] is incorrect, this action does not comply with T.S. for inoperable rods.

References:

Technical Specifications 3.5.2.2 & 3.0.3.

History:

Modified QID 1823 for use in 2001 RO/ SRO exam.

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

TUOI: ANO-1-LP-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:

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

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

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

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

c. To maintain pressurizer heaters off during testing.

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

Answer:

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

Notes:

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

References:

1105.006, Rev. 009-00-0, Reactor Coolant System NNI, page 12, 13
STM 1-69 rev. 4, Non Nuclear Instrumentation system, page 14, 15

History:

Direct from regular exambank QID#5545 for 2001 RO/SRO Exam.

QID: 0396 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-1-LP-RO-ICS **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 015 **System Title:** Reactor Coolant Pump (RCP) Malfunctions

Description: Ability to operate and/or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCS flow.

K/A Number: AA1.05 **CFR Reference:** 41.7 / 45.5 / 45.6

Tier: 1 **RO Imp:** 3.8 **RO Select:** Yes **Difficulty:** 3

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

Question:

The plant is steady state at 70% power per Dispatcher request.
Subsequently, you observe the following indications:
"A" MFW flow 4.7 e 6 lbm/hr
"B" MFW flow 2.3 e 6 lbm/hr

What event would cause this MFW flow discrepancy?

a. "B" MFW pump trip

b. "A" T cold instrument failed high

c. "A" RCP trip

d. "D" RCP trip

Answer:

c. "A" RCP trip

Notes:

Answer [c] is correct, a RCP trip caused FW to re-ratio with the highest flow in the opposite loop.
Answer [a] is incorrect, a MFW trip would cause FW flow to decrease to both SGs.
Answer [b] is incorrect, this would cause FW to decrease to the "A" SG and increase to the "B" SG.
Answer [d] is incorrect, this would cause FW to decrease to the "A" SG and increase to the "B" SG.

References:

STM 1-64 rev. 6, Intergrated Control System, page 34, 35

History:

New question created for 2001 RO/SRO Exam.

QID: 0397 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-RO-NOP **Objective:** 4 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

System Number: 001 **System Title:** Continuous Rod Withdrawal

Description: Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place.

K/A Number: AA2.03 **CFR Reference:** 43.5 / 45.13

Tier:	1	**RO Imp:**	4.5	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	4.8	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

During approach to criticality the control rods continue to withdraw after the manual command switch has been released.

The following indications are observed:

- Sustained SUR of 2 DPM
- Continued rod outward motion without a command.

What actions should you take?

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

b. Commence Emergency Boration per RT-12.

c. Swap controlling rod group to the Auxiliary Power supply.

d. Select "JOG" on rod speed selector switch.

Answer:

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

Notes:

Answer [a] is correct, the Source Range rod hold on high SUR of 2 DPM should have stopped outward rod motion. A system failure has occurred, control of reactivity has been lost, and reactor should be tripped.
Answer [b] is incorrect, this will add negative reactivity but it is also slow.
Answers [c] and [d] are incorrect, attempts to stop rod motion with these actions will not work.

References:

1102.008, Rev. 018-00-0, Approach to Criticality, page 4, 11

History:

New created for 2001 SRO Exam.

QID: 0398 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-SRO-ADMIN **Objective:** 6 **Point Value:** 1

Section: 4.2 **Type:** Generic APEs

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

Description: Ability to determine and interpret the following as they apply to the Plant Fire On Site: Requirements for establishing a fire watch.

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

Tier: 1 **RO Imp:** 2.9 **RO Select:** No **Difficulty:** 3

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

Question:

A fire in the T-57A Emergency Diesel Fuel Oil Storage vault has been in progress but is now extinguished. The intensity of the fire warped the water tight door and destroyed the smoke detection in the room.

What are the minimum compensatory measures required by 1000.152 for this area?

a) Restore the inoperable door and detection to operable status within 14 days or initiate a condition report.

b) Within one hour establish an hourly fire watch since flame detection for this room is still operable.

c) Within one hour establish a continuous fire watch with standby suppression equipment.

d) Immediately test T-57B vault's fire detection and suppression equipment to prove operability.

Answer:

c) Within one hour establish a continuous fire watch with standby suppression equipment.

Notes:

Answer [c] is correct, the fire barrier is inoperable and no detection is available on either side of the door.
Answer [a] is incorrect, although this is similar to other 1000.152 actions, it is incorrect for this scenario.
Answer [b] is incorrect, no flame detection exists in the EDG F.O. vaults, so no detection exists and a continuous watch is required.
Answer [d] is incorrect, this is similar to Tech Spec action statements but is not applicable to Fire Systems.

References:

1000.152, Rev. 003-01-0, Unit 1 & 2 Fire Protection System Specifications, pages 11, 12, 21, 22

History:

Created for 2001 SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0399 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** New **Originator:** R.Soukup

TUOI: ANO-1-LP-RO-EOP06 **Objective:** 4 **Point Value:** 1

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

System Number: E03 **System Title:** Loss of Subcooling Margin

Description: Knowledge of how the event based emergency/abnormal operating procedures are used in conjunction with the symptom based EOPs.

K/A Number: 2.4.8 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier:	1	**RO Imp:**	3.0	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.7	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

A plant shutdown is in progress due to a primary-to-secondary tube leak.
Several tubes fail catastrophically resulting in RCS pressure dropping to
1450 psig and CET's indicate 560°F.
Reactor Building pressure and sump level are trending upward.

Which of the following procedures should be in use?

a) Reactor Trip and Overheating

b) Tube Rupture and ESAS

c) Loss of Subcooling Margin and ESAS

d) Reactor Trip and Tube Rupture

Answer:

d) Reactor Trip and Tube Rupture

Notes:

Answer [d] is correct, the Tube Rupture procedure is used in conjunction with Reactor Trip when abnormal conditions other than a SGTR exist, i.e., RB sump and pressure.
Answer [a] is incorrect, although the RCS temperature is high for the pressure, this does not meet the entry conditions for the Overheating EOP.
Answer [b] is incorrect, although an ESAS would have occurred, it is not entered with a SGTR.
Answer [c] is incorrect, although both ESAS and loss of SCM has occurred, they are not entered with a SGTR.

References:

1202.006, Rev. 007-02-0, Tube Rupture, page 1

History:

Created for 2001 SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0400 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Direct **Originator:** R.Soukup

TUOI: ANO-1-LP-RO-CRD **Objective:** 7 **Point Value:** 1

Section: 3.1 **Type:** Reactivity Control

System Number: 001 **System Title:** Control Rod Drive

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

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

Tier:	2	**RO Imp:**	3.4	**RO Select:**	No	**Difficulty:**	3
Group:	1	**SRO Imp:**	3.8	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

Of the following CRD operating limits which one is NOT due to heat removal considerations?

a. Maximum CRD travel is 420 inches per hour.

b. Maximum CRD running time is 30 minutes per hour.

c. No more than 2 phases are energized when movement is stopped.

d. Latching control rods in JOG speed.

Answer:

d. Latching control rods in JOG speed.

Notes:

Answer [d] is correct, this limit is to prevent damage to CRDM spider.
Answers [a], [b], [c] are all limits associated with heat removal from CRDM and are therefore incorrect.

References:

1105.009, Rev. 016-02-0, CRD System Operating Procedure, page 11

History:

Direct from regular exambank, QID #2322, for 2001 SRO Exam.

QID: 0401 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-1-LP-RO-RBS **Objective:** 8 **Point Value:** 1

Section: 3.5 **Type:** Containment Integrity

System Number: 026 **System Title:** Containment Spray

Description: Knowledge of annunciator response procedures.

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

Tier: 2 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 4

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

Question:

A large break LOCA is in progress.
RCS pressure is ~ 50 psig.
RB pressure is 55 psia and trending down.
Annunciator K11-C6, "RB SPRAY P-35A ES FAILURE" is in alarm.

What condition brought in this alarm?

a) "A" RB Spray pump high motor winding temperature >300°F after ES Channel 7 actuation

b) "A" RB Spray flow <1050 gpm 55 seconds after ES Channel 7 actuation.

c) "A" RB Spray flow >1650 gpm 55 seconds after ES Channel 7 actuation.

d) "A" RB Spray pump high bearing temperature >200°F after ES Channel 7 actuation.

Answer:

b) "A" RB Spray flow <1050 gpm 55 seconds after ES Channel 7 actuation.

Notes:

Answer [b] is correct, this is the correct logic for this annunciator.
Answer [a] is incorrect, high motor winding temperature does not input into this alarm.
Answer [c] is incorrect, a high flow condition will not actuate this alarm.
Answer [d] is incorrect, high bearing temperature does not input into this alarm.

References:

1203.012J, Rev. 034-00-0, Annunciator K11 Corrective Action, page 29

History:

Created for 2001 SRO Exam.

QID: 0402 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Direct **Originator:** R.Soukup

TUOI: ANO-S-LP-RO-CCM08 **Objective:** 1 **Point Value:** 1

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

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

Description: Knowledge of the process for determining the internal and external effects on core reactivity.

K/A Number: 2.2.34 **CFR Reference:** 43.6

Tier: 2 **RO Imp:** 2.8 **RO Select:** No **Difficulty:** 3

Group: 2 **SRO Imp:** 3.2 **SRO Select:** Yes **Taxonomy:** C

Question:

Regarding void formation in the reactor coolant system during an accident, which area of void formation will have the greatest affect on the source range monitor response?

a. Vessel head

b. Vessel downcomer

c. Core shroud and periphery

d. Central core region

Answer:

b. Vessel downcomer

Notes:

Answer [b] is correct, a voided downcomer will allow more neutron leakage.
Answer [a] is incorrect, voids in this region will not affect source range indication.
Answer [c] is incorrect, the downcomer region surrounds this area.
Answer [d] is incorrect, this region is surrounded by water.

References:

General Physics "Mitigating Reactor Core Damage" training material, p.4-89,90

History:

Direct from regular exambank QID#5305, used in 2001 SRO Exam.

QID: 0403 **Rev:** 0 **Rev Date:** 11/21/00 **Source:** Direct **Originator:** D.Slusher

TUOI: ANO-1-LP-RO-NNI **Objective:** 10 **Point Value:** 1

Section: 3.7 **Type:** Instrumentation

System Number: 016 **System Title:** Non-nuclear Instrumentation

Description: Knowledge of annunciator response procedures.

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

Tier: 2 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 4

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

Question:

Initial Conditions:

SASS Mismatch Annunciator
Rx is Feedwater Limited Annunciator
Feedwater is Rx Limited Annunciator
Unit Tave TI-1032 is 583 degrees F
Loop A Tave TI-1020 is 588 degrees F
Loop B Tave TI-1043 is 578 degrees F

Which of the following actions would clear the cross limits and return temperature indications to normal?

a. Select the NNI-Y signal for RCS loop A hot leg temperature.

b. Place the Controlling Tave Selector Switch in the Loop B position.

c. Place Loop A Feedwater Demand in hand and raise Loop A feedwater flow.

d. Select the NNI-Y signal for RCS loop B cold leg temperature.

Answer:

a. Select the NNI-Y signal for RCS loop A hot leg temperature.

Notes:

Answer [a] is correct, because cross limits indicate a problem with a false high temperature feeding into the unit Tave.
Answer [b] is incorrect, due to instrument failure, the controlling Tave would not be selected until ICS components were placed in Manual.
Answer [c] is incorrect, ICS would be trying to raise A FW flow due to the cross limits.
Answer [d] is incorrect, "B" T cold is not the problem, the loop with the higher temperature has the failed instrument.

References:

STM 1-64, rev. 6, Integrated Control System, page 44, 45

History:

Direct from exambank 4574, used in 2001 SRO Exam.

QID: 0404 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-1-LP-RO-EOP02 **Objective:** 4 **Point Value:** 1

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

System Number: 035 **System Title:** Steam Generator

Description: Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Small break LOCA.

K/A Number: A2.06 **CFR Reference:** 41.5 / 43.5 / 45.3 / 45.5

Tier:	2	**RO Imp:**	4.5	**RO Select:**	No	**Difficulty:**	3
Group:	2	**SRO Imp:**	4.6	**SRO Select:**	Yes	**Taxonomy:**	Ap

Question:

Given:

- Small Break LOCA has occurred and Subcooling Margin was lost.
- ESAS actuated with all components operating properly.
- RCS pressure is now 1200 psig and rising slowly.
- CETs are 500°F and dropping slowly.
- Source of LOCA has been isolated.

Which of the following conditions will allow you to transition to the Reactor Trip EOP, 1202.001?

a) Uncontrolled RCS cooldown is occurring due to HPI/Break flow.

b) Primary to secondary heat transfer is NOT established.

c) Steam generator tube leakage is indicated.

d) Primary to secondary heat transfer is in progress.

Answer:

d) Primary to secondary heat transfer is in progress.

Notes:

Answer [d] is correct, the Reactor Trip EOP is only entered if the cause of the LOCA is isolated, SCM is restored, and primary to secondary heat transfer has been established.
Answer [a] is incorrect, this condition would lead to a transition to Small Break LOCA Cooldown, 1203.041.
Answer [b] is incorrect, the Loss of SCM EOP would still be in use.
Answer [c] is incorrect, the operator would transition to the Tube Rupture EOP if SCM was restored.

References:

1202.002, Rev. 004-00-0, Loss of Subcooling Margin, page 12

History:

New created for 2001 SRO Exam.

QID: 0405 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-1-LP-RO-MSSS **Objective:** 2 **Point Value:** 1

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

System Number: 076 **System Title:** Service Water

Description: Ability to analyze the affect of maintenance activities on LCO status.

K/A Number: 2.2.24 **CFR Reference:** 43.2 / 45.13

Tier: 2 **RO Imp:** 2.6 **RO Select:** No **Difficulty:** 3

Group: 3 **SRO Imp:** 3.8 **SRO Select:** Yes **Taxonomy:** Ap

Question:

Given:

- Plant is at 100% power.
- P-4B SW pump is inoperable due to high vibrations on last surveillance.
- P-4A SW pump is supplying Loop 1 SW
- P-4C SW pump is supplying Loop 2 SW
- Maintenance is complete on P-4B.
- Pre-evolution brief has been conducted and the surveillance is ready to be performed to prove operability of P-4B.

Are any compensatory measures needed to be in place while running the P-4B SW pump on Loop 1?

a) No actions needed due to P-4A and P-4C operable and powered from independent buses.

b) A dedicated licensed operator stationed to monitor SW, with no other duties.

c) No actions needed since P-4A will start on ESAS signal.

d) Enter Tech Spec 3.3.6 for inoperable SW loop during the surveillance.

Answer:

b) A dedicated licensed operator stationed to monitor SW, with no other duties.

Notes:

Answer [b] is correct, per 1104.029 operability section which states that an inoperable Service Water pump can be run on an operable loop of service water as long as the following compensatory measures are taken: pre-job brief and dedicated operator stationed to monitor SW, with no other duties.
Answer [a] is incorrect, due to if ESAS acutates and no operator action is taken an inoperable pump will be supplying loop 1 SW.
Answer [c] is incorrect, due to P-4A will not start on ESAS signal due to P4B was the pump running prior to the acutation.
Answer [d] is incorrect, if operations chooses to run P-4B without taken any measures then all equipment supplied by SW Loop 1 would be inoperable making LCO 3.0.3 the approriate LCO to enter.

References:

1104.029, Rev. 054-00-0, Service Water and Auxiliary Cooling System, page 29

History:

New created for 2001 SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0406 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-S-LP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of shift turnover practices.

K/A Number: 2.1.3 **CFR Reference:** 41.10 / 45.13

Tier: 3 **RO Imp:** 3.0 **RO Select:** No **Difficulty:** 2

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

Question:

You are the oncoming Shift Manager for night shift following your days off after a training week.

How many days of Station Logs are you required to review?

a) three

b) four

c) six

d) seven

Answer:

d) seven

Notes:

Answer [d] is correct, per 1015.001 the SM is required to review the station logs for the last 7 days or the last time on shift, whichever is shorter.
Answers [a], [b], [c] are other intervals the candidate might choose, [a] being the most probable since three days had expired since the SM was last at the plant but the SM was not on shift.

References:

1015.001, Rev. 052-05-0, Conduct of Operations, page 38

History:

New created for 2001 SRO Exam.

QID: 0407 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** New **Originator:** J.Cork

TUOI: ANO-S-LP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Knowledge of shift staffing requirements.

K/A Number: 2.1.4 **CFR Reference:** 41.10 / 43.2

Tier: 3 **RO Imp:** 2.3 **RO Select:** No **Difficulty:** 2

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

Question:

On New Year's Eve night shift, the on-duty CRS has a heart attack and must be transported to St. Mary's at 0210.

What is the latest time at which a replacement CRS must be in the Control Room
BEFORE the requirement of 1015.001, Conduct of Operations is violated?

a) 0300

b) 0400

c) 0500

d) 0600

Answer:

b) 0400

Notes:

Answer [b] is the correct answer since the maximum time the shift can be below the minimum complement is two hours.
Answers [a], [c], [d] are one hour increments around the correct answer.

References:

1015.001, Rev. 052-05-0, Conduct of Operations, page 36

History:

New created for 2001 SRO Exam.

QID: 0408 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-S-LP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.1 **System Title:** Conduct of Operations

Description: Ability to manage short term information such as night and standing orders.

K/A Number: 2.1.15 **CFR Reference:** 45.12

Tier: 3 **RO Imp:** 2.3 **RO Select:** No **Difficulty:** 2

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

Question:

During a refueling outage, the RP Manager approaches the Shift Manager about issuing a Night Order he has prepared concerning radiological precautions during fuel handling.
The Shift Manager takes the Night Order and reviews it with his crew.

Do you agree or disagree with his actions?

a) Agree, the Shift Manager has the authority to issue Night Orders.

b) Disagree, a procedure change is required to implement guidance.

c) Agree, any Manager has the authority to issue Night Orders.

d) Disagree, only the Ops Manager or his designee may issue Night Orders.

Answer:

d) Disagree, only the Ops Manager or his designee may issue Night Orders.

Notes:

Answer [d] is correct per 1015.001.
Answers [a], [b], [c] do not comply with 1015.001.

References:

1015.001, Rev. 052-05-0, Conduct of Operations, page 33

History:

New created for 2001 SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0409 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** New **Originator:** Jcork

TUOI: ANO-S-LP-SRO-ADMIN **Objective:** 3 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of process for making changes in the facility as described in the Safety Analysis Report.

K/A Number: 2.2.5 **CFR Reference:** 43.3 / 45.13

Tier: 3 **RO Imp:** 1.6 **RO Select:** No **Difficulty:** 2

Group: **SRO Imp:** 2.7 **SRO Select:** Yes **Taxonomy:** C

Question:

Which of the following would require a 10CFR50.59 Evaluation per 1000.131, 10CFR50.59 Review Program?

a) Replacing EDG's Air Receiving Tanks with larger tanks.

b) A switch to a different manufacturer for a valve operator for a safety system.

c) A change in the title of Shift Superintendent to Shift Manager.

d) A drawing change to correct a valve number on a P&ID used in the SAR.

Answer:

a) Replacing EDG's Air Receiving Tanks with larger tanks.

Notes:

Answer [a] is correct, any time you are changing out Plant Eupiment use must perform a 50.59 review.
Answers [b], [c], [d] are excluded from evaluation as listed in Att. 1 in 1000.131.

References:

1000.131, Rev. 003-04-0, 10CFR50.59 Review Program page.29-30,

History:

Created for 2001 SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0410 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** New **Originator:** S.Pullin

TUOI: ANO-S-LP-SRO-ADMIN **Objective:** 4 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.2 **System Title:** Equipment Control

Description: Knowledge of the process for managing maintenance activities during power operation.

K/A Number: 2.2.17 **CFR Reference:** 43.5 / 45.13

Tier:	3	**RO Imp:**	2.3	**RO Select:**	No	**Difficulty:**	3
Group:		**SRO Imp:**	3.5	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

Given:

- Plant is at 100% power.
- Mechanic wants to sign in a refueling work package for Main Turbine.
- The reason for doing this is to draw parts from warehouse for prep work.

What would the CRS-Admin use (if possible) to allow MM to draw the parts they need without actually starting work?

a) This is not allowed prior to the start of the refueling outage.

b) CRS must initiate a separate MAI for parts acquisition only.

c) Sign in work package and place an Operations Hold in package.

d) Sign in work package and conduct a pre-job brief.

Answer:

c) Sign in work package and place an Operations Hold in package.

Notes:

Answer [c] is correct IAW 1000.024.
Answer [a] is incorrect, the Ops Hold allows this type of activity.
Answer [b] is incorrect, a separate MAI sounds good but is not needed.
Answer [d] is incorrect, a pre-job brief is used in most cases but would not prevent other MM personnel from starting work.

References:

1000.024, Rev. 047-00-0, Control of Maintenance, p.13

History:

QID: 0411 **Rev:** 0 **Rev Date:** 12/1/00 **Source:** Modified **Originator:** E-Plan

TUOI: ANO-S-LP-EP-A0082 **Objective:** 4 **Point Value:** 1

Section: 2 **Type:** Generic Knowledges and Abilities

System Number: 2.4 **System Title:** Emergency Procedures/Plan

Description: Knowledge of which events related to system operations/status should be reported to outside agencies.

K/A Number: 2.4.30 **CFR Reference:** 43.5 / 45.11

Tier: 3 **RO Imp:** 2.2 **RO Select:** No **Difficulty:** 2

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

Question:

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 requirement for notification to the NRC?

a) Within 15 minutes of declaration of an emergency class.

b) Immediately following notification of the state and within 1 hour of the declaration of an emergency class.

c) Immediately following notification of the local agencies.

d) Within 4 hours of declaration of an emergency class.

Answer:

b) Immediately following notification of the state 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:

1903.011, Rev. 025-04-0, Emergency Response/Notifications, page 67

History:

Modified E-Plan exambank QID#61 for use in 2001 SRO Exam.

QID: 0412 **Rev:** 0 **Rev Date:** 12/5/00 **Source:** New **Originator:** S.Pullin

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

Section: 4.2 **Type:** Generic APEs

System Number: 057 **System Title:** Loss of Vital AC Inst. Bus

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

K/A Number: 2.4.1 **CFR Reference:** 41.10 / 43.5 / 45.13

Tier: 1 **RO Imp:** 4.3 **RO Select:** No **Difficulty:** 3

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

Question:

Given:
- Plant at 100% power.
- RPS Channel "C" inoperable and in the tripped state.
- I&C are performing RPS Channel "B" monthly calibration.
- Inverter Y-11 has a fault and the static switch fails to operate.

Which of the following procedures should be in use?

a) Loss of NNI Power, 1203.047

b) Reactor Trip, 1202.001

c) Loss of 125 VDC, 1203.036

d) ESAS, 1202.010

Answer:

b) Reactor Trip, 1202.001

Notes:

Answer [b] is correct since a loss of Y11 will cause a loss of RS-1 which will trip "A" RPS channel. This will cause a reactor trip since "C" is in the tripped state.
Answer [a] is incorrect, NNI is powered from RS-1 but will transfer to alternate power.
Answer [c] is incorrect, although Y11 is normally powered from a DC source, the Rx Trip EOP has the highest priority.
Answer [d] is incorrect, although ESAS is partially powered from RS-1, no actuations will result.

References:

1105.001, Rev.019-00-0, NI & RPS Operating Procedure, pages 7,8

History:

New created for 2001 SRO Exam.

Arkansas Nuclear One - Unit One
Initial RO/SRO Exam Question Data

QID: 0413 **Rev:** 0 **Rev Date:** 12/06/00 **Source:** New **Originator:** S. Pullin

TUOI: ANO-1-LP-AO-ELECD **Objective:** 4 **Point Value:** 1

Section: 3.6 **Type:** Electrical

System Number: 062 **System Title:** AC Electrical Distribution System

Description: Knowledge of AC distribution system design feature(s) and/or interlock(s) which provide for the following: Bus Lockouts.

K/A Number: K4.01 **CFR Reference:** 41.7

Tier:	2	**RO Imp:**	2.6	**RO Select:**	Yes	**Difficulty:**	2
Group:	2	**SRO Imp:**	3.2	**SRO Select:**	Yes	**Taxonomy:**	C

Question:

The plant is in Hot Shutdown with a normal electrical alignment.
Due to an electrical fault, K02-A6 "A1 L.O. RELAY TRIP" comes into alarm.

How will the electrical system respond?

a. #1 EDG will auto-start and will supply bus A3.

b. Bus A1 will auto-transfer to transformer SU#1.

c. Bus A1 will auto-transfer to transformer SU#2.

d. #1 EDG will auto-start, but will not supply bus A3.

Answer:

a. #1 EDG will auto-start and will supply bus A3.

Notes:

(a) is the correct answer, the A1 bus lock-out will cause a loss of A3, EDG#1 will auto-start and supply A3.
(b) and (c) are incorrect, the A1 lockout will prevent any transformer from supplying the bus.
(d) is incorrect, the A1 lockout will not prevent #1 EDG from supplying bus A3.

References:

1203.012B, Rev. 025-02-0, page 36.

History:

New Question for 2000 RO/SRO NRC Exam.