

Form 4.1-PWR Pressurized-Water Reactor Examination Outline

Facility:		Millstone 2		K/A Catalog Rev. 3				Rev. 8/8/2024				Date of Exam: 9/9/2024					
Tier	Group	RO K/A Category Points												SRO-Only Points			
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total	
1. Emergency and Abnormal Plant Evolutions	1	3	3	3				3	3				3	18	3	3	6
	2	1	1	2				2	1				1	8	2	2	4
	Tier Totals	4	4	5				5	4				4	26	5	5	10
2. Plant Systems	1	2	3	3	1	3	2	3	2	3	3	3	28	2	3	5	
	2	2	0	0	2	1	1	1	1	0	1	0	9	0	2	1	3
	Tier Totals	4	3	3	3	4	3	4	3	3	4	3	37	4	4	8	
3. Generic Knowledge and Abilities Categories	CO	EC			RC			EM			6	CO	EC	RC	EM	7	
	2	2			1			1				2	2	1	2		
4. Theory	Reactor Theory			Thermodynamics									6				
	3			3													
Notes: CO = Conduct of Operations; EC = Equipment Control; RC = Radiation Control; EM = Emergency Procedures/Plan * These systems/evolutions may be eliminated from the sample when Revision 2 of the K/A catalog is used to develop the sample plan. ** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan.																	

ES-4.1-PWR		PWR Examination Outline (Millstone 2)									
Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)											
Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#	
1	(007) (EPE 7; BW E02 & E10; CE E02) Reactor Trip, Stabilization, Recovery				X			(CE02EA1.09) Ability to operate and/or monitor the following as they apply to (CE E02) STANDARD POST-TRIP ACTIONS AND REACTOR TRIP RECOVERY (CFR: 41.7 / 45.5 / 45.6): PZR PCS	3.9	1	
2	(008) (APE 8) Pressurizer Vapor Space Accident				X			(008AA1.05) Ability to operate and/or monitor the following as they apply to (APE 8) PRESSURIZER VAPOR Space Accident (CFR: 41.7 / 45.5 / 45.6): ECCS	4.0	2	
3	(009) (EPE 9) Small Break LOCA	X						(009EK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (EPE 9) SMALL-Break LOCA (CFR: 41.8 / 41.10 / 45.3): Natural circulation and cooling, including reflux boiling	3.8	3	
4	(011) (EPE 11) Large Break LOCA						X	(011) (EPE 11) Large Break LOCA (G2.4.50) EMERGENCY PROCEDURES/PLAN: Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure (CFR: 41.10 / 43.5 / 45.3)	4.2	4	
5	(022) (APE 22) Loss of Reactor Coolant Makeup					X		(022AA2.02) Ability to determine and/or interpret the following as they apply to (APE 22) LOSS OF REACTOR Coolant Makeup (CFR: / 43.5 / 45.13): Charging pump problems	3.5	5	
6	(025) (APE 25) Loss of Residual Heat Removal System		X					(025AK2.20) Knowledge of the relationship between (APE 25) LOSS OF RESIDUAL Heat Removal System and the following systems or components (CFR: 41.7 / 45.7): ECCS	3.8	6	
7	(026) (APE 26) Loss of Component Cooling Water		X					(026AK2.02) Knowledge of the relationship between (APE 26) LOSS OF Component Cooling Water and the following systems or components (CFR: 41.8 / 41.10 / 45.3): RCPs	4.2	7	
8	(029) (EPE 29) Anticipated Transient Without Scram				X			(029EA1.16) Ability to operate and/or monitor the following as they apply to (EPE 29) ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) (CFR: 41.7 / 45.5 / 45.6): CVCS	3.6	8	
9	(038) (EPE 38) Steam Generator Tube Rupture		X					(038EK2.10) Knowledge of the relationship between (EPE 38) STEAM GENERATOR Tube Rupture and the following systems or components (CFR: 41.7 / 41.8 / 45.4 / 45.7 / 45.8): ECCS	4.1	9	
10	(040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer			X				(040AK3.02) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 40) STEAM LINE RUPTURE (CFR: 41.5 / 41.10 / 45.6 / 45.13): Reactor trip and/or ESFAS initiation	4.2	10	
11	(054) (APE 54; CE E06) Loss of Main Feedwater						X	(054) (APE 54; CE E06) Loss of Main Feedwater (G2.4.51) EMERGENCY PROCEDURES/PLAN: Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required) (CFR: 41.10 / 43.5 /45.13)	3.0	11	
12	(055) (EPE 55) Station Blackout					X		(055EA2.07) Ability to determine and/or interpret the following as they apply to (EPE 55) Station Blackout (CFR: 43.5 / 45.13): AFW flow	4.1	12	
13	(056) (APE 56) Loss of Offsite Power					X		(056AA2.60) Ability to determine and/or interpret the following as they apply to (APE 56) Loss of Offsite Power (CFR: 43.5 / 45.13): Status of MSIVs	3.2	13	
14	(057) (APE 57) Loss of Vital AC Instrument Bus	X						(057AK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 57) LOSS OF VITAL AC ELECTRICALINSTRUMENT BUS (CFR: 41.8 / 41.10 / 45.3): Effect of a loss of power to instruments powered by a vital instrument bus on plant operation	4.3	14	

15	(058) (APE 58) Loss of DC Power	X						(058AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 58) LOSS OF DC Power (CFR: 41.8 / 41.10 / 45.3): Effect of battery discharge rate on capacity	3.7	15
16	(062) (APE 62) Loss of Nuclear Service Water			X				(062AK3.03) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 62) LOSS OF SERVICE WATER (CFR: 41.4 / 41.8 / 45.7): Guidance actions contained in AOPs for loss of service water	3.9	16
17	(065) (APE 65) Loss of Instrument Air						X	(065) (APE 65) Loss of Instrument Air (G2.2.44) EQUIPMENT CONTROL: Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions (CFR: 41.5 / 43.5 / 45.12)	4.2	17
18	(077) (APE 77) Generator Voltage and Electric Grid Disturbances			X				(077AK3.02) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 77) GENERATOR VOLTAGE AND ELECTRIC Grid Disturbances (CFR: 41.4 / 41.5 / 41.7 / 41.10 / 45.8): Actions contained in AOPs for voltage and grid disturbances	3.5	18
19	(015) (APE 15) Reactor Coolant Pump Malfunctions						X	(015) (APE 15) Reactor Coolant Pump Malfunctions (G2.3.11) RADIATION CONTROL: Ability to control radiation releases (CFR: 41.11 / 43.4 / 45.10)	4.3	76
20	(027) (APE 27) Pressurizer Pressure Control System Malfunction					X		(027AA2.11) Ability to determine and/or interpret the following as they apply to (APE 27) PRESSURIZER PRESSURE Control System Malfunction (CFR: 43.5 / 45.13): RCS pressure	3.8	77
21	(038) (EPE 38) Steam Generator Tube Rupture					X		(038EA2.15) Ability to determine and/or interpret the following as they apply to (EPE 38) STEAM GENERATOR Tube Rupture (CFR: 43.5 / 45.13): RCS pressure	3.9	78
22	(040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer						X	(040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer (G2.4.29) EMERGENCY PROCEDURES/PLAN: Knowledge of the emergency plan implementing procedures (CFR: 41.10 / 43.5 / 45.11)	4.4	79
23	(058) (APE 58) Loss of DC Power					X		(058AA2.02) Ability to determine and/or interpret the following as they apply to (APE 58) LOSS OF DC Power (CFR: 43.5 / 45.13): 125-V DC bus voltage	3.8	80
24	(065) (APE 65) Loss of Instrument Air						X	(065) (APE 65) Loss of Instrument Air (G2.4.12) EMERGENCY PROCEDURES/PLAN: Knowledge of operating crew responsibilities during emergency and abnormal operations (CFR: 41.10 / 43.2 / 45.12)	4.3	81
	(W E04) LOCA Outside Containment / 3									
	(W E11) Loss of Emergency Coolant Recirculation / 4									
	(BW E04; W E05) Inadequate Heat Transfer – Loss of Secondary Heat Sink / 4									
K/A Category Totals:		3	3	3	3	6	6	Group Point Total:	24	

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(059) (APE 59) Accidental Liquid Radwaste Release / 9									
(060) (APE 60) Accidental Gaseous Radwaste Release / 9									
(061) (APE 61) Area Radiation Monitoring System Alarms / 7									
(068) (APE 68; BW A06) Control Room Evacuation / 8									
(076) (APE 76) High Reactor Coolant Activity / 9									
(W E01 & E02) Rediagnosis & SI Termination / 3									
(W E13) Steam Generator Overpressure / 4									
(W E15) Containment Flooding / 5									
(W E16) High Containment Radiation /9									
(BW A01) Plant Runback / 1									
(BW A02 & A03) Loss of NNI-X/Y/7									
(BW A04) Turbine Trip / 4									
(BW A05) Emergency Diesel Actuation / 6									
(BW A07) Flooding / 8									
(BW E03) Inadequate Subcooling Margin / 4									
(BW E08; W E03) LOCA Cooldown – Depressurization / 4									
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4									
(BW E13 & E14) EOP Rules and Enclosures									
(CE A11**; W E08) RCS Overcooling – Pressurized Thermal Shock / 4									
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
K/A Category Totals:	1	1	2	2	3	3	Group Point Total:		12

ES-4.1-PWR																
PWR Examination Outline (Millstone 2)																
Plant Systems—Tier 2/Group 1 (RO/SRO)																
Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#	
37	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM						X						(003K6.08) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF4P RCP) REACTOR COOLANT PUMP SYSTEM (CFR: 41.7 / 45.5): Antireverse rotation device	2.8	27	
38	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM			X									(004K3.03) Knowledge of the effect that a loss or malfunction of the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM will have on the following systems or system parameters (CFR: 41.5 to 41.7): CCWS	2.7	28	
39	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM							X					(005A1.09) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM, including (CFR: 41.5 / 45.5): RCS temperature	3.7	29	
40	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM					X							(005K5.14) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM (CFR: 41.5 / 45.7): Refueling operations	3.5	30	
41	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM							X					(006A1.15) Ability to predict and/or monitor changes in parameters associated with operation of the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM, including (CFR: 41.5 / 45.3 to 45.5): RWST level and temperature	3.6	31	
42	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM		X										(006K2.02) Knowledge of electrical power supplies to the following: (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM (CFR: 41.7) Valve operators for safety injection tanks	3.2	32	
43	(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM					X							(007K5.07) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM (CFR: 41.5 / 45.7): PZR safety tailpipe temperature variations with PRT/quench tank pressure	3.7	33	

44	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM									X			(008A3.08) Ability to monitor automatic features of the (SF8 CCW) COMPONENT COOLING WATER SYSTEM, including (CFR: 41.7 / 45.5): Automatic actions associated with the CCWS that occur as a result of an ESFAS signal	4.1	34
45	(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM							X					(010A1.07) Ability to predict and/or monitor changes in parameters associated with operation of the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM, including (CFR: 41.5 / 45.5): RCS or PZR pressure	3.9	35
46	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM						X						(012K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF7 RPS) REACTOR PROTECTION SYSTEM (CFR: 41.7 / 45.7): Bistables and bistable test equipment	3.7	36
47	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM										X		(012A4.03) (SF7 RPS) REACTOR PROTECTION SYSTEM Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.7): Channel blocks and bypasses	3.8	37
48	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM									X			(013A3.07) Ability to monitor automatic features of the (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, including (CFR: 41.6 to 41.8 / 45.5): Main steam isolation system actuation	4.1	38
49	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM											X	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (G2.1.8) CONDUCT OF OPERATIONS: Ability to coordinate personnel activities outside the control room (CFR: 41.10 / 43.1 / 45.5 / 45.12 / 45.13)	3.4	39
50	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM								X				(022A2.04) Ability to (a) predict the impacts of the following on the (SF5 CCS) CONTAINMENT COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 43.5 / 45.3 / 45.13): Cooling water system malfunction	3.7	40

51	(026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM										X		(026A4.01) Ability to manually operate and/or monitor the (SF5 CSS) CONTAINMENT SPRAY SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): CSS controls	3.9	41
52	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM	X											(039K1.08) Knowledge of the physical connections and/or cause and effect relationships between the (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 / 45.8): MFW	3.6	42
53	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM			X									(039K3.05) Knowledge of the effect that a loss or malfunction of the (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.6): RCS	3.9	43
54	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM										X		(059A4.12) Ability to manually operate and/or monitor the (SF4S MFW) MAIN FEEDWATER SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Initiation of automatic MFW isolation	3.8	44
55	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM									X			(059A3.09) Ability to monitor automatic features of the (SF4S MFW) MAIN FEEDWATER SYSTEM (CFR: 41.7 / 45.5): MFW pump trips	3.6	45
56	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM											X	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM (191005K1.05) MOTORS AND GENERATORS (CFR: 41.7): Explanation of the difference between starting current and operating (running) current in a motor	2.8	46
57	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM				X								(062K4.03) Knowledge of (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Interlocks between automatic bus transfer and breakers	3.6	47
58	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM	X											(063K1.05) Knowledge of the physical connections and/or cause and effect relationships between the (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM and the following systems (CFR: 41.3 to 41.8 / 45.7 / 45.8): EDG	4.4	48

59	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM		X									(064K2.01) Knowledge of electrical power supplies to the following: (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM (CFR: 41.7) Starting air compressor	3.0	49
60	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM							X				(073A2.01) Ability to (a) predict the impacts of the following on the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 43.5 / 45.3 / 45.13 / 8-9): PRM component failures	3.9	50
61	(076) (SF4S SW) SERVICE WATER SYSTEM		X									(076K2.01) Knowledge of electrical power supplies to the following: (SF4S SW) SERVICE WATER SYSTEM (CFR: 41.7): SWS pumps (Class 1E)	3.8	51
62	(076) (SF4S SW) SERVICE WATER SYSTEM					X						(076K5.02) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4S SW) SERVICE WATER SYSTEM (CFR: 41.4 / 41.7 / 45.5): Pump runout	3.2	52
63	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM			X								(078K3.07) Knowledge of the effect that a loss or malfunction of the (SF8 IAS) INSTRUMENT AIR SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.6): CDS	2.7	53
64	(103) (SF5 CNT) CONTAINMENT SYSTEM									X		(103) (SF5 CNT) CONTAINMENT SYSTEM (G2.1.19) EQUIPMENT CONTROL: Ability to use available indications to evaluate system or component status (CFR: 41.10 / 45.12)	3.9	54
65	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM							X				(003A2.02) Ability to (a) predict the impacts of the following on the (SF4P RCP) REACTOR COOLANT PUMP SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 43.5 / 45.3 / 45.13): Conditions that exist for an abnormal shutdown of an RCP compared to a normal shutdown of an RCP	3.5	86

66	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM											X	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM (G2.1.6) CONDUCT OF OPERATIONS: Ability to manage the control room crew during plant transients (CFR: 43.5 / 45.12 / 45.13)	4.8	87
67	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM											X	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM (G2.1.7) CONDUCT OF OPERATIONS: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 43.5 / 45.12 / 45.13)	4.7	88
68	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM								X				(061A2.01) Ability to (a) predict the impacts of the following on the (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Inadvertent actuation of AFW at power	3.3	89
69	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM											X	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM (G2.4.20) EMERGENCY PROCEDURES/PLAN: Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions, and notes (CFR: 41.10 / 43.5 / 45.13)	4.3	90
	(025) (SF5 ICE) ICE CONDENSER SYSTEM														
	(053) (SF1; SF4P ICS*) INTEGRATED CONTROL SYSTEM														
K/A Category Totals:		2	3	3	1	3	2	3	4	3	3	6	Group Point Total:		33

Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
70	(011) (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM	X											(011K1.02) Knowledge of the physical connections and/or cause and effect relationships between the (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 / 45.8): RCS	4.1	55
71	(017) (SF7 ITM) IN CORE TEMPERATURE MONITOR SYSTEM							X					(017A1.01) Ability to predict and/or monitor changes in parameters associated with operation of the (SF7 ITM) IN CORE TEMPERATURE MONITOR SYSTEM, including (CFR: 41.5 / 45.7): Core exit temperature	4.0	56
72	(002) (SF2; SF4P RCS) REACTOR COOLANT SYSTEM						X						(002K6.06) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF2; SF4P RCS) REACTOR COOLANT SYSTEM (CFR: 41.7 / 45.7): Sensors and detectors	3.3	57
73	(029) (SF8 CPS) CONTAINMENT PURGE SYSTEM				X								(029K4.03) Knowledge of (SF8 CPS) CONTAINMENT PURGE SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Automatic purge isolation	3.6	58
74	(033) (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM	X											(033K1.08) Knowledge of the physical connections and/or cause and effect relationships between the (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 / 45.8): LRS	2.6	59
75	(045) (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM								X				(045A2.11) Ability to (a) predict the impacts of the following on the (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 43.5 / 45.3 / 45.5): Control problems in primary (e.g., axial flux imbalance); need to reduce load on secondary	3.1	60

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	(015) (SF7 NI) NUCLEAR INSTRUMENTATION SYSTEM														
	(027) (SF5 CIRS) CONTAINMENT IODINE REMOVAL SYSTEM														
	(028) (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM														
	(034) (SF8 FHS) FUEL HANDLING EQUIPMENT SYSTEM														
	(035) (SF4P SG) STEAM GENERATOR SYSTEM														
	(041) (SF4S SDS) STEAM DUMP / TURBINE BYPASS CONTROL SYSTEM														
	(050) (SF9 CRV*) CONTROL ROOM VENTILATION														
	(075) (SF8 CW) CIRCULATING WATER SYSTEM														
	(079) (SF8 SAS**) STATION AIR SYSTEM														
	(086) (SF8 FPS) FIRE PROTECTION SYSTEM														
K/A Category Totals:		2	0	0	2	1	1	1	3	0	1	1	Group Point Total:		12

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON		COMMON Examination Outline (Millstone 2)					
Facility: Millstone 2				Date of Exam: 9/9/2024			
Generic Knowledge and Abilities Outline (Tier 3) (RO/SRO)							
Category	K/A #	Topic	Item #	RO		SRO-Only	
				IR	Q#	IR	Q#
1. Conduct of Operations	G2.1.21	(G2.1.21) CONDUCT OF OPERATIONS: Ability to verify that a copy of a controlled procedure is the proper revision (CFR: 41.10 / 45.10 / 45.13)	82	3.5	64		
	G2.1.37	(G2.1.37) CONDUCT OF OPERATIONS: Knowledge of procedures, guidelines, or limitations associated with reactivity management (CFR: 41.1 / 41.5 / 41.10 / 43.6 / 45.6)	83	4.3	65		
	G2.1.23	(G2.1.23) CONDUCT OF OPERATIONS: Ability to perform general and/or normal operating procedures during any plant condition (CFR: 41.10 / 43.5 / 45.2 / 45.6)	84			4.4	94
	G2.1.25	(G2.1.25) CONDUCT OF OPERATIONS: Ability to interpret reference materials, such as graphs, curves, and tables (reference potential) (CFR: 41.10 / 43.5 / 45.12)	85			4.2	95
	Subtotal			N/A	2	N/A	2
2. Equipment Control	G2.2.12	(G2.2.12) EQUIPMENT CONTROL: Knowledge of surveillance procedures (CFR: 41.10 / 43.2 / 45.13)	86	3.6	66		
	G2.2.39	(G2.2.39) EQUIPMENT CONTROL: Knowledge of less than or equal to 1 hour TS action statements (does not include action statements of 1 hour or less that follow the expiration of a completion time for a TS condition for which an action statement has already been entered) (CFR: 41.7 / 41.10 / 43.2 / 45.13)	87	4.5	67		
	G2.2.5	(G2.2.5) EQUIPMENT CONTROL: Knowledge of the process for making design or operating changes to the facility, such as 10 CFR 50.59, "Changes, Tests and Experiments," screening and evaluation processes, administrative processes for temporary modifications, disabling annunciators, or installation of temporary equipment (CFR: 41.10 / 43.3 / 45.13)	88			3.2	96
	G2.2.45	(G2.2.45) EQUIPMENT CONTROL: Ability to determine and/or interpret TS with action statements of greater than 1 hour (SRO Only)(CFR: 41.7 / 41.10 / 43.2 / 45.13)	89			4.5	97
	Subtotal			N/A	2	N/A	2
3. Radiation Control	G2.3.12	(G2.3.12) RADIATION CONTROL: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters (CFR: 41.12 / 43.4 / 45.9 / 45.10)	90	3.2	68		
	G2.3.5	(G2.3.5) RADIATION CONTROL: Ability to use RMSs, such as fixed radiation monitors and alarms or personnel monitoring equipment (CFR: 41.11 / 41.12 / 43.4 / 45.9)	91			2.9	98
	Subtotal			N/A	1	N/A	1
4. Emergency Procedures / Plan	G2.4.25	(G2.4.25) EMERGENCY PROCEDURES/PLAN: Knowledge of fire protection procedures (CFR: 41.10 / 43.5 / 45.13)	92	3.3	69		
	G2.4.18	(G2.4.18) EMERGENCY PROCEDURES/PLAN: Knowledge of the specific bases for emergency and abnormal operating procedures (CFR: 41.10 / 43.1 / 45.13)	93			4.0	99
	G2.4.32	(G2.4.32) EMERGENCY PROCEDURES/PLAN: Knowledge of operator response to loss of annunciators (CFR: 41.10 / 43.5 / 45.13)	94			4.0	100
	Subtotal			N/A	1	N/A	2
Tier 3 Point Total				N/A	6	N/A	7

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON		COMMON Examination Outline (Millstone 2)				
Facility: Millstone 2		Date of Exam: 9/9/2024				
Theory (Tier 4) (RO)						
Category	K/A #	Topic	Item #	RO		
				IR	Q#	
Reactor Theory	192006	(192006K1.14) FISSION PRODUCT POISONS (CFR: 41.1): Explain the methods and reasons for the reactor operator to compensate for the time-dependent behavior of xenon-135 concentration in the reactor	95	3.3	70	
	192007	(192007K1.04) FUEL DEPLETION AND BURNABLE POISONS (CFR: 41.1): Describe how and why boron concentration changes over core life	96	3.4	71	
	192008	(192008K1.03) REACTOR OPERATIONAL PHYSICS (CFR: 41.1): (STARTUP AND APPROACH TO CRITICALITY) Describe count rate and instrument response that should be observed for rod withdrawal during the approach to criticality	97	4.0	72	
	Subtotal			N/A	3	
Thermodynamics	193003	(193003K1.24) STEAM (CFR: 41.14): Explain the usefulness of steam tables to the control room operator	98	3.1	73	
	193004	(193004K1.15) THERMODYNAMIC PROCESS (CFR: 41.14): (THROTTLING AND THE THROTTLING PROCESS) Determine the exit conditions for a throttling process based on the use of steam and/or water	99	2.8	74	
	193009	(193009K1.09) CORE THERMAL LIMITS (CFR: 41.14): Describe the effects of quadrant power tilt (symmetric offset), including long-range effects	100	3.2	75	
	Subtotal			N/A	3	
Tier 4 Point Total				N/A	6	

Form 4.1-1 Record of Rejected Knowledge and Abilities

Refer to Examination Standard (ES)-4.2, “Developing Written Examinations,” Section B.3, for deviations from the approved written examination outline.

[illegible]

Question #1

A plant trip occurred from 100% power.

- RC-100E, Loop 1A Pressurizer Spray valve, is stuck open.

What action(s) are directed by EOP 2525, Standard Post Trip Actions?

- A. Stop RCPs as necessary to stabilize RCS pressure.
- B. Close both MSIVs to maintain RCS temperature and pressure.
- C. Start all available charging pumps to maintain Pressurizer pressure.
- D. Perform a containment entry and manually isolate the Pressurizer Spray Valve.

Question #1

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 02EA1.09

Importance Rating: 3.9

K/A Statement: Ability to operate and/or monitor the following as they apply to STANDARD
POST-TRIP ACTIONS AND REACTOR TRIP RECOVERY: PZR PCS

Proposed Answer: A

Justification: If no operator action is taken, a stuck open spray valve results in an uncontrolled depressurization of the RCS due to Pressurizer spray flow. Differential pressure across the reactor is the driving force for spray flow, and the D/P is generated by the RCP driven forced circulation through the reactor Vessel. EOP 2525, Standard Post Trip Actions, Step 5 → RNO 5.2 specifies that any pressurizer spray valve which is open, and the valve will not close requires stopping RCPs as necessary. This action in the EOPs will stop spray flow and assist in regaining control of RCS pressure.

Plausibility:

- A. Correct.
- B. Closing both MSIVs will isolate the SGs, but the Spray Flow comes from the RCS Cold leg, so spray flow will continue. Plausible; Candidate may assume that closing the MSIVs and stopping heat removal from the RCS would cause RCS to heat up and cause pressure to rise with the subsequent in-surge.
- C. Starting all charging pumps will mitigate or correct the pressurizer level loss on an RCS leak which also would be a cause of pressure lowering. Plausible; Candidate may associate the RCS pressure loss with a leak and think it can be mitigated with charging pumps.
- D. A manual Containment entry cannot be done quickly enough to make a difference. By the time the preparations are made, the pressure will have lowered to the SI setpoint. Plausible; If there were more time before the pressure would lower to the Safety Injection setpoint, then in theory, the spray valves could be manually isolated.

Technical Reference(s):

- EOP 2525, Standard Post Trip Actions
- Vision Question 415086

Provided reference(s): None

Learning Objective: 283649 – Outline the Instruction and Contingency actions for the Immediate
Actions in EOP 2525, Standard Post Trip Actions

Source: Bank Vision #415086

Cognitive Level: Memory

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Question #2

The plant was tripped due to a LOCA.

- Pressurizer Safety 2-RC-201 is partially open.
- Pressurizer level = 19% and lowering.
- Reactor vessel head level = 80%.
- RCS pressure = 1300 psia and stable.
- $T_{CET} = 530$ °F and stable.
- Both S/G levels are 55% and rising.

In accordance with EOP 2532, Loss of Coolant Accident, is...

1. HPSI throttle criteria met?

AND

2. LPSI pump stop criteria met?

A. 1. YES.
2. YES.

B. 1. YES.
2. NO.

C. 1. NO.
2. YES.

D. 1. NO.
2. NO.

Question #2

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 008AA1.05

Importance Rating: 4.0

K/A Statement: Ability to operate and/or monitor the following as they apply to PRESSURIZER
VAPOR Space Accident: ECCS

Proposed Answer: C

Justification: HPSI throttle criteria are NOT met due LOWERING Pressurizer level. LPSI stop criteria are met due to RCS pressure > 360 psia and stable. The examinee will need to use steam tables to determine the RCS > 30 °F is subcooled

Plausibility:

- A. Examinee may think with 19% in the pressurizer and 80% in the Vessel head, the Inventory Control Safety function is met, allowing HPSI to be throttled. LPSI can be stopped since pressure is stable.
- B. Examinee may think with 19% in the pressurizer and 80% in the Vessel head, the Inventory Control Safety function is met, allowing HPSI to be throttled. The examinee may think since the plant is going to be cooled down and depressurized that LPSI cannot be stopped because it's required for the transition to Shutdown Cooling.
- C. Correct.
- D. The examinee may think that since pressurizer level is lowering, pressure will soon be lowering and LPSI stop criteria will not be met in the future.

Technical Reference(s):

- EOP 2532, Loss of Coolant Accident, steps 24 and 26
- Vision Question #415300

Provided reference(s): None

Learning Objective: 283791 – List the plant conditions required for the following as specified in EOP 2532, Loss of Coolant Accident: B. HPSI/LPSI termination criteria.

Source: Modified

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Unit 2 experienced an automatic trip on High Pressurizer Pressure due to an inadvertent closure of all Main Turbine Control Valves. The following conditions exist:

- The crew reports that Bus 24D is deenergized due to a fault.
- Power Operated Relief Valve (PORV), RC-404, is stuck open and cannot be isolated.
- After completion of EOP 2525, the crew entered EOP 2532, "Loss of Coolant Accident".
- All other equipment operated as designed.

The following conditions now exist:

- Containment pressure is 4.6 psig, slowly increasing.
- Reactor vessel level is 43%, decreasing.
- CET temperatures are 573°F - 577°F, and stable.
- RCS pressure is 1243 psia, stable.
- Pressurizer level is 100%.
- Steam generator levels are both 41%, slowly increasing.

WHICH action(s) must the Unit Supervisor/Shift Manager perform to preserve a Safety Function?

- A. Direct the Technical Support Center to develop a plan to restore RWST level.
- B. Direct the Balance of Plant Operator to align Condenser Air Removal to the Unit 2 Stack.
- C. Direct the Reactor Operator to place the SI/CS Pump Miniflow "OPERATE/INHIBIT" switches in "OPERATE" .
- D. Direct the crew to commence a controlled cooldown and depressurization.

Answer: D

Answer Explanation

D IS CORRECT; With RCS pressure stable at 1243 psia and the PORV still open, RCS inventory is being lost faster than Charging can restore it. The steps for the cooldown and subsequent depressurization must be pulled forward (performed out of sequence) to allow RCS pressure to be reduced below HPSI shut off head to allow adequate Safety Injection flow.

A is incorrect; Although RWST level is lowering, there is NO need to develop a plan to restore RWST level at this time (perform step out of sequence).

Plausible because step 8 of EOP 2532 directs the US or SM to have the TSC develop a plan for restoring level in the RWST if the LOCA is determined to be outside of Containment. Examinee may not remember that this step is performed ONLY if the LOCA is outside of Containment.

B is incorrect; With Containment pressure 4.42 psia, MSI has actuated and the MSIVs are closed resulting in a loss of Condenser vacuum; therefore, there is no need to align Condenser Air Removal to the Unit 2 stack.

Plausible because this is a procedurally directed step. Step 15 states, If EBFAS has initiated and the Condenser is available, then align Condenser Air Removal to Unit 2 stack. SIAS, CIAS, EBFAS, and MSI have all automatically actuated on CTMT pressure. If the examinee does not realize that MSI has actuated, then this step may be performed out of sequence.

C is incorrect; The SI/CS Pump Miniflow switches are not placed in 'OPERATE' until RWST level is 20%.

Plausible because the examinee may feel that a Sump Recirc Actuation Signal is imminent; therefore, it would be appropriate to perform this step out of sequence.

Question 1 Info

Question Type:	Multiple Choice
Status:	
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	1.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	415300
Version ID:	497973
User-Defined ID:	0086426-MB
Cross Reference Number:	
Topic:	MB-00391 E32-01-C LOIT Loss of Coolant Accident EOP 2532 Enabling Objective MB-05937
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	

Question #3

Millstone Unit 2 has experienced a Small Break LOCA with inadequate HPSI flow and has transitioned to two phase natural circulation (reflux boiling).

The crew can enhance reflux boiling heat removal by increasing _____?

- A. RCS T_{COLD} to > 550 °F.
- B. PZR level from 15% to 55%.
- C. S/G Narrow Range level from 10% to 50%.
- D. PZR pressure from 1500 psia to 1600 psia.

Question #3

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 009EK1.01

Importance Rating: 3.8

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to SMALL-Break LOCA: Natural circulation and cooling, including reflux boiling

Proposed Answer: C

Justification: Reflux boiling is the process of the RCS THOT voiding (boiling) to become steam, then traveling up the THOT leg side of the S/G tubes, and subsequently condensing in the tubes and falling back down into the RCS THOT leg. As (secondary side) S/G water level rises, the S/G tubes become more covered by cooler water, and this results in a greater cooling effect.

Plausibility:

- A. Changing RCS parameters will have negligible effect on reflux boiling. Plausible; May think the reflux boiling is a result of uncoupled natural circulation (where the RCS ΔT has grown too large) and attempt to raise cold leg temperature.
- B. Changing RCS parameters will have negligible effect on reflux boiling. Plausible; May think that raising pressurizer level will add cooler water to the RCS and have a greater cooling effect.
- C. Correct.
- D. Changing RCS parameters will have negligible effect on reflux boiling. Plausible; May think that raising pressure might shift conditions towards saturation, and that this may increase cooling.

Technical Reference(s):

- EOP 2532, Loss Of Coolant Accident, Tech Guide
- Vision Question 451896

Provided reference(s): None

Learning Objective: 283789 - Outline and explain the bases for the major actions in EOP 2532, Loss of Coolant Accident.

Source: Bank Vision #451896, NRC 2014 Q3

Cognitive Level: Memory

10CFR55.41(b)(14): Principles of heat transfer thermodynamics and fluid mechanics.

Comments:

Question #4

The plant experienced a LOCA 6 hours ago.

- Containment Spray IS NOT required for containment cooling.
- Containment pressure = 5.5 psig and lowering.
- RM-8240 & 8241, Containment High Range Radiation Monitors, ARE NOT in alarm.
- RM-8123 & 8262, Containment Air Radiation Monitors, ARE in alarm

In accordance with EOP 2532, LOCA, is Containment Spray termination criteria met?

- A. Yes, spray termination criteria is met.
- B. No, not enough time has elapsed.
- C. No, radiation levels are too high.
- D. No, Containment pressure is too high.

Question #4

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # EPE 11 G2.4.50 Importance Rating: 4.2

K/A Statement: Large Break LOCA EMERGENCY PROCEDURES/PLAN: Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure (CFR: 41.10 / 43.5 / 45.3)

Proposed Answer: A

Justification: EOP 2532 has four criteria which must be met prior to terminating Containment Spray: Four hours elapsed (for iodine scrubbing, satisfied in question stem), spray not required for containment cooling (satisfied based on containment temperature trend), containment pressure below 7 psig and stable or lowering, and RM-8240/8241 not in alarm.

Plausibility:

- A. Correct.
- B. Wrong, enough time has elapsed to allow spray to be secured. Plausible, the examinee may think spray is required to be run for > 8 hours.
- C. Wrong, CTMT area radiation monitors RM-8240/41 are NOT in alarm. The examinee may think the CTMT process radiation monitors RM-8123/8262 are used for CTMT radiation levels but they are isolated on high CTMT pressure.
- D. Wrong, containment pressure is below 7 psig and lowering. The examinee may think CTMT pressure needs to be less than the SIAS setpoint of 4.42 psig in order to secure spray

Technical Reference(s):

- EOP 2532, "Loss of Coolant Accident"

Provided reference(s): None

Learning Objective: Predict how operator action or inaction affects plant and system conditions concerning a Loss of Primary Coolant (283788)

Source: Modified Vision 415312

Cognitive Level: Memory

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: From 2/22 phone call - Added a time component and determination that Spray can be secured (other spray termination criteria met); also, changed answers to annunciators to meet K/A more closely. DF 05MAR Modified question to make it more operationally valid (give board indications, make operator determine actions. DF 7/26. Changed from alarm window nomenclature to rad monitor names based on validator feedback. DF 7/29

Unit 2 experienced a Large Break LOCA twelve (12) hours ago and is taking actions in EOP 2532 "Loss of Coolant Accident".

Which ONE of the following groups of criteria describe when Containment Spray may be secured?

- A. When CTMT Spray pumps have run for greater than 2 hours.
When CTMT pressure is less than 4.0 psig.
When CTMT Spray is no longer needed for cooling.
When Radiation Monitors RM-8123 & RM-8262 are not alarming.
- B. When CTMT Spray pumps have run for greater than 4 hours.
When CTMT pressure is less than 7.0 psig.
When CTMT Spray is no longer needed for cooling.
When Radiation Monitors RM-8240 & RM-8241 are not alarming.
- C. When CTMT Spray pumps have run for greater than 4 hours.
When CTMT pressure is less than 7.0 psig.
When CTMT Spray is no longer needed for cooling.
When Radiation Monitors RM-8123 & RM-8262 are not alarming.
- D. When CTMT Spray pumps have run for greater than 2 hours.
When CTMT pressure is less than 4.0 psig.
When CTMT Spray is no longer needed for cooling.
When Radiation Monitors RM-8240 & RM-8241 are not alarming.

Answer: B

Answer Explanation	
	A is WRONG. CTMT Spray must run for 4 hrs. minimum and RM-8240/8241 are the basis. B is CORRECT. Per EOP 2532 STep 59. C is WRONG. CTMT Press is < 7.0 psig and RM-8240 and 8241. D is WRONG. CTMT Spray run for 4 hrs minimum and CTMT pressure < 7.0 psig.

Question 1 Info	
Question Type:	Multiple Choice
Status:	
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	5
Difficulty:	1.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	415312
Version ID:	497986
User-Defined ID:	0086330-MB
Cross Reference Number:	
Topic:	MP2 LOIT, MB-00391, E32-01-C, MB-05939
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	

Associated objective(s):

ILT Predict how operator action or inaction affects plant and system conditions concerning a Loss of Primary Coolant.

Question #5

The plant is operating at 85% power with the 'C' charging pump operating

- The 'A' charging pump is started in manual.
- Charging and letdown balanced per 2304E, Charging Pumps.

Subsequently,

- The 'A' charging pump TRIPS.

Describe the system response to the pump trip.

- A. The 'B' charging pump will immediately start on the trip of the 'A' charging pump, and pressurizer level will remain stable at its setpoint.
- B. VCT level will slowly lower as letdown flow throttles down, and pressurizer level will remain stable at its setpoint
- C. Pressurizer level will initially lower until the 'B' charging pump starts and returns pressurizer level to setpoint.
- D. Pressurizer level will slowly lower and be maintained by the control system below normal setpoint.

Question #5

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 022AA2.02

Importance Rating: 3.5

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF REACTOR Coolant Makeup (CFR: / 43.5 / 45.13): Charging pump problems

Proposed Answer: D

Justification: The pressurizer LICs are proportional controllers, they generate a deviation signal when they are off their established program setpoint. The setpoint is adjusted based on number of charging pumps that are in operation. The stem states this adjustment would have been completed and set for two charging pumps in operation.

Since more than one pump had been operating at the time of the “A” Charging pump trip, and charging and letdown was previously balanced, then the Letdown Flow Controller, HIC-110, bias would need to be adjusted to “re-balance” charging and letdown flow and maintain pressurizer level at setpoint. If this were not done (as stated in the stem), pressurizer level would drop based on the deviation between charging and letdown (~ 44 gpm difference for loss of one charging pump). As soon as level starts to drop below setpoint, the level control system will begin to throttle down on letdown flow, based on a generated deviation from the proportional controller.

However, the level deviation required to lower letdown flow by 44 gpm (capacity of the lost charging pump) is greater than that required to trigger the second backup charging pump. Since the second backup charging pump (“B” pump in this case) is still available for automatic operation, then it will start when pressurizer level drops to the second backup trigger point. The “B” pump will then secure automatically as level recovers, and level will cycle in that fashion until operator action is taken.

Plausibility:

- A. There is no direct charging pump start based off a different charging pump tripping. Charging pump starts are based on LIC-110-1(-2) deviate from established setpoint (set on HIC-110) and the established charging pump backup control switch configuration. Plausible; May think that charging flow must be maintained to prevent the occurrence of flashing in the regenerative heat exchanger and may think that a charging pump automatically starts to ensure this does not occur.
- B. If pressurizer level maintains stable at setpoint, letdown will not throttle down because there will not be any proportional deviation signal generated. Pressurizer level cannot maintain stable at its setpoint for this transient. Plausible; May confuse the proportional controller with the function of the proportional integral controller which would adjust to maintain pressurizer level at setpoint.
- C. The “B” charging pump will start on a low proportional controller deviation signal, but it will also secure at some pressurizer level below setpoint and will cycle below program pressurizer level setpoint. Plausible; May confuse the controller operation with that of a

Proportional-Integral controller that would return the system to setpoint using the backup charging pump and the letdown flow control valve.

D. Correct.

Technical Reference(s):

- OP 2304E “Charging Pumps” R25
- OP 2204, Load Changes
- PLC-01-C, Pressurizer Level and Pressure Control System Lesson Plan
- Vision Question 451463

Provided reference(s): None

Learning Objective: 281166 - Describe the effects, including a loss or malfunction, of the Chemical and Volume Control System on the following:

- A. RCS
- B. Pressurizer Level
- C. Pressurizer Pressure (Aux Spray)
- D. Safety Injection System
- E. Shutdown Cooling System

Source: Bank Vision #451463

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(3): Mechanical components and design features of the reactor primary system.

Comments:

Question #6

The plant is in MODE 5.

- The "B" LPSI pump is operating on Shutdown Cooling.
- A411, RSST Supply to Bus 24D, trips open.

HOW will the "B" LPSI pump respond when power is restored by the "B" EDG, and WHY?

The "B" LPSI pump ____1.____ start automatically because ____2.____.

- A. 1. will
 2. the open "fingers" on C07R allow auto restart of the pump
- B. 1. will NOT
 2. the anti-pumping circuit prevents pump restart until UV is reset
- C. 1. will
 2. auto restart of the pump is enabled with Shutdown Cooling in service
- D. 1. will NOT
 2. no SIAS is present to start the pump signal from the sequencer

Question #6

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 025AK2.20

Importance Rating: 3.8

K/A Statement: Knowledge of the relationship between LOSS OF RESIDUAL Heat Removal System and the following systems or components: ECCS

Proposed Answer: D

Justification: There is no automatic restart of the LPSI pumps on an LNP, but they do get a load shed signal. Because of the load shed signal, the pump will not restart when power is restored, even though the control board handswitch is still in the Normal-After-Close position. Therefore, to restore SDC flow, the LPSI pump must be manually restarted after power is restored to the vital bus.

Plausibility:

- A. Wrong: The "fingers" on C07R do NOT affect the load shed signal, therefore the pump will still be turned off and must be manually restarted. The C07R fingers prevent tripping of the LPSI pumps due to Main Turbine testing or 345 kV breaker operations. Plausible: The student may recall that the fingers on C07R are designed to prevent inadvertent tripping of the LPSI pump due to various breaker control circuit operations and assume this will allow the pump to restart normally.
- B. Wrong: Although the LPSI pump has an "anti-pump" circuit, it is only armed if an automatic start signal is present and the pump is then shutdown by a signal other than a load shed. Therefore, it is NOT interlocked off under these conditions. Plausible: The student may misinterpret the interrelationship between the Load Shed signal and the Anti-Pump interlock. This often occurs with the Aux. Feedwater Pumps, if they are turned off with an AFAS signal present in preparation for re-energizing their dead bus. When the vital bus is reenergized in this situation, the AFW pump CANNOT be restarted until the AFAS signal is reset.
- C. Wrong: ; Even though the LPSI pump handswitch is in the "Normal-After-Closed" position, there is no constant "breaker close" or start signal present for the pump. Only a ESAS SIAS would send a start signal to the LPSI pump. Plausible: The student may recall that in all other situations (accident) where the LPSI pump is running, it is automatically restarted by the sequencer. In the SDC function, the LPSI pump is being used as the only means of core heat removal, leading one to believe it is acting in a similar role as in accident mitigation.

D. Correct.

Technical Reference(s):

- SDC-01-C, Shutdown Cooling Lesson Text

Provided reference(s): None

Learning Objective: Predict how operator action or inaction affects plant and system conditions concerning a Loss of Shutdown Cooling (283353).

Source: Bank Vision 452209 NRC 2008/Q7 2018/Q15

Cognitive Level: Memory

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Question #7

The plant is at 100% power when the following annunciators are received:

- “RBCCW SURGE TK LEVEL HI/LO” alarm (C-06/7, A-8)
- “RBCCW SURGE TK AUTO MAKEUP” alarm (C-06/7, B-8)
- “RBCCW HDR A PRES LO” alarm (C-06/7, B-7)
- “RBCCW HDR B PRES LO” alarm (C-06/7, D-7)

What condition requires tripping the affected RCP's?

- A. RCP Controlled Bleed Off temperature rises to 170 °F.
- B. RCP Lower Seal Temperature rises to 165 °F.
- C. RCP Stator winding temperature rises to 250 °F
- D. RCP RBCCW Flow has been lost for > 5 minutes.

Question #7

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 026AK2.02

Importance Rating: 4.2

K/A Statement: Knowledge of the relationship between LOSS OF Component Cooling Water and the following systems or components: RCPs

Proposed Answer: D

Justification: AOP 2564, Loss of RBCCW, Rev.6 step 3.3 states the reactor trip criteria.

RBCCW flow must be restored to both headers within five minutes or the contingency action, TRIP reactor and ENSURE affected RCPs are secured, must be performed.

Plausibility:

- A. RCP trip is required when any RCP CBO temperature rises above 195 °F. Plausible; May confuse the 170 °F in the distractor with the setpoint for RCP trip of ALL lower seal temperatures less than 170 °F.
- B. RCP trip is required when any RCP Seal Temperature rises above 170 °F. Plausible; The distractor value given is near the trip value and is a trip parameter.
- C. RCP trip is required when any RCP Stator temperature rises above 260 °F. Plausible; the value of 250 °F in the distractor could be confused with the require RCP trip on RCP Anti Reverse Bearing Temperature of 260 °F.
- D. Correct.

Technical Reference(s):

- AOP 2564, Loss of RBCCW
- AOP 2586, RCP Malfunctions

Provided reference(s): None

Learning Objective: 283218 - Describe the actions required when RCP temperature limits are exceeded.

Source: New

Cognitive Level: Memory

10CFR55.41(b)(3): Mechanical components and design features of the reactor primary system.

Comments:

Question #8

The plant is operating at 100% power.

- An inadvertent MSI occurs
- Pressurizer pressure is 2400 psia and stable.
- The TCB TRIP buttons were pushed at C-04.
- All Core Mimic RED lights are lit.
- TCB and MG contactor RED lights are lit on RPS mimic.

What action, directed by EOP 2525, Standard Post Trip Actions, WILL MITIGATE this event?

- A. Emergency borate from the BASTs.
- B. Trip the Main Turbine.
- C. Sequentially insert all CEAs.
- D. Depressurize and inject with HPSI.

Question #8

☒ RO

☐ SRO

Tier # 1 Group # 1 K/A # 029EA1.16 Importance Rating: 3.6

K/A Statement: Ability to operate and/or monitor the following as they apply to ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS): CVCS

Proposed Answer: A

Justification: In EOP 2525, Step 1 (Reactivity Control—Reactor Trip), if more than 1 CEA is not fully inserted after actions are taken at C04 and local (local trip of TCBs), emergency boration is required. In this case, 61 rods remain not fully inserted (the red lights indicate they are fully withdrawn). The RO is expected not to wait for the field operator to locally trip TCBs before performing the Emergency Boration.

Plausibility:

- A. Correct.
- B. Wrong: Although the BOP is required to ensure the turbine is tripped, since the MSIVs are already closed, tripping the turbine will do nothing additional to mitigate the ATWS. Plausible: OP 2260 directs the BOP to trip the turbine without delay if the RO is unsuccessful at tripping the reactor from C04.
- C. Wrong: EOP 2525 contains no provision for simply inserting CEAs via the normal method. If any CEAs remain out after efforts to trip during step 1, then emergency boration is required. Plausible: Inserting CEAs to the bottom through normal means would insert negative reactivity and mitigate the event.
- D. Wrong: Emergency boration using Appendix 3 is the prescribed method during EOP 2525. Plausible: when charging is not available and emergency boration is required, the Diagnostic Flowchart kicks out directly to EOP 2540 (based on reactivity safety function not met), and Success Path RC-3 gives guidance to reduce RCS pressure and borate with SI.

Technical Reference(s):

- EOP 2525, “Standard Post Trip Actions”

Provided reference(s): None

Learning Objective: Given a set of plant conditions, determine the action required in accordance with the Immediate Actions of EOP 2525, Standard Post Trip Actions (283650)

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Modified stem to add ‘inadvertent MSI’ as a bullet so it’s not skimmed over.

Changed C to “sequentially insert” for specificity. Edited question for clarity. DF 5/28 added bullet to show TCB TRIP buttons were pushed based on validator feedback. DF 8/15

Question #9

A plant cooldown is in progress with the following conditions:

- 'A' LPSI Pump is supplying Shutdown Cooling
- RCS Temperature = 190 °F
- RCS Pressure = 150 psia

- A Steam Generator Tube Rupture occurs.

What are the required actions in accordance with OP 2207, Plant Cooldown?

- A.
 - 1. Push SIAS pushbuttons on C-01 and CHECK for proper Safety Injection.
 - 2. GO TO EOP 2534, Steam Generator Tube Rupture.

- B.
 - 1. Start available HPSI pump(s) and manually ALIGN Safety Injection components.
 - 2. GO TO EOP 2534, Steam Generator Tube Rupture.

- C.
 - 1. Push SIAS pushbuttons on C-01 and CHECK proper Safety Injection.
 - 2. Start available LPSI pump aligned to the RWST.

- D.
 - 1. Push SIAS pushbuttons on C-01 and manually INITIATE HPSI.
 - 2. Start available LPSI pump aligned to the RWST.

Question #9

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 038EK2.10

Importance Rating:

K/A Statement: Knowledge of the relationship between STEAM GENERATOR Tube Rupture and the following systems or components: ECCS

Proposed Answer: D

Justification: OP 2207, Attachment 10 "Plant Cooldown Conditional Actions" section F states "If SIAS is not blocked, go to EOP 2534 "Steam Generator Tube Rupture". Based on conditions given in the stem, SIAS is currently blocked for cooldown, therefore, entering EOP 2534 would be an incorrect action to take. In addition, with SIAS inhibited, only depressing SIAS pushbuttons is not adequate to ensure the system is aligned properly. The system will require manual realignment in addition to pushbutton initiation in this plant condition.

Plausibility:

- A. WRONG; There is no step in OP 2207 that directs the operator to 'GO TO' EOP 2534 with SIAS blocked. Plausible; Possible to forget that SIAS inhibit requires manual operation of the ECCS components under 1850 psig. Also, With SIAS blocked, the SGTR is handled in OP 2207.
- B. WRONG; One HPSI pump has its breaker racked down in these conditions, and the distractor does not mention the required LPSI alignment action. Additionally, there is no step in OP 2207 that directs the operator to 'GO TO' EOP 2534 with SIAS blocked. Plausible; May remember than manual valve alignment for ECCS injection but may not realize LPSI is required also.
- C. WRONG; SIAS is inhibited during cooldown and requires manual HPSI/LPSI component manipulation by operators. Also, with shutdown cooling in service, the 'A' LPSI pump would not be aligned with a flow path from the RWST. Plausible; May remember that a normal SIAS initiation at power, this answer would be correct.
- D. Correct.

Technical Reference(s):

- OP-2207, Plant Cooldown, ATT 10 REV. 054
- OP-2207, Plant Cooldown, Section 4.6
- Vision question #414008

Provided reference(s): None

Learning Objective: 282613 - Given a set of plant conditions, determine OP 2207, "Plant Cooldown" applicability.

Source: Bank Vision #414008

Cognitive Level: Comprehension or Analysis

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Changed B to start both HPSI pumps (5/8 phone call). Changed temperature to 190 F based on validator feedback. DF 8/15

Question #10

Describe the reason for a SIAS during a Main Steam Line Break (MSLB)?

- A. Maintain RCS pressure to prevent loss of subcooling and reactor vessel head voiding during the RCS cooldown.
- B. Minimize the probability of a Pressurized Thermal Shock event by limiting the pressure transient on the RCS.
- C. Ensure that a CIAS will be generated in the event of a MSLB inside Containment.
- D. Ensure borated water is added to the RCS to offset the positive reactivity added during the RCS cooldown.

Question #10

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 040AK3.02

Importance Rating: 4.2

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to STEAM LINE RUPTURE: Reactor trip and/or ESFAS initiation

Proposed Answer: D

Justification: FSAR 6.3.1.1 states that the safety injection system must inject borated water into the RCS to limit fuel damage and increase shutdown margin during a main steam line rupture.

Plausibility:

- A. Wrong, inventory is not lost in a MSLB and subcooling goes up, therefore head voiding is not likely. Plausible: RCS pressure does drop during a MSLB. Additionally, voiding is a concern in many other events addressed by the EOP set.
- B. Wrong, the additional inventory actually raises the probability of a PTS event, particularly if the RCS is allowed to heat back up. Plausible because the inventory does limit the pressure transient, and the student may not recall the concern with the RCS expanding and overpressurizing (with the extra inventory) if allowed to heat back up.
- C. Wrong, the desire for CIAS is not the reason for SIAS in a MSLB. Plausible: a CIAS will actuate at the same setpoint as a SIAS.
- D. Correct.

Technical Reference(s):

- ECC-01-C ECCS PowerPoint slide 168
- FSAR 6.3.1.1
- FSAR 14.1.5.1.5.1.4
- Ginna 2010 NRC exam Q47

Provided reference(s): None

Learning Objective: Describe the expected plant response to an Excess Steam Demand (283587)

Source: Ginna 2010 Q47

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Question #11

A Loss of All Feedwater (LOAF) event is in progress.

- EOP 2537, Loss Of All Feedwater, is in progress.
- The 'C' Charging Pump is unavailable.
- 'A' SG Wide Range level = 155" and lowering.
- 'B' SG Wide Range level = 74" and lowering.
- RCS T_{COLD} = 540°F and rising.

In accordance with EOP 2537, Loss of All Feedwater, operators are required to initiate Once Through Cooling (OTC) 1. .

AND

Once OTC is initiated, the crew will then 2. for follow-up actions.

- A.
 - 1. immediately
 - 2. exit EOP 2537, Loss of All Feedwater and enter EOP 2540, Functional Recovery.
- B.
 - 1. immediately
 - 2. proceed with EOP 2537, Loss of All Feedwater.
- C.
 - 1. after 'B' SG Wide Range lowers below 70 inches
 - 2. exit EOP 2537, Loss of All Feedwater and enter EOP 2540, Functional Recovery.
- D.
 - 1. after 'B' SG Wide Range lowers below 70 inches
 - 2. proceed with EOP 2537, Loss of All Feedwater.

Question #11

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # APE 54 G2.4.51

Importance Rating: 3.0

K/A Statement: (054) (; CE E06) Loss of Main Feedwater () EMERGENCY

PROCEDURES/PLAN: Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required) (CFR:)

Proposed Answer: A

Justification: With RCS TCOLD slowly rising despite efforts to stabilize, EOP 2537 step 6 RNO must be entered. Since a specific value of TCOLD temperature rise is not specified in the stem, the candidate must use the information given about 1) the SG WR levels, 2) the 'C' charging pump unavailability, and 3) the CAUTION statement on EOP 2537 page 7. The Caution states that OTC should be initiated prior to ONE steam generator wide range level reaching 70 inches, AND ONE steam generator wide range level reaching 165 inches if less than 3 charging pumps are available. In this case, only two are available, and that is why OTC should be initiated immediately, vice waiting until the 'B' SG WR lowers below 70". Second, any situation where OTC is invoked in EOP 2537, requires a transition from EOP 2537 to EOP 2540.

Plausibility:

- A. Correct.
- B. when the RNO is entered for step 6 of EOP 2537, Loss of All Feedwater, to initiate OTC, then the operator must complete the step to proceed to EOP 2540, Functional Recovery. Plausible; Candidate may not remember that the very last sub step of the EOP 2537 RNO column specifies the procedure transition to EOP 2540.
- C. The caution before EOP 2537 Step 6 specifically states that if there are less than 3 charging pumps available, then OTC should be initiated prior to the 'B' SG WR level lowering to 70 inches or below. Plausible; Candidate may correctly identify that OTC is required in this situation but may not have knowledge of the caution that directs the operator to initiate OTC prior to meeting the conditions specified in EOP 2537 Step 6.1 if less than 3 charging pumps are available.
- D. when the RNO is entered for step 6 of EOP 2537, Loss of All Feedwater, to initiate OTC, then the operator must complete the step to proceed to EOP 2540, Functional Recovery. Plausible; Candidate may not remember that the very last sub step of the EOP 2537 RNO column specifies the procedure transition to EOP 2540.

Technical Reference(s):

- EOP 2537 Rev. 026, Loss Of All Feedwater.

Provided reference(s): None

Learning Objective: LOIT Describe the condition dependent actions and their bases for the following as specified in EOP 2537, Loss of All Feedwater: Initiation of once through cooling.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: editorial change to specify part 2 performance after Once Through Cooling is initiated based on validator feedback. DF 5/28 Changed temperature to 540 and rising to remove any ambiguity and added procedure EOP 2537 based on validator feedback. DF 7/29

Question #12

Millstone Unit 2 has experienced a Station Blackout (SBO).

- EOP 2530, “Station Blackout,” has been entered to mitigate the event.

WHAT indication reflects a challenge to the safety function which is required to be addressed FIRST with operator action?

- A. AFW flow = 0 GPM to both steam generators.
- B. Pressurizer Level = 21% and lowering.
- C. Containment temperature = 121°F and slowly rising.
- D. Pressurizer pressure = 1890 psia and slowly lowering.

Question #12

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 055EA2.07

Importance Rating: 4.1

K/A Statement: Ability to determine and/or interpret the following as they apply to Station

Blackout: AFW flow

Proposed Answer: A

Justification: AFW flow is required for RCS heat removal and the TDAFP is required to be started within 10 minutes of a loss of Main Feedwater if steam generators are lowering (they would be with no AFW flow) and the MDAFPs did not start (they would not have started without power). Additionally, RCS heat removal is the priority safety function from the standpoint of operator action. From OP 2260, "EOP User's Guide:" All Safety Functions are challenged by a SBO. The ability to maintain vital auxiliaries (power to AC equipment) is most affected. **Establishing RCS heat removal capability is a priority** as reactivity control, inventory control, and pressure control SFSCs all require AC power or control of RCS heat removal such that other Safety Functions can still be maintained. Failure to maintain natural circulation (NC) flow could impact the other Safety Functions. Natural Circulation and heat transfer from primary to secondary via the SGs is the preferred method of RCS and core heat removal for this event since RCPs are unavailable to provide forced circulation and the SDC is unavailable to provide RCS heat removal.

Plausibility:

- A. Correct.
- B. Wrong, Inventory Control requires power to address, and control of heat removal to mitigate in the meantime. Plausible: 21% in the pressurizer indicates that inventory control needs to be addressed.
- C. Wrong, Containment Pressure and Temperature Control requires power to address. Plausible: 121°F in Containment is above the TS maximum of 120°F, and thus, indicates Containment Temperature and Pressure Control is challenged.
- D. Wrong, Pressure Control requires power to address, and control of heat removal to mitigate in the meantime. Plausible: 1890 psia is below the SPTA band of 1900-2350 psia, indicating Pressure Control is challenged.

Technical Reference(s):

- OP 2260, "EOP User's Guide"
- Vision Question 415253

Provided reference(s): None

Learning Objective: Identify and explain the Safety Functions most affected by Station Blackout (283757).

Source: Modified Vision 415253

Cognitive Level: Memory

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Determined to be Cognitive level – Memory 7/25 phone call. DF 7/26. Added “and rising” based on NRC comments (when is “and stable” ever a bad thing?) DF 8/13

Which ONE of the following Safety Functions must be addressed first, utilizing operator action, during a Station Blackout event? (assume no other failures)

- A. RCS Heat Removal.
- B. RCS Inventory Control.
- C. Containment Temperature & Pressure Control.
- D. RCS Pressure Control.

Answer: **A**

Answer Explanation

CEN 152 Bases for Station Blackout--establishing natural circulation flow (RCS Heat Removal) is needed (as well as restoration of AC power) for the other Safety Functions to be addressed. OP 2260, EOP 2530, "Station Blackout" Implementation Guide.

Question 1 Info

Question Type:	Multiple Choice
Status:	
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	1.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	415253
Version ID:	497921
User-Defined ID:	0071928-MB
Cross Reference Number:	
Topic:	MP2 LOIT 2530, 2260 Important Safety Function
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	

Question 1 History

Exams Administered:	0
Student Encounters:	0
Answered Right:	0
Answered Wrong:	0
Partially Correct:	0
Answer Invalid:	0
Unanswered:	0
Ignore Response:	0
Avg Points Awarded:	0.00
... As % of Point Value:	0
Standard Deviation:	0.00

Question #13

A grid disturbance resulting in a Loss of Offsite Power (LOOP) has occurred.

The following conditions are present:

- Crew has entered EOP 2525, Standard Post Trip Actions.
- #1 Steam Generator is 780 psia and lowering.
- #2 Steam Generator is 805 psia and lowering.

Based on these conditions and the actions in EOP 2525...

The likeliest reason for this steam generator pressure is ____1.____ .

AND

Main Steam Isolation Valves (MSIVs) ____2.____ procedurally required to be shut.

- A. 1. steam flow via 2nd stage reheat causing cooldown
2. are NOT
- B. 1. T_{cold} dropping while natural circulation is building in
2. are NOT
- C. 1. steam flow via 2nd stage reheat causing cooldown
2. are
- D. 1. T_{cold} dropping while natural circulation is building in
2. are

Question #13

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 056AA2.60

Importance Rating: 3.2

K/A Statement: Ability to determine and/or interpret the following as they apply to (APE 56)

Loss of Offsite Power (CFR: 43.5 / 45.13): Status of MSIVs

Proposed Answer: C

Justification: 1. Second stage reheat drain tank high level dump valves fail open at 15%-20% turbine load decreasing. Loss of a non-vital bus (24A or 24B) cause the MOVs which supply second stage reheat steam to the affected side to fail as is (open) due to the power loss. On a LOOP, since both 24A and 24B are lost, this makes an open path for steam to flow to the condenser via both sides of 2nd stage reheat. This path is commonly referred to as the “sneak path.” 2. Main Steam Isolation Valves (MSIVs) are required to be shut. EOP 2525, Standard Post Trip Actions, Step 7, RNO c.4 states: if ANY steam generator pressure is less than 800 psia AND lowering, close BOTH MSIVs.

Plausibility:

- A. WRONG; MSIVs are required to be shut. EOP 2525, Standard Post Trip Actions, Step 7, RNO c.4 states: if ANY steam generator pressure is less than 800 psia AND lowering, close BOTH MSIVs. Plausible; Candidate may think that BOTH steam generators must be below 800 psia in EOP 2525 to manually close the MSIVs.
- B. WRONG; Natural circulation builds in with a rising T_{hot} while T_{cold} remains essentially constant. Additionally, MSIVs are required to be shut if ANY steam generator is below 800 psia and lowering. Plausible; Candidate may not recall the details as to how natural circulation builds in, and may believe both Steam Generators need to be below 800 psia before MSIVs are required to be shut.
- C. Correct.
- D. WRONG; Natural circulation builds in with a rising T_{hot} while T_{cold} remains essentially constant. Plausible; Candidate may not recall the details as to how natural circulation builds in.

Technical Reference(s):

- EOP 2525 Rev. 029, Standard Post Trip Actions.
- EOP 2528 Rev. 022, Loss of Offsite Power/Loss of Forced Circulation.
- E28-01-C PowerPoint, Slide 51

Provided reference(s): None

Learning Objective: 283650 - Given a set of plant conditions, determine the action required in accordance with the immediate actions of EOP 2525, Standard post trip actions.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Changed first part of the 2x2 to ask student why steam generator pressure is likely to drop during a LOOP, to ensure plausibility of all distractors. SRM 8/15.

Question #14

What action occurs on a Loss of VA-10 that requires securing Charging and Letdown?

- A. CH-501, VCT OUT ISOL, closes, causing the charging pumps to trip on low suction pressure.
- B. CH-512, MAKEUP VLV STOP, opens, causing an uncontrolled dilution of the RCS from the PWST.
- C. CH-514, BA ISOL, opens, causing an uncontrolled boration of the RCS from the BASTs.
- D. CH-192, RWST ISOL, opens, causing an uncontrolled boration of the RCS from the RWST.

Question #14

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 057AK1.01

Importance Rating: 4.3

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to LOSS OF VITAL AC ELECTRICAL INSTRUMENT BUS: Effect of a loss of power to instruments powered by a vital instrument bus on plant operation

Proposed Answer: D

Justification: Charging and Letdown are secured on a Loss of VA-10, VA-20, VR-11 and VR-21. On a loss of Vital Instrument bus VA-10, VCT level column, LC-227, loses power causing the VCT outlet, CH-501 to close and the RWST isolation to the Charging Pump Suction, CH-192, to open causing an uncontrolled boration to the RCS.

Plausibility:

- A. CH-501 may close but since CH-192 opens, the charging pumps will remain running.
- B. The examinee might forget that the reason for isolating charging and letdown on a loss of VA-10 is due to boration and not dilution.
- C. The examinee remembers the RCS is being borated but forgets by which path.
- D. Correct.

Technical Reference(s):

- AOP 2504C, Loss of 120 VAC Vital Instrument Panel VA-10
- Vision #451845

Provided reference(s): None

Learning Objective: 282856 – Given a set of plant conditions, identify TS & TRM implications concerning AOP 2504A/B/C/D/E/F, Loss of instrument Panels VR-11/21 and Vital Instrument Panels VA-10/20/30/40.

Source: Modified Vision #451845

Cognitive Level: Memory

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Editorial change to make the syntax of the answers match the stem. DF 8/14/24

The plant is operating at 100% power.

Then, VCT level transmitter, LC-227, fails low.

Which of the following will automatically occur and what action is necessary?

- A. VCT Outlet Isolation, CH-501, closes and would cause Charging pumps to trip on low suction pressure.
Isolate Letdown flow if Charging pumps trip or manually secure both Charging and Letdown.
- B. Makeup Valve Stop, CH-512, opens if in automatic, and would initiate a blended makeup to the VCT.
Monitor Reactor power for any change if allowed to progress, or switch to Manual Makeup mode.
- C. Boric Acid Isolation, CH-514, opens and would align Charging suction to the Boric Acid Storage Tanks.
Secure Charging and Letdown and monitor reactor power for changes due to possible boration.
- D. RWST Isolation, CH-192 opens, and would initiate boration from the Refueling Water Storage Tank.
Secure Charging and Letdown and adjust Turbine load as necessary to stabilize RCS temperature.

Answer: D

Answer Explanation

A is wrong; The VCT outlet isolation (Charging Pump suction) will close; however, the Charging Pump suction from the RWST will open. With any Charging Pump flow path, the Charging Pumps will not trip on low suction pressure.

Plausible: The examinee may believe that the Charging Pump suction pressure trip is prevented due to the higher pressure from the VCT than the RWST or that the Charging Pump suction flow path is isolated when CH-501 closes.

B is wrong; CH-512 will NOT automatically open on a VCT lo-lo level.

Plausible: With the Makeup Controls aligned for automatic makeup, CH-512 will automatically open when the VCT reaches the low level auto makeup setpoint; however, this is from a different level transmitter. The examinee may not realize that these two functions are controlled by two different level transmitters.

C is wrong; CH-514, Charging Pump suction from the Boric Acid Storage Tanks, will NOT open on a lo-lo level in the VCT.

Plausible: The examinee believe that it's logical for the Charging Pumps to take a suction from the Boric Acid Storage Tanks on a lo-lo level in the VCT. The Boric Acid Tanks are aligned to the Charging Pump suctions during a normal blended makeup; therefore, it would be logical to assume the suction flow path is the same for a lo-lo level in the VCT.

D is correct; at 8% decreasing, LC-227 sends a signal to close the VCT outlet valve, 2-CH-501, and open the Charging Pump suction flow path to the RWST; 2-CH-504, RWST to Charging Suction (normally open), and 2-CH-192, RWST Isolation.

Question #15

The plant tripped from 100% power.

The following alarms are locked in:

- RSST LOCKOUT CHANNEL 1 (C-08, A-3).
- DIESEL GEN 13U BKR DIFFERENTIAL LOCKOUT (C-08, D-36).

When compared to an uncomplicated trip,

Battery 201A discharge rate is 1. for this trip.

AND

Battery 201B discharge rate is 2. for this trip.

- A. 1. the same
2. the same
- B. 1. the same
2. higher
- C. 1. higher
2. the same
- D. 1. higher
2. higher

Question #15

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 058AK1.03

Importance Rating: 3.7

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to LOSS OF DC Power: Effect of battery discharge rate on capacity

Proposed Answer: B

Justification: The RSST LOCKOUT CHANNEL 1 indicates that the RSST and North Bus are isolated and the DIESEL GEN 13U BKR DIFFERENTIAL LOCKOUT shows that the 'B' EDG is TRIPPED, not powering the 'B' vital bus. With these conditions present, the 201B Battery Charger is NOT powered while the 201A Charger IS powered. Normally, both Chargers would be powered after a trip. As such, the 201A battery discharge rate is the same as an uncomplicated trip while the 201B battery discharge rate is higher.

Plausibility:

- A. Battery 201B Discharge rate is higher for this trip. With RSST LOCKOUT combined with DG 13U BKR DIFF LOCKOUT, Battery charger 201B is not being powered. Without the battery charger, the DC loads on 125 VDC Bus 201B are powered by the battery, resulting in a higher discharge rate. Plausible; May not realize that this combination of conditions results in no power being supplied to the battery charger 201B.
- B. Correct.
- C. This is the reverse of the correct answer. See justification for details. Plausible; candidate may confuse the numbering nomenclature for the 'A' (12U) and 'B' (13U) EGD. Normally facility 2 components are even numbers while facility 1 components are odd numbers. In this case it is reversed.
- D. Power is only lost to the Battery Charger 201B, not 201A. Plausible; Candidate may think both chargers are lost due to the RSST LOCKOUT

Technical Reference(s):

- ARP 2590F-009, RSST LOCKOUT CHANNEL 1
- ARP 2590F-140, DIESEL GEN 13U BKR DIFFERENTIAL LOCKOUT
- LVD-00-C, 125VDC/120VAC Lesson plan
- Vision Question 525996

Provided reference(s): None

Learning Objective: 281636 - Given a plant condition, and a sequence of indications, alarms or malfunctions; predict the effect on the following plant systems and/or plant equipment: B) 4160 VAC, D) 120 VAC, and E) 125 VDC.

Source: Bank Vision # 525996 2022 NRC Exam #48

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(8): Safely operate the facilities' auxiliary and emergency systems, including operation of those controls associated with plant equipment that could affect reactivity or the release of radioactive materials to the environment.

Comments:

Question #16

The plant is in MODE 6.

- The 'A' Service Water header is secured.
- The 'A' Service Water pump is available.
- 'B' Service Water pump is NOT available.
- 'B' LPSI pump is providing SDC flow.

Subsequently,

- The 'C' Service Water pump trips.
- Annunciator 'SW PUMP C OVERLOAD/TRIP' (C-06/7, BA-4) is locked in.

What procedure provides the guidance to mitigate this situation?

- A. OP 2326A, Service Water System.
- B. OP 2326C, Off-Normal Service Water System Operations.
- C. AOP 2565, Loss of Service Water.
- D. AOP 2585, Immediate Operator Actions.

Question #16

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 062AK3.03

Importance Rating: 3.9

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to LOSS OF SERVICE WATER: Guidance actions contained in AOPs for loss of service water

Proposed Answer: C

Justification: AOP 2565, Loss of Service Water, Section 10 provides guidance for a loss of a SW pump when only one SW header is in operation.

Plausibility:

- A. OP 2326 does not have actions to supply the 'B' SW header from the 'A' SW pump.
- B. OP 2326C is only applicable to MODEs 1-4. The examinee may forget what MODEs the procedure is used in and figure a loss of a service water header is 'off-normal'.
- C. Correct.
- D. The AOP 2585 does not have actions for a loss of a SW pump (only loss of RB pump). The examinee may think the procedure has actions for mitigating the loss of a SW header.

Technical Reference(s):

- AOP 2565, Loss of Service Water
- Vision Question #477374

Provided reference(s): None

Learning Objective: 283250 – Given a set of plant conditions, determine AOP 2565, Loss of Service Water, applicability.

Source: Modified Vision question #477374

Cognitive Level: Memory

10CFR55.41(b)(8): Safely operate the facilities' auxiliary and emergency systems, including operation of those controls associated with plant equipment that could affect reactivity or the release of radioactive materials to the environment.

Comments: Removed second part of question (SRO level with 2nd part in). Cog level changed to memory. Changes made based on NRC phone call 7/25. DF 7/26

The plant is in MODE 5 with the following conditions:

- Day 23 of a 28 day refueling outage.
- "B" Service Water (SW) Pump is tagged out for maintenance.
- "A" SW Pump was just secured in preparation for inspecting the "A" SW Header with an underwater camera.
- "B" SW header is in operation.

Then, the "C" SW Pump suddenly trips on overload.

Which one of the following actions must be taken?

- A. Secure the running LPSI Pump, enter AOP 2572, Loss of Shutdown Cooling, and immediately initiate gravity feed makeup from the RWST.
- B. Emergency trip both EDGs and use the guidance in AOP 2583, Loss of All AC Power During Shutdown Condition, to align Fire Water to one EDG.
- C. Per 2326C, Off-Normal Service Water System Operations, align and start the "A" Service Water Pump to supply both Service Water Headers.
- D. Per AOP 2565, Loss of Service Water, open SW-97A and SW-97B, Service Water header Crosstie Valves, and start the "A" Service Water Pump.

Answer: D

Answer Explanation
A is incorrect. Although SDC is technically lost when Service Water is lost, in this case Service Water can be restored.
B is incorrect. No need to trip Diesel Generators; Service Water can be restored. The primary concern is cooling for SDC.
C is incorrect. The referenced procedure provides the appropriate guidance; however, it only applies to MODE 1-4.
D is correct. AOP 2565, Loss of Service Water, Section 10.0 provides the appropriate guidance for a loss of Service Water in this configuration.

Question #17

The plant is operating at 100%

- Annunciator INST AIR HDR PRES LO (C06/7, CA-1) alarms.
- PI 7079, IA AIR PRESSURE = 75 psig and lowering.
- Letdown isolates unexpectedly.

In accordance with AOP 2563, Loss of Instrument Air, what actions are required to be performed?

- A. Place Charging Pump handswitches in Pull-To-Lock.
- B. Trip the Reactor AND GO TO EOP 2525.
- C. Trip the Reactor AND secure all RCPs.
- D. Direct a PEO to align backup air from Unit 3.

Question #17

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # APE 65 G2.2.44

Importance Rating: 4.2

K/A Statement: Loss of Instrument Air EQUIPMENT CONTROL: Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions

Proposed Answer: B

Justification: ARP 2590E-005 for INST AIR HDRPRES LO alarm directs operator to AOP 2563, Loss of Instrument Air. AOP 2563 Reactor trip criteria states: IF both of the following are met (1) Instrument Air Header Pressure – Remains less than or equal to 80 psig C-06, PPC and (2) Air Operated Valves inappropriately or unexpectedly changing position, then TRIP the reactor. Loss of letdown was unexpected in this scenario so it must be due to the loss of Instrument Air

Plausibility:

- A. Although the loss of Letdown immediate actions requires securing charging, a reactor trip is required which will require charging during the trip recovery. Plausible; Securing charging is an Immediate Action for a loss of letdown, not a loss of instrument air.
- B. Correct.
- C. Tripping the RCPs is not required directly for a loss of Instrument Air. Plausible; May determine a loss of RCP Bleed off would require securing the RCPs following a Reactor Trip. The stem says nothing about losing RCP Bleed Off capability.
- D. Aligning an alternate source of Instrument Air is not recommended for a faulted air system that may be ruptured. Plausible; Candidate may confuse with actions required for a loss of Instrument Air in EOP 2525, Standard Post Trip Actions, for a loss of Instrument Air Compressors or a loss of power to compressors NOT for a faulted Instrument Air header.

Technical Reference(s):

- MP-PROC-OPS-ARP 2590E-005, INST AIR HDR PRES LO, REV. 000.
- MP-PROC-OPS-AOP 2563, Loss of Instrument Air, REV. 011.
- MP-PROC-OPS-EOP 2525, Standard Post Trip Actions, REV. 029.
- Vision Question 522536

Provided reference(s): None

Learning Objective: Given plant conditions and/or Station Air and Instrument Air Systems parameter values, determine if Station Air and Instrument Air Systems operating parameters meet entry conditions for EOPs and/or AOPs.

Source: Bank Vision #522536

Cognitive Level: Comprehension or Analysis

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Changed correct answer to announce loss of Instrument Air. Announcing loss of letdown is not directed in AOP 2563. Capitalized AND in B & C due to feedback from validator. DF 5/28. Changed correct answer → got rid of “announce Loss of IA and replaced it with GO TO EOP 2525 DF 7/8

Question #18

The crew has entered AOP 2580, Degraded Voltage.

The procedure directs tripping the unit when the 4.16 kV buses reach 1. in order to prevent safety-related pumps and motors from 2..

- A.
 - 1. 3700 VAC (88% nominal voltage)
 - 2. tripping on overcurrent.
- B.
 - 1. 3700 VAC (88% nominal voltage)
 - 2. overheating their windings.
- C.
 - 1. 2912 VAC (70% nominal voltage)
 - 2. tripping on overcurrent.
- D.
 - 1. 2912 VAC (70% nominal voltage)
 - 2. overheating their windings.

Question #18

☒ RO

☐ SRO

Tier # 1

Group # 1

K/A # 077AK3.02

Importance Rating: 3.5

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to GENERATOR VOLTAGE AND ELECTRIC Grid Disturbances:
Actions contained in AOPs for voltage and grid disturbances

Proposed Answer: B

Justification: The AOP directs a plant trip if safety bus voltage remains at 3700 VAC (after procedural actions are taken) to prevent motor windings from overheating due to increased current.

Plausibility:

- A. The examinee may think the increased current due to lowering voltage will cause the pumps to trip on overcurrent.
- B. Correct.
- C. The examinee may confuse the directed trip setpoint with the LNP setpoint; also they may think the increased current due to lowering voltage will cause the pumps to trip on overcurrent.
- D. The examinee may confuse the directed trip setpoint with the LNP setpoint

Technical Reference(s):

- AOP 2580, Degraded Voltage
- Vision question 452226

Provided reference(s): None

Learning Objective: 283479 - Determine the effects of a degraded voltage condition on plant components.

Source: Bank #452226 NRC 2016

Cognitive Level: Memory

10CFR55.41(b)(4): Secondary coolant and auxiliary systems that affect the facility.

Comments:

Question #19

The following conditions are observed while operating at 100% power:

- $T_{\text{COLD}} = 543$ °F and lowering.
- CET High is 567 °F and lowering.
- NI Power channels read as follows:
 - Channel A = 93.4 %.
 - Channel B = 93.0 %.
 - Channel C = 99.2 %.
 - Channel D = 96.5 %.

What would cause these indications?

- A. Xenon oscillation
- B. Dropped CEA
- C. Inadvertent boration
- D. Open Atmospheric Dump Valve

Question #19

☒ RO

☐ SRO

Tier # 1

Group # 2

K/A # 003AA2.08

Importance Rating: 2.7

K/A Statement: Ability to determine and/or interpret the following as they apply to DROPPED
Control Rod: ITM

Proposed Answer: B

Justification: A dropped CEA will add negative reactivity. RCS Temperature will lower to add positive reactivity to compensate. The core flux will be suppressed in the quadrant with the dropped rod, and flux will be higher in the other three quadrants to compensate.

Plausibility:

- A. A xenon oscillation would affect all four quadrants of power and temperature. Plausible; Candidate may think that a xenon oscillation can oscillate from a single quadrant to the other three quadrants.
- B. Correct.
- C. An inadvertent boron injection would affect all 4 quadrants of power and temperature. Plausible; Candidate may not consider the mixing effect of the RCS coolant as it enters the core and think the injection only enters the quadrant nearest to where the injection leg is connected to the vessel.
- D. An open ADV would result in lower temperature and higher power in all quadrants, not just 3 out of 4 quadrants. Plausible; Candidate may think that an ADV open on a single SG may effect a single quadrant where that loops cold leg is attached to the core.

Technical Reference(s):

- MP-PROC-OPS-AOP 2556 REV. 022, CEA Malfunctions.
- MP-PROC-OPS-ARP 2590C REV. 000, CA-19 CEA DROPPED ROD SW.
- Waterford 3 2018 RO exam Q-20

Provided reference(s): None

Learning Objective: 283045 – Given a set of plant conditions, determine AOP 2556, CEA Malfunctions, applicability.

Source: Modified (Waterford 3 2018 RO 20)

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects

Comments: Rewrote initial conditions, changed T cold to 543 and lowering, removed CET bullet (no value added) based on 7/25 NRC phone call. DF 7/26

**2018 NRC Exam
RO Written Exam Worksheet**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000003 AA2.03	
	Importance Rating	3.6	

K/A Statement

000003 Dropped Control Rod AA2.03 Ability to determine and interpret the following as they apply to the Dropped Control Rod: **Dropped rod, using in-core/ex-core instrumentation, in-core or loop temperature measurements**

Proposed Question: RO 20 Rev: 0

Given:

- Plant power ascension to 100% is in progress
- The following conditions are observed:
 - T_{ave} has lowered 1.5°F
 - Core Exit thermocouple indication has lowered by 10-13°F near Loop 2
 - CPC B power indication is 63%
 - CPC A, C and D power indication is 74%

Which of the following describes the cause of these indications?

- A. Xenon oscillation
- B. Dropped CEA
- C. Inadvertent boration
- D. Open Atmospheric Dump Valve

**2018 NRC Exam
RO Written Exam Worksheet**

Proposed Answer: B

Explanation: (Optional)

- A. Incorrect. A xenon oscillation would affect all 4 quadrants of power and temperature.
B. **CORRECT:** Temp will lower to compensate for – reactivity added by dropped CEA, adding + reactivity. The – reactivity will show up in one core quadrant, and the other 3 will rise to compensate as well.
C. Incorrect. An inadvertent boron injection would affect all 4 quadrants of power and temperature.
D. Incorrect. Would result in lower temperature and higher power in all quadrants, not just 3 of 4.

Technical Reference(s):
(Attach if not previously provided)
(including version/revision number)

WLP-OPS-TYR10 Rev 12, slides 181 & 182
WLP-OPS-TYR05 Rev 5, slides 121 & 123
OP-901-102 Rev 304, page 5
OP-002-005 Rev 61, page 23
WLP-OPS-MS00 Rev 30, slide 266

Proposed references to be provided
to applicants during examination: None

Learning Objective: WLP-OPS-TYR05 Rev 5, EO-10 (As available)

Question Source: Bank # RO57
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2008 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis 2

10 CFR Part 55 Content: 55.41 2
55.43

Comments:

Question #20

The plant is performing an up-power.

- Reactor power = 35% and increasing.

Subsequently,

- Annunciator NIS CHANNEL DEVIATION HI (C-04, BA-12) alarms.
- Channel D Nuclear Instrumentation 'HI DEV' LED is lit.

What would cause this condition?

- A. CEAs are positioned too low for the power level.
- B. CEAs are positioned too high for the power level.
- C. A Turbine Control valve is stuck closed.
- D. A CEA has separated from its extension shaft.

Question #20

☒ RO

☐ SRO

Tier # 1

Group # 2

K/A # 005AK2.07

Importance Rating: 3.9

K/A Statement: Knowledge of the relationship between INOPERABLE/STUCK Control Rod and the following systems or components: NIS

Proposed Answer: D

Justification: The combination of the alarm and the 'HI DEV' LED would indicate that the 'D' nuclear instrumentation safety channel has deviated from the NI Grand Average by ~ 4%. A CEA separated from its extension shaft would allow the extension shaft would move with the CEA would remaining on the bottom of the core, depressing power in its quadrant.

Plausibility:

- A. The examinee may think the CEAs being too low for the power level would push ASI to the bottom of the core. They could think the 'D' HI DEV LED is just the first alarm to activate.
- B. The examinee may think the CEAs being too high for the power level would push ASI to the top of the core. They could think the 'D' HI DEV LED is just the first alarm to activate.
- C. The examinee may think a turbine control valve being closed would cause an axial imbalance in the core. Since the control valves are downstream of the steam chest, closing one would not have an upstream impact.
- D. Correct.

Technical Reference(s):

- Nuclear Instrumentation Lesson Plan, NIS-00-C

Provided reference(s): None

Learning Objective: 281834 – State the purpose and describe the operation of the following major Nuclear Instrumentation components:
G) Comparator Averager

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Question #21

The plant is at 100% power.

- Pressurizer level channel Y is the controlling channel.
- LIC-110X fails high.
- LIC-110Y is trending down.

- AOP 2568, RCS Leak is entered.
- The 'A' Charging Pump is started.
- Letdown flow = 80 gpm and stable
- Pressurizer level = 64% and stable
- Containment temperature and pressure are trending up.

Based on these indications, the leak is on the 1 leg of the Channel X instrument line.

AND

Based on the size of the leak, AOP 2568, RCS Leak, requires you to 2.

- A.
 - 1. Reference
 - 2. Trip the reactor and go to EOP 2525, Standard Post Trip Actions.
- B.
 - 1. Reference
 - 2. Initiate shutdown per AOP 2575, Rapid Down power.
- C.
 - 1. Variable
 - 2. Trip the reactor and go to EOP 2525, Standard Post Trip Actions.
- D.
 - 1. Variable
 - 2. Initiate shutdown per AOP 2575, Rapid Down power.

Question #21

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 028AK1.01

Importance Rating: 3.5

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to PRESSURIZER (PZR) Level Control Malfunction: PZR reference leg leak abnormalities

Proposed Answer: B

Justification: Pressurizer reference leg break is an RCS leak and fails the indication high. Channel X is pegged high, but channel Y is controlling, so PLCS responds normally based on channel Y. Per AOP 2568 RCS Leak, step 4, unidentified leakage is greater than 1 gpm. Refer to TS LCO 3.4.6.2. Step 17 sends operators to AOP 2575 for the downpower and OP 2207 for a cooldown to MODE 5.

Plausibility:

- A. Variable leg causes low detector failure. Plausible; applicant may confuse detector response with failure of the variable leg. Per AOP 2568 foldout page, if leak exceeds charging capacity, then trip the reactor. The stem states that the starting of the second charging pump has stabilized level.
- B. Correct.
- C. Variable leg causes low detector failure. Plausible; applicant may confuse detector response with failure of the variable leg.
- D. Given parameters do not require a Rx Trip and EOP 2525 entry. Plausible; Per AOP 2568 foldout page, if leak exceeds charging capacity, then trip the reactor. The stem states that the starting of the second charging pump has stabilized level.

Technical Reference(s):

- AOP 2568, Reactor Coolant System Leak.
- TS LCO 3.4.6.2, REACTOR COOLANT LEAKAGE
- Vision Question #414727

Provided reference(s): None

Learning Objective: 283281 - Given a set of plant conditions indicative of an RCS leak, determine the most likely location of leakage.

Source: Bank Vision #414727 2015 NRC Q76

Cognitive Level: Comprehension or Analysis

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Question #22

The plant is in MODE 6 with a full core offload in progress.

- There is no fuel assembly attached to the refuel machine.
- 'D' Channel of nuclear instrumentation is NOT OPERABLE.
- Audible Count Rate (RC05E) is selected to Channel 'C'.

Subsequently,

- The plant experiences a loss of VA-30.

How does the loss of VA-30 impact the core offload?

- A. ALL fuel handling (CTMT & SFP) must be immediately suspended.
- B. Fuel can still be offloaded from, but NOT loaded into the core.
- C. Fuel offload from the core must immediately be suspended.
- D. ALL fuel handling (CTMT & SFP) may continue.

Question #22

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 032AK3.02

Importance Rating: 3.5

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to LOSS OF SOURCE RANGE Nuclear Instrumentation: Guidance contained in procedures for loss of source range NI

Proposed Answer: C

Justification: TS 3.9.2 requires 2 OPERABLE source range neutron detector channels and one with audible indication in CTMT and Control Room. The loss of VA-30 still leaves 2 OPERABLE source range channels but the audible indication is lost (along with the third channel). Without audible counts, CORE ALTERATIONS are immediately suspended

Plausibility:

- A. The examinee may think all fuel handling is suspended. Fuel movement is still allowed in the Spent Fuel Pool and refueling cavity.
- B. The examinee may think that since fuel is being removed from the core and there are two OPERABLE source range channels, a conservative action would be to continue removing fuel from the core (removing reactivity).
- C. Correct.
- D. The examinee may think with two OPERABLE source range channels, all fuel movement, including CORE ALTERATIONS, is permissible without audible counts.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications 3.9.2, INSTRUMENTATION
- OP 2209, Refueling Operations

Provided reference(s): None

Learning Objective: 281829 - Given any operating condition for the Nuclear Instrumentation System, state whether the condition requires entry into the Technical Specifications

Source: Bank Vision #452191

Cognitive Level: Comprehension

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: 725 phone call. Added bullet to show no fuel assembly attached to refuel machine. DF 7/26

Question #23

The plant is operating at 100% power.

- Main Condenser backpressure = 3.0 “HgA and slowly degrading.
- ‘A’ Circulating water pump is running at 100% speed.
- ‘B’, ‘C’, and ‘D’ Circulating water pumps are running in ‘BYPASS’.

For the given conditions, what action does AOP 2574, Loss of Condenser Vacuum, direct to improve condenser vacuum?

- A. Raise circulating water pump speeds.
- B. Start mechanical vacuum pumps.
- C. Raise gland seal steam pressure.
- D. Lower reactor power and main turbine load.

Question #23

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # APE 51 G2.1.20

Importance Rating: 4.6

K/A Statement: Loss of Condenser Vacuum CONDUCT OF OPERATIONS: Ability to interpret and execute procedure steps

Proposed Answer: D

Justification: Lower reactor power and main turbine load is correct. AOP 2574, "Loss of Condenser Vacuum", states 'if at any time efforts to restore vacuum are not successful, refer to AOP 2575, "Rapid Downpower", and LOWER reactor power and turbine load at the maximum attainable rate until pressure stabilizes.

Plausibility:

- A. Raise circulating water pump speeds is not correct since in the "Bypass" mode of operation the circulating water pump can only be at 100% speed. Raising speed is not possible. Plausible; AOP 2574, "loss of condenser vacuum", under the heat transfer section checks the circulating water system. The contingency actions for this section refers to AOP 2517, "Circulating Water Malfunctions". This procedure has you raise speeds of all circulating water pumps operating in the VFD mode.
- B. Start mechanical vacuum pumps is not correct since starting the pumps would initially make vacuum get worse. AOP 2574, Loss of condenser vacuum, directs the start of the mechanical vacuum pumps if condenser backpressure is greater than 4.0 inches. Plausible; AOP 2574, Loss of condenser vacuum, does direct the start of the mechanical vacuum pumps if condenser backpressure is greater than 4.0 inches. But if the mechanical vacuum pumps were started vacuum would get at worse and at best stabilize at 4.0 inches..
- C. Raise gland seal steam pressure is not correct. Gland seal pressure cannot be raised when the turbine is at greater than 10% power since the turbine glands are self-supplying and the main steam supply closes. Gland seal pressure is maintained by unloader valves that dump excess steam to a feedwater heater. Plausible; AOP 2574, loss of condenser vacuum does have the operator check steam seal pressure and at lower loads the steam seal bypass is used to raise gland seal pressure.

D. Correct.

Technical Reference(s):

- AOP 2574, Loss of Vacuum
- CWS-00-C, Circulating Water Lesson Plan
- Vision Question 452008
- Reference 3

Provided reference(s): None

Learning Objective: 283397 – Outline the major actions for AOP 2574, Loss of Condenser Vacuum.

Source: Bank Vision Q#452008 2018 NRC Q22

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(4): Secondary coolant and auxiliary systems that affect the facility.

Comments: 7/25 phone call. Added clarifier “for the given conditions” to rule out starting

Mechanical Vacuum pumps. Changed verbiage to read ‘backpressure’ and ‘degrading’ DF 7/26

Question #24

Annunciator, FIRE SYSTEM TROUBLE (C-06/7, AB19), is in alarm.

Zone 43, EAST DC SWGR ROOM FLP-5, is illuminated on the Fire Panel.

A PEO reports the following from FLP-5:

- Strobe lights and horn are pulsating slowly.
- 1 (one) smoke detector is in alarm.

What is the status of the East DC Switchgear Room Halon system?

- A. The Halon is not yet being discharged into the Switchgear Room.
- B. Halon is being discharged into the Switchgear Room.
- C. The system has completed discharging into the Switchgear Room.
- D. The system is NOT available to discharge into the Switchgear Room.

Question #24

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 067AA1.06

Importance Rating: 3.3

K/A Statement: Ability to operate and/or monitor the following as they apply to PLANT Fire On
Site: Fire alarm

Proposed Answer: B

Justification: The activation of ONE smoke detector (either ion or photoelectric) will cause the system's horn and strobe lights to pulse slowly. The activation of a 2nd detector of an opposite type as the first will cause the indicators to pulse rapidly for 1 minute prior to discharging into the room.

Plausibility:

- A. Correct.
- B. The system's strobe lights and horns will sound/light steadily once the Halon is being discharged into the room. The system status will NOT change after the Halon stops discharging into the room. The student may think one detector is enough to cause the Halon to discharge.
- C. The system's strobe lights and horns will sound/light steadily once the Halon is being discharged into the room. The system status will NOT change after the Halon stops discharging into the room. The student may think the indicators pulse slowly AFTER the Halon discharge is complete.
- D. The system has no indicator for being not able to discharge into the room. The student may think the system is failed and that the strobe lights and horns pulsate quickly when a smoke detector activates

Technical Reference(s):

- ARP 2590I, Alarm Response for Fire Panel, C-26, ZONE 45
- Vision Question 418919

Provided reference(s): None

Learning Objective: 287608 - State the purpose and describe the operating characteristics of the following major Fire Protection system components: G) Local Fire Detection Panels

Source: Bank #418919

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: 7/25 phone call, replaced question. DF 7/26

Question #25

Unit 2 has experienced a SIAS due to high Containment pressure.

- Facility 2 CIAS failed to actuate.
- Facility 2 CIAS operated valves did NOT close.

Which one of the following describes...

1. an additional malfunction that would prevent an isolation of a penetration

AND

2. The specific ESAS signal, and the valve that is actuated by that signal, to prevent the additional malfunction.
-
- A.
 1. CH-515, Reactor Coolant Letdown Valve, is open.
 2. Facility 1 SIAS should have closed CH-515.
 - B.
 1. P-33B, CTMT Sump Pump, is running with SSP-16.1, CTMT Sump Isolation Valve, open.
 2. Facility 1 CIAS should have closed SSP-16.1.
 - C.
 1. CH-516, Reactor Coolant Letdown Valve, is open.
 2. Facility 1 CIAS should have closed CH-516.
 - D.
 1. P-33B, CTMT Sump Pump, is running with SSP-16.2, CTMT Sump Isolation Valve, open.
 2. Facility 1 SIAS should have closed SSP-16.2.

Question #25

☒ RO

☐ SRO

Tier # 1

Group # 2

K/A # 069AA1.01

Importance Rating: 4.0

K/A Statement: Ability to operate and/or monitor the following as they apply to LOSS OF Containment Integrity: Containment isolation valves, dampers, and/or other devices

Proposed Answer: A

Justification: CH-515 Letdown Loop Isolation Valve, closed by FAC 1 SIAS, is in series with Inboard Containment Letdown Isolation CH-516 and Outboard Containment Letdown Isolation CH-089. Both CH-516 and CH-089 receive FAC 2 CIAS to close. CH-515 therefore provides the redundant FAC 1 containment isolation function for the letdown line penetration.

Plausibility:

- A. Correct.
- B. P-33B receives a stop signal on a FAC 2 CIAS. SSP-16.1 receives a close signal on a FAC 2 CIAS. This choice is incorrect because the listed components receive a FAC 2, not a FAC 1, CIAS. The redundant valve, SSP-16.2, which receives a FAC 1 CIAS, would have closed, isolating the penetration. Plausible; Pumping down the containment sump during a LOCA is a loss of containment integrity. Examinee may think FAC 1 CAS closes SSP-16.1.
- C. CH-515, not CH-516, will close on FAC 1 SIAS as the redundant containment isolation to CH-516 and CH-089 isolation valves. Also, the isolation signal for CH-516 is FAC 2 CIAS. Plausible; Examinee may think CH-516 is the redundant containment isolation for this penetration and that it receives a FAC 1 CIAS.
- D. P-33B receives a stop signal on a FAC 2 CIAS. SSP-16.2 receives a close signal on a FAC 1 CIAS. This alignment (pump running, SSP-16.2 open), in conjunction with non-functioning FAC 2 CIAS given in the stem would result in a loss of containment integrity. However, the choice is incorrect because SSP-16.2 receives a CIAS, not a SIAS close signal. Plausible; Pumping down the containment sump during a LOCA is a loss of containment integrity. Examinee may recognize the alignment as a loss of integrity but may incorrectly think SSP-16.2 receives a SIAS rather than a CIAS close signal.

Technical Reference(s):

- LP ESA-01-C. R4C2, Engineering Safety Features Actuation System.
- Millstone Unit 2 TRM Table 3.6-1
- Vision Question #451909

Provided reference(s): None

Learning Objective: 281374 - Describe the potential effects of a loss or malfunction of the Engineered Safety Features Actuation System (ESAS) on the following: Containment.

Source: Bank Vision #451909 2015 NRC Exam Q25

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(3): Mechanical components and design features of the reactor primary system.

Comments:

Question #26

Millstone 2 is analyzed as an 8 hour coping plant. This means that the plant can cope with a Station Blackout if ____1.____ is restored to maintain the ____2.____ Safety Function.

- A. 1. Charging
 2. Reactivity Control
- B. 1. Charging
 2. RCS Inventory Control
- C. 1. CST Make-up
 2. RCS Heat Removal
- D. 1. DC Room Ventilation
 2. Maintenance of Vital Auxiliaries

Question #26

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # CE13EK3.10

Importance Rating: 3.8

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to LOSS OF FORCED CIRCULATION AND/OR LOOP AND/OR A BLACKOUT: Restoring offsite power

Proposed Answer: B

Justification: An 8 hour coping plant requires charging be restored within 1 hour to prevent the core from being uncovered after 8 hours.

Plausibility:

- A. The examinee may think that the Reactivity safety function is challenged during a plant cooldown during a Station Blackout.
- B. Correct.
- C. The examinee may think the CST is rated for maintaining the plant in Hot Standby for 8 hours after a Station Blackout.
- D. The examinee may think that the procedural requirement to align supplemental cooling within 30 minutes is to allow for the plant to cope with the Station Blackout.

Technical Reference(s):

- Station Blackout Lesson Plan, E30-01-C
- Vision Question #526198

Provided reference(s): None

Learning Objective: 283759 - Predict how operator action or inaction affects plant and system conditions concerning a Station Blackout.

Source: Modified Vision # 526198

Cognitive Level: Memory

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The plant has experienced a Station Blackout (SBO). The Turbine Driven Auxiliary Feedwater pump is feeding both Steam Generators at 300 gpm each.

1. Millstone Unit 2 is analyzed to cope with a Station Blackout (SBO) for 1. .

AND

2. A 2. pump must be started, within one hour, to ensure the core will remain covered for the duration of the coping time.

- A. 1. 8 hours
 2. HPSI
- B. 1. 8 hours
 2. Charging
- C. 1. 10 hours
 2. HPSI
- D. 1. 10 hours
 2. Charging

Answer: B

Answer Explanation

Question #27

When securing a Reactor Coolant Pump (RCP)...

The anti-reverse rotation device 1. ,

AND

The purpose of the anti-reverse rotation device is to prevent 2..

- A.
 - 1. must be manually engaged
 - 2. primary coolant backflow through the RCP
- B.
 - 1. will automatically engage
 - 2. primary coolant backflow through the RCP
- C.
 - 1. must be manually engaged
 - 2. excessive motor current on RCP start
- D.
 - 1. will automatically engage
 - 2. excessive motor current on RCP start

Question #27

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 003K6.08

Importance Rating: 2.8

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR COOLANT PUMP SYSTEM:
Anti-reverse rotation device

Proposed Answer: D

Justification: The anti-rotation device automatically engages when securing the RCP. The purpose of the anti-rotation device is to prevent excessive starting current surge in the RCP motor that would occur if the RCP was rotating in the reverse direction and an RCP pump start were attempted.

Plausibility:

- A. WRONG; The anti-reverse rotation device automatically engages when RCP is secured. The purpose is to prevent reverse rotation to limit starting current surge. Plausible; The candidate may think that preventing reverse rotation of the RCP will prevent backflow through the pump and ensure more flow goes through the core, which is not true. There will be backflow through the RCP when it is secured regardless of the status of reverse rotation.
- B. WRONG; The purpose is to prevent reverse rotation to limit starting current surge. Plausible; The candidate may think that preventing reverse rotation of the RCP will prevent backflow through the pump and ensure more flow goes through the core, which is not true. There will be backflow through the RCP when it is secured regardless of the status of reverse rotation.
- C. WRONG; The anti-reverse rotation device automatically engages when RCP is secured. Plausible; candidate may believe the anti-rotation device must be manually engaged like some turning gears in the plant that must be engaged when a piece of equipment is secured.
- D. Correct.

Technical Reference(s):

- ARP 2590C-013 REV. 000, RC LO FLOW TRIP CH A
- ARP 2590B-073 REV. 000, AA-18 RCP A REV ROTATION.
- RCS-00-C REV. 009 CH. 003, Reactor Coolant System Lesson Plan.
- 2022 Palo Verde RO Q36.

Provided reference(s): None

Learning Objective: 281983 Describe the purpose of each of the following major RCS components:

A) RCP Anti-Reverse Device.

Source: Modified 2022 Palo Verde RO Q36

Cognitive Level: Memory

10CFR55.41(b) (3) Mechanical components and design features of the reactor primary system.

Source: Palo Verde 2022 NRC exam Q36

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Coolant Pump: Knowledge of Reactor Coolant Pump System design features and/or interlocks that provide for the following: Prevention of reverse rotation	Tier	2		
	Group	1		
	K/A	003 K4.05		
	IR	3.0		

Question 36

When starting a Reactor Coolant Pump (RCP), the Anti-Reverse Rotation Device (ARRD) ____ (1) ____ and when an RCP is stopped, the AARD ____ (2) ____ .

- A. (1) will automatically disengage
(2) will automatically engage
- B. (1) will automatically disengage
(2) must be manually engaged from the Control Room
- C. (1) must be manually disengaged from the Control Room
(2) will automatically engage
- D. (1) must be manually disengaged from the Control Room
(2) must be manually engaged from the Control Room

Comments:

Question #28

The plant is operating at 100% power.

- VCT level starts trending down.

The lowering trend can be due to a tube leak in the 1..

AND

The leak location can be verified within minutes by observing an increasing...

- A.
 - 1. Regenerative Heat Exchanger
 - 2. Charging temperature (TI-229).
- B.
 - 1. Regenerative Heat Exchanger
 - 2. Letdown temperature (TI-221).
- C.
 - 1. Letdown Heat Exchanger
 - 2. 'A' RBCCW Header activity.
- D.
 - 1. Letdown Heat Exchanger
 - 2. 'B' RBCCW Header activity.

Question #28

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 004K3.03

Importance Rating: 2.7

K/A Statement: Knowledge of the effect that a loss or malfunction of the CHEMICAL AND VOLUME CONTROL SYSTEM will have on the following systems or system parameters: CCWS

Proposed Answer: D

Justification: In the Letdown Heat Exchanger, the RCS pressure (~340 psig) is greater than RBCCW pressure (~120 psig). A tube leak would introduce RCS into the RBCCW system causing RBCCW activity to increase. The Letdown Hx is cooled by the 'B' RBCCW header. Although the RBCCW system has only one radiation monitor, AOP 2568, RCS Leak, directs the operator to have Chemistry sample the RBCCW headers if the RM is NOT reading normal values.

Plausibility:

- A. Plausible if the examinee thinks of the leak only by the leak causing fluid to leave a system (not where it's going to). Also, depending on the location of the leak (near the charging inlet), charging temperature would increase.
- B. Plausible if the examinee thinks of the leak only by the leak causing fluid to leave a system (not where it's going to). Also, depending on the location of the leak (near the letdown inlet), letdown temperature could increase.
- C. The examinee may forget what RBCCW header cools the Letdown Heat Exchanger
- D. Correct.

Technical Reference(s):

- RBCCW Lesson Plan RBC-00-C

Provided reference(s): None

Learning Objective: 281942 - ILT Given indications of leakage into the RBCCW System, identify the possible sources.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments:

Question #29

The plant is performing a cooldown in accordance with OP 2207, Plant Cooldown.

- Section 4.18, Initiating SDC With Concurrent RCP Operation, is in progress.
- The crew is preparing to throttle open LPSI Injection valves to initiate SDC flow.

OP 2207, Plant Cooldown directs...

Components of the SDC system must not be exposed to RCS temperature greater than 1. ,

AND

WHEN “SDC to RCS temperature, T351Y” is within 2. of operating Loop T_{HOT},
INITIATE SDC flow.

- A. 1. 265 °F
2. 20 °F
- B. 1. 265 °F
2. 48 °F
- C. 1. 300 °F
2. 20 °F
- D. 1. 300 °F
2. 48 °F

Question #29

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 005A1.09

Importance Rating: 3.9

K/A Statement: Ability to monitor automatic features of the RESIDUAL HEAT REMOVAL SYSTEM, including: RCS temperature

Proposed Answer: C

Justification: 1. OP 2207, Section 4.18 caution statement: Components of the SDC system must not be exposed to either RCS pressure greater than 265 psia or RCS temperature greater than 300 °F.
2. OP 2207, Step 4.18.22 states: WHEN “SDC to RCS temperature, T351Y” is within 20 °F of operating Loop T_{HOT}, PERFORM the following: THROTTLE open each LPSI injection valve to establish between 800 and 1,000 gpm flow on each injection header.

Plausibility:

- A. Components of the SDC system must not be exposed to RCS temperature greater than 300 °F. Plausible; 265 psia is the maximum pressure limit per the caution.
- B. Components of the SDC system must not be exposed to RCS temperature greater than 300 °F. Also, T351Y and operating Loop T_{HOT} must be within 20 °F. Plausible; 265 psia is the maximum pressure limit per the caution. Also, 48 °F/hr is the administrative value cooldown rate limit.
- C. Correct.
- D. WHEN SDC to RCS temperature, T351Y is within 20 °F of operating Loop T_{HOT}, throttle open each LPSI injection valve to establish flow. Plausible; 48 °F/hr is the administrative value cooldown rate limit.

Technical Reference(s):

- MP-PROC-OPS-OP 2207 REV. 054, Plant Cooldown.
- MP-PROC-OPS-OP 2310 REV. 037, Shutdown Cooling System.

Provided reference(s): None

Learning Objective:

Source: New

Cognitive Level: Memory

10CFR55.41(b)(8): Components, capacity, and functions of emergency systems.

Comments: Added clarification in stem to ask “highest permissible” difference between T_{HOT} and T351Y, to make 48°F more clearly wrong. SM 6/21/24. Reworded stem to mimic OP 2207 verbiage DF 7/11/24

Question #30

The plant is in MODE 6 preparing to conduct a core off-load.

- 175 hours have elapsed since entering MODE 3.
- RBCCW header temperature = 87° F and stable.

1. Can the core be off-loaded to the Spent Fuel Pool?

AND

2. Why?

- A.
 - 1. YES.
 - 2. RBCCW temperature IS low enough and The reactor HAS been off-line long enough.
- B.
 - 1. NO.
 - 2. RBCCW temperature IS NOT low enough AND the reactor HAS NOT been off-line long enough.
- C.
 - 1. NO.
 - 2. RBCCW temperature IS NOT low enough.
- D.
 - 1. NO.
 - 2. The reactor HAS NOT been off-line long enough.

Question #30

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 005K5.14

Importance Rating: 3.5

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the RESIDUAL HEAT REMOVAL SYSTEM: Refueling operations

K/A justification: RBCCW can be considered part of RHR since it provides the heat sink for the system. RBCCW flows through the Shutdown Cooling heat exchangers.

Proposed Answer: C

Justification: If the reactor has been subcritical greater than 160 hours, RBCCW temperature is required to be less than or equal to 85 ° F; otherwise, fuel off-load is suspended

Plausibility:

- A. The examinee may think that RBCCW temperature is low enough and the unit has been shutdown long enough.
- B. The examinee may forget about the RBCCW temperature requirement think the RBCCW temperature requirement is not applicable after a certain point (i.e., 150 hours which is a legacy number).
- C. Correct.
- D. The examinee may think the RBCCW temperature requirement is not applicable after a certain point (i.e., 150 hours which is a legacy number).

Technical Reference(s):

- OP 2209A, Refueling Operations

Provided reference(s): None

Learning Objective: 282676 – Describe the objective and scope of OP 2209A, Refueling Operations

Source: New

Cognitive Level: Comprehensive OR Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments: Changed stem from 145 to 165 hours and RB temperature 77 ° F to 87 ° F. with original numbers, there were two correct answers. Changes made based on validation results and feedback. DF 7 MAY24. Changed time shutdown to 175 hours, 'cold' to 'low' based on feedback from validators and NRC. DF 8/14

Question #31

The plant was operating at 100% power when a LB-LOCA occurred.

- CIAS, EBFAS and MSI initiated from ESAS signals.
- SIAS was required to be manually initiated.

- The following two alarms are received:
 - RWST CH A LEVEL LO/LO (C-01, A-17)
 - RWST CH C LEVEL LO/LO (C-01, A-19)

Describe the status of the ECCS pumps.

- A. HPSI and LPSI pumps are running, aligned to the RWST.
- B. HPSI and LPSI pumps are running, aligned to the containment sump.
- C. HPSI pumps are running, aligned to the RWST. LPSI pumps have tripped.
- D. HPSI pumps are running, aligned to the containment sump. LPSI pumps have tripped.

Question #31

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 006A1.15

Importance Rating: 3.6

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the EMERGENCY CORE COOLING SYSTEM, including: RWST level and temperature

Proposed Answer: D

Justification: Sump Recirculation Actuation Signal (SRAS) initiates on 2 of 4 RWST LOLO level bistable trips, as indicated by the RWST CH A LEVEL LO/LO and the RWST CH C LEVEL LO/LO alarm. The SRAS signal will swap ECCS pump suction to the containment sump and trip the LPSI pumps.

Plausibility:

- A. This would be the alignment if SRAS had not initiated. Plausible; Candidate may not recall the SRAS operates on 2/4 logic or may assume you need an automatic SIAS for SRAS to actuate correctly.
- B. The sump is the correct source, but SRAS trips the LPSI pumps. Plausible; candidate may think that manual initiation of SIAS will prevent LPSI pumps being secured on a valid SRAS signal.
- C. The RWST is the incorrect source of water once SRAS has initiated. The pump alignment is correct. Plausible; Candidate may think a manual SIAS requires manual operation of valves when an SRAS signal is received.
- D. Correct.

Technical Reference(s):

- EOP 2532 rev 37, Loss of Coolant Accident.
- ESA-01-C REV. 004 Ch. 002, ESFAS Lesson plan.
- ECC-01-C REV. 006., ECCS lesson plan

Provided reference(s): None

Learning Objective: 281305 - Given a LOCA, assess how a loss of any systems that comprise the ECCS will affect the RCS.

281307 - State the following; Major components that receive an SRAS actuation signal. (MB-06568)

Source: Bank 2014 NRC exam Q32 (Vision #451913)

Cognitive Level: Comprehension /Analysis

10CFR55.41(b) (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features:

Comments:

Question #32

The plant is operating at 100% power.

SIT Outlet valve indications are as follow:

SI-614	SI-624	SI-634	SI-644
<u>#1 SIT OUTLET</u>	<u>#2 SIT OUTLET</u>	<u>#3 SIT OUTLET</u>	<u>#4 SIT OUTLET</u>
lit	lit	not lit	not lit

These indications could be a result of an issue with what MCC?

- A. B-51
- B. B-52
- C. B-61
- D. B-62

Question #32

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 006K2.02

Importance Rating: 3.2

K/A Statement: Knowledge of electrical power supplies to the following: EMERGENCY CORE COOLING SYSTEM Valve operators for safety injection tanks

Proposed Answer: C

Justification: On a loss of MCC B-61 (Facility 2), the Facility 2 SIT outlet Motor-Operator Valves SI-6314 and SI 644 will NOT have power available to their breakers. Position indication comes from control power which is powered from the 480V breaker.

Plausibility:

- A. Examinee may think position indication power comes from vital 120VAC or 125VDC instrument panels.
- B. The examinee may think Each valve is powered from a different MCC. Conventional thinking would have SI-644 powered from MCC B-62
- C. Correct.
- D. The examinee may think the SITs follow conventional numbering where even numbered components are powered from Facility 2 so only SI-614 and SI-634 would have power.

Technical Reference(s):

- SIT Lesson Plan SIT-00-C

Provided reference(s):

Learning Objective: 282825 - ILT Given a set of plant conditions during a loss of 480 VAC Bus 22A, 22B, 22C, 22D, 22E or 22F, determine equipment limitations caused by those conditions.

Source: New

Cognitive Level: Memory

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Question #33

Thirty (30) minutes after a plant transient which caused the Pressurizer Safety Valves to lift, the following conditions exist:

- RCS is at a saturated condition.
- RCS pressure is 1000 psia.
- The Quench Tank Rupture Disc has blown.
- Containment pressure is 25 psig.

What Pressurizer Safety Valve tailpiece temperature would indicate that fluid is still leaking by the Safety valves?

- A. 240 °F
- B. 312 °F
- C. 348 °F
- D. 545 °F

Question #33

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 007K5.07

Importance Rating: 3.7

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the PRESSURIZER RELIEF/QUENCH TANK SYSTEM: PZR safety tailpipe temperature variations with PRT/quench tank pressure

Proposed Answer: B

Justification: The open safety valve constitutes an isenthalpic (constant enthalpy) process. From the Mollier Diagram the temperature for 1000 psia expanded to 40 psia (25 psig + 15 = 40 psia) is 312 °F. This is obtained by following the 1000 psia constant pressure line down to The Saturation line. Then straight across (constant enthalpy) to the 40 psia constant pressure line.

Plausibility:

- A. 240 °F is the temperature for 25 psia (10 psig) instead of 40 psia (25 psig + 15 psi). Plausible; It is reasonable for the candidate to not convert psig to psia to obtain the correct temperature. And to follow the Saturation Line down to 25 psia instead of straight across, which is correct for a constant enthalpy process.
- B. Correct.
- C. 348 °F is the temperature of 1000 psia expanded down to the Saturation Line and to then expanded straight across (constant enthalpy) but stopping at the other side of the Saturation Line, instead of going further to 40 psia. Plausible; This is similar to how the temperature is determined if pressure starts higher and is expanded to the end pressure below the Saturation Line. In this case, the end point constant pressure line is followed up to the Saturation Line to determine the correct temperature.
- D. 545 °F is the saturation temperature for 1000 psia. Plausible; This is a common misconception. That the fluid will be at a saturation condition, when in fact it is superheated.

Technical Reference(s):

- Mollier Diagram, GFES Thermodynamic Processes

Provided reference(s): Steam Tables w/ Mollier diagram

Learning Objective: 281 – Describe the indications available to detect a leaking code safety valve.

Source: Bank 2020 NRC exam Q2 (Vision #486065)

Cognitive Level: Comprehension/Analysis

10CFR55.41(b) (14) Principles of heat transfer thermodynamics and fluid mechanics.

Comments:

Question #34

A SIAS signal has been initiated.

Which RBCCW component shows its correct SIAS signal position?

<u>RBCCW component</u>	<u>SIAS signal to:</u>
A. 2-RB-8.1B, Spent Fuel Pool Hx Outlet valve	CLOSE
B. 2-RB-13.1A, Shutdown Cooling Hx Outlet valve	OPEN
C. 2-RB-402, Letdown Hx RBCCW Outlet TCV	CLOSE
D. 2-RB-28.1B, CAR Cooler RBCCW Inlet valve	OPEN

Question #34

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 008A3.08

Importance Rating: 4.1

K/A Statement: Ability to monitor automatic features of the COMPONENT COOLING WATER SYSTEM, including: Automatic actions associated with the CCWS that occur as a result of an ESFAS signal

Proposed Answer: A

Justification: On a SIAS, RBCCW is isolated to the Spent Fuel Pool Heat Exchangers to ensure adequate RBCCW cooling flow to the CAR coolers.

Plausibility:

- A. Correct.
- B. The SDC Heat Exchanger RBCCW outlet valves do not receive a SIAS signal, they receive a SRAS signal to open.
- C. The Letdown Heat Exchanger outlet valve is interlocked with 2-CH-089 which receives a CIAS closed signal.
- D. The CAR Cooler inlet valves receive no accident signal; they are locked open.

Technical Reference(s):

- Vision question #480664
- RBCCW Lesson Plan RBC-01-C

Provided reference(s): None

Learning Objective: 281949 – Describe the effects of the following on RBCCW system and operating status:

- A) Safety Injection Actuation System (SIAS)
- B) Sump Recirculation Actuation System (SRAS)
- C) Containment Isolation Actuation System (CIAS)
- D) Loss of Normal Power (LNP)

Source: Bank

Cognitive Level: Memory

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Reworded stem for clarity. Some validators thought the question was a matching question. DF 5/28. Per 7/25 phone call, added 'signal' to eliminate second correct answer. DF 7/26

Question #35

Given the following conditions:

- Unit 2 is at 100% power.
- RCS pressure is slowly **RISING** due to an ongoing event.

As RCS Pressure rises, PIC-100Y, PRES CNTL-Y, output will 1..

AND

The PORV(s) will reach their setpoint to OPEN at 2..

- A. 1. rise
2. 2397 psia
- B. 1. rise
2. 2500 psia
- C. 1. lower
2. 2397 psia
- D. 1. lower
2. 2500 psia

Question #35

☒ RO

☐ SRO

Tier # 1

Group # 2

K/A # 010A1.07

Importance Rating: 3.9

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the PRESSURIZER PRESSURE CONTROL SYSTEM, including: RCS or PZR pressure

Proposed Answer: A

Justification: As pressure rises, the controller output will rise (higher output means more demand for pressure reduction) which will result in the spray valves fully opening to reduce pressure and the heaters de-energizing to slow the pressure rise. RPS High pressure trip will both trip the reactor and open both PZR PORVs at 2397 psia.

Plausibility:

A. Correct.

B. WRONG; Both PORVs will open upon a high-pressure reactor trip signal at 2380 psia. Plausible; May think the PORV setpoint with the Safety valve setpoint.

C. WRONG; As pressure rises, the demand signal on the pressure controller also rises. As the demand signal rises, the system is designed to fully open spray valves and secure pressurizer heaters to mitigate the pressure increase. Plausible; Candidate may think that controller demand will lower as pressure rises, to reflect that it is demanding a lower pressure.

D. WRONG; As pressure rises, the demand signal on the pressure controller also rises. As the demand signal rises, the system is designed to fully open spray valves and secure pressurizer heaters to mitigate the pressure increase. Also, Both PORVs will open upon a high-pressure reactor trip signal at 2380 psia. Plausible; Candidate may think that controller demand will lower as pressure rises, to reflect that it is demanding a lower pressure AND confuse PORV setpoint with Safety setpoint.

Technical Reference(s):

- MP-DWG-000-25203-26014 (50.59 DWG M2) SH.000002 REV. 050.
- MP-PROC-OPS-ARP 2590B-216 REV. 000-01, PZR CH X PRES HI/LO.
- MP-PROC-OPS-ARP 2590B-043 REV. 001, PORV RC-402 OPEN.
- PLC-01-C REV. 007, Pressurizer Level and Pressure Control System Lesson Plan.
- RPS-01-C REV. 007, Reactor Protection System Lesson Plan.

Provided reference(s): None

Learning Objective:

Source: Bank SL 2020 NRC RO Q 36

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: 7/25 phone call changed 2nd part to PORV setpoint, the number of PORVs opening was not discriminating. DF 7/26

Question #36

The plant is at 13.5% power performing a turbine startup.

- RPS Channel 'C' is de-energized.

Subsequently,

- #4 Turbine Control Valve closes.

With no operator action,

The turbine would be 1..

AND

The reactor would be 2..

- A. 1. tripped
2. tripped
- B. 1. tripped
2. NOT tripped
- C. 1. NOT tripped
2. tripped
- D. 1. NOT tripped
2. NOT tripped

Question #36

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 012K6.01

Importance Rating: 3.7

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR PROTECTION SYSTEM:
Bistables and bistable test equipment

Proposed Answer: D

Justification: The conditions given will cause a one out of three logic for tripping the reactor (C Channel de-energized). The Control Valve closing will complete the logic for a reactor trip due to turbine trip (2 Control Valves closed as sensed by RPS); however at power levels less than 15%, a turbine trip will not cause a reactor trip.

Plausibility:

- A. The examinee may forget the reactor trip from turbine tripped is bypassed < 15% power. Also, the examinee may think the turbine will be tripped due to the 2 bistables being de-energized. However, the valves aren't closed, their RPS bistables are de-energized.
- B. The examinee may remember the reactor will not trip on a turbine trip < 15% power yet think the turbine is tripped on due to 2 control valves closing. However, the valves aren't closed, their RPS bistables are de-energized.
- C. The examinee may get the turbine trip/reactor trip relationship backwards and think the reactor can trip without a turbine trip.
- D. Correct.

Technical Reference(s):

- RPS lesson plan RPS-01-C

Provided reference(s): None

Learning Objective: 282144 - Describe the conditions that actuate each of the following Reactor Protection System trips or interlocks:
C) Turbine-Reactor trip

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Changed part two to turbine control valve closing due to confusion during validation.
DF 6/12

Question #37

A reactor startup is in progress.

- The plant is at $1\text{E-}5$ % power.
- The Channel 'A' Wide Range Nuclear Instrument fails high.

Describe the plant response.

- A. Channel 'A' Zero Power Mode Bypass is automatically disabled.
- B. The CEAPDS PDIL alarm and interlock remains bypassed.
- C. Channel 'A' Level 1 and Level 2 LEDs extinguish.
- D. Channel 'A' Power Trip Test Interlock (PTTI) arms.

Question #37

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 012A4.03

Importance Rating: 3.8

K/A Statement: REACTOR PROTECTION SYSTEM Ability to manually operate and/or monitor in the control room: Channel blocks and bypasses

Proposed Answer: A

Justification: Initially, all Wide Range Nuclear Instruments are below $9.4 \times 10^{-5}\%$ power so the LVL 1 and LVL2 bistables are in their 'reset' condition. When the failure occurs on channel 'A', indicated WR power goes above $1\text{E-}4\%$ power, the LVL 1 bistable trips which causes the LVL 1 LED to be lit. Channel 'A' Power above $1\text{E-}4\%$ also causes the LVL 2 bistable to trip, which causes the LVL 2 LED to be lit. In addition, at greater than $1\text{E-}4\%$ power, the Zero Power Mode Bypass (ZPMB) for channel 'A' will be automatically disabled regardless of the ZPMB key switch status.

Plausibility:

- A. Correct.
- B. PDIL is now ENABLED for all channels since it only requires a single LVL 2 bistable to be tripped ($\frac{1}{4}$ logic). Plausible; Candidate may not realize this feature only requires one out of four logic.
- C. The LVL1 and LVL 2 bistables will be in a tripped condition when power goes above $1\text{E-}4\%$ power and their LEDs light. Plausible; Candidate may confuse the condition of the bistable with the condition of the LVL1 and LVL 2 LEDs which extinguish when the bistables are reset.
- D. The PTTI interlock would indeed be armed for this channel IF the failed detector power supply were on a Linear Channel. Plausible; Student may confuse the NI failure mechanism that would trigger a PTTI and the direction of failure required.

Technical Reference(s):

- NIS-01-C REV. 005 Ch. 002, Nuclear Instrumentation System Lesson Plan.
- RPS-01-C REV. 007 Ch. 002, Reactor Protection System Lesson Plan.

Provided reference(s): None

Learning Objective: 281844 - Describe the purpose, actuating conditions, and components affected for each of the following Nuclear Instrument System interlocks or bistables: Wide Range Monitor nuclear instrumentation LVL 1 and LVL 2 bistable.

Source: Modified Vision #452219

Cognitive Level: Memory

10CFR55.41(b) (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments: Changed cognitive level to Memory based on source question being designated as Memory on the 2018 exam, and this version with modifications still has a similar stem. Changed C to read LEDs extinguish from reset. From an operator's perspective, they see the LEDs light when the bistables trip, the difference between these bistables

The plant is at 2%, starting up.

The Channel "A" Wide Range Nuclear Instruments fails to 1 E-6 % power.

Describe the plant response?

- A. The Zero Power Mode Bypass will arm on Channel "A".
- B. The Level 1 and Level 2 Bistables will reset on Channel "A".
- C. The PDIL alarm and interlock on CEAPDS is now bypassed.
- D. The Power Trip Test Interlock (PTTI) will arm on Channel "A".

Answer: B

Answer Explanation	
A - WRONG:	A bypass key for the channel must be in the "bypass" position when the Wide Range NI for that channel drops below 1X10-4% in order to arm the Zero Power Mode Bypass. The key is procedurally required to be removed prior to criticality.
PLAUSIBLE:	Student may remember that the WRNI dropping below 1X10-4% power will ALLOW initial arming, <u>or</u> re-arming, of the Zero Power Mode Bypass. If the key were left in the bypass position and power dropped below the trigger value, the ZPMB would re-arm automatically.
B - CORRECT:	The Level 1/2 bistables will RESET on this channel when the signal drops below 1X10-4% and would set up the circuits that would allow for the conditions described in choices "A" and "C" to occur.
C - WRONG:	The PDIL bypass would NOT activate on a single WRNI failing low, as more than one channel below 1X10-4% is required to activate the bypass. However, if the bypass is armed, only <u>one</u> channel failing above 1X10-4% would clear the bypass and re-arm the PDIL interlock.
PLAUSIBLE:	The student may confuse the failure mechanism that will "re-arm" the interlock with the one that bypasses it.
D - WRONG:	The PTTI interlock would indeed be armed for this channel, IF the failed detector power supply were on a Linear Channel.
PLAUSIBLE:	Student may confuse the NI failure mechanism that would trigger a PTTI.

Question 1 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	452219
Version ID:	590537
User-Defined ID:	8600020-MB-1
Cross Reference Number:	032/AK2.01
Topic:	MP2*LOIT NIS, 2202, Failure, NRC-2008, NRC-2018
Num Field 1:	
Num Field 2:	
Text Field:	NRC-2018, Q-21, R-2, RO
Comments:	2018 NRC RO/SRO, MP2*LOIT NIS, 2202, Failure, NRC-2008, NRC-2018 - Ques. #21, Designated: RO - Memory

Question #38

The plant has experienced a steam line rupture.

- The crew is in EOP 2536, Excess Steam Demand Event.
- Step 7b reads “**CHECK** at least **one** train of MSI – properly ACTUATED (C-01X)”

How is Step 7b performed?

Check all the MSI 1. lights illuminated on C-01X.

AND

For any of the above lights NOT illuminated...

- A.
 - 1. WHITE
 - 2. MANUALLY align the component (C-05)
- B.
 - 1. WHITE
 - 2. Send a PEO to align the component (local)
- C.
 - 1. BLUE
 - 2. MANUALLY align the component (C-05)
- D.
 - 1. BLUE
 - 2. Send a PEO to align the component (local)

Question #38

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 013A3.07

Importance Rating: 4.1

K/A Statement: Ability to monitor automatic features of the ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, including: Main steam isolation system actuation

Proposed Answer: C

Justification: On panel C-01X, a BLUE light indicates a component is in its accident position. For any component with its BLUE light extinguished (after an actuation signal), the component is positioned from the control room.

Plausibility:

- A. A WHITE light indicates that component position cannot be determined.
- B. A WHITE light indicates that component position cannot be determined. Operators are not sent out into the field at this point in the procedure.
- C. Correct.
- D. Operators are not sent out into the field at this point in the procedure.

Technical Reference(s):

- Engineered Safety Actuation System Lesson Plan ESA-01-C

Provided reference(s): None

Learning Objective: 281378 - Describe the functions of the following Engineered Safety Features Actuation System (ESAS) Control Room controls and indications, including how controlled component(s) is/are affected by each mode or position of the control:
D) Status lights at Panel C-01X

Source: New

Cognitive Level: Memory

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Question #39

The plant was tripped from 100% power due to a 'A' Main Steam line rupture inside containment.

- VA-20 is DEAD.
- The crew has transitioned to EOP-2536, Excess Steam Demand Event.
- #1 S/G level = 50" and lowering

Where is #2 ADV operated from to stabilize RCS temperature?

- A. The ADV controller at C21.
- B. The ADV locally at the valve.
- C. The ADV controller at C05.
- D. The ADV controller at C10.

Question #39

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 013 G2.1.8

Importance Rating: 3.4

K/A Statement: ENGINEERED SAFETY FEATURES ACTUATION SYSTEM CONDUCT OF OPERATIONS: Ability to coordinate personnel activities outside the control room

Proposed Answer: B

Justification: VA-20 powers the entire #2 ADV control circuit outside of the control room. With a loss of VA-20, the #2 ADV can NOT be operated remotely from ANY location. There is a loss of Pressure Indication on C10, CO5F, C21, and Plant Computer PCA4 AND Loss of controller operations on C05, C21 and C10. The valve can be operated locally due to the location of the steam rupture (CTMT).

Plausibility:

- A. The loss of VA-20 also de-energizes the C21 part of the ADV control circuit. Plausible; Candidate may confuse loss of VR-21 requires the #2 ADV be operated from C-21.
- B. Correct.
- C. The #2 ADV C05 controller is powered from VA-20 and would be de-energized. Plausible; The #1 ADV control power has been modified to allow operation in manual only, from the C05 controller upon a loss of VR-11.
- D. The C10 Fire Shutdown panel is designed for used when the control room must be evacuated during an Appendix "R" type fire. Plausible; Although it is very protected due to its function, the loss of VA-20 will still prevent the operation of the #2 ADV from here.

Technical Reference(s):

- AOP 2504D REV. 011, Loss of 120VAC Vital Instrument Panel VA-20.
- LVD-00-C REV. 0007 Ch. 002, 125VDC/120VAC lesson Plan.

Provided reference(s): None

Learning Objective: Describe the expected plant response to an Excess Steam Demand (283857)

Source: Bank Vision 451749

Cognitive Level: Comprehension or Analysis

10CFR55.41(b) (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Question #40

The plant is operating at 100% power.

- The 'A' RBCCW header temperature controller experienced a malfunction.
- PPC point CVCONTM, PRIORITY – CTMT AIR TEMP AVG, alarms.
- Containment temperature = 116 °F

What procedural action is taken?

- A. Start the idle CAR fan in FAST speed IAW OP 2313A, Containment Air Recirculation and Cooling System.
- B. Start the idle CAR fan in SLOW speed IAW OP 2313A, Containment Air Recirculation and Cooling System.
- C. Start both CTMT Aux. Circ. Fans in FAST speed IAW OP 2313B, Containment Auxiliary Circulation System.
- D. Start the idle CEDM cooling fan IAW OP 2313D, CEDM Cooling System

Question #40

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 022A2.04

Importance Rating: 3.7

K/A Statement: Ability to (a) predict the impacts of the following on the CONTAINMENT COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Cooling water system malfunction

Proposed Answer: A

Justification: A high ambient temperature in CTMT of 116°F requires the idle CAR fan to be started in SLOW speed.

Plausibility:

- A. Procedurally, all 4 CAR fans cannot be running in FAST speed to prevent overpressurizing the CAR fan ductwork. The examinee may forget this precaution or think that cooling CTMT over rides the ducting concern.
- B. Correct.
- C. Starting the CTMT Aux Circ fans in fast won't cool down the CTMT temperature because heat isn't being removed from CTMT. The examinee may think that more air flow will help cool the building
- D. Starting the third CEDM Cooling fan is not permitted by procedure. The examinee may think more air flow to the CEDMs will prevent the CEA coils from overheating and causing CEAs to drop.

Technical Reference(s):

- OP 2313A

Provided reference(s): None

Learning Objective: 281946 – Describe the operating implications of a loss of RBCCW flow to a given cooling load.

Source: New

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments: per 725 phone call, replaced question to match K/A. DF 7/26 Added bullet for CTMT temperature = 116 F based on validator feedback. DF 8/15

Question #41

The following conditions exist five (5) hours after a plant trip, due to a Large-Break LOCA:

- EOP 2532, Loss of Coolant Accident, is in progress.
- SIAS, CIAS, CSAS, MSI and SRAS have been verified fully actuated.
- CTMT pressure = 12 psig and lowering slowly.
- Reactor Vessel Level = 19% and stable.

Subsequently, the RSST is lost.

- Bus 24C is DEAD.
- Bus 24 D is LIVE, powered from the 'B' EDG
- Facility 2 Sequencer failed at Sequence 0 (zero).

What pumps are required to be started?

- A.
 1. LPSI.
 2. Charging.
- B.
 1. LPSI.
 2. Containment Spray.
- C.
 1. HPSI.
 2. Charging.
- D.
 1. HPSI.
 2. Containment Spray.

Question #41

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 026A4.01

Importance Rating: 3.9

K/A Statement: Ability to manually operate and/or monitor the CONTAINMENT SPRAY SYSTEM in the control room: CSS controls

Proposed Answer: D

Justification: With containment pressure > 9.2 psig, Containment spray is required by procedure. Also, in sump recirculation (SRAS signal) the HPSI pumps are used for RCS or Core cooling and the LPSI pumps are tripped.

Plausibility:

- A. HPSI is required for flow through the core because LPSI is not used during sump recirc. Charging pumps will not be running during SRAS due to low BAST and RWST levels. Plausible; May think that the loss of the 24D ECCS pumps may require running charging pumps to compensate for the loss.
- B. In sump recirc (SRAS) the LPSI pumps are not used for RCS or core cooling. Plausible; May not recall that the SRAS signal results in a direct trip of the LPSI pumps and they are not required in recirculation.
- C. Post – SRAS actions require charging pumps to be placed in pull-to-lock. Plausible; May think that the loss of the 24D ECCS pumps may require running charging pumps to compensate for the loss.
- D. Correct.

Technical Reference(s):

- EOP 2532 REV. 037, Loss of Coolant Accident.
- CSS-00-C REV. 008, Containment Spray System Lesson Plan (55371/58131)
- ECC-01-C REV. 006, Emergency Core Cooling System Overview.

Provided reference(s): None

Learning Objective: 281307 - State the following; Containment sump level at the start of SRAS, RWST level associated with SRAS, Major components that receive a SRAS actuation signal.

Source: Bank Vision 477224

Cognitive Level: Comprehension or Analysis

10CFR55.41(b) (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.:

Comments:

Question #42

The plant is starting up after a refueling outage.

The Main Feedwater Pump is initially driven by steam from 1.,

AND

Above 25 % Generator load, the pump is driven by steam from 2..

- A. 1. the Main Steam header
2. the Main Steam header
- B. 1. the Main Steam header
2. Cross-over Steam
- C. 1. Cross-over Steam
2. the Main Steam header
- D. 1. Cross-over Steam
2. Cross-over Steam

Question #42

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 039K1.08

Importance Rating: 3.6

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the MAIN AND REHEAT STEAM SYSTEM and the following systems: MFW

Proposed Answer: B

Justification: At low power operations, the cross-over steam pressure is not high enough to drive a MFW pump, above 25%, the steam pressure is high enough to drive the pump.

Plausibility:

- A. The examinee may think Main Steam is always used, that would explain the pump running after a trip.
- B. Correct.
- C. The examinee may think cross-over steam is high enough to drive the pumps at low power but want main steam at higher power levels in case of a plant trip isolating cross-over steam.
- D. The examinee may forget that cross-over steam pressure is too low to drive the pumps during lower power conditions.

Technical Reference(s):

- Main Feedwater Lesson plan, MFW-00-C

Provided reference(s): None

Learning Objective: 281647 – Summarize the operating characteristics of the following major Main Feedwater system components as given in MFW-00-C:
A) Steam Generator Feedwater pumps (SGFP)

Source: New

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments:

Question #43

The plant is operating at 100% power

- The 2A Feedwater Heater Extraction Steam Supply closes

Describe the effect on the plant.

- A. Delta-T power will rise.
- B. RCS T_{COLD} will rise.
- C. Generator MWe output will lower.
- D. Main Condensate flow will lower.

Question #43

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 039K3.05

Importance Rating: 3.9

K/A Statement: Knowledge of the effect that a loss or malfunction of the MAIN AND REHEAT STEAM SYSTEM will have on the following systems or system parameters:
RCS

Proposed Answer: A

Justification: The loss of steam supply to the feedwater heater will result in lower feedwater temperatures entering the SG. This will result in a drop in RCS T_{COLD} . A lowering T_{COLD} will cause Delta-T power to rise.

Plausibility:

- A. Correct.
- B. T_{COLD} will lower due to the colder feedwater entering the SG. Plausible; 'Extraction Steam' concept could be construed to mean there is now less energy leaving the feedwater heater, so the feed will now be more preheated than before.
- C. The loss of steam flow to the extraction results in a gain of steam flow directly to the main turbine. This extra steam will cause generator electrical output to rise slightly. Plausible; may think that the loss of feedwater preheating will result in less steam being generated.
- D. Main condensate flow will rise. Plausible; The extra steam going through the main turbine equates to extra water in the main condenser and a RISE in condensate flow

Technical Reference(s):

- FHD-00-C REV. 006, Extraction Steam and Feedwater Heater System Lesson Plan.
- Vision Question 451761

Provided reference(s): None

Learning Objective: 281406 – Describe the effects of a loss or malfunction of the Extraction Steam System on the following: 3) Feedwater System.

Source: Modified Vision 451761

Cognitive Level: Comprehension or Analysis

10CFR55.41(b) (4) Secondary coolant and auxiliary systems that affect the facility.

Comments: Changed cognitive level to Memory based on being designated as Memory on the 2008 exam. Changed back to Comprehension or Analysis based on 75-day discussion. Changed to Modified from Bank. SRM 7/26.

The plant is operating at 100% power

- The 2A FeedWater Heater Extraction Steam Supply closes.

Describe the effect on the plant.

- A. RCS Tcold will lower.
- B. Delta-T power will lower.
- C. Generator MWe output will lower.
- D. Main Condensate Flow will lower.

Answer: A

Answer Explanation

A - Correct; The loss of steam supply to the feedwater heater will result in lower feedwater temperatures entering the SG. This will result in a drop in RCS Tcold.
 B - Wrong; The lowering of Tcold combined with an unchanged steam demand will result in a RISE in delta-T power.
 C - Wrong; The loss of steam flow to the extraction results in a gain of steam flow directly to the main turbine. This extra steam will cause generator electrical output to RISE.
 D - Wrong; The extra steam going through the main turbine equates to extra water in the main condenser and a RISE in condensate flow.

Question 1 Info

Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	451761
Version ID:	573658
User-Defined ID:	8078895-MB
Cross Reference Number:	039/K3.05
Topic:	MP2*LOIT, MSR, Reheat, NRC-2008
Num Field 1:	
Num Field 2:	
Text Field:	NRC-2008, Q-43, R-2, RO
Comments:	2008 NRC RO/SRO, MP2*LOIT, MSR, Reheat, NRC-2008 - Ques. #43, Designated: RO - Memory

Question #44

The plant is performing a Cooldown in accordance with OP-2207, Plant Cooldown.

- Facility 1 MSI is BLOCKED.
- Facility 2 MSI is NOT BLOCKED.

A secondary side casualty occurs, causing the following conditions:

- #1 S/G pressure = 568 psia and lowering.
- #2 S/G pressure = 625 psia and lowering.

1. What is the status of the MSIVs?

AND

2. Assuming no operator actions, what is the path for RCS Heat Removal?

- A.
 - 1. Both OPEN
 - 2. 'A' Steam Dump only.
- B.
 - 1. #1 CLOSED, #2 OPEN
 - 2. 'A' ADV and 'A' Steam Dump
- C.
 - 1. #1 CLOSED, #2 OPEN
 - 2. 'A' Steam Dump only.
- D.
 - 1. Both CLOSED
 - 2. 'A' and 'B' ADV.

Question #44

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 059A4.12

Importance Rating: 3.8

K/A Statement: Ability to manually operate and/or monitor the MAIN FEEDWATER SYSTEM in the control room: Initiation of automatic MFW isolation

Proposed Answer: D

Justification: With the given conditions, a Facility 1 MSI will be generated (S/G press. < 572 psia). Either facility MSI will close both MSIVs. Heat removal will be from the ADVs when S/G pressure increases to setpoint. K/A justification: MSI occurs in the course of this question, which **includes** MFW isolation (they are both in the same actuation).

Plausibility:

- A. The examinee may not remember the MSI setpoint and think the MSIVs remain open.
- B. The examinee may think the MSIVs are facility specific for MSI (the MSIV bypasses are facility specific). With the 'A' S/G isolated, it's ADV would remove heat along with the 'A' Condenser dump
- C. The examinee may think the MSIVs are facility specific for MSI (the MSIV bypasses are facility specific). Even with the 'A' S/G isolated, the 'A' Condenser steam dump would remove heat from the RCS and keep the 'A' SG below the ADV setpoint.
- D. Correct.

Technical Reference(s):

- Vision Question #452016
- Engineered Safeguards Actuation System Lesson Plan ESA-01-C

Provided reference(s): None

Learning Objective: 281376 – Given a simplified logic diagram, trace the logic associated with each Engineered Safeguards Actuation System trip function from sensors to actuation relays.

Source: Modified

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The plant was manually tripped when a pressurizer spray valve failed open.
The following conditions exist:

- All RCPs were secured to stop the RCS depressurization.
- EOP 2528, Loss Of Offsite Power/Loss of Forced Circulation is in progress.
- A Natural Circulation cooldown to Mode 4 has been started.
- "B" Main Feed pump is supplying feed flow.
- The "A" Condenser steam dump valve is being used for the cooldown.
- Facility 1 MSI has just been manually blocked.
- Facility 2 MSI has NOT yet been blocked.

Then, a problem with #1 SG feed control causes a sudden excess feed flow to the SG.
Before the crew can stabilize the plant, SG pressures drop to the following values:

- #1 SG pressure drops to 568 psia.
- #2 SG pressure drops to 670 psia.

Which of the following describes the impact of the SG pressure drop, and the actions taken to control RCS heat removal?

- A. MSI actuation has closed the #2 MSIV only. Override open #2 MSIV, continue using Main Feed Water and the "A" condenser steam dump valve.
- B. MSI actuation has closed both MSIVs. Verify both MSIVs closed, switch to Auxiliary Feed Water and use the atmospheric dump valves.
- C. MSI actuation has closed the #2 MSIV only. Block and reset MSI, continue using Main Feed Water and the "A" condenser steam dump valve.
- D. MSI actuation has closed both MSIVs. Block and reset MSI, open the MSIVs, switch to Auxiliary Feed Water and use the Atmospheric dump valves.

Answer: B

Answer Explanation

A - WRONG; The MSI block was activated for only facility 1 during the controlled cooldown. However, either SG dropping below the ESAS MSI setpoint of 572 psia will trigger an MSI actuation on Facility 2 because it is not yet blocked. Either Facility of MSI actuating will close BOTH MSIVs.

PLAUSIBLE; Student may recall MSI can still occur if one SG is isolated due to an event and the other continues to depressurize. However, they may feel because Facility 1 was already blocked, that Facility's MSIV should not get a close signal.

B - CORRECT; The MSI block was activated for Facility 1 ONLY during the controlled cooldown. As a result of the excess feed flow event, #1 SG pressure dropped below the ESAS MSI setpoint of 572 psia, which will trigger an MSI actuation on Facility 2 of ESAS because it is not yet blocked. Either ESAS Facility of MSI actuating will close BOTH MSIVs.

C - WRONG; Either Facility of MSI actuating will close BOTH MSIVs.

PLAUSIBLE; Student may confuse the action of ESAS on an MSI trigger of one Facility and be considering the actions taken if an MSI were to occur during a cooldown for a SG Tube Rupture.

D - WRONG; EOP 2528 does not have an RNO step to "block and reset" MSI if it occurred because the RCS cooldown was not controlled. The EOP requires MSI be fully actuated.

PLAUSIBLE; Student may confuse the mitigating actions for an undesired MSI during this scenario with that of an undesired MSI during a SG Tube Rupture.

Question #45

The Unit is operating at 55% power.

- The 'A' and 'B' SGFPs are both operating in automatic control.
- Annunciator SGFP 'A' TURBINE TRIP (C-05, A-4), alarms.
- 'A' SGFP is NOT tripped.
- 'A' SGFP speed is 5550 rpm and rising.

In accordance with ARP 2590D-013, SGFP A TURBINE TRIP, which of the following describes the correct operator response?

- A. PLACE 'A' SGFP in Manual and ENSURE FRV D/P between 40 and 150 psid.
- B. TRIP 'A' SGFP and the Reactor, then GO TO EOP 2525, Standard Post Trip Actions.
- C. TRIP 'A' SGFP, then OPEN CNM-2, COND DEMIN BYP.
- D. PLACE 'A' SGFP in Manual, CHECK PPC alarm typer for SGFP related alarms.

Based on review of post-examination comments submitted, question 45 was determined to have no correct answer and was deleted from the exam

Millstone 2 proposes that answer A is the only correct answer and that answer C is not a correct answer. Answer C was the original correct answer.

The question stem gives the plant conditions as 55% power with the 'A' and 'B' SGFPs operating in automatic control. Subsequently, annunciator SGFP 'A' TURBINE TRIP (C-05, A-4), alarms. 'A' SGFP is NOT tripped. 'A' SGFP speed is 5550 rpm and rising.

Question 45 asks, "In accordance with ARP 2590D-013, SGFP A TURBINE TRIP, which of the following describes the correct operator response?"

Answer A (Place 'A' SGFP in Manual and Ensure FRV D/P between 40 and 150 psid). Millstone 2 proposes that this is a correct action because OP-AA-100, "Conduct of Operations, Section 5, Manual Control of Automatic Systems," would have the operator take manual control of a controller that is not operating properly in automatic. Additionally, they justify adjusting FRV D/P to between 40 and 150 psid because OP 2204, "Load Changes," has the operator maintain Feed Reg. Valve Δ P between 40 and 150 psid. These actions are not directed from ARP 2590D-013, SGFP A TURBINE TRIP, and therefore the NRC does not accept 'A' as a correct answer.

Given the stem condition that the SGFP 'A' speed is 5550 rpm and rising and NOT tripped, the applicants were expected to recognize that SGFP 'A' speed was above the electrical trip setpoint (5400 rpm), trending toward the mechanical trip setpoint (5600 rpm), and take action to manually trip SGFP 'A'. This would lead to possible correct answers of 'B' and 'C'. Based upon the stem condition of 55% power, IAW ARP 2590D-013, SGFP A TURBINE TRIP, the correct choice would be 'C' due to reactor power being less than 70%. IAW OP-AA-1800, Operator Fundamentals, Attachment 1, 2e, "Anticipate automatic trips and equipment protective features; and take manual actions if possible without haste, to avoid challenging automatic actuations."

Answer C was to TRIP 'A' SGFP, then OPEN CNM-2, COND DEMIN BYP. ARP 2590D-013, SGFP A TURBINE TRIP directs, OBSERVE C-05 indications and DETERMINE if "A" SGFP tripped. Step 3 states "IF "A" SGFP not tripped, PERFORM the following:

3.1 REVIEW PPC alarm typer for "WGA-DIF" (initiated if "A" SGFP Trip Header oil pressure switch PS-16 failed open and the "A" SGFP has not tripped).

3.2 REQUEST I&C to troubleshoot PS-16.

Since there is no direction to troubleshoot the condition or take the controller to manual, and no direction to trip the pump in the ARP, the NRC agrees that answer 'C' is incorrect. Therefore, the NRC concludes that there is no correct answer and question 45 should be removed from the exam.

Question #45

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 059A3.09

Importance Rating: 3.6

K/A Statement: Ability to monitor automatic features of the Main Feedwater System, including:
MFW pump trips

Proposed Answer: C

Justification: The Electrical Overspeed Trip setpoint is 5400 rpm, so the SGFP must be tripped manually since it has failed to automatically trip (this guidance is found in OP-AA-1800, Operator Fundamentals). SGFP 'A' TURBINE TRIP A-4 step 4 is N/A because one SGFP is still operating, and reactor power is less than 70% (a reactor trip is not required). Step 5 says to OPEN CNM-2, COND DEMIN BYP.

Plausibility:

- A. Wrong, the A SGFP must be tripped because it has exceeded its overspeed setpoint of 5400 rpm. Plausible; Candidate may confuse the mechanical overspeed trip setpoint with the electrical overspeed trip setpoint or may not realize the electrical overspeed trip happens at a lower speed than the mechanical trip. Also alarm A-1, SGFP A TROUBLE, states the operator "May place the 'A' SGFP in "MANUAL" speed control, but that would be wrong based on the current MFP speed.
- B. Wrong, the action to trip the reactor is not required because the 'B' SGFP is still operating, and the reactor power is less than 70 %. Plausible; Candidate may not interpret the power conditions that are given in the stem correctly, which allow continued operation with a single SGFP below 70%.
- C. Correct.
- D. Wrong, the A SGFP must be tripped because it has exceeded its overspeed setpoint of 5400 rpm. It would be correct to check the additional indications once the more pressing actions are taken. Plausible; Candidate may not identify the 'A' SGFP has exceeded a trip criteria, and may think manual operation and additional investigation is the correct path forward per A-4 alarm response step 3.

Technical Reference(s):

- ARP 2590D-013, A-4 SGFP A TURBINE TRIP
- OP-AA-1800, Operator Fundamentals

Provided reference(s): None

Learning Objective: List the automatic Steam Generator Feedwater pump trips, and relate each trip to the indications associated with that trip as given in MFW-00-C (281650).

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(4): Secondary coolant and auxiliary systems that affect the facility.

Comments: Changed power level to 55% based on validator feedback. DF. Changed SGFP speed to 5550 rpm based on validator feedback. DF 8/14

Question #46

The plant tripped from 100% power.

- The 'A' Motor Driven Auxiliary Feed Pump starts on AFAS.

Compared to a normal surveillance run, the 'A' Motor Driven Auxiliary Feed pump's...

starting current on a AFAS start 1. as/than a surveillance run.

AND

running current on a AFAS start 1. as/than a surveillance run.

- A.
 - 1. decays more quickly
 - 2. is the same
- B.
 - 1. decays more quickly
 - 2. is higher
- C.
 - 1. takes longer to decay
 - 2. is the same
- D.
 - 1. takes longer to decay
 - 2. is higher

Question #46

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 061191005 K1.05 Importance Rating: 2.8

K/A Statement: AUXILIARY / EMERGENCY FEEDWATER SYSTEM MOTORS AND GENERATORS: Explanation of the difference between starting current and operating (running) current in a motor

Proposed Answer: D

Justification: An AFAS signal causes the 'A' Auxiliary Feedwater (AFW) Regulating valve to fail open. Starting an AFW pump in this condition will cause the starting current to last longer and the running current to be higher than if the AFW Regulating valve were closed when the pump started (as is the case during a surveillance run).

Plausibility:

- A. The examinee may think starting current decays quicker on a AFAS start due to the smaller difference between starting current and running current. Plus, they may think the running current in both cases is so much lower than starting current, they can be considered equal
- B. The examinee may think starting current decays quicker on a AFAS start due to the smaller difference between starting current and running current.
- C. The examinee may think the starting current lasts longer because the discharge valve is open but also think the running current is the same due to higher post-trip S/G pressures.
- D. Correct.

Technical Reference(s):

- INPO GFES 191004 Pumps Student Guide

Provided reference(s): None

Learning Objective: GFES 191004, Pumps, ELO 1.7 Explain the operation of centrifugal pumps to include priming, starting, and net positive suction head.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Reworded question to compare motor amperage during an AFAS actuation with a surveillance run. DF 5/28. Still had 2 correct answers after first change. Modified A & B to have starting current decay time opposite than C & D. Changed quicker to more quickly in distractors so they are less "labored" to read. SRM 7/26.

Question #47

A surveillance run on the 'B' Diesel Generator (DG) is being performed.

- While removing the 'B' DG from service, its output breaker trips on reverse power.
- A subsequent Loss of Normal Power (LNP) occurs on bus 24D.
- The following annunciators on C-08 are acknowledged and reported to the US:
 - DIESEL GEN 13U BKR TRIP (C-08, A31)
 - DIESEL GEN 13U TROUBLE (C-08, B36)
 - DIESEL GEN 13U BKR CLOSING CKT BLOCKED (C-08, D-36)

Which of the following actions must be taken before the 'B' DG can power bus 24D?

- A. Reset the 'B' DG breaker lockout in the 4160 electrical room.
- B. Press the alarm reset button on the 'B' DG skid mounted panel.
- C. Reset the 'B' DG output breaker using the breaker hand switch on C-08.
- D. Manually stop and then restart the 'B' DG from the controls on C-08.

Question #47

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 062K4.03

Importance Rating: 3.6

K/A Statement: Knowledge of AC ELECTRICAL DISTRIBUTION SYSTEM design features and/or interlocks that provide for the following: Interlocks between automatic bus transfer and breakers

Proposed Answer: C

Justification: Reset the 'B' DG output breaker using the breaker hand switch on C08. As specified in the ARP, the breaker is reset using the output breaker hand switch on C08. A caution in the DG procedure states, "It is acceptable to reset the reverse power relay and the breaker as specified in the ARP and make a second attempt." The ARP specifies IF breaker tripped on reverse power while removing DG from service OR maintenance has been performed on the breaker, PERFORM the following to reset alarm (C-08): 2.1 ENSURE SYN SW 15G-13U-2 (A401) in OFF, 2.2 TURN DG B FDR BKR 15G-13U-2 (A401) to CLOSE. and 2.3 TURN DG B FDR BKR 15G-13U-2 (A401) to TRIP. This resets the breaker. The breaker will now close in response to the LNP and power bus 24D.

Plausibility:

- A. Wrong, have a PEO reset the 'B' DG breaker lockout in the 4160 electrical room. The breaker is reset using the breaker hand switch on C08 and not locally at the breaker. Plausible; Some breaker trips are reset by the relay targets. The DIESEL GEN 13U DIFFERENTIAL LOCKOUT ARP requires resetting relay targets locally at the breaker. It is normal practice to send a PEO to check breaker relay targets.
- B. Wrong, have a PEO press the alarm reset button on the 'B' DG skid mounted panel. This will not reset the DG breaker hand switch to allow breaker closure. Plausible; If a LNP condition exists and the diesel engine has been shut down by an emergency trip, the engine can be restarted by pressing master RESET button on the diesel engine skid mounted gage board. In this case there is a LNP but the diesel engine has not been shut down. The diesel is running with the output breaker open.
- C. Correct.
- D. Wrong, Manually stop and then restart the 'B' DG from the controls on C08 will not reset the breaker. Plausible; It is reasonable for the examinee to think that if they shut the DG down fully that it will clear all interlocks and reset the breaker.

Technical Reference(s):

- ARP 2590F-121, A-31 DIESEL GEN 13U BKR TRIP
- ARP 2590F-116, D-29 DIESEL GEN 13U BKR TROUBLE
- ARP 2590F-140, D-35 DIESEL GEN 13U DIFFERENTIAL LOCKOUT
- ARP 2590F-144, D-36 DIESEL GEN 13U BKR CLOSING CKT BLOCKED
- OP 2346C, 'B' Emergency Diesel Generator

Provided reference(s): None

Learning Objective: 281331 Describe the functions of each Emergency Diesel Generator System control at panel C-08, including how the components are affected by each mode or position.

Source: Bank Vision 459264

Cognitive Level: Memory

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Changed cognitive level to Memory based on being designated as Memory on the 2018 exam.

Question #48

The plant tripped from 100% power.

- Bus 24C is DEAD.
- ALL bus 24C indicating lights are extinguished (C-08).

The 'A' Diesel 1. be loaded.

AND

What actions are directed by EOP 2525, Standard Post Trip Actions?

- A.
 - 1. CAN
 - 2. Close A312 locally.
- B.
 - 1. CAN
 - 2. Operate the 'A' EDG locally.
- C.
 - 1. CANNOT
 - 2. Trip the 'A' EDG from C-08.
- D.
 - 1. CANNOT
 - 2. Trip the 'A' EDG locally.

Question #48

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 063K1.05

Importance Rating: 4.4

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the DC ELECTRICAL DISTRIBUTION SYSTEM and the following systems: EDG

Proposed Answer: D

Justification: The loss of DC bus DV-10 is indicated by no lights lit on A312. A loss of DV-10 requires the 'A' EDG to be secured locally by tripping the fuel racks and closing the air start valves

Plausibility:

- A. The examinee may think the A312 lights are out due to a blown fuse and the breaker can be operated locally.
- B. The examinee may think remote control of the EDG is lost, so local control can be used to operate the EDG.
- C. The examinee may think the A EDG can be tripped remotely from C-08.
- D. Correct.

Technical Reference(s):

- AOP 2506A, Loss of Vital 125 VDC Instrument Panel DV10.

Provided reference(s): None

Learning Objective: Given a set of plant conditions, explain the mitigating actions taken for the EDG per AOP 2506A or B before and after re-energizing the associated DV10 or DV20 load circuit.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(8): Safely operate the facilities' auxiliary and emergency systems, including operation of those controls associated with plant equipment that could affect reactivity or the release of radioactive materials to the environment.

Comments: from 222 phone call added EOP 2525 and changed 'all A312' lights to 'bus 24C lights'. The lights were changed to provide absolute indication that 125VDC bus DV-20 was DEAD. DF 05MAR

Question #49

The plant is operating at 100% power.

- 'B' EDG Starting Air header pressure = 210 psig and lowering.
- The leak size is within the capacity of one (1) Starting Air compressor.
- Annunciator 480V BUS 22E TO B51, B52 BKR TRIP (C-08, A-19), alarms.
- **MCC B-52** is DEAD.

With these plant conditions, when the starting air tank pressure reaches 1.,

the 'B' EDG 2. Starting Air compressor will start.

- A. 1. 200 psig
2. DC
- B. 1. 180 psig
2. DC
- C. 1. 180 psig
2. AC
- D. 1. 200 psig
2. AC

Question #49

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 064K2.01

Importance Rating: 3.0

K/A Statement: Knowledge of electrical power supplies to the following: EMERGENCY
DIESEL GENERATOR SYSTEM Starting air compressor

Proposed Answer: D

Justification: B-52 Vital MCC has become de-energized. This is the power supply for the 'A' EDG AC Starting Air Compressor. The 'B' EDG is the header (with slowly lowering starting air header pressure) is separate and independent from the 'A' header. Therefore, the 'B' AC Starting Air compressor will start at 200 psig as designed.

Plausibility:

- A. Wrong, The DC air compressor will not start because pressure will not lower to its 180 psig setpoint. The AC compressor starts at 200 psig and will raise pressure to 230 psig before securing. Plausible; Candidate may confuse B52 for B-62, and think the DC compressor will need to start, and they also confuse the starting setpoint between the AC and DC compressors.
- B. Wrong, The DC air compressor will not start because pressure will not lower to its 180 psig setpoint. The AC compressor starts at 200 psig and will raise pressure to 230 psig before securing. Plausible; Candidate may confuse B-52 for B-62, and think the DC compressor will need to start.
- C. Wrong, The AC compressor will start at 200 psig, not 180 psig. Plausible; Candidate may confuse the auto start setpoints between the AC and DC compressors.
- D. Correct.

Technical Reference(s):

- ARP 2590F-073, 480V BUS 22E TO B51, B52 BKR TRIP (A-19)
- OP 2344B, 480 Volt Motor Control Centers
- EDG-00-C, Emergency Diesel Generators Lesson Plan
- IHE-00-C, In-House Electrical System Lesson Plan

Provided reference(s): None

Learning Objective: 281345 - Describe the effect on Diesel Generator operation of a malfunction or loss of the following: b) AC Electrical Distribution System

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(8): Components, capacity, and functions of emergency systems.

Comments: Modified question: removed initial power level and moved 'B' EDG air leak as initial condition. Removed bullet about leak size within the capacity of a starting air compressor. All based on validation review and comments. DF 5/7/24. Put bullet about leak size back into question to remove possibility of two correct answers. DF 6/7/24. Swapped pressures and

compressor types in distractors to align with the stem as written SM 6/21/24. Bolded MCC-B52 to emphasize what MCC is de-energized. DF 8/28

Question #50

RM 4262, Steam Generator Blowdown Radiation Monitor, fails HIGH.

Which of the following contains ONLY those valves which will receive a close signal?

- A.
 - 2-MS-135, SGBD Quench Tank pump discharge valve
 - 2-MS-15, SGBD Blowdown Tank Discharge valve
 - 2-MS-191A & B, SG sample isolation valves
- B.
 - 2-MS-135, SGBD Quench Tank pump discharge valve
 - 2-MS-15, SGBD Blowdown Tank Discharge valve
 - 2-MS-220A & B, SGBD isolation valves
- C.
 - 2-MS-220A & B, SGBD isolation valves
 - 2-HV-4287 & 4288, blowdown sample discharge to secondary sample sink
 - 2-MS-117A & B, DIS TO QUENCH TK
- D.
 - 2-HV-4287 & 4288, blowdown sample discharge to secondary sample sink
 - 2-MS-145A & B, Discharge to blowdown tank
 - 2-MS-135, SGBD Quench Tank pump discharge valve

Question #50

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 073A2.01

Importance Rating: 3.9

K/A Statement: Ability to (a) predict the impacts of the following on the PROCESS

RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: PRM component failures

Proposed Answer: B

Justification: SGBD RM high closes all potential discharge paths to prevent the release of activity. Not closing the sample isolations allows continued monitoring of SGBD.

Plausibility:

- A. WRONG; MS-191A&B do not receive an isolation signal on a Radiation monitor alarm. Plausible; Examinee may confuse the signal that closes the SG sample isolation valves 2-MS-191A & B which is a CIAS.
- B. Correct.
- C. WRONG; DIS TO QUENCH TK MS-117A/B does not receive an isolation signal. Plausible; Examinee may think the Quench Tank gets isolated to prevent an unmonitored release.
- D. WRONG; Discharge to blowdown tank MS-145A/B does not receive an isolation signal. Plausible; Examinee may think the Blowdown Tank gets isolated to prevent an unmonitored release.

Technical Reference(s):

- ARP 2590H-005A, RIC-2300A CH.1 S/G BLOWDOWN GROSS ACTIVITY RM - 4262.

Provided reference(s): None

Learning Objective: 282110 – Describe the automatic actions and logic associated with the SJAE and/or SG Blowdown Radiation Monitoring Subsystem(s).

Source: Bank Vision 452113, NRC 2016 Q51

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments: Modified question to apply to Clean Waste because Aerated Waste discharges are no longer normally performed. SRM 7/26. Replaced with Vision question 452113 based on previous question's similarity to Q93. Made the stem say "fails high" instead of "exceeding alarm setpoint" to better align with the K/A. SRM 8/14.

Question #51

The plant is at 100% power.

- Service Water (SW) Pump 'A' and SW pump 'C' are running.
- Bus 24E is aligned to bus 24C.
- A Loss of Offsite Power (LOOP) has occurred.
- The 'B' Emergency Diesel Generator (EDG) output breaker has tripped.

Assuming no operator action, how many Service Water Pumps are running after two minutes?

- A. 0
- B. 1
- C. 2
- D. 3

Question #51

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 076K2.01

Importance Rating: 3.8

K/A Statement: Knowledge of electrical power supplies to the following: SWS pumps (Class 1E)

Proposed Answer: B

Justification: The 'A', 'B', and 'C' SW pumps are powered by the 24C, 24D, 24E 4160 V vital buses respectively. The normal configuration is to run the 'A' and 'C' SW pumps and have the 'B' SW pump in Pull-to-Lock (PTL), with 24E aligned to either the 24C or 24D. On a LOOP, the 24C bus is eventually powered by the 'A' EDG but the 24D bus remains deenergized due to the 'B' EDG output breaker fault. This configuration results in only the 'A' SW pump running.

Plausibility:

- A. Wrong, the 'A' EGD LOOP sequencer will start the 'A' SW pump on a loss of power to the 24C vital bus. The 'B' and 'C' SW pumps will not be running. Plausible; Candidate may know the 'B' SW pump is in PTL and the 'C' SW pump has no power. They may not identify the 'A' SW pump as a load automatically restored by the 'A' EDG.
- B. Correct.
- C. Wrong, the 'A' EGD LOOP sequencer will start the 'A' SW pump on a loss of power to the 24C vital bus. The 'B' and 'C' SW pumps will not be running. Plausible; Candidate may not recall that the 'B' SW pump is in PTL when the 24E is aligned to the 24C bus to prevent overloading the 'A' EDG in the event of a LOOP/LOCA scenario.
- D. Wrong, The 'A' EGD LOOP sequencer will start the 'A' SW pump on a loss of power to the 24C vital bus. The 'B' and 'C' SW pumps will not be running. Plausible; Candidate may not recall that the 'B' SW pump is in PTL when the 24E is aligned to the 24C bus to prevent overloading the 'A' EDG in the event of a LOOP/LOCA scenario. They also may not see that the loss of power to the 24D bus caused by the 'B' EDG output breaker overcurrent has occurred.

Technical Reference(s):

- 345-00-C, 345 KV Transmission System Lesson Plan
- EDG-00-C, Emergency Diesel Generators Lesson Plan

Provided reference(s): None

Learning Objective: 282306 Describe the effects on the Service Water System of a loss or malfunction of the following: A) AC Electrical Distribution System (4.16 kVAC, 480 VAC).

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(4): Secondary coolant and auxiliary systems that affect the facility.

Comments:

Question #52

The plant is performing OP 2207, Plant Cooldown.

- Shutdown Cooling is in service with Concurrent RCP Operation.
- Service Water injection temperature = 50° F.
- T_{COLD} = 220° F and stable.
- 2-SI-306, SDC Total Flow Control valve, is in manual at 50% open.
- 2-SI-657, SDC System Hx Flow Control valve, is in manual at 50% open.
- Both SDC Heat Exchangers are in service.

The control power fuses blow for 2-SW-8.1A, 'A' RBCCW HX Service Water Outlet TCV.

'A' Service Water pump will be 1. runout conditions.

AND

'A' RBCCW header flow will 2..

- A. 1. closer to
2. stay the same
- B. 1. closer to
2. decrease
- C. 1. further from
2. stay the same
- D. 1. further from
2. decrease

Question #52

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 076K5.02

Importance Rating: 3.2

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the SERVICE WATER SYSTEM:

Pump runout

Proposed Answer: A

Justification: A failure of control power will cause SW-8.1A to fail open, causing a cooldown of the RCS. The higher flow will bring the 'A' service water pump closer to runout. RBCCW flow will remain the same since there are no loads with Temperature Control valves on the 'A' RBCCW header.

Plausibility:

- A. Correct.
- B. Service water flow will increase (closer to runout); the examinee may think there is temperature-controlled load on the 'A' RBCCW header.
- C. The examinee may think a loss of control power would cause SW-8.1A to fail closed.
- D. the examinee may think a loss of control power would cause SW-8.1A to fail closed and there is temperature-controlled load on the 'A' RBCCW header.

Technical Reference(s):

- Service Water System Lesson Plan pg. 15
- RBCCW system drawing

Provided reference(s): None

Learning Objective: 282302 – Describe the effects of a loss or malfunction of the Service Water system on the following:

A) Reactor Building Closed Cooling Water system

Source: Modified 451977, 2016 NRC exam #53

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Added references to pump runout to better align with the K/A. SRM 7/26.

RO and SRO Exam Questions (No "Parents" Or "Originals")

Question #: **53**

Question ID: **2016022**

☒ **RO**

☐ **SRO**

☐ Student Handout?

☐ Lower Order?

Rev. **1**

☒ Selected for Exam

Origin: **New**

☐ Past NRC Exam?

The Plant is currently performing a cooldown for a Mid Cycle Shutdown with Service Water injection temperature at 50°F.

Current conditions:

- "A" and "B" RCPs operating
- "A" Shutdown Cooling pump running
- Both SDC heat exchangers in service.
- 2-SI-306 "SDC Total Flow Control Valve" is 50% open
- 2-SI-657 "SDC System HX Flow CNTL" is 50% open
- TCOLD is at 220°F and Stable

The Control Room has suspended the Cooldown while racking down 2 RCP Breakers.

An Operator mistakenly removes Fuse Block for 2-SW-8.1A "A" "RBCCW HX Service Water Outlet TCV" for the in service HX while performing action for a Clearance on Service Water.

What effect would this action have on Reactor Coolant temperature with no Operator action?

-
- ☐ **A** No effect on RCS temperature because the RBCCW Service Water HX outlet valves are throttled to prevent run out of the Service Water pumps.
- ☐ **B** No effect to RCS temperature because RBCCW flow to the SDC HX is throttled to minimize temperature transients on the RCS when SDC is in service.
- ☒ **C** RCS temperature would start to lower because RBCCW temperature will lower when Service Water flow to the RBCCW HX increases.
- ☐ **D** RCS temperature would start to rise because RBCCW temperature would rise when Service Water cooling is isolated to the RBCCW HX.

Question Misc. Info: MP2*LOIT SW, SDC, 2207, NRC-2016

Justification

"A" WRONG; The Service Water Valves are throttled to maintain RBCCW header temperature and fail open on a loss of power and for the conditions stated in the stem a raise in SW flow will reduce RBCCW header temperature.

Plausible; A loss of power to SW-8.1A fails open and SW-9A/B/C is throttled to the non-service RB HX for minimum flow during the winter.

"B" WRONG; While RBCCW flow does not change, SW header flow rises overcooling RBCCW thus cooling the RCS

Plausible; It is correct that the RBCCW flow is manually throttled to the SDC HX could cause a misunderstanding to the Examinee that temperature would not change.

"C" CORRECT; SI-306 and SI-657 at 50% coupled with stable RCS temperatures informs the Examinee that the loading on the Service Water System is nowhere near capacity, therefore de-energizing SW-8.1A to its LOCA position will raise the flowrate through the on service RBCCW HX therefore lowering RBCCW temperatures.

"D" WRONG; The TCV/SIAS SW outlet valve for the RBCCW HX fails open therefore SW header flow rises overcooling RBCCW thus cooling the RCS.

Plausible; The Examinee may think that the Temperature Control Valve for the RBCCW HX fails closed.

References

25203-26008 SH-02

Comments and Question Modification History

Per Validation; Added SW injection to stem and removed plural of temperature and HX.

NRC; changed "on service HX" in stem to "in service HX".

NRC K/A System/E/A System 076 Service Water System (SWS)

Number K1.08 **RO** 3.5* **SRO** 3.5* **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: RHR system

Question #53

Unit 2 is operating at 90 % power.

The following valves have experienced a loss of instrument air to their valve operators:

- 2-HD-103E, Feedwater Heater 4A Normal Level Control Valve
- 2-HD-102E, Feedwater Heater 4A High Level Control Valve

Describe the effect on the plant.

- A. Main turbine efficiency will rise.
- B. Reactor power will lower.
- C. Overall plant efficiency will lower.
- D. Heater Drain Tank level will rise.

Question #53

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 078K3.07

Importance Rating: 2.7

K/A Statement: Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following systems or system parameters: CDS

Proposed Answer: C

Justification: 2-HD-103E, FWH 4A Normal Level Control Valve fails closed on a loss of instrument air and 2-HD-102E, FWH 4A High Level Control Valve fails open on a loss of instrument air. When the valves reposition, FWH 4A High Level Dump valve drains the shell side water to the condenser which severely reduces the amount of heat being transferred to the condensate flowing through the heat exchanger's tube side. Less preheating of condensate leads to cooler feedwater, requiring the reactor to produce more thermal MW to maintain the same S/G pressure and steam flow that existed before the event. Overall plant efficiency will lower.

Plausibility:

- A. WRONG; Main turbine efficiency will not change. Plausible; Examinee may think that the extraction steam will be isolated and the isolation of that steam will result in more steam to the main turbine.
- B. WRONG; Reactor power will rise, if any change is noted, due to the loss of preheating of feedwater and its effect on reactivity in the core. Plausible; Examinee may mistake the fail positions of HD-103E and HD-102E and this could cause them to correlate a change in reactor power in the wrong direction.
- C. Correct.
- D. WRONG; FWH 4A(B) does not drain to the Heater Drains Tank, there should be no observable change in HDT level from this failure. Plausible; Examinee may think FWH 4A High Level Control Valve fails open and this will dump FWH 4A to the HDT and cause HDT level to rise.

Technical Reference(s):

- MP-PROC-OPS-OP 2320 REV. 023, Feedwater Heater Drains and Vents
- Course 55716 REV. 006, Extraction Steam & Feedwater Heater Drains

Provided reference(s): None

Learning Objective: 281404 - Describe the effects of a loss or malfunction of the Heater Drains and Vents system on the following: 1. Condensate System 2. Feedwater System

Source: New

Cognitive Level: Comprehension/Analysis

10CFR55.41 Content: b(4) Secondary coolant and auxiliary systems that affect the facility.

Comments:

Question #54

The plant is operating at 100% power.

- CTMT pressure = + 30 "H₂O (1.1 psig).
- CTMT temperature = 125 °F.

According to LCO 3.6.1.4, INTERNAL PRESSURE and LCO 3.6.1.5, TEMPERATURE, what is the status of Containment OPERABILITY?

- A. OPERABLE
- B. NOT OPERABLE due to pressure only.
- C. NOT OPERABLE due to temperature only.
- D. NOT OPERABLE due to both pressure and temperature.

Question #54

☒ RO

☐ SRO

Tier # 2

Group # 1

K/A # 103 G2.1.19 Importance Rating: 3.9

K/A Statement: CONTAINMENT SYSTEM EQUIPMENT CONTROL: Ability to use available indications to evaluate system or component status

Proposed Answer: D

Justification: Pressure exceeds the Technical Specification of 1.0 psig and temperature is greater than the 120° F limit.

Plausibility:

- A. The examinee may think both temperature and pressure are within normal limits. Temperature because it's close to normal and pressure because it's less than the SIAS/MSI setpoint.
- B. The examinee may think temperature is within normal limits because it's close to normal operating temperature (~110° F).
- C. The examinee may think pressure is within normal limits because it's 4x less than the SIAS/MSI setpoint (4.42 PSIG).
- D. Correct.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications 3.6.1.4, INTERNAL PRESSURE
- Millstone Unit 2 Technical Specifications 3.6.1.5, AIR TEMPERATURE

Provided reference(s): none

Learning Objective: 280970 – Given any operating condition for the primary Containment or Enclosure Building, state whether the condition requires entry into the Technical Specifications.

Source: New

Cognitive Level: Memory

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Question #55

The unit is operating at 45% power.

- Pressurizer Level Control is selected to channel 'Y'.
- Annunciator 'PRESSURIZER CH Y LEVEL HI/LO' (C-02/3 A-38), alarms.
- Pressurizer Level indicator, LI-110Y, indicates 100%

Due to the plant response, Reactor Coolant inventory will 1..

AND

In accordance with ARP 2590B-217, A-39 PRESSURIZER CH Y LEVEL HI/LO, the board operator will 2..

- A.
 - 1. increase
 - 2. select pressurizer level control to channel 'X'
- B.
 - 1. increase
 - 2. deselect the faulty input for RRS on the Foxboro IA
- C.
 - 1. decrease
 - 2. select pressurizer level control to channel 'X'
- D.
 - 1. decrease
 - 2. deselect the faulty input for RRS on the Foxboro IA

Question #55

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 011K1.02

Importance Rating: 4.2

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the PRESSURIZER LEVEL CONTROL SYSTEM and the following systems: RCS

Proposed Answer: C

Justification: When the controlling level instrument channel of pressurizer level control (CH Y in this instance) fails high it will show 100% pressurizer level. The Pressurizer Level Control System will automatically respond to the perceived erroneous high-level indication, which will immediately send a signal to the letdown flow control flow to maximize letdown flow. This results in a decrease in reactor coolant inventory. In accordance with A-39 PRESSURIZER CH Y LEVEL HI/LO step 9.7 the operator will shift pressurizer level control to channel 'X'.

Plausibility:

- A. Wrong, RCS inventory will decrease. Plausible: may confuse the system response with an instrument failure in the minimum direction.
- B. Wrong, RCS inventory will decrease and RRS has no faulty input to deselect. Plausible: student may believe there is a problem with the setpoint from RRS rather than the instrument output.
- C. Correct.
- D. Wrong, RRS has no faulty input to deselect. Plausible: student may believe there is a problem with the setpoint from RRS rather than the instrument output.

Technical Reference(s):

- PLC-01-C, Pressurizer Level and Pressure Control System Lesson Plan
- ARP 2590B-217, A-39 PRESSURIZER CH Y LEVEL HI/LO
- OP 2382, Pressurizer Pressure and Level Control System

Provided reference(s): None

Learning Objective: 281907 Given the plant with a steam bubble in the pressurizer, and given a pressurizer level or pressure transmitter failure (high or low) on either control channel (selected or non-selected), describe: a) The system response that would result from this failure, b) The actions necessary to mitigate this failure, c) The plant response if no operator actions are taken.

Source: Modified Bank St. Lucie 2023 exam

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(3): Mechanical components and design features of the reactor primary system.

Comments: Changed "select RRS selector switch to RRS-1 position" to "deselect faulty input for RRS at the Foxboro IA" and removed references to RRS selector switch in the stem and plausibility statements due to Millstone 2 not having a RRS selector

switch. Changed source to Modified Bank to be consistent in designating industry bank questions as Bank or Modified Bank. SRM 7/26.

PSL L-23-1 NRC INITIAL WRITTEN EXAM - RO

56.

Given the following:

- Unit 1 is 45% power and STABLE
- Pressurizer (PZR) Level Control Selector Switch is selected to the 'X' position
- Reactor Regulating System (RRS) Selector is in the RRS 1 position

Subsequently:

- LI-1110X indicates 100%
- 1-AOP-01.10, Pressurizer Pressure and Level, is in progress

Which ONE of the following completes the statements?

Reactor Coolant Inventory will ____ (1) ____.

In accordance with 1-AOP-01.10, the event will be mitigated by placing ____ (2) ____.

A.	(1)	increase
	(2)	RRS selector switch to the RRS-2 position
B.	(1)	increase
	(2)	PZR level control selector switch to the 'Y' position
C.	(1)	decrease
	(2)	RRS selector switch to the RRS-2 position
D.	(1)	decrease
	(2)	PZR level control selector switch to the 'Y' position

Question #56

The plant is operating at 100% power.

The 4 highest Channel 1 Core Exit Thermocouples are reading as follows:

- L16-T030 = 584 °F and stable.
- L4-T270 = 577 °F and stable.
- R4-T280 = 567 °F and stable.
- N11-T010 = 563 °F and stable.

Saturation temperature = 653 °F and stable.

If L4-T270 fails LOW and the PPC removes it from calculations, what subcooled margin would be indicated for CET_{HIGH}?

- A. 63 °F and stable.
- B. 76 °F and stable.
- C. 86 °F and stable.
- D. 88 °F and stable.

Question #56

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 017A1.01

Importance Rating: 4.0

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the IN-CORE TEMPERATURE MONITOR SYSTEM, including:
Core exit temperature

Proposed Answer: C

Justification: The Subcooled Margin Monitor (SMM) portion of ICCMS calculates the margin to saturation using the highest of the two available T_{HOT} and T_{COLD} signals and the second highest CET for CET_{HIGH} . If the second highest CET fails, the new second highest CET is used for calculating CET_{HIGH} . The difference between 653 °F and 567 °F is 86 °F.

Plausibility:

- A. The student may think CET_{HIGH} is the highest reading CET. The highest reading CET is CET_{MAX} .
- B. The student may think the CET value 'locks in' on a failure until the operator deselects the failed CET.
- C. Correct.
- D. The student may think CET_{HIGH} is the average of the second and third highest CET.

Technical Reference(s):

- OP 2387G, Inadequate Core Cooling System

Provided reference(s): None

Learning Objective: 281505 – Given a plant condition/mode, describe the expected indications of the Core Exit Thermocouples for the following conditions:

- A. Normal full power operations
- B. On SDC with RCS temperature at 200 °F
- C. LOCA with no RCS voiding
- D. LB LOCA with core uncover

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments: Clarified the failure mode of the failed CET. SRM 7/26.

Question #57

The plant is operating at 100% power, then:

- RPS Channel B 'LOW RCS FLOW' trips.
- All other parameters on Channel B are functioning normally
- All other channels of RPS are functioning normally.

What condition caused only the Channel B 'LOW RCS FLOW' to trip?

- A. Channel 'B' S/G D/P is reading half the value of the other three channels.
- B. Channel 'B' Core D/P is reading half the value of the other three channels.
- C. The 'B' RCP motor-impeller coupling has failed, but the motor is still operating.
- D. The 'B' RCP amps are reading zero and its breaker has the green and amber lights energized.

Question #57

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 022K6.06

Importance Rating: 2.6

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the REACTOR COOLANT SYSTEM: Sensors and detectors

Proposed Answer: A

Justification: The displayed 'B' S/G D/P is an average of the 'B' channel D/Ps across the S/G in each loop. This is the D/P used by RPS to determine proper RCS flow. The trip setpoint is 92% of actual flow. Indication of each safety channel's flow is displayed on C-03 as D/P.

Plausibility:

- A. Correct.
- B. Wrong, core D/P is for indication only, it has no input into RCS. Plausible; Candidate may think that core flow is the parameter used by RPS to generate trip signals.
- C. Wrong, this would cause a low flow trip on all four channels of RPS as all 4 channels would see a lower D/P. Plausible; Candidate may think that RCP speed is used for the trip and may think that shaft speed is used.
- D. Wrong, this would cause a low flow trip on all four channels of RPS as all 4 channels would see a lower D/P. Plausible; candidate may think this is a loss of indication of the 'B' RCP and that somehow is also going to generate a 'B' Channel low flow trip.

Technical Reference(s):

- RPS-01-C, Reactor Protection System Lesson Plan

Provided reference(s): None

Learning Objective: List the inputs to the Reactor Protection System from each of the following instrumentation systems (282155): E) Steam Generator

Source: Bank Vision 451481 NRC 2001 Q7

Cognitive Level: Comprehension/Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Swapped answers C & B to put dP answers together and RCP answers together.

Modified Core dP distractor to read the same as S/G dP to ensure examinee is differentiating only between Core and S/G, not magnitude of failure. Changes made based on validation results and review. DF 7MAY24

Question #58

Refueling operations are in progress.

- Containment Purge and Enclosure Building Purge are in service IAW OP 2314B.
- There are NO Temporary Modifications installed.
- RM-8262B, Containment Air Gaseous Radiation Monitor, alarms.

The Containment Purge valves will 1. and F-23, Purge Supply fan, will 2.

- A. 1. close
2. trip
- B. 1. close
2. stay running
- C. 1. remain open
2. trip
- D. 1. remain open
2. stay running

Question #58

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 029K4.03

Importance Rating: 3.6

K/A Statement: Knowledge of CONTAINMENT PURGE SYSTEM design features and/or interlocks that provide for the following: Automatic purge isolation.

Proposed Answer: B

Justification: CTMT Purge valves will close on a 1 out of 4 logic signal from the CTMT radiation monitors, RM-8123A/B and RM-8262A/B (CPVIS). The purge supply fan, F-23 will remain running (receives a trip signal from CIAS)

Plausibility:

- A. The examinee may think the Purge Supply fan will trip on a CPVIS in order to protect the fan. From a ESAS perspective, only a CIAS (Containment Isolation Actuation Signal) will trip the fan
- B. Correct.
- C. The examinee may think the CPVIS is 2 out of 4 logic, they also may think the fan trips due to no temporary modification (temp mod) being installed. There is a temp mod installed to prevent the fan from tripping on low ambient temperature.
- D. The examinee may think the CPVIS is 2 out of 4 logic.

Technical Reference(s):

- Engineered Safeguards Actuation System Lesson Plan, ESA-01-C

Provided reference(s): None

Learning Objective: 281380 - Describe each of the 11 Engineered Safeguards Actuation System (ESAS) actuations or trips with respect to:

- A) Conditions sensed and setpoints
- B) Purpose
- C) Functions

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Fuel handling facilities and procedures.

Comments: added procedure reference based on validator feedback. DF 8/15

Question #59

The plant is in MODE 5.

- 'B' Spent Fuel Pool (SFP) Cooling Pump is cooling the SFP.
- The RWST is NOT available as a make-up water source to the SFP.
- 'A' SFP Cooling Pump is out of service.

Subsequently,

- The 'B' SFP cooling pump trips on overload.

What is the PREFERRED method to cool the SFP as directed by AOP 2582, Loss of Spent Fuel Pool Cooling?

- A. Make up to the SFP using PMW when level lowers due to evaporation.
- B. Feed and bleed the SFP using AFW and the Clean Radwaste System.
- C. Feed and bleed the SFP using PMW and the Clean Radwaste System.
- D. Align Containment Spray and the SDC Hx to supply cooling to the SFP.

Question #59

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 033K1.08

Importance Rating: 2.6

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the SPENT FUEL POOL COOLING SYSTEM and the following systems: LRS

Proposed Answer: C

Justification: AOP 2582, Loss of Spent Fuel Pool Cooling, provides options for spent fuel pool cooling. Without the ability to run a SFP cooling pump, and with the RWST being unavailable, the correct procedural option is to feed and bleed the SFP using PMW and clean Radwaste system IAW AOP 2582 step 11.

Plausibility:

- A. Wrong, waiting for losses due to evaporation is not an approved form of SFP cooling. Makeup due to evaporation is performed occasionally, but not as an approved alternate means of cooling the SFP. Plausible; Candidate may think an elevated SFP temperature would accelerate the evaporation and make it reasonable to just cool via refilling the SFP.
- B. Wrong, using Auxiliary Feedwater is the last option for SFP cooling only to be done in emergency situations as a last effort to maintain cooling. Plausible; with multiple pieces of equipment unavailable, candidate may assume this is an emergency requiring AFW cooling to the SFP.
- C. Correct.
- D. Wrong, Containment Spray cannot be used to cool the SFP after the core has been reloaded, there is no procedural guidance for this. Plausible; CS system can be used to cool the SFP but only if the core is currently offloaded.

Technical Reference(s):

- AOP 2582, "Loss of Spent Fuel Pool Cooling"
- ARP 2590E-065, SFP PUMPS OVERLOAD/TRIP

Provided reference(s): None

Learning Objective: Identify the alternate sources of cooling for the SFP as described in AOP 2582, "Loss of Spent Fuel Pool Cooling." (283498)

Source: Modified Vision 414998

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(8): Components, capacity, and functions of emergency systems.

Comments: Adjusted stem to make it clear that it is not possible to open RW-280 to the RFP (changed conditions to MODE 5). SRM 7/26.

The plant is in a refueling outage with the following conditions:

- Core reload was completed and 2-RW-280 (transfer tube isolation valve) is CLOSED.
- "A" SFP Cooling Pump is out of service due to an emergent problem.

Then, the "B" SFP cooling pump seizes and trips on overload.

Which of the following would be the PREFERRED method to supply cooling to the SFP?

- A. Feed and bleed of the SFP using the RWST to fill the SFP to the Level HI alarm.
- B. Feed and bleed of the SFP using PMW and the Clean Radwaste System.
- C. Feed and bleed of the SFP using Aux Feedwater and the Clean Radwaste System.
- D. Align the Shutdown Cooling System to supply cooling to the SFP.

Answer: **A**

Answer Explanation

D is Wrong. AOP 2582 directs using SDC if it was supplying or was in service prior to the loss.
B & C wrong. Feed and Bleed would only be used after all other means of cooling have failed.
A is correct. Refuel water Purification pumps supplied from the RWST to the SFP.

Question 1 Info

Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	1.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	414998
Version ID:	661405
User-Defined ID:	0071637-MB
Cross Reference Number:	
Topic:	MP2*LORT*2582
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	

Question #60

The plant is operating at 100% power.

- The crew enters AOP 2518, Loss of Plant Process Computer.
- The PWR RATIO HI/LO switch is placed in the 'OPERATE' position (RC-22).
- Annunciator 'PWR RATIO HI/LO (C-04, BB-12) is in alarm.

What actions are directed by AOP 2518, Loss of Plant Process Computer?

REFER to 1.,

AND

PERFORM actions to reduce turbine load to 2..

- A.
 - 1. OP 2204, Load Changes
 - 2. clear the alarm
- B.
 - 1. OP 2204, Load Changes
 - 2. < 95% power
- C.
 - 1. AOP 2575, Rapid Downpower
 - 2. clear the alarm
- D.
 - 1. AOP 2575, Rapid Downpower
 - 2. < 95% power

Question #60

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 045A2.11

Importance Rating: 3.1

K/A Statement: Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Control problems in primary (e.g., axial flux imbalance); need to reduce load on secondary

Proposed Answer: C

Justification: With the PPC inoperable, AXIAL FLUX is determined by the ex-core nuclear instruments. With the alarm PWR RATIO HI/LO active, power is reduced to clear the alarm (within limits of ASI 'tent').

Plausibility:

- A. OP 2204 is not correct. The examinee may think that CEA motion to clear the alarm only requires a few steps so there is no reason to enter AOP 2575.
- B. OP 2204 is not correct. The examinee may think that CEA motion to clear the alarm only requires a few steps so there is no reason to enter AOP 2575. < 95% is not correct, the examinee may think 95% puts the plant squarely inside the ASI tent, preventing any need to perform further power manipulations.
- C. Correct.
- D. < 95% is not correct, the examinee may think 95% puts the plant squarely inside the ASI tent, preventing any need to perform further power manipulations.

Technical Reference(s):

- AOP 2518, Loss of Plant Process Computer

Provided reference(s): none

Learning Objective: 282987 – Determine the effects of a loss or malfunction of the Plant Process Computer on continued operation of the plant.

Source: New

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments: per 2/22 phone call: swapped 'turbine load' for 'power' to more closely match K/A.
DF 05MAR

Question #61

The plant is operating at 100% power

- A Loss of Coolant Accident (LOCA) occurs.
- SIAS, CIAS and EBFAS actuate.
- R8169RA, SPDS U2 WR EBFS RAD (Millstone Stack Radiation Monitor), indicates **greater than** ($>$) 1 decade above pre-event value.

What actions are directed by EOP 2532, Loss of Coolant Accident, to maintain condenser vacuum?

- A. Override and open 2-EB-55 and 2-EB-56, Condenser Air Removal isolation dampers.
- B. Swap Condenser Air Removal to F-55A and open EB-171, F-55A makeup damper.
- C. Bypass the Gland Seal steam regulator using 2-MS-182A, Bypass Feed MOV, and restore GS steam pressure to normal.
- D. Open 2-EB-57, Condenser Air Removal to Unit 2 Stack, and start one Main Exhaust Fan.

Question #61

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 055A4.04

Importance Rating: 3.0

K/A Statement: Ability to manually operate and/or monitor the CONDENSER AIR REMOVAL SYSTEM in the control room: Realign CARS exhaust path

Proposed Answer: D

Justification: Opening EB-57 and starting the exhaust fan provides Condenser Air Removal fan flow path, which was automatically isolated by the CIAS (ME fan) and EBFAS (stack flowpath), backing up non-condensable gases in the main condenser.

Plausibility:

- A. Wrong, EB-55 & 56 automatically close on EBFAS, reopening would parallel condenser air removal fan with EBFS for the discharge path, not procedurally allowed and wouldn't work. Plausible; Discharging radioactive steam (SGTR) out the station stack is considered an "air release" and is preferred to discharging out the Unit 2 stack, which is considered a "ground release."
- B. Wrong, No discharge flow path is available for the either Condenser Air Removal fan unless EB-57 is opened. Plausible; Higher capacity fan may overcome EBFS fan back pressure.
- C. Wrong, gland seal steam never was interrupted by the given ESAS signals. This would automatically happen on a MSI signal. Plausible; The normal gland seal regulator is known to stick closed on a trip as it is not open above ~20% power (glands self-seal then). This would be the expected action if the stem did not state that all equipment functioned as designed.
- D. Correct.

Technical Reference(s):

- EOP 2532, "Loss of Coolant Accident"

Provided reference(s): None

Learning Objective: Describe the effects of a loss or malfunction of the Condenser Air Removal System on the Main Condenser (280968)

Source: Bank Vision 451873, NRC 2011 Q63

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(4): Secondary coolant and auxiliary systems that affect the facility.

Comments: Restored 3rd bullet based on 5/8 phone call. Without 8169RA readings, the operator actions would be different.

Question #62

The condensate system is being filled in accordance with OP 2319A, Section 4.2, Filling Condensate System Downstream of Drains Cooler 'A' and 'B' Inlet, 2-CN-11A and 2-CN-11B.

What is the PRIMARY purpose of Step 4.2.8, "VENT Condensate System from the following points:"?

- A. To minimize oxygen transport to the Steam Generators.
- B. To minimize the chance of a oxide layer crud burst.
- C. To minimize the chance of pipe failure due to water hammer.
- D. To minimize air blanketing of Feedwater heater tubes.

Question #62

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 056K5.11

Importance Rating: 2.6

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the CONDENSATE SYSTEM: Reasons for venting all high points in CDS

Proposed Answer: C

Justification: The condensate system is vented while filling to prevent water hammer. After filling, the system is placed on long recycle which allows clean-up of the condensate and flushing of air to the condenser where it can be removed by the condenser air removal system.

Plausibility:

- A. Plausible, air is never desired in any fluid system (except air systems, of course). If the examinee doesn't think of the sequence of putting condensate in service, they may forget the system is put on long recycle to help clean-up the system. This includes flushing air to the condenser.
- B. Typically, there is an excess amount of iron from the oxide layer found in the condensate during system start-up. The primary reason the system is put on long recycle is to remove iron and other impurities prior to feeding the generator.
- C. Correct.
- D. After filling the system is placed on long recycle which flows water through the LP feedwater heaters. The flow provides a higher velocity (~3200 gpm) flushing which removes air from the FWH tubes and transports it back to the condenser.

Technical Reference(s):

- GFES 193006, Thermodynamics – Fluid Statics
- OP 2319A, Condensate

Provided reference(s): None

Learning Objective: GFES Fluid Statics and Dynamics (193006) ELO 4.3 – List actions that operators take to minimize the occurrence of water hammer

Source: New

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments: Emphasized PRIMARY purpose in stem. SRM 7/26.

Question #63

What radiation levels will result in an isolation of hydrogen purge?

- A. Control Room area radiation monitor (RM-7899) reading greater than 5 mR/hr.
- B. Containment Refuel Floor area radiation monitor (RM-7891) reading greater than 200 mR/hr.
- C. Containment High Range Area monitor (RM-8240) reading greater than 5 R/hr.
- D. Spent Fuel Pool area radiation monitor (RM-8142) reading greater than 100 mR/hr.

Question #63

☒ RO

☐ SRO

Tier # 2

Group # 2

K/A # 072K4.03

Importance Rating: 3.2

K/A Statement: Knowledge of AREA RADIATION MONITORING SYSTEM design features and/or interlocks that provide for the following: PVS isolation

Proposed Answer: C

Justification: Containment High Range Area monitors RM-8240/8241 Alarm 1 has a setpoint of 5 R/hr. At this point, when either monitor alarms, hydrogen purge will be isolated. RM-8240 closes EB-92 & EB-99.

Plausibility:

- A. Wrong, the Control Room area rad monitor has no automatic functions. Plausible: student may believe that activity from hydrogen purge may start to affect the Control Room envelope, and automatic isolation would be appropriate.
- B. Wrong, the Refuel Floor area rad monitor has no automatic functions. Plausible: the Refuel Floor area rad monitor radiation levels will rise when the High Range radiation levels rise.
- C. Correct.
- D. Wrong, Spent Fuel Pool area rad monitors need 2 out of 4 to actuate, and they actuate AEAS, but do nothing to Hydrogen Purge. Plausible: student may believe that hydrogen purge is isolated with an AEAS, and forget the 2 out of 4 actuation.

Technical Reference(s):

- ARP 2590A-119, "HYDROGEN PURGE ISOL VLVs HI RAD"

Provided reference(s): None

Learning Objective: Describe the automatic actions and logic associated with the Containment High Range Radiation Monitoring Subsystem (282105) (Question as originally written tied to objective 281110)

Source: Modified Vision 452064, NRC 2016 Q52

Cognitive Level: Memory

10CFR55.41(b)(11): Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments: Changed 'B' distractor from the proposed Control Room RM-8011 to Containment Refuel Floor RM-7891 to align with the other distractors (also area rad monitors) and to make it more plausible. Rewrote plausibilities to align better with the modified question, as the original plausibilities are more appropriate for the CRAC recirc question. Changed Learning Objective to align with the modified question.

RO and SRO Exam Questions (No "Parents" Or "Originals")

Question #: **52**

Question ID: **53448**

☒ **RO** ☐ **SRO**

☐ Student Handout?

☒ Lower Order?

Rev. **1**

☒ Selected for Exam

Origin: **Bank**

☐ Past NRC Exam?

Which of the following conditions, by itself, will result in the Control Room Ventilation System shifting automatically into the Recirculation Mode of operation?

- ☐ **A** Control Room area radiation monitor (RM-7899) reading greater than 10 mr/hr.
- ☐ **B** Control Room gaseous process radiation monitor (RM-8011) reading greater than 10 mr/hr.
- ☒ **C** Control Room ventilation duct radiation monitor (RM-9799A) reading greater than 100 mr/hr.
- ☐ **D** Spent Fuel Pool area radiation monitor (RM-8142) reading greater than 100 mr/hr.

Question Misc. Info: MP2*LORT*3134 [088 BPV-01-C 2760] (11/25/97) 2315A, CRAC, RM, NRC-2016

Justification

Meets the design limits for control room habitability for the operators during accident conditions. This is the only Rad monitor that the setpoint is related to design limits and not radiation release limits.

"A" WRONG; Although the Rad monitor is located inside the Control Room envelope it has no automatic function.

Plausible; The rad monitor located next to C08 alarms locally may cause the examinee to wrongly deduce an automatic function of shifting CRAC to recirculation mode.

"B" WRONG; Although the Rad monitor samples inside the Control Room envelope it has no automatic function.

Plausible; The process rad monitor samples the Control Room envelope may cause the examinee to wrongly deduce an automatic function of shifting CRAC to recirculation mode.

"C" CORRECT; RM-9799A/B, SIAS/CIAS/EBFAS signal and AEAS signal are the only inputs to automatically place CRAC in recirc mode.

"D" WRONG; Although the Rad monitor is part of the AEAS signal to CRAC it requires 2 of 4 area rad monitors to initiate AEAS therefore CRAC to recirc mode.

Plausible; The area rad monitor is part of the AEAS signal to CRAC

References

ARP 2590A-159

Comments and Question Modification History

Placed commas before and after "by itself"

NRC K/A System/E/A System 073 Process Radiation Monitoring (PRM) System

Number A1.01 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR: 41.5 / 45.7)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels

Question #64

You have been assigned to return a pump to service following maintenance.

- The procedure is in progress and was last used the previous day.

In accordance with AD-AA-102, Procedure Use and Adherence, the procedure 1. require to be revision verified, 2..

- A.
 - 1. DOES NOT
 - 2. the procedure was revision verified when it was printed.
- B.
 - 1. DOES NOT
 - 2. less than 24 hours has elapsed since pausing the procedure.
- C.
 - 1. DOES
 - 2. the procedure must be verified with the Procedure Index app.
- D.
 - 1. DOES
 - 2. the procedure must be verified with DocTop.

Question #64

☒ RO

☐ SRO

Tier # 3

Group #

K/A # G2.1.21

Importance Rating: 3.5

K/A Statement: CONDUCT OF OPERATIONS: Ability to verify that a copy of a controlled procedure is the proper revision

Proposed Answer: D

Justification: AD-AA-102, Section 3.3 states: Prior to starting task daily, CHECK procedure to be used is correct revision using one of the following sources:

- DocTop or electronic index-of-record (e.g., DMIS, FileNet)
- Forms only - Nuclear E-forms in EDMS (DocTop)
- Department's controlled procedure file
- Records Management

Plausibility:

- A. The examinee may think that since the procedure is rev verified when printed, that should suffice.
- B. The examinee may think since the procedure was verified the previous day and it's still in progress, it doesn't need to be verified again.
- C. The Procedure Index app provides a method to verify procedures but that verification is through DocTop. The procedures linked to the app are not updated on a regular basis.
- D. Correct.

Technical Reference(s):

- AD-AA-102, Section 3.3, Checking Latest Revision
- Vision Question 369877

Provided reference(s): None

Learning Objective: 251849 – STATE two methods acceptable to verify that a document to be used/in use is the current revision

Source: Bank

Cognitive Level: Memory

10CFR55.41(b) (10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Question #65

Unit 2 is performing a plant heat-up in accordance with OP 2201, Plant Heatup.

- 3 RCPs are running ('A', 'B', and 'C').
- RCS temperature = 505 °F and rising.
- RCS Pressure = 2250 psia and stable.

Preparations are in progress to energize Control Element Drive Mechanisms (CEDMs).

Identify the plant condition which, by itself, would allow CEDMs to be energized in accordance with LCO 3.1.3.7.

- A. High Power Trip is OPERABLE.
- B. A boron concentration ≥ 1720 ppm.
- C. A boron dilution is NOT in progress.
- D. Local Power Density Trip is OPERABLE.

Question #65

☒ RO

☐ SRO

Tier # 3

Group #

K/A # G2.1.37

Importance Rating: 4.3

K/A Statement: CONDUCT OF OPERATIONS: Knowledge of procedures, guidelines, or limitations associated with reactivity management

Proposed Answer: B

Justification: TS 3.1.3.7 states that the control rod drive mechanisms shall be de-energized in MODES 3*, 4, 5 and 6, whenever the RCS boron concentration is less than refueling concentration of Spec 3.9.1. TS 3.9.1 states that refueling boron concentration is > 1720 ppm or boron concentration that ensures $K_{eff} < 0.95$. The note for MODE 3* states that the control rod drive mechanisms may be energized for MODE 3 as long as 4 reactor coolant pumps are OPERATING, the reactor coolant system temperature is greater than 500°F , pressurizer pressure is > 2000 psia and the high power trip is OPERABLE. Since only 3 RCPs are running, this note does not apply; however, if RCS Boron concentration is greater ensures boron concentration that ensures $K_{eff} < 0.95$. OP 2201 step 4.10.4 which directs that if any of the conditions listed in step 4.10.2 do NOT exist (4 RCPS running, RCS Temp $> 500^{\circ}\text{F}$, Pressurizer Pressure > 2000 psia and High Power Trip Operable) AND RCS boron concentration is less than refueling boron concentration, VERIFY CEDMS de-energized. From the bases for TS 3.1.3.7, the drive mechanisms may be energized with the boron concentration greater than or equal to the refueling concentration since, under these conditions, adequate SHUTDOWN MARGIN is maintained, even if all CEAs are fully withdrawn from the core.

Plausibility:

- A. Wrong, because 4 Reactor Coolant Pumps are not running, the MODE 3 conditions to energize CEDMs are still not met. Plausible; TS 3.1.3.7 bases provides this as the reason for allowing CEDMs to be energized with 4 RCPs running, Pressurizer pressure > 2000 psia, RCS temperature $> 500^{\circ}\text{F}$ and the High Power Trip operable, the plant is within the bounds of the safety analysis for a CEA withdraw accident from subcritical core.
- B. Correct.
- C. Wrong, this does not satisfy the Technical Specification requirement for energizing CEDMs in MODE 3. Plausible; Preventing situations that cause two consecutive positive reactivity additions is a common reactivity management principle.
- D. Wrong, this does not satisfy the Technical Specification requirements for energizing CEDMs in MODE 3. Plausible; Applicant may confuse the High Power Trip operability requirement with the Local Power Density Trip.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications

Provided reference(s): None

Learning Objective: Given a set of plant conditions and a copy of Technical Specifications, identify the associated LCO ACTION requirements and plant operating procedures required actions (281014)

Source: Bank Vision 451950, NRC 2014 Q96

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: added ≥ 1720 ppm to correct answer in order to match TS. Simplified question to make it RO level. Per 725 phone call, removed second part to ensure not an SRO question. DF 7/26

Question #66

The unit is in MODE 4.

- The 'A' Emergency Diesel Generator (EDG) is declared NOT OPERABLE.

What ACTION is required to be performed within one (1) hour?

- A. Perform SP 2619G, AC Electrical Sources Inoperability.
- B. Verify the Terry Turbine is OPERABLE.
- C. Demonstrate the B EDG is not inoperable due to a common cause failure.
- D. Verify the Unit 3 EDGs are OPERABLE and the SBO is available.

Question #66

☒ RO

☐ SRO

Tier # 3

Group #

K/A # G2.2.12

Importance Rating: 3.6

K/A Statement: EQUIPMENT CONTROL: Knowledge of surveillance procedures

Proposed Answer: A

Justification: With a EDG declared inoperable, the operator is required to verify Electrical Source operability within one (1) hour and every 8 hours thereafter.

Plausibility:

- A. Correct.
- B. The Terry Turbine has to be verified OPERABLE within 2 hours (MODES 1, 2, and 3 only).
- C. The OPERABLE EDG has to be determined not inoperable due to a common cause failure within 24 hours.
- D. The Unit 3 EDGs are verified OPERABLE and SBO determined available within 1 hour only if the 14 day allowed outage time is utilized. Since the EDG is inoperable due to an emergent issue, the 14-day allowance cannot be used.

Technical Reference(s):

- EDG-00-C, Emergency Diesel Generators, Lesson Plan
- Millstone Unit 2 Technical Specifications 3.8.1.1.b, one diesel generator inoperable
- Comanche Peak 2011 NRC exam Q20

Provided reference(s): None

Learning Objective: state the Technical Specifications Action Statements associated with the following Emergency Diesel Generator operating conditions:

- a) One Emergency Diesel Generator inoperable .
- b) Two Emergency Diesel Generators inoperable .

Source: Comanche Peak 2011 NRC exam Q20

Cognitive Level: Memory

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments:

Question #67

LCO 3.5.1, SAFETY INJECTION TANKS, requires immediate entry into LCO 3.0.3 if two or more SITs are inoperable.

LCO 3.0.3 states “Within one (1) hour ACTION shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in ...”

Assuming that two (2) SITs are declared inoperable while the plant is in MODE 1, select the highest MODE the unit must be placed in to comply with LCO 3.5.1.

- A. MODE 5, COLD SHUTDOWN.
- B. MODE 4, HOT SHUTDOWN.
- C. MODE 3*, HOT STANDBY (* with RCS pressure below 1750 psia).
- D. MODE 3*, HOT STANDBY (* with RCS pressure below 600 psia).

Question #67

☒ RO

☐ SRO

Tier # 3

Group #

K/A # G2.2.39

Importance Rating: 4.5

K/A Statement: EQUIPMENT CONTROL: Knowledge of less than or equal to 1 hour TS action statements (does not include action statements of 1 hour or less that follow the expiration of a completion time for a TS condition for which an action statement has already been entered)

Proposed Answer: C

Justification: LCO 3.0.3 must be entered because two SITs are INOPERABLE. The SIT Technical Specification does not have an allowance for two SITs being inoperable, and requires 'Immediate' TS 3.0.3 entry. LCO 3.0.3 requires mode reduction until SIT TS is no longer applicable. Since the SIT TS states MODE 3 < 1750 psia is outside of the applicability, then those are the conditions that are required to be met.

Plausibility:

- A. Wrong, the plant need only be moved until LCO 3.5.1 for SITs is no longer applicable, which is HOT STANDBY with RCS pressure below 1750 psia. Plausible: LCO 3.0.3 has provisions to drive all the way to COLD SHUTDOWN, when applicable.
- B. Wrong, the plant need only be moved until LCO 3.5.1 for SITs is no longer applicable, which is HOT STANDBY with RCS pressure below 1750 psia. Plausible: Student may recall that 3.5.1 is applicable in MODE 3 and not recall the part about only being applicable above 1750 psia.
- C. Correct.
- D. Wrong, RCS pressure is only required to be reduced below 1750 psia. Plausible: Student may recall that administratively, the SITs are required to be OPERABLE to less than 600 psia.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications

Provided reference(s): None

Learning Objective: As given in the Technical Specifications, state the actions required or permitted (282392): A) When a condition exists which is less conservative than a Limiting Condition for Operation and the corresponding Action

Source: Bank Vision 413742

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Changed initial condition from "100% power" to MODE 1 to be consistent with other Technical Specification questions. Distractors rewritten to enhance plausibility and make all four within RO level knowledge. DF 5/20. Changed stem by copying and pasting portion of 3.0.3. This aligns the question more with the K/A. Not a 3.0.3 question but a SIT one. DF 5/28.

Clarified in stem to select the highest MODE indefinitely permissible, so that the remaining distractors would be unambiguously wrong. SM 6/21/24.

Question #68

The plant is at 100% power.

Which one of the following radiation monitors (RMs) would be the first to alarm during a 75 gpd Steam Generator tube leak?

- A. RM-4262, Steam Generator Blowdown RM.
- B. RM-4299A, Main Steam Line RM.
- C. RM-5099, Steam Jet Air Ejector RM.
- D. RM-8132A/B, Unit 2 Wide Range Stack RM.

Question #68

☒ RO

☐ SRO

Tier # 3

Group #

K/A # G2.3.12

Importance Rating: 3.2

K/A Statement: RADIATION CONTROL: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters

Proposed Answer: C

Justification: The Steam Jet Air Ejector Rad Monitor (RM-5099) is the first radiation monitor (from the ones given) to indicate the presence of a SGTL due to the very high velocities within the Main Steam System. A tube leak would be indicated within minutes of occurring.

Plausibility:

- A. Wrong, The Blowdown RM response to a SGTL is delayed due to the volume of and the slow velocities within the SG B/D piping. Typically, the Blowdown RM will indicate increased rads around 20-30 minutes after the onset of a SGTL.
- B. The Main Steam RMs would not detect a SGTL due the relatively low activity of the RCS being below the detectable range of the rad monitors.
- C. Correct.
- D. The Unit 2 Stack Rad Monitor is downstream of the SJAЕ RM so would indicate the presence of a leak after the SJAЕ RM.

Technical Reference(s):

- Steam Generator Tube Leak Lesson text, A69-01-C
- Vision Q #414750

Provided reference(s): none

Learning Objective: 283305 – Describe those parameters which will most likely be the first to indicate the presence of a steam generator tube leak.

Source: Vision Q #414750

Cognitive Level: Memory

10CFR55.41(b)(12): Radiological safety principles and procedures

Comments: Replaced question to better align with the K/A. SRM 7/26.

Question #69

A fire in Appendix R Fire Area R-1 is in progress.

- The Control Room is being evacuated due to the fire.
- The crew is performing AOP 2579A, Fire Procedure for Hot Standby Appendix R Fire Area R-1.

In accordance with AOP 2579A, what action is required to be completed within the 30 minutes of evacuating the Control Room?

- A. Establish power to a vital 4160 Volt bus.
- B. Establish Feedwater Flow to a steam generator.
- C. Cross-tie Instrument Air with Unit 3 Service Air.
- D. Restore a battery charger to Facility 2.

Question #69

☒ RO

☐ SRO

Tier # 3

Group #

K/A # G2.4.25

Importance Rating: 3.3

K/A Statement: EMERGENCY PROCEDURES/PLAN: Knowledge of fire protection procedures

Proposed Answer: B

Justification: The caution prior to step 1 of AOP 2579A states, "Failure to initiate Auxiliary Feedwater flow to any SG within 30 minutes of a loss of normal feedwater may result in that SG boiling dry."

Plausibility:

- A. Wrong, power must be restored within 4 hours of the reactor shutdown. However, lack of power should not delay feeding the S/Gs because the Turbine Driven AFW pump is assumed available. Plausible; May recall that power is required to use the electric AFW pumps and that EOPs expect two pumps to be used to feed the SGs.
- B. Correct.
- C. Wrong, instrument Air restoration is not a TCOA of AOP 2579. Plausible; May recall that IA restoration is performed in AOP 2579A but confuse it with the TCOAs that are required. It is also an action that would be performed as soon as possible.
- D. Wrong, a Battery Charger is required to be aligned to Facility 2 prior to depletion of the 'B' battery. This is assumed to take longer than 30 minutes. Plausible; Student may recall that the station batteries have very limited capacity and would be drained faster in an accident situation, therefore, believe that this is a requirement that must be done expeditiously.

Technical Reference(s):

- AOP 2579A, Fire Procedure for Hot Standby Appendix R Fire Area R-1

Provided reference(s): None

Learning Objective: Knowing the fire location, determine the Appendix "R" AOP what should be used to achieve Hot Standby and Cold Shutdown conditions when given the Fire Area/Procedure Cross Reference in AOP 2559, "Fire."

Source: Bank Vision 452091, NRC 2011 Q74

Cognitive Level: Memory

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Question #70

The plant has been operating at 100% power for one (1) month.

- The crew reduces power to 50% using AOP 2575, Rapid Downpower.
- The power reduction takes 30 minutes using CEAs only.
- Only blended makeup is available. (Assume no boration or dilution).

Over the next 24 hours how are CEAs manipulated in order to maintain reactor power at 50%?

For the first 5 to 6 hours, CEAs will be 1..

AND

For the next 18 to 19 hours, CEAs will be 2..

- A. 1. withdrawn
2. withdrawn
- B. 1. withdrawn
2. inserted
- C. 1. inserted
2. inserted
- D. 1. inserted
2. withdrawn

Question #70

☒ RO

☐ SRO

Tier # 4

Group #

K/A #192006 K1.14

Importance Rating: 3.3

K/A Statement: FISSION PRODUCT POISONS: Explain the methods and reasons for the reactor operator to compensate for the time-dependent behavior of xenon-135 concentration in the reactor

Proposed Answer: B

Justification: During a step downpower, Xenon initially increases due to a reduction in the Xenon removal term due to neutron absorption, requiring positive reactivity addition (CEA withdrawal) to maintain power. After 5-6 hours, the Xenon production term due to Iodine decay is reduced, causing a reduction in Xenon requiring negativity addition (CEA insertion) to maintain power stable.

Plausibility:

- A. These actions would be required during an up-power maneuver. The student may not remember the Xenon profile for a downpower when power is going to be stabilized.
- B. Correct.
- C. These are the normal actions for a rapid downpower. The student may not remember the Xenon burn out term during a step downpower.
- D. These would be the correct actions for a step up-power. The student may get the Xenon profile for an up-power confused with a downpower.

Technical Reference(s):

- GFES Lesson 192006, Fission Product Poisons
- NRC GFES question P2061

Provided reference(s): None

Learning Objective: GFES Fission Product Poisons (192006) ELO 1.5 – Explain how Xenon-135 reacts during the following nuclear reactor operations: initial reactor startup (xenon free), reactor shutdown, decrease in reactor power, increase in reactor power, and reactor startup with xenon present in the core.

Source: NRC GFES question P2061

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects

Comments: per 2/22 phone call: Added AOP 2575 to the stem; reworded stem to show only CEAs used during the downpower (no boration or dilution). DF 05MAR

Question #71

If a CEA drops at Millstone Unit 2, the crew will need to reduce turbine load more at 1. due to 2. .

- A.
 - 1. beginning of life
 - 2. the higher RCS boron concentration causing a smaller MTC
- B.
 - 1. end of life
 - 2. a higher CEA worth from diminished boron competition
- C.
 - 1. beginning of life
 - 2. the resonance capture of neutrons from burnable poisons
- D.
 - 1. end of life
 - 2. the increased amount of plutonium in the core

Question #71

☒ RO

☐ SRO

Tier # 4

Group #

K/A # 192007 K1.04

Importance Rating: 3.4

K/A Statement: FUEL DEPLETION AND BURNABLE POISONS: Describe how and why boron concentration changes over core life

Proposed Answer: A

Justification: The higher boron concentration at BOL will result in a very low negative value for MTC. This would translate to requiring a greater RCS temperature drop to add the necessary amount of positive reactivity to compensate for the negative reactivity added by the dropped CEA. Therefore, a larger secondary power reduction is required to stabilize RCS temperature at the new lower reactor power level and restore RCS temperature to program.

Plausibility:

- A. Correct.
- B. Wrong, the effect from the higher negative value of MTC at EOL easily overrides the higher CEA worth at EOL. Plausible; The student may focus on the fact that CEAs are worth more at EOL.
- C. Wrong, although burnable poisons are only present at BOL, they are designed to capture thermal neutrons in competition with the fresh uranium fuel and not necessarily in the resonance region. Therefore, their neutron absorption effect does not change appreciably with temperature. Plausible: Student may recognize that the burnable poisons due add to neutron capture at BOL and assume their presence has a large enough impact on FTC to require a greater load reduction.
- D. Wrong, although the power coefficient increasing over core life is the key, the PC increase is mainly due to the increase in the MTC value, which far exceeds the increase in FTC. Plausible; The student may focus on the fact that higher amounts of plutonium building in over core life do increase the value of FTC (power coefficient) and equate this to the need for a larger drop in RCS temperature to compensate.

Technical Reference(s):

- General Fundamentals Handbook

Provided reference(s): None

Learning Objective: Predict how operator action or inaction affects plant and system conditions concerning a CEA malfunction (283049)

Source: Bank Vision 452007

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Comments: Changes made from proposed version include formatting into 2x2 template and shortening the correct answer so it doesn't stand out from the distractors.

Question #72

The reactor tripped from 100% power 72 hours ago.

- A reactor startup using CEAs is in progress.

Describe the response of nuclear instrumentation after CEA motion is stopped.

As the core approaches criticality (not yet critical), counts will 1. and Startup Rate (SUR) will 2..

- A. 1. continue to increase
2. stabilize at a positive value
- B. 1. increase, then stabilize
2. decrease to zero
- C. 1. increase, then decrease
2. decrease to zero
- D. 1. increase, then decrease
2. stabilize at a negative value

Question #72

☒ RO

☐ SRO

Tier # 4

Group #

K/A # 192008 K1.03 Importance Rating: 4.0

K/A Statement: REACTOR OPERATIONAL PHYSICS: (STARTUP AND APPROACH TO CRITICALITY) Describe count rate and instrument response that should be observed for rod withdrawal during the approach to criticality

Proposed Answer: B

Justification: As the reactor approaches criticality, after CEA motions stops, reactor power will continue to increase at a decreasing rate asymptotically approaching a new power level as SUR will decrease to zero.

Plausibility:

- A. This describes a critical (supercritical) reactor. An excess of positive reactivity will cause power to continue to increase until the POAH combined with a positive SUR.
- B. Correct.
- C. This describes a subcritical reactor with the effects of xenon building in (reactor with power history).
- D. This describes a subcritical reactor. An excess of negative reactivity will cause power to continue to lower combined with a negative SUR.

Technical Reference(s):

- N02-00-C, OP 2202, Reactor Startup, Lesson Plan, pg. 12
- GFES Lesson 192008, Reactor Operational Physics
- Vision question 455244

Provided reference(s): None

Learning Objective: GFES Reactor Operational Physics (192008) ELO 1.2 – Describe the nuclear instrumentation response during a reactor startup to criticality.

Source: Vision question 455244

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects

Comments:

Question #73

Given the following initial conditions at Millstone Unit 2:

- RCS Leak is in progress.
- RCS Temperature is 526 °F and steady.
- RCS Pressure is 2000 psia and lowering.

Assuming that temperature remains constant, at what pressure will RCS subcooling equal 0 °F?

- A. 856 psia
- B. 871 psia
- C. 931 psia
- D. 946 psia

Question #73

☒ RO

☐ SRO

Tier # 4

Group #

K/A # 193003 K1.24 Importance Rating: 3.1

K/A Statement: STEAM: Explain the usefulness of steam tables to the control room operator

Proposed Answer: A

Justification: Saturation pressure for 526 °F is 856 psia.

Plausibility:

- A. Correct.
- B. Wrong, Saturation pressure for 526 °F is 856 psia. Plausible; may get this answer if 15 psia is added to the pressure for unit conversion.
- C. Wrong, Saturation pressure for 526 °F is 856 psia. Plausible; Table may be misread, and the pressure directly below the correct answer may be used erroneously.
- D. Wrong, Saturation pressure for 526 °F is 856 psia. Plausible; May get this answer if 15 psia is added to the pressure for unit conversion and table is misread, and the pressure directly below the correct answer may be used.

Technical Reference(s):

- Mollier Diagram and Steam Tables

Provided reference(s): Mollier Diagram and Steam Tables

Learning Objective:

Source: Modified Bank

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(14): Principles of heat transfer thermodynamics and fluid mechanics.

Comments: Modified the stem, distractors and justification for a temperature of 526°F instead of 536°F. Changed source to “Modified Bank.” SM 6/21/24.

**2023 NRC Exam
RO Written Exam Worksheet**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	193003 K1.24	
	Importance Rating	3.1	

K/A Statement

(193003K1.24) **STEAM:** Explain the usefulness of steam tables to the control room operator

Proposed Question: RO 73 Rev: 0

Given:

- RCS leak in progress.
- RCS Temperature is 536°F and steady
- RCS pressure is 2000 psia and lowering

As RCS pressure lowers, when will subcooling equal 0°F? (Assume RCS Temperature is remains constant)

- A. 930 psia
- B. 945 psia
- C. 962 psia
- D. 977 psia

Question #74

The plant has shut down due to a 20 gpm RCS leak through an ideal hole.

- CTMT pressure = 5 psig.
- RCS Pressure = 2250 psia.

Given that flow is proportional to the square root of D/P, what would the leak rate be once the RCS is depressurized to 1225 psia with NO change in Containment pressure?

- A. 20 – 21 gpm
- B. 14 – 15 gpm
- C. 10 – 11 gpm
- D. 5 – 6 gpm

Question #74

☒ RO

☐ SRO

Tier # 4

Group #

K/A # 193004 K1.15 Importance Rating: 2.8

K/A Statement: THERMODYNAMIC PROCESS: (THROTTLING AND THE THROTTLING PROCESS) Determine the exit conditions for a throttling process based on the use of steam and/or water

Proposed Answer: B

Justification: The leak rate is a function of the square root of the pressure differential between the RCS and CTMT. Since the pressure differential lowers, the leak rate will also lower. $V_f = V_i \times (\sqrt{\Delta P_f} \div \sqrt{\Delta P_i})$

Plausibility:

- A. The student might think leak rate won't change at all.
- B. Correct.
- C. The student may forget (or miss the cue in the question stem) the change in leak rate is proportional to the square root of the pressure function. $V_f = V_i \times (\Delta P_f \div \Delta P_i)$
- D. The student may think the leak rate changes as a square (or, similarly, read the relationship backwards in the question stem). $V_f = \sqrt{V_i} \times (\Delta P_f \div \Delta P_i)$

Technical Reference(s):

- GFES Lesson 193006, Fluid Statics and Dynamics
- Vision question 513921

Provided reference(s): Steam Tables w/ Mollier Diagram

Learning Objective: GFES Fluid Statics and Dynamics (193006) TLO 1 – Explain the fundamental properties of fluids and closed fluid systems.

Source: Modified Question 513921

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(14): Principles of heat transfer thermodynamics and fluid mechanics

Comments: Supplied students with the relationship between flowrate and D/P. Changed values in distractor D to align with the intended misconception. Removed reference to prior NRC question. SRM 7/26.

A 75 gpm leak to atmosphere has developed from a cooling water system that is operating at 80 psig. Which one of the following will be the approximate leak rate when system pressure has decreased to 40 psig?

- A. 38 gpm
- B. 44 gpm
- C. 53 gpm
- D. 59 gpm

Answer: C

Question 1 Info	
Question Type:	Multiple Choice
Status:	
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	0.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	513921
Version ID:	666892
User-Defined ID:	P1382
Cross Reference Number:	FEBRUARY 2017 GFE
Topic:	A 75 gpm leak to atmosphere has developed from a cooling water system that is operating at 80 psig.
Num Field 1:	2.9
Num Field 2:	3.0
Text Field:	193006 K1.05
Comments:	

Question 1 History	
Exams Administered:	0
Student Encounters:	0
Answered Right:	0
Answered Wrong:	0
Partially Correct:	0
Answer Invalid:	0
Unanswered:	0
Ignore Response:	0
Avg Points Awarded:	0.00
... As % of Point Value:	0
Standard Deviation:	0.00

Associated objective(s):

Fluid Statics and Dynamics

Question #75

Unit 2 is operating at 100% power.

- An 'A' quadrant CEA drops to the bottom of the core.
- Reactor power stabilizes at 98% (assuming no actions taken).

The dropped CEA causes the value of AZIMUTHAL POWER TILT (T_Q) in the core to 1. .

AND

Per AOP 2556, CEA Malfunctions, a power reduction must be completed within a MAXIMUM of 2. to ensure fuel integrity is maintained.

- A.
 - 1. lower
 - 2. one (1) hour
- B.
 - 1. lower
 - 2. two (2) hours
- C.
 - 1. rise
 - 2. one (1) hour
- D.
 - 1. rise
 - 2. two (2) hours

Question #75

☒ RO

☐ SRO

Tier # 4

Group #

K/A # 193009 K1.09 Importance Rating: 3.2

K/A Statement: CORE THERMAL LIMITS: Describe the effects of quadrant power tilt (symmetric offset), including long-range effects

Proposed Answer: C

Justification: TS → AZIMUTHAL POWER TILT shall be the difference between the maximum power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core TILT is used to monitor radial neutron flux distribution in the core. Flux will be significantly lower in the core quadrant with the dropped rod. Overall, reactor power will stabilize at a slightly lower value, but the flux will rise in the three other core quadrants. This will result in a higher QPTR. Since the CEA dropped fully into the core, both upper and lower half detectors will see the change in flux so both upper and lower core values for QPTR will rise. AOP 2556, CEA Malfunctions, appendix D, step D.6.b.1 directs a rapid downpower to less than 70% within one hour, using only boration.

Plausibility:

- A. WRONG; TILT will rise. Plausible; Candidate incorrectly reverses the TILT ratio.
- B. WRONG; TILT will rise and power reduction is required within a maximum of 60 minutes. Plausible; Candidate incorrectly reverses the TILT ratio, and remembers the time of 2 hours required in AOP 2556 for the maximum allowable time for recovering the dropped CEA.
- C. Correct.
- D. WRONG; Power reduction must occur within a maximum of 60 minutes. Plausible; remembers the time of 2 hours required in AOP 2556 appendix D step 11 for the maximum allowable time for recovering the dropped CEA.

Technical Reference(s):

- MP-PROC-OPS-AOP 2556 REV. 022, CEA Malfunctions
- General Fundamentals Handbook QPTR (TILT)

Provided reference(s): None

Learning Objective: GFES Control Rods ELO 2.4 – Describe Quadrant Power Tilt Ratio (QPTR) and the effects of operating with a high ratio.

Source: Modified, Palo Verde 2023 NRC exam Q75

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Thermodynamics: Describe the effects of quadrant power tilt (symmetric offset), including long-range effects	Tier	4		
	Group			
	K/A	193009 K1.09		
	IR	3.2		

Question 75

Given the following conditions:

- Unit 1 is operating at 100% power, all CEAs fully withdrawn

Subsequently:

- An 'A' quadrant 4-finger CEA dropped to the bottom of the core
- Prior to any operator action being taken, Reactor power stabilized at 98%

(1) The dropped CEA will cause the value of azimuthal tilt in the core to...

(2) Per 40AO-9ZZ11, CEA Malfunctions, in order to ensure the integrity of the fuel is protected, the crew must commence a power reduction within a MAXIMUM of...

- A. (1) rise
(2) 10 minutes
- B. (1) rise
(2) 30 minutes
- C. (1) lower
(2) 10 minutes
- D. (1) lower
(2) 30 minutes

Question #76

The plant is in MODE 5 with a cooldown in progress.

- SDC and 'A' & 'B' RCPs are concurrently operating.
- Containment Purge is aligned to Main Exhaust.
- The 'A' RCP vapor seal begins leaking.

What actions are directed FIRST to minimize the RADIATION RELEASE to the public?

- A. Align Containment Purge to EBFS in accordance with OP 2314B, "Containment and Enclosure Building Purge."
- B. Perform Conditional Actions of OP 2207, "Plant Cooldown," to secure 'A' and 'B' RCPs.
- C. Implement the Closure Plan IAW OP 2264, "Conduct of Outages" and close Containment Purge valves.
- D. Enter AOP 2515, "Loss of Containment Integrity" and restore CONTAINMENT INTEGRITY.

Question #76

☐ RO

☒ SRO

Tier # 1

Group # 1

K/A # APE 15 G2.3.11

Importance Rating: 4.3

K/A Statement: Reactor Coolant Pump Malfunctions RADIATION CONTROL: Ability to control radiation releases

Proposed Answer: C

Justification: When Containment Purge is in service, a Closure Plan would be in effect for the Containment Purge valves, which are Containment penetrations. AOP 2568A, "RCS Leak, MODE 4, 5, 6 and Defueled," has guidance to direct setting Containment Closure.

Plausibility:

- A. Purge valves would need to be closed IAW the closure plan. Crew would then follow up by referring to OP 2314B to complete securing of Containment Purge. Only after conditions are stable and the leak is mitigated would the crew determine if Containment Purge to EBFS is desired. Plausible: Containment Purge to EBFS would send the purge air to the EBFS charcoal filter, thus minimizing release. Containment Purge to EBFS may be desired after the RCS leak is mitigated, to facilitate cleaning the Containment atmosphere to implement repairs.
- B. Securing 'A' and 'B' RCPs will not stop the leak nor will it keep the activity already released to the Containment atmosphere from the leak from going to Main Exhaust via purge. Plausible: a known leaking RCP would certainly be secured, and OP 2207 would require the remaining RCP to be secured (concurrent operations is not allowed on 1 RCP).
- C. Correct.
- D. CONTAINMENT INTEGRITY is required in MODES 1 through 4, and AOP 2515 is only applicable in MODES 1 through 4 and MODE 6. Plausible: AOP 2515 contains guidance to establish Containment Closure, but it is only applicable in MODE 6.

Technical Reference(s):

- OP 2264, "Conduct of Outages"
- OU-AA-200, "Shutdown Risk Management"
- AOP 2568A, "RCS Leak, MODE 4, 5, 6 and Defueled"

Provided reference(s): None

Learning Objective: Discuss the requirements of a Containment Closure Plan (282277)

Source: New

Cognitive Level: Memory

10CFR55.43(b)(4): Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments: Specified cognitive level of Memory. Clarified stem to say actions directed FIRST, to remove ambiguity and make 'D' unambiguously incorrect. SRM 7/26.

Question #77

The plant is operating at 100% power.

- PZR PRESSURE SELECTED CHANNEL DEVIATION HI/LO (C-02/3, D-37), alarms.
- Pressurizer pressure = 2225 psia lowering.
- Pressure controllers (PIC-100X&Y) are in AUTO with setpoints = 2250 psia.
- Both Pressure controllers (PIC-100X&Y) are operating properly.
- Facility 1 Proportional Heater Amps = 0 amps.
- Facility 2 Proportional Heater Amps = 170 amps.

1. Describe the actions directed to the Reactor Operator.

AND

2. Describe the administrative impact of the situation.
 - A.
 1. Energize Pressurizer Backup Heaters as necessary to maintain pressure at 2250 psia.
 2. Pressurizer pressure is maintained to ensure the proper operation of the code safety valves and the PORVs.
 - B.
 1. Raise the output of the in-service pressure controller to maximize Proportional Heater output and maintain pressure at 2250 psia.
 2. Pressurizer pressure is maintained above 2225 psia to ensure accident and transient analysis specified limits remain valid.
 - C.
 1. Energize Pressurizer Backup Heaters as necessary to maintain pressure at 2250 psia.
 2. Total Pressurizer heater output is below the T.S. limit needed to ensure Pressurizer pressure required for establishing and maintaining natural circulation.
 - D.
 1. Place the in-service pressure controller in manual and energize all available Pressurizer Backup Heaters to force Pressurizer Sprays.
 2. The loss of automatic Pressurizer pressure control would result in exceeding an LSSS setpoint if power were to drop 10% instantaneously or ramp $\geq 5\%$ per minute.

Question #77

☐ RO

☒ SRO

Tier # 1

Group # 1

K/A # 027AA2.11

Importance Rating: 3.8

K/A Statement: Ability to determine and/or interpret the following as they apply to

PRESSURIZER PRESSURE Control System Malfunction: RCS pressure

Proposed Answer: C

Justification: Facility 1 proportional heaters are lost, which equates to half of the required TS heaters. Basis: The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish and maintain natural circulation. The bases can no longer be met. This question is SRO only as it requires assessing plant conditions and deciding based on knowledge of the Technical Specification bases.

Plausibility:

- A. WRONG; The proper operation of the code safety valves, and the power operated relief valves are ensured by PZR level <70%. Plausible; 2225 psia is the limit for the DNB TS and taking action to maintain pressure would increase the available margin to DNB, which does sound like an appropriate action but is not correct.
- B. WRONG; FAC 1 Proportional heaters are failed and FAC 2 Proportional heaters are already putting out max output. Raising the output will not give desired result. Plausible; May not realize that the failed proportional heaters are already receiving a demand for maximum output, and they are at zero amps because they have failed.
- C. Correct.
- D. WRONG; Auto pressurizer pressure control has not been lost. The pressure controller should not be taken to manual. Spray will be able to control pressure automatically with backup heaters energized. Plausible; May think that manual pressure control is required to maintain margin to LSSS setpoint.

Technical Reference(s):

- ARP 2590B-212, D-37 PZR PRESS SELECTED CHANNEL DEVIATION HI/LO.
- MPS-2 Technical Specifications, TS 3.4.4, Reactor Coolant System- Pressurizer.
- TS 3.4.4, Reactor Coolant System- Pressurizer BASIS

Provided reference(s): None

Learning Objective: 281903 – State the following requirements of Pressurizer Technical Specification 3.4.4: a. Maximum allowed pressurizer level. B. Pressurizer heater requirements. C. Applicability. D. BASES for each.

Source: Bank #451943

History: 2014 NRC exam Q#89

Cognitive Level: Comprehension/Analysis

10CFR55.43(b)(3) Facility operating limitations in the technical specifications and their bases.

Comments: Changed pressure controller bullet in stem to say the controllers are operating properly and remove reference to controller output, to eliminate possible double jeopardy with Question 35. SRM 7/26.

Question #78

The unit experienced a Steam Generator Tube Rupture.

- The affected S/G is isolated.
- #2 S/G pressure = 825 psia and lowering.
- Pressurizer Pressure = 850 psia and lowering.
- The crew is conducting a natural circulation cooldown.
- RCP restart conditions are met.

WHAT action does EOP 2534, Steam Generator Tube Rupture, direct first?

AND

WHY is this performed first?

- A.
 - 1. Lower RCS pressure below the affected S/G pressure.
 - 2. To more easily depressurize and cooldown the S/G.
- B.
 - 1. Start a RCP in the unaffected loop.
 - 2. Minimize reactivity effect of S/G water flowing into the RCS.
- C.
 - 1. Lower RCS pressure below the affected S/G pressure.
 - 2. To minimize RCS inventory loss.
- D.
 - 1. Start a RCP in the unaffected loop.
 - 2. Forced circulation allows for better control of the Cooldown.

Question #78

☐ RO

☒ SRO

Tier # 1

Group # 1

K/A # 038EA2.15

Importance Rating: 3.9

K/A Statement: Ability to determine and/or interpret the following as they apply to STEAM
GENERATOR Tube Rupture: RCS pressure

Proposed Answer: B

Justification: Affected Steam Generator level will eventually reach 90%. Backflowing is the path to success to lower this level. However, in natural circulation, backflow is not desired due to adding a source of RCS dilution without adequate mixing. OP 2260 requires at least 1 RCP running to provide the required RCS mixing so that the crew can then proceed to lower RCS pressure to backflow from the affected S/G. EOP 2534 step for RCP restart starts one in each loop, unaffected loop first.

Plausibility:

- A. Wrong, lowering affected Steam Generator level through backflow in natural circulation adds dilution water without adequate mixing. Plausible: it is desired to keep the affected steam generator below 90% and maintaining NR level in the indicating range allows monitoring leak rate.
- B. Correct.
- C. Wrong, lowering affected Steam Generator level through backflow in natural circulation adds dilution water without adequate mixing. Plausible: inventory is being lost via the tube rupture.
- D. Wrong, the EOP set is written to allow cooldown in natural circulation all the way to Shutdown Cooling entry conditions. The more pressing concern is providing RCS mixing to allow backflow from the affected Steam Generator without creating a slug of diluted water in the affected loop. Plausible: cooldown control is more responsive in forced circulation.

Technical Reference(s):

- EOP 2534, "Steam Generator Tube Rupture"
- EOP 2541, Appendix 2, "Figures"
- OP 2260, "EOP User's Guide"

Provided reference(s): None

Learning Objective: 306287 - Outline and explain the bases for RCS depressurization during a SGTR

Source: New

Cognitive Level: Memory

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Simplified stem of question. Primary and secondary pressure bullets were retained to maintain alignment with the K/A.

Question #79

Unit 2 is operating at 100% power when the following occur:

- A seismic event has just occurred.
- An ESDE has occurred in Containment on the #2 Steam Generator.
- Radiation levels associated with potential core damage have been reported.
- The TSC and EOF have not yet been activated.

What DECISION can be delegated by the Control Room Director of Station Emergency Operations (CR-DSEO) according to MP-26-EPI-FAP01, Control Room Emergency Operations?

- A. Declaration of a Site Area Emergency.
- B. Evacuation of non-essential personnel from site.
- C. Authorization of emergency exposure limits for field activities.
- D. Invocation of 10CFR50.54(x) departure from a license condition.

Question #79

☐ RO

☒ SRO

Tier # 1

Group # 1

K/A # APE 40 G2.4.29

Importance Rating: 4.4

K/A Statement: Steam Line Rupture – Excessive Heat Transfer EMERGENCY

PROCEDURES/PLAN: Knowledge of the emergency plan implementing procedures

Proposed Answer: D

Justification: 10CFR50.54(x) specifically states, that at a minimum, a licensed senior operator must approve the action that deviates from the license condition. The CR-DSEO is a role that would typically be filled by the affected unit SM until the TSC/EOF is fully activated and ready for turnover of the DSEO role. While the CR-DSEO position could be held by someone with a senior license, the CR-DSEO does NOT explicitly need to be the person that approves a 10CFR50.54(x) deviation from license conditions. Distractors A, B, and C are all listed in MP-26-EPI-FAP01 as non-delegable actions that the CR-DSEO is responsible.

Plausibility:

- A. WRONG; The CR-DSEO cannot delegate event classification. Plausible; Candidate may not be familiar with the list of non-delegable actions of the CR-DSEO and may think any license holder can make an event declaration.
- B. WRONG; The CR-DSEO cannot delegate approval for site evacuation. Plausible; Candidate may not be familiar with the list of non-delegable actions of the CR-DSEO and may think the Station Duty Officer can evacuate nonessential personnel from site.
- C. WRONG; The CR-DSEO cannot delegate approval for authorization of emergency exposure limits. Plausible; Candidate may not be familiar with the list of non-delegable actions of the CR-DSEO and may think the Radiation Protection Manager must approve emergency exposure limits for field activities.
- D. CORRECT.

Technical Reference(s):

- MP-26-EPI-FAP01 REV. 008, Control Room Emergency Operations.

Provided reference(s): None

Learning Objective:

Source: Modified Calvert Cliffs NRC exam 2022

Cognitive Level: Memory

10CFR55.43(b)(5): Assessment of plant conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Added initial condition and event which included an ESDE to the stem, to better align question with the K/A. SM 6/21/24. changed stem to put question first and procedure at the end. DF 7/11

Exam Material

2022 ILT NRC SRO EXAM

Test Key

Question 100

ID: 2229781

Points: 1.00

****SRO ONLY****

- A General Emergency has been declared
- The Shift Manager has assumed responsibility as the Emergency Director
- The Emergency Offsite Facility and Technical Support Center are NOT yet activated

Which of the following is a non-delegable action of the Shift Emergency Director?

- A. Notify Offsite Agencies
- B. Authorize issuance of thyroid blocking agents
- C. Activate the Emergency Response Organization using Everbridge
- D. Initiate an Emergency Plant Announcement for the notification of the event

Answer

B

Answer Explanation

A. Incorrect. Plausible if operator misinterprets that since the SM is responsible to notify offsite agencies that he personally must do it.

B. Correct. Per EP-AA-112, issuance of KI cannot be delegated.

C. Incorrect. Plausible if operator misinterprets that since the SM is responsible to recall the ERO that he personally must do it.

D. Incorrect. Plausible since this is an action in the Shift Emergency Director Checklist that has a short time limit.

Question #80

The plant experienced a trip from 100% power.

- Facility 1 buses failed to fast transfer to the RSST.
- Annunciator 125 VDC BUS 201D UNDERVOLTAGE (C06/07, D-44) alarms.
- Bus 201D voltage = 128 VDC and lowering.

Bus 201D is currently 1.

AND

The basis for the determination is 2.

- A.
 - 1. OPERABLE
 - 2. Equipment which requires 201D is in its accident condition.
- B.
 - 1. OPERABLE
 - 2. 201D charger is not required for design function of 201D.
- C.
 - 1. NOT OPERABLE
 - 2. Bus voltage is insufficient for OPERABILITY.
- D.
 - 1. NOT OPERABLE
 - 2. An OPERABLE battery bus requires at least one side of a charger.

Question #80

☐ RO

☒ SRO

Tier # 1

Group # 1

K/A # 058AA2.02

Importance Rating: 3.8

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF DC Power: 125VDC bus voltage

Proposed Answer: B

Justification: Technical Specification basis for OPERABILITY of bus 201D requires the bus and its respective battery. The battery is considered inoperable below 125 VDC. The Turbine Battery Charger is not required because the Technical Specification design function (feedwater isolation) occurs within seconds of the onset of the event.

Plausibility:

- A. Wrong, only the FRV (not the bypass) goes closed on a turbine trip, and nothing is preventing it from being re-opened. Plausible because the FRV does close on a turbine trip.
- B. Correct.
- C. Wrong, 128 VDC is more than 125 VDC, the minimum voltage required for the bus to be considered OPERABLE. Plausible because the undervoltage annunciator is locked in.
- D. Wrong, because the Turbine Battery Charger is not required to consider bus 201D OPERABLE. Plausible because busses 201A and 201B require at least one side of their respective charger to be considered OPERABLE.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications

Provided reference(s): None

Learning Objective: Given a set of plant conditions, determine Technical Specification applicability during a Loss of Vital 125 VDC Bus 201D (282892).

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Adjusted format for consistency. Updated justification to reflect 128 VDC instead of 126 VDC. SRM 7/26.

Question #81

The plant is operating at 90% power.

- Annunciator FEEDWATER REGULATING VALVE 1 LOCKED (C-05, B-7), is alarming.
- The 'A' SG water level = 80% and stable.

- Instrument Air pressure = 82 psig and stable.
- 2-SA-10.1, SA/IA X-TIE, is closed.
- 2-SA-11.1, SA HDR ISOL, is closed.
- AOP-2563, Loss of Instrument Air, is in progress.

Per LCO 3.7.1.6 Basis, the Feedwater Regulating Valve is required to be isolated 1..

AND

Given the conditions above, AOP 2563, Loss of Instrument Air, directs opening 2.,

- A.
 - 1. immediately, to prevent moisture carryover from the S/G to the main turbine
 - 2. SA-11.1, SA HDR ISOL
- B.
 - 1. within 72 hours, to prevent feed line flashing during a steam line break
 - 2. SA-10.1, SA/IA X-TIE
- C.
 - 1. within 72 hours, to prevent feed line flashing during a steam line break
 - 2. SA-11.1, SA HDR ISOL
- D.
 - 1. immediately, to prevent moisture carryover from the S/G to the main turbine
 - 2. SA-10.1, SA/IA X-TIE

Question #81

☐ RO

☒ SRO

Tier # 1

Group # 1

K/A # APE 65 G2.4.12

Importance Rating: 4.3

K/A Statement: Loss of Instrument Air EMERGENCY PROCEDURES/PLAN: Knowledge of operating crew responsibilities during emergency and abnormal operations.

Proposed Answer: B

Justification: As Instrument Air header pressure falls below 85 psig, SA-10.1 automatically opens and SA-11.1 automatically closes. The given air pressure is 82 psig, and SA-10.1 is still closed, so it must be opened manually. TS 3.7.1.6 basis explains that an inoperable Main Feedwater Regulating Valve must be made operable or isolated within 72 hours to have protection against a steam line break which could result in flashing of the feed line volume. Closing FW-42A will isolate the inoperable Main Feedwater Regulating Valve as required in TS 3.7.1.6 action a.2.

Plausibility:

- A. WRONG; SA-11.1 closes automatically when IA pressure falls below 85 psig, as given in the stem. FW-42A is not required to be closed immediately since S/G water level is < 85% per AOP 2563. Plausible; Candidate may confuse the required position of SA-11.1 with SA-10.1 at the IA pressure given in the stem and may incorrectly recall the S/G water level at which closing FW-42A is required. In addition, the basis for preventing carryover is a plausible concern but is not the TS basis for the operation.
- B. Correct.
- C. WRONG; SA-11.1 closes automatically when IA pressure falls below 85 psig, as given in the stem. Plausible; Candidate may confuse the required position of SA-11.1 with SA-10.1 at the IA pressure given in the stem.
- D. WRONG; FW-42A is not required to be closed immediately since S/G water level is < 85% per AOP 2563. Plausible; candidate may incorrectly recall the S/G water level at which closing FW-42A is required. In addition, the basis for preventing carryover is a plausible concern but is not the TS basis for the operation.

Technical Reference(s):

- MPS-2 Technical Specifications, TS 3.7.1.6 & TS 3.7.1.6 Basis.
- AOP 2563 REV. 011, Loss of Instrument Air.
- ARP 2590D-026, FEEDWATER REGULATING VALVE 1 LOCKED.

Provided reference(s): None

Learning Objective: 283187 – Identify those IA/SA valves which automatically reposition due to lowering air pressure and explain the effects on the Instrument Air system.

Source: Modified MP2 NRC exam 2022 Q#81

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(1): Conditions and limitations in the facility license.

Comments: NRC changed K/A from 2.4.49 to 2.4.12 for better match. DF 8/14

Question #81

The plant is operating at 100% power.

A major air leak occurs.

1. What Instrument Air system automatic actions occur as air pressure lowers?

AND

2. What LCO would be affected by a complete loss of Instrument Air?

- A.
 1. Station Air Supply to Instrument Air (2-SA-10.1) opens.
 2. 3.7.1.7, Atmospheric Dump Valves
- B.
 1. Station Air Supply to Instrument Air (2-SA-10.1) opens.
 2. 3.7.1.6, Main Feedwater Isolation Components
- C.
 1. Instrument Air Supply to Station Air (2-IA-641) isolates.
 2. 3.7.1.7, Atmospheric Dump Valves
- D.
 1. Instrument Air Supply to Station Air (2-IA-641) isolates.
 2. 3.7.1.6, Main Feedwater Isolation Components

Question #82

The core is being on-loaded.

- Containment Purge is aligned per OP 2314B, Section 4.1, Initiating Containment Purge using EBFS and Containment Cleanup Flow Path (MODE 5 or 6).
- RM-8132A/B, Unit 2 Stack Radiation Monitor, alarms.
- A fuel bundle is dropped and gas bubbles are observed shortly afterwards.

1. Where is the Fuel Handling Accident?

AND

2. What system alignments are directed per AOP 2577, Fuel Handling Accident?

- A.
 - 1. Spent Fuel Pool Area.
 - 2. VERIFY EBFS is operating in the AEAS mode
CHECK CRAC operating in Recirculation mode.
- B.
 - 1. Containment.
 - 2. CHECK CRAC operating in Recirculation mode.
VERIFY CTMT Purge is isolated (AC-4, 5, 6, 7 are closed).
- C.
 - 1. Spent Fuel Pool Area.
 - 2. VERIFY EBFS is operating in the AEAS mode.
VERIFY CTMT Purge is isolated (AC-4, 5, 6, 7 are closed).
- D.
 - 1. Containment.
 - 2. VERIFY EBFS is operating in the EBFS mode.
VERIFY CTMT Purge is isolated (AC-4, 5, 6, 7 are closed).

Question #82

☐ RO

☒ SRO

Tier # 1

Group # 2

K/A # 036AA2.01

Importance Rating: 3.7

K/A Statement: Ability to determine and/or interpret the following as they apply to FUEL HANDLING INCIDENTS: Radiation monitoring equipment indications

Proposed Answer: A

Justification: The fuel handling accident occurred in the SFP area due to the Unit 2 stack rad monitor in alarm. With CTMT Purge aligned for Containment Cleanup using EBFS, the exhaust is directed to the site stack. AOP 2519, Fuel Handling Accident directs the operator to verify the EBFS system is aligned to Auxiliary Exhaust and the Control Room Ventilation System (CRAC) is recirculating.

Plausibility:

- A. Correct Answer
- B. The examinee may think the accident occurred in the CTMT so CTMT Purge needs to be isolated and EBFS running in AEAS to address the issue in the Spent Fuel Pool.
- C. The examinee may think the Equipment Hatch is open so isolating CTMT Purge would prevent a release to the environment.
- D. The examinee may think the fuel handling accident occurred in CTMT so EBFS is initiated and CTMT Purge isolated to prevent a release to the environment.

Technical Reference(s):

- AOP 2577, Fuel Handling Accident

Provided reference(s): None

Learning Objective: 283453 – Describe the general sequence of operations for AOP 2577, Fuel Handling Accident

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(7): Fuel handling facilities and procedures.

Comments: per 2/22 phone call changed stem condition from area rad monitors to atmospheric process rad monitor readings. Also, added an initial condition for Containment Purge alignment.

Adjusted format for consistency. SRM 7/26.

Question #83

The plant is operating at 100% power.

- An S/G Tube Leak is diagnosed.
- The crew is working through AOP 2569, Steam Generator Tube Leak.
- A second Charging pump is started
- Charging and Letdown are balanced.
- Pressurizer level = 65% and steady
- Letdown flow rate = 72 gpm

1. What is the primary to secondary LEAKAGE rate?

AND

2. Technical Specifications requires the plant be placed in HOT STANDBY in what amount of time?

- A. 1. 12 gpm.
2. Six (6) hours.
- B. 1. 12 gpm.
2. Four (4) hours.
- C. 1. 8 gpm.
2. Six (6) hours.
- D. 1. 8 gpm.
2. Four (4) hours.

Question #83

☐ RO

☒ SRO

Tier # 1

Group # 2

K/A # 037AA2.04

Importance Rating: 3.6

K/A Statement: Ability to determine and/or interpret the following as they apply to STEAM Generator Tube Leak: Comparison of RCS fluid inputs and outputs to detect leaks

Proposed Answer: A

Justification: Normal Letdown flow with Charging and Letdown balanced would be 84 gpm. Charging is providing 88 gpm into the RCS, RCP Controlled Bleed-Off is removing 4 gpm. With a 12 gpm leak and Charging and Letdown balanced, Letdown flow is then 72 gpm. Technical Specification LCO 3.4.6.2 requires the plant be placed in MODE 3 within 6 hours of not meeting the primary to secondary LEAKAGE specification (75 gpd)

Plausibility:

- A. Correct.
- B. WRONG; 12 gpm is the correct leak rate but 4 hours is not the correct ACTION time. LCO 3.4.6.2 allows 4 hours to reduce LEAKAGE but that ACTION does not apply to primary to secondary LEAKAGE.
- C. WRONG; 8 gpm leak rate is incorrect. The examinee may think normal letdown is 40 gpm, if a second charging pump is started, letdown is doubled to 80 gpm ($80 - 8 = 72$)
- D. WRONG; 8 gpm leak rate is incorrect. The examinee may think normal letdown is 40 gpm, if a second charging pump is started, letdown is doubled to 80 gpm ($80 - 8 = 72$). 4 hours is not the correct ACTION time. LCO 3.4.6.2 allows 4 hours to reduce LEAKAGE but that ACTION does not apply to primary to secondary LEAKAGE.

Technical Reference(s):

- AOP 2569, Steam Generator Tube Leak
- Vision Question # 452134

Provided reference(s): None

Learning Objective: 283307 – Describe the general sequence of operations for AOP 2569, Steam Generator Tube Leak.

Source: Modified Vision Question #452134 2014 NRC exam Q#82

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(1): Conditions and limitations in the facility license.

Comments: Replaced question with Vision 452204 to be better aligned with the K/A. New question requires student to assess the indications to know that a Steam Generator Tube Leak is in progress and small enough not to be a Steam Generator Tube Rupture. SRM 7/26. Rewrote question based NRC input. DF 8/14

A Steam Generator tube leak develops while the plant is at 100% power.

- Letdown flow = 32 gpm and stable.
- Pressurizer level = 64% and stable

- A second charging pump is started.
- Letdown flow is being adjusted

1. Which one of the following conditions would indicate that the Steam Generator tube leak rate has increased?
2. When is the plant required to be in HOT STANDBY per the Technical Specifications?

- A.
 1. After letdown flow is adjusted until pressurizer level stabilizes at 65%. Letdown flow rises to 70 gpm and stabilizes.
 2. Be In HOT STANDBY within 4 hours.
- B.
 1. After letdown flow is adjusted to 84 gpm, pressurizer level then lowers to 63.5% and letdown flow lowers to 76 gpm before both stabilize.
 2. Be In HOT STANDBY within 4 hours.
- C.
 1. After letdown flow is adjusted until pressurizer level stabilizes at 65%. Letdown flow rises to 70 gpm and stabilizes.
 2. Be In HOT STANDBY within 6 hours.
- D.
 1. After letdown flow is adjusted to 84 gpm, pressurizer level then lowers to 63.5% and letdown flow lowers to 76 gpm before both stabilize.
 2. Be In HOT STANDBY within 6 hours.

Answer: C

Answer Explanation

Question #84

A Loss of All Feedwater has occurred.

- A Feedwater Regulating Valve (FRV) Bypass Valve will *not* open.
- The US directs throttling open the respective FRV.

According with OP 2260, Unit 2 EOP User's Guide, the direction to open the FRV is considered a procedure 1. .

AND

Concurrence of the directed action by the OMOC 2. required in accordance with OP-AP-104, Emergency and Abnormal Operating Procedures.

- A.
 - 1. variance (action DOES NOT deviate from license condition)
 - 2. IS NOT
- B.
 - 1. variance (action DOES NOT deviate from license condition)
 - 2. IS
- C.
 - 1. deviation (action DOES deviate from license condition)
 - 2. IS NOT
- D.
 - 1. deviation (action DOES deviate from license condition)
 - 2. IS

Question #84

☐ RO

☒ SRO

Tier # 1

Group # 2

K/A # EPE 74 G2.2.14

Importance Rating: 4.3

K/A Statement: Inadequate Core Cooling EQUIPMENT CONTROL: Knowledge of the process for controlling equipment configuration or status

Proposed Answer: A

Justification: OP 2260, "EOP User's Guide," states that a variance can be directed by the US IAW OP-AP-104, "Emergency and Abnormal Operating Procedures," for component failures, and specifically calls out, as an example, using a FRV for a LOAF when the FRV bypass will not move. OP-AP-104 states "OBTAIN concurrence from a second SRO, if possible.

Plausibility:

- A. Correct.
- B. Wrong, although a procedure variance is correct, a concurrence from the OMOC is NOT required. Plausible because concurrence from the crew is required for performing steps out of sequence.
- C. Wrong, a deviation is only required when a departure from a licensed condition or technical specification is required. Plausible: student may believe opening the FRV to be outside the licensed condition or technical specifications.
- D. Wrong, a deviation is only required when a departure from a licensed condition or technical specification is required and a concurrence from the OMOC is NOT required.. Plausible: examinee may believe opening the FRV to be outside the licensed condition or technical specifications and concurrence from the crew is required for performing steps out of sequence.

Technical Reference(s):

- OP 2260, "EOP User's Guide"
- OP-AP-104, "Emergency and Abnormal Operating Procedures"

Provided reference(s): None

Learning Objective: Outline each crew member's responsibilities during EOP usage as described in OP 2260, Unit 2 EOP User's Guide

Source: New

Cognitive Level: Memory

10CFR55.43(b)(3): Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments: Rewrote the stem to simplify. SRM 7/26. Changed 2nd SRO to OMOC. 2nd SRO concurrence is a Department expectation. Procedurally, the variance requires concurrence by the crew which could mean 2nd SRO depending on viewpoint. OMOC is a reasonable distractor as he/she is the 1st POC to address issues

Question #85

The plant is operating at 100% power.

- A pressurizer Safety Valve starts leaking by its seat.
- The leak rate = 11 gpm and stable.
- Pressurizer Pressure = 2250 psia and stable.

Identify the type of LEAKAGE and the basis behind its' Technical Specification limit.

- A. UNIDENTIFIED LEAKAGE; leakage could cause further deterioration, resulting in higher leakage
- B. UNIDENTIFIED LEAKAGE; leakage is of an amount that containment leakage monitoring systems can detect within a reasonable time
- C. IDENTIFIED LEAKAGE; leakage is from a known source that does not interfere with unidentified leakage detection and is within RCS makeup capacity
- D. PRESSURE BOUNDARY LEAKAGE; leakage could cause further deterioration, resulting in higher leakage.

Question #85

☐ RO

☒ SRO

Tier # 1

Group # 2

K/A # APE 78* G2.2.23

Importance Rating: 4.6

K/A Statement: RCS Leak EQUIPMENT CONTROL: Ability to track TS limiting conditions for operation

Proposed Answer: C

Justification: The TS 3.4.6.2 leak limits are as follows: No PRESSURE BOUNDARY LEAKAGE permitted, 1 gpm UNIDENTIFIED LEAKAGE, and 10 gpm IDENTIFIED LEAKAGE. Pressurizer Safety valve leakage which is classified as IDENTIFIED LEAKAGE. IDENTIFIED LEAKAGE is defined as leakage into closed systems that are captured and conducted into closed sump or collecting tank. Safety valve leakage is directed to the Quench Tank. The calculated 11 gpm leakage is IDENTIFIED LEAKAGE, the basis for which is (from TS bases) “from a known source that does not interfere with unidentified leakage detection and is within RCS makeup capacity.”

Plausibility:

- A. WRONG; Pressurizer Safety leakage is IDENTIFIED LEAKAGE. The basis given is the basis for PRESSURE BOUNDARY LEAKAGE limit. Plausible; Candidate may know that both the identified and unidentified leak limits are exceeded but not know the definition of each leakage to be able to classify it correctly.
- B. WRONG; Pressurizer Safety leakage is IDENTIFIED LEAKAGE. The basis given is for the UNIDENTIFIED LEAKAGE limit. Plausible; Candidate may know that both the identified and unidentified leak limits are exceeded but not know the definition of each leakage to be able to classify it correctly.
- C. Correct.
- D. WRONG; Pressurizer Safety leakage is IDENTIFIED LEAKAGE. The given basis is the basis for the limit of PRESSURIZER BOUNDARY LEAKAGE. Plausible; May incorrectly classify the leakage as pressure boundary leakage.

Technical Reference(s):

- AOP 2586 REV. 017, Reactor Coolant System Leak.
- MPS-2 Technical Specification LCO 3.4.6.2 and Basis.

Provided reference(s): None

Learning Objective: 283284 – Given a set of plant conditions, determine Technical Specification applicability during a Reactor Coolant leak.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases

Comments:

Question #86

The unit is in MODE 2, performing a reactor startup.

- The 'B' RCP upper motor guide bearing temperature = 195°F and rising.

The 'B' RCP 2..

AND

The US will 1..

- A.
 - 1. may remain running
 - 2. continue with OP 2202, "Reactor Startup ICCE"
- B.
 - 1. must be stopped
 - 2. enter TSAS 3.4.1, "Coolant Loops and Coolant Circulation"
- C.
 - 1. may remain running
 - 2. direct ARP 2590B-124, "RCP B UPPER GUIDE TEMP HI"
- D.
 - 1. must be stopped
 - 2. implement EOP 2525, "Standard Post Trip Actions"

Question #86

☐ RO

☒ SRO

Tier # 2

Group # 1

K/A # 003A2.02

Importance Rating: 3.8

K/A Statement: Ability to (a) predict the impacts of the following on the REACTOR COOLANT PUMP SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Conditions that exist for an abnormal shutdown of an RCP compared to a normal shutdown of an RCP

Proposed Answer: D

Justification: ARP directs trip of reactor/turbine, stop of RCP and reference to EOP-2525 at >194°F guide bearing temperature. SRO justification: The alarm setpoints and trip criteria are RO knowledge, however, selection of procedure and path for the crew is SRO level knowledge.

Plausibility:

- A. Wrong: ARP-2590B provides necessary guidance. RCP must be stopped. Plausible: May not know the temperature limit on RCP upper guide bearing.
- B. Wrong: TS 3.4.1 for MODE 3 only requires one RCP in operation. No TS impact to stopping RCP. Plausible: May think TS LCO will require action with less than 4 RCPs in operation.
- C. Wrong: Bearing temperature is above 194°F. ARP requires immediate trip of reactor/turbine and RCP. Plausible: May think bearing temperature is below maximum allowed value. ARP does provide necessary guidance on high temperature.
- D. Correct.

Technical Reference(s):

- ARP 2590B-124, "RCP B UPPER GUIDE TEMP HI"

Provided reference(s): None

Learning Objective: Given a set of plant conditions determine if any RCP trip criteria is met (286064).

Source: Bank Vision 451719 NRC 2005/Q92

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: per 2/22 phone call: Swapped order of answers to what must be done with the RCP, then what procedure will the US enter. DF 05MAR. Added discussion to justification to clarify that the procedure selection makes this question SRO level.

Question #87

The plant is in MODE 3 when 125 VDC bus DV-10 de-energizes.

The crew enters AOP 2506A, Loss of Vital 125 VDC Instrument Panel DV10.

- Step 1 is completed, the 'A' EDG is secured.
- Step 2 is completed, a PEO takes manual control of 2-FW-43A.

The crew is performing step 4 and "taking actions necessary due to the potential effects on plant operation from the circuits lost". The following conditions are noted at C-02/3:

- 'A' Charging pump is running.
- Charging flow = 44 gpm and stable
- RCS pressure = 2250 psia and stable.
- Pressurizer level = 40% and stable.

Subsequently,

- Annunciator VCT LEVEL LO (C-02/3, B-7) alarms.

The Unit Supervisor will direct the crew to...

- restore letdown flow per OP 2304A, Volume Control Portion of CVCS, in order to recover and maintain level control in the Volume Control Tank (VCT).
- enter AOP 2512, Loss of All Charging, place all three charging pumps in PTL, and close CH-518, CHG ISOL and CH-519, CHG ISOL to prevent air binding charging pumps.
- perform manual makeup to the VCT per OP 2304C, Make Up Portion of CVCS, before charging pump suction swaps to the RWST.
- enter AOP 2568, RCS Leak, for a leak rate that is within capacity of charging pumps and issue attachment E, Potential RCS Leak Isolations.

Question #87

☐ RO

☒ SRO

Tier # 2

Group # 1

K/A # 004 G2.1.6

Importance Rating: 4.8

K/A Statement: CHEMICAL AND VOLUME CONTROL SYSTEM CONDUCT OF OPERATIONS: Ability to manage the control room crew during plant transients (SRO Only)

Proposed Answer: D

Justification: Loss of power to DV-10 causes 1. CH-515 to fail closed (isolating letdown), 2. CH-518 and CH-519, charging header isolation valves, to fail open, and 3. removes ability to perform blended VCT makeup. The conditions of no letdown flow, charging at 44gpm, stable pressurizer level, and a VCT Low Level alarm are indicative of a leak from an unknown location. The indications provide 44 gpm from charging pump, 4 gpm RCP Bleedoff, and stable pressurizer level, therefore there is approximately a 40 gpm leak from an unknown location. A 40 gpm leak is within the capacity of the charging pumps capability to handle so AOP 2568 is the appropriate procedure to select with leak isolation in accordance with attachment E the appropriate action to take.

Plausibility:

- A. WRONG; Letdown flow cannot be restored because of the loss of DV-10. Letdown Isolation valve CH-515 fails closed. Plausible; Loss of DV-10 has isolated letdown, and in order to restore letdown, DV-10 power must be restored. Since an automatic VCT makeup has not occurred when it was required, the examinee may consider directing a restoration of letdown flow, which could mitigate the VCT level drop.
- B. WRONG; Charging header isolation valves cannot be closed due to the loss of DV-10. CH-518 and CH-519 fail open. Plausible; Examinee may note the VCT LO level think the loss of DV-10 will prevent an automatic charging pump suction swap on VCT LO-LO level, and this will result in a complete loss of charging pump suction flowpath. Examinee may believe action must be prioritized to protect the charging pumps from damage.
- C. WRONG; Manual VCT make up cannot be performed on a loss of DV-10. CH-512 fails closed. Plausible; Examinee may not identify there is a leak in progress based on an analysis of the given conditions, in addition, they may not understand how loss of DV-10 will prevent a manual VCT makeup. Performing a manual VCT makeup would be a required action if it were possible based on plant conditions.

D. Correct.

Technical Reference(s):

- ARP 2590B-026, VCT LEVEL LO
- MP-PROC-OPS-AOP 2506A REV. 005, Loss of Vital 125 VDC Instrument Panel DV10

Provided reference(s): None

Learning Objective:

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Changed stem to MODE 3 with MSIVs closed since loss of DV-10 will close them.

Based on 5/8 phone call. Put bullet regarding charging flow = 44 gpm back in to remove possibility of charging pump relief lifting. DF 6/7/24. Added phrase, "The plant stabilizes with the following conditions" prior to the bulleted list of conditions. Added the word "Subsequently" prior to the bullet for the annunciator. Added affected valves to the "B" distractor. SRM 7/26. Added verbiage that shows the crew entering 2506A and working through that procedure to remove the issue of entering AOP 2585. Changed Pzr level to 40% per program at HZP. DF 8/21

Question #88

The plant is in MODE 1.

- An inadvertent Facility 2 SRAS occurs.

In order to restore the ____1.____ to OPERABLE status, SRAS must be reset AND ____2.____.

- A.
 - 1. Facility 2 RBCCW header
 - 2. 2-RB-13.1B, "B" SDC HX RBCCW outlet valve, must be reclosed
- B.
 - 1. Facility 2 ECCS Safety Injection subsystem
 - 2. the "B" LPSI pump must be started
- C.
 - 1. Facility 2 Containment Spray header
 - 2. 2-SI-660, SI Pump Minimum Flow Isolation valve, must be reopened
- D.
 - 1. Containment INTEGRITY
 - 2. 2-CS-13.1B, CS Header "B" Isolation valve, must be reclosed

Question #88

☐ RO

☒ SRO

Tier # 2

Group # 1

K/A # 008 G2.1.7

Importance Rating: 4.7

K/A Statement: COMPONENT COOLING WATER SYSTEM CONDUCT OF OPERATIONS:

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Proposed Answer: A

Justification: With RB-13.1B open (in Mode 1-3), the applicable RBCCW header is not OPERABLE. The added flow demand on the header would limit its capability as a heat sink for the "Design Base Accident".

Plausibility:

- A. Correct.
- B. Wrong, with CS-16.1B open, all SI pumps would be vapor bound on any accident in CTMT. Plausible: Fac. 2 ECCS is inoperable and one of the reasons is the inability of the LPSI pump to start as designed.
- C. Wrong, SI-660 cannot reposition unless the switch on C-01 is placed in "Operate". Plausible: Student may not recall the necessity of placing key switch in "Operate" to reposition SI-660, and SI-660 closed would cause Containment Spray to be not OPERABLE.
- D. Wrong, CS-13.1B is a normally open valve which does not get an open signal on an SRAS, so no action has to be taken on this valve to restore CTMT Integrity. Plausible: CS-13.1B open at the same time as CS-16.1B provides no closed (non-check) valve between the RWST and Containment.

Technical Reference(s):

- OP 2330A, "RBCCW System"
- AOP 2571, "Inadvertent ESFAS Actuation"

Provided reference(s): None

Learning Objective: Given any operating condition for the RBCCW System, state whether the condition requires entry into the Technical Specifications (281951).

Source: Bank Vision 477364

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Question #89

The plant is operating at 100% power.

- An inadvertent Facility 1 and Facility 2 AFAS actuation has occurred.
- The AFAS actuation signals have failed to reset.
- 'A' S/G Level = 70.9 % and stable.
- 'B' S/G Level = 71.2 % and stable.
- CVMWTH = 2756.0 MW_{TH}
- CVHRCAL = 2699.2 MW_{TH}
- CV12HCAL = 2699.0 MW_{TH}

What actions are required in accordance with...

1. OP 2204, Load Changes

AND

2. Technical Specifications?

- A.
 1. Immediately reduce power to less than 2,699.8 MW_{TH}.
 2. be in MODE 3 within 6 hours.
- B.
 1. Immediately trip the reactor.
 2. be in MODE 4 within 13 hours.
- C.
 1. No action required since CV12HCAL ≤ 2700 MW_{TH}.
 2. be in MODE 3 within 6 hours.
- D.
 1. Immediately reduce power to less than 2,699.8 MW_{TH}.
 2. DO NOT change MODEs until an AFW pump is OPERABLE.

Question #89

☐ RO

☒ SRO

Tier # 2

Group # 1

K/A # 061A2.01

Importance Rating: 3.3

K/A Statement: Ability to (a) predict the impacts of the following on the AUXILIARY/EMERGENCY FEEDWATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Inadvertent actuation of AFW at power

Proposed Answer: A

Justification: OP 2204 section 4.2.6 directs an immediate reduction in power to less than or equal to 2,699.8 MWth if PPC point CVMWTH exceeds 102 %. Since the inadvertent AFW actuation signals are unable to be reset on either train, both Motor Driven AFW pumps are declared inoperable since they can no longer meet the surveillance requirement TS 4.7.1.2.d. Additionally, TS 3.7.1.2 action d is entered because two AFW pumps are inoperable in MODEs 1, 2, 3. This requires transition to MODE 3 within 6 hours and MODE 4 within the following 12 hours.

Plausibility:

- A. Correct.
- B. WRONG; There is no requirement to immediately trip the reactor. Mode 4 in 13 hours would be correct if TS 3.0.3 was entered, and they thought they were able to do a mode change with the TDAFP available. Plausible; may think that power at 102.1 % is indication of a significant issue and requires more severe action.
- C. WRONG; 'CVMWTH' power has risen to 102.1 % and requires immediate action to reduce power to less than 2,699.8 MWth. Plausible; MODE 3 within 6 hours is correct for two Motor Driven AFW pumps inoperable. Also, OP 2204 4.2.7 does have requirements to reduce power based on 12-hour average power, but the distractor states that those are within limits, so candidate may think no action is required.
- D. WRONG; TS 3.7.1.2 action d requires MODE 3 in 6 hours. Plausible; With two Motor Driven AFW pumps inoperable, the candidate may think a MODE change is not allowed. 3.7.1.2 does not allow a MODE change if 3 AFW pumps are inoperable.

Technical Reference(s):

- MP-PROC-OPS-OP 2204 REV. 048, Load Changes.
- AOP 2571, Inadvertent ESFAS Actuation
- MPS-2 Technical Specifications TS 3.0.3, TS 3.7.1.3 and TS basis.

Provided reference(s): None

Learning Objective: 282548 – Determine the operator action required when core thermal power exceeds the license limit of 2700 MWth

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Question #90

The plant is in MODE 1.

- Meters on the right side of all control boards have failed.
- The core mimic is dark.
- A loss of VR-21 is diagnosed.
- The crew has implemented AOP 2504B, “Loss of Non-Vital Instrument Panel VR-21.”

What effect does this event have on Technical Specification LEAKAGE DETECTION SYSTEMS?

LCO 3.4.6.1, LEAKAGE DETECTION SYSTEMS, is 1.,

AND

The required ACTION for continued operation is 2. (if applicable).

- A.
 - 1. met
 - 2. none, all the LCO required equipment is unaffected by this event.
- B.
 - 1. NOT met
 - 2. perform an RCS inventory AND CTMT atmosphere grab samples every 24 hours.
- C.
 - 1. met
 - 2. none, the LCO only requires one CTMT particulate Radiation Monitor.
- D.
 - 1. NOT met
 - 2. perform an RCS inventory balance every 24 hours.

Question #90

☐ RO

☒ SRO

Tier # 2

Group # 1

K/A # 073 G2.4.20 Importance Rating: 4.3

K/A Statement: PROCESS RADIATION MONITORING SYSTEM EMERGENCY

PROCEDURES/PLAN: Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions, and notes

Proposed Answer: D

Justification: A loss of VR-21 removes power from the Containment Sump level indication. Both Containment atmosphere particulate rad monitors are vitally powered (VA-10 for RM-8123, VA-20 for RM-8262). One containment rad monitor and the containment sump level monitor are required to meet the LCO. Losing only containment sump level indication requires only an inventory balance. K/A justification: Assessment of lost radiation monitors (Radiation Monitoring System) during the Loss of VR-21 (Emergency Procedures/Plan) is required to answer the question, and there is a note prior to Step 11 (EOP/AOP warnings, cautions and notes) of the AOP for assessing technical specifications which clarifies assessment of Technical Specification applicability including things to consider.

Plausibility:

- A. Wrong, containment sump level is lost, and is required by the LCO. Plausible: a loss of VR-11 would not affect containment sump level, or either one of the containment atmosphere rad monitors.
- B. Wrong, grab samples are not required as long as either one of the containment atmosphere particulate rad monitors remain OPERABLE. Plausible: a loss of VA-10 or VA-20 would de-energize one containment atmosphere particulate rad monitor, and student may not recall that only one is required.
- C. Wrong, containment sump level is lost, and is required by the LCO. Plausible: a loss of VA-10 or VA-20 would de-energize one containment atmosphere rad monitor, and not affect containment sump level indication.
- D. Correct.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications
- AOP 2504B, "Loss of Non-Vital Instrument Panel VR-21"

Provided reference(s): None

Learning Objective: Given any plant Operating Mode and the operability status of any Radiation Monitoring system instrument channel, use information in the Unit 2 Technical Specifications or other Administrative requirements, to determine required actions (282098).

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Added justification to clarify the tie to the K/A. Added a bullet to state that the crew entered AOP 2504B. Changed “daily” to “every 24 hours” to align with the language in TSAS 3.4.6.1. SRM 7/26.

Question #91

The plant is operating at 100% power.

- The turbine tripped due to a Loss of Offsite Power (LOOP).
- The reactor did not automatically trip.
- The reactor was manually tripped.
- 3 CEAs did not fully insert.
- Emergency boration is in progress at 88 gpm.
- Reactor power is 5×10^{-4} % and lowering
- SUR = -0.3 DPM and stable.

EOP 2541, Standard Appendices, Appendix 1, Diagnostic Flow Chart, directs entering 1.,
because 2..

- A.
 - 1. EOP 2540, Functional Recovery
 - 2. the reactor did not trip when the turbine tripped.
- B.
 - 1. EOP 2540, Functional Recovery
 - 2. more than 1 CEA is not fully inserted.
- C.
 - 1. EOP 2526, Reactor Trip Recovery
 - 2. all safety functions are satisfied.
- D.
 - 1. EOP 2528, Loss of Offsite power/Loss of Forced Recirculation
 - 2. RCPs were lost on the Loss of Offsite Power.

Question #91

☐ RO

☒ SRO

Tier # 2

Group # 2

K/A # 001 G2.4.5

Importance Rating: 4.3

K/A Statement: SF1 CRDS) CONTROL ROD DRIVE SYSTEM EMERGENCY

PROCEDURES/PLAN: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions

Proposed Answer: D

Justification: According to the diagnostic flow chart, reactivity control is met even though three CEAs are not fully inserted, because an adequate emergency boration is in progress. No RCPs are running due to the LOOP so the diagnostic directs the SRO to consider EOP 2528, Loss of Offsite power/Loss of Forced Recirculation.

Plausibility:

- A. WRONG; EOP 2525, Standard Post Trip Actions (SPTA), step 1 RNO directs manually tripping the reactor which successfully mitigates that failure. Plausible; RPS failing to automatically trip the reactor when required is part of the entry criteria into EAL MU6.1 and may be confused with the Safety Function not being met.
- B. WRONG; EOP 2525 step 1 RNO directs an emergency boration per EOP 2541 App 3, Emergency Boration if more than one CEA is not fully inserted. The diagnostic flowchart requires power is 10^{-4} and lowering with negative SUR and EITHER Emergency boration in progress, OR a max of one CEA not fully inserted. Since all other criteria are met other than CEA insertion, EOP 2540 is not required. Plausible; Candidate may not remember the 'OR' statement that allows emergency boration to accommodate for more than the max of one CEA not being fully inserted.
- C. WRONG; All safety functions are satisfied, but there still was a LOOP that caused loss of all running RCPs and a loss of forced circulation. In this situation, EOP 2526 is not the appropriate procedure selection. Plausible; after completing the diagnostic flow chart, the candidate may acknowledge that all safety functions are being met, but not remember to consider forced circulation has been lost, and EOP 2528 entry is required.

D. Correct.

Technical Reference(s):

- MP-PROC-OPS-EOP 2525 REV. 029, Standard Post Trip Actions.
- MP-PROC-OPS-EOP 2541 App. 1 REV. 002, Diagnostic Flowchart.
- MP-PROC-OPS-EOP 2540 REV. 027, Functional Recovery.

Provided reference(s): None

Learning Objective: 283658 – Given a set of plant conditions and the Diagnostic Flowchart, determine the most appropriate EOP to mitigate the event.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Question #92

The plant is in MODE 1.

- The following ESAS SENSOR CABINET 'C' switches are in 'INHIBIT':
 - CTMT PRESS SIAS/CIAS.
 - CTMT PRESS CSAS.
- Within a minute of the Channel 'C' switches being placed in 'INHIBIT', ESAS Channel 'D' Containment (CTMT) Pressure fails HIGH (i.e. CTMT PRESS SIAS/CIAS and CTMT PRESS CSAS tripped on channel 'D').
- Action is immediately taken to positively maintain the 'D' channel of CTMT PRESS SIAS/CIAS and CTMT CSAS in the tripped condition.

Operation in this configuration may continue for up to 1. .

AND

An ESF actuation 2. occur due to the 'D' Channel CTMT pressure instrument failure.

- A.
 - 1. 1 hour
 - 2. did
- B.
 - 1. 48 hours
 - 2. did
- C.
 - 1. 1 hour
 - 2. did NOT
- D.
 - 1. 48 hours
 - 2. did NOT

Question #92

☐ RO

☒ SRO

Tier # 2

Group # 2

K/A # 016A2.01

Importance Rating: 3.4

K/A Statement: Ability to (a) predict the impacts of the following on the NONNUCLEAR INSTRUMENTATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Detector/transmitter failure

Proposed Answer: D

Justification: Containment Pressure RPS trip, and Containment Pressure SIAS, CIAS, EBFAS, MSI and CSAS are all listed with 4 as the “Total number of channels” and 3 as the “Required number of channels” in Table 3.3-1 (for RPS) and 3.3-3 (for ESAS). The failure of channel ‘C’ will meet the required number, but go 1 below the total number of channels, for all of these actuations, requiring the channel to be bypassed or tripped within 1 hour, or tripped within 24. This is met for ESAS with the SIAS/CIAS and CSAS switches in INHIBIT (the stem does not speak to RPS because that knowledge is not being tested in this question, but procedurally it would be bypassed at RPS, as well). Table 3.3-1 (for RPS) and 3.3-3 (for ESAS) action 2c (same action in both tables) allows a second channel out of service for up to 48 hours as long as one of the out of service channels is taken to trip. ESAS would NOT be actuated on “D” failing high because “C” channel is already in inhibit for SIAS/CIAS and for CSAS.

Plausibility:

- A. Wrong, 48 hours is allowed as long as channel “C” or “D” is placed in a trip condition. Additionally, ESAS actuation will NOT occur because channel “C” is in INHIBIT for SIAS/CIAS and CSAS, i.e. in a bypassed condition. Plausible: student may believe there is no guidance in Technical Specifications for the second channel failure, placing the unit in TS 3.0.3 with a 1 hour AOT. Additionally, student may not recall that INHIBIT on ESAS is the position for bypassing the channel.
- B. Wrong, ESAS actuation will NOT occur because channel “C” is in INHIBIT for SIAS/CIAS and CSAS, i.e. in a bypassed condition. Plausible, student may not recall that INHIBIT on ESAS is the position for bypassing the channel.
- C. Wrong, 48 hours is allowed as long as channel “C” or “D” is placed in a trip condition. Plausible: student may believe there is no guidance in Technical Specifications for the second channel failure, placing the unit in TS 3.0.3 with a 1 hour AOT.
- D. Correct.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications
- ARP 2590A-084, CONT PRES HI D D-21
- North Anna Power Station 2010 NRC SRO Exam, Question 81

Provided reference(s): None

Learning Objective: Given any operating condition or status for the Engineered Safety Features Actuation System (ESAS), state whether the condition requires entry into the Technical Specifications (281368)

Source: Modified (North Anna Bank)

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Removed the reason why 'C' channel is in INHIBIT (not relevant). Edited question for readability and specificity. Added a time requirement for 'C' channel out of service to meet TS requirements. DF 5/28. Changed source from New to Modified to consistently apply philosophy that industry bank questions count as Bank. SRM 7/26. Changed answers in part 2 to 'di' and 'did NPT' to reflect time component of question based on validator feedback. DF 7/27

NAPS 2010 NRC SRO Exam

81.

Given the following conditions:

- Unit 1 is at 100% power.
- Containment Pressure Protection Channel I, 1-P-LM-100A, loses power.
- The crew has completed 1-AP-3, Loss of Vital Instrumentation.
- The crew has NOT yet performed any actions of the applicable MOP.

Which ONE of the following identifies the Technical Specification required action, **AND** includes the plant response if Containment Pressure Protection Channel II subsequently fails HIGH prior to any actions being taken?

Place 1-P-LM-100A in BYPASS within _____.

- A. 1 hour; an ESF actuation will occur
- B. 72 hours; an ESF actuation will occur
- C. 1 hour; an ESF actuation will NOT occur
- D. 72 hours; an ESF actuation will NOT occur

Question #93

The plant is operating at 100% power.

- A Clean Liquid Radioactive Waste discharge is in progress.
- After 30 minutes, RM-9049, Liquid Waste Radiation Monitor, alarms.
- RM-9049 is reading off-scale high.
- 2-LRR-32.1 & 32.2, Clean Liquid Discharge Isolation valves, closed automatically.
- I&C determines RM-9049 has failed and will not be returned to service for 30 days.

Per the Offsite Dose Calculation Manual...

1. Releases _____ continue via this pathway

AND

2. What are the requirements for the situation?

- A.
 1. MAY
 2. Perform gamma analyses on two independent samples, independent verification of calculation and valve lineups.
- B.
 1. MAY NOT
 2. Perform a grab sample to independently confirm that high RM-9049 reading is not due to a high radiation condition.
- C.
 1. MAY
 2. During releases obtain grab samples once every 12 hours and estimate the flow rate using pump curves once every 4 hours.
- D.
 1. MAY NOT
 2. A release is not permitted until RM-9049 is repaired.

Question #93

☐ RO

☒ SRO

Tier # 2

Group # 2

K/A # 068A2.02

Importance Rating: 3.8

K/A Statement: Ability to (a) predict the impacts of the following on the LIQUID RADWASTE SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Failure to comply with the conditions in release permit

Proposed Answer: A

Justification: MP-22-REC-BAP01, Offsite Dose Calculation Manual, TABLE IV.C.-1 Action A states: With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirements, effluent releases may continue provided that best efforts are made to repair the instrument and that prior to initiating a release: (1) At least two independent samples are analyzed for gamma radioactivity as specified in Table I.C.-2, and; (2) The original release rate calculations and discharge valving are independently verified by a second individual.

Plausibility:

A. Correct.

B. WRONG; Releases via this path are not required to be suspended if the appropriate ODCM actions are taken. Plausible; Action B on ODCM Table IV.C.-1 has an action to suspend all effluent releases as a possible approach, but ONLY for Steam Generator Blowdown Monitoring.

C. WRONG; must perform gamma analyses on two independent samples, independent verification of calculation and valve lineups. Plausible; Obtaining grab samples once per 12 hours and estimating flow rate every 4 hours during releases sounds correct, but is a combination of actions incorrectly taken from Actions B,C and D.

D. WRONG; Releases via this path are not required to be suspended if the appropriate ODCM actions are taken. Plausible; Repairing the RM-9049 is something that would eventually occur regardless, but in the meantime, releases may continue based on allowances in ODCM Table IV.C.-1 Action A.

Technical Reference(s):

- MP-22-REC-BAP01 REV. 032, Offsite Dose Calculation Manual
- RLD-04-C REV. 001, Radioactive Liquid Waste System Discharges Lesson Plan
- CLR-04-C REV. 004, Clean Liquid Radwaste System Lesson Plan

Provided reference(s): None

Learning Objective: 331949 – Review and approve radioactive waste permitted discharges

Source: Modified Saint Lucie NRC exam Q #98

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases

Comments: Validator feedback is that we don't do Aerated Liquid Waste discharges anymore. Changed to Clean Liquid Radwaste. Changed RM-9116 to RM-9049. Changed reference 3 to CLR-04C DF 5/28. Added noun name "Liquid Waste Radiation Monitor" to RM-9049. This is the noun name listed in OP 2383A, "Process Radiation Monitoring Systems." SRM 7/26.

98.

Given the following conditions:

- Unit 1 is at 100% power
- Gaseous Release Permit # 19-12 was issued to authorize this release
- A gaseous release is in progress from the 1A Gas Decay Tank(GDT) in accordance with 1-NOP-06.20, Controlled Gaseous Batch Release to Atmosphere

Subsequently:

- Radiation Monitor Channel #42, Waste Gas Monitor fails HIGH

Which ONE of the following describes the MINIMUM required actions to properly complete the discharge from the 1A GDT, in accordance with 1-NOP-06.20?

- A. Restart the release using permit #19-12
- B. Issue a new release permit with independent samples and valve lineup verifications
- C. Complete an additional independent GDT release rate calculation, then restart the release using permit #19-12
- D. Complete an additional independent GDT grab sample, verifying tank activity has not changed, then restart the release using permit #19-12

Question #94

The 'A' SIT requires sampling in MODE 1.

OP 2306O, Safety Injection Tanks, RCS >1,750 psi, directs an Operator be stationed at 2-SI-463, SIT'S RECIRC HEADER STOP, with instructions to close the valve when directed.

The operator at 2-SI-463 1. perform other duties.

AND

Entry into TSAS 3.6.3.1, CONTAINMENT ISOLATION VALVES, 2. required.

- A.
 - 1. may NOT
 - 2. is NOT
- B.
 - 1. may NOT
 - 2. is
- C.
 - 1. may
 - 2. is NOT
- D.
 - 1. may
 - 2. is

Question #94

☐ RO

☒ SRO

Tier # 3

Group #

K/A # G2.1.23

Importance Rating: 4.4

K/A Statement: CONDUCT OF OPERATIONS: Ability to perform general and/or normal operating procedures during any plant condition

Proposed Answer: A

Justification: 2-SI-463 is a Containment isolation valve and is required to be locked closed per Containment integrity requirements. Per OP 2306O, whenever the valve is operated in Modes 1 - 4, a Dedicated Operator must be stationed at the valve, in direct communications with the control room, for the entire time the valve is unlocked or open. During this time, per TS bases, there must be an assurance that access to close the valve is not precluded and the action will prevent the release of radioactivity outside the containment. Additional duties of this operator would fall under the heading of "precluding access to close the valve," and are, therefore, not allowed. Additionally, the SRO must make a Narrative Log entry for the existing valve position required.

Plausibility:

- A. Correct.
- B. Wrong, opening a Containment Isolation valve under the control of a dedicated operator does NOT require entry into the TSAS 3.6.3.1, Note 1: "Containment Isolation Valves may be opened on an intermittent basis under administrative controls." Plausible: If the examinee does NOT remember the note concerning the administrative control of Containment Isolation Valves, then it would be logical to assume TSAS 3.6.3.1 would apply.
- C. Wrong, operation of 2-SI-463 requires a "Dedicated Operator" (whose access to close SI-463 is not precluded, i.e. no other duties), due to the administrative impact of a Containment Isolation Valve being open with a subsequent CIAS. Plausible: There are many Technical Specification controlled valves, that do not require a "Dedicated Operator" be present. The examinee may feel that this is one of them.
- D. Wrong, opening a Containment Isolation valve under the control of a dedicated operator does NOT require entry into the TSAS 3.6.3.1, Note 1: "Containment Isolation Valves may be opened on an intermittent basis under administrative controls." Additionally, operation of 2-SI-463 requires a "Dedicated Operator" (whose access to close SI-463 is not precluded, i.e. no other duties), due to the administrative impact of a Containment Isolation Valve being open with a subsequent CIAS. Plausible: If the examinee does NOT remember the note concerning the administrative control of Containment Isolation Valves, then it would be logical to assume TSAS 3.6.3.1 would apply. Additionally, there are many Technical Specification controlled valves, that do not require a "Dedicated Operator" be present. The examinee may feel that this is one of them.

Technical Reference(s):

- OP 2306O, "Safety Injection Tanks, $RCS \geq 1750$ psi"
- Millstone Unit 2 Technical Specifications
- Vision Question 451889

Provided reference(s): None

Learning Objective: Given a set of plant conditions, determine Technical Specification applicability concerning a Loss of Containment Integrity (282956)

Source: Bank Vision 451889 NRC 2011 Q90

Cognitive Level: Memory

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Question #95

The plant is in MODE 1.

- RWST Concentration = 1800 ppm
- RWST Level = 95%
- 'A' BAST concentration = 3.25 wt%
- 'A' BAST volume = 4400 gal
- 'B' BAST concentration = 3.00 wt%
- 'B' BAST volume = 5000 gal

The 'A' Boric Acid Pump is declared NOT OPERABLE.

To meet the requirements for BAST contents per TR 3.1.2.8, the SRO directs 1..

- A. no additional action be taken, TR 3.1.2.8 is met.
- B. raising RWST concentration to 1900 ppm
- C. raising 'A' BAST concentration to 3.50 wt%
- D. raising 'B' BAST concentration to 3.25 wt%

Question #95

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Tier # 3

Group #

K/A # G2.1.25

Importance Rating: 4.2

K/A Statement: CONDUCT OF OPERATIONS: Ability to interpret reference materials, such as graphs, curves, and tables (reference potential)

Proposed Answer: D

Justification: Raising the 'B' BAST concentration from 3.0 wt% to 3.25 wt% at a constant volume of 5000 gallons would put the 'B' BAST in the Acceptable Region of TRM Figure 3.1-1. This makes the 'B' BAST able to meet the requirements of the TRM ALONE without the 'A' BAST. The 'A' and 'B' BAST cannot be considered together using a weighted average per TR 3.1.2.8.a.2 because the 'A' BA pump will not function.

Plausibility:

- A. WRONG; Action is required to operate per the TRM Figure 3.1-1 acceptable region. Using TR 3.1.2.8 and Figure 3.1-1 shows no one single BAST meeting the requirement. In addition, without the 'A' BA pump available, the 'A' and 'B' BAST volumes cannot be combined in a weighted average per TR 3.1.2.8.a.2. Plausible; candidate may misinterpret 3.1.2.8.a.2 or not recognize the effect between the 'A' BA pump being unavailable and the Technical Requirement. This could cause them to believe a weighted average of the two tanks will meet the TR.
- B. WRONG; Raising RWST concentration to 1900 ppm will change which Figure 3.1-1 line is used, which lowers the required gallons that a BAST must contain. The 'A' BAST at a 3.25wt% requires over 4600 gal, the 'B' BAST at a 3.0 wt% requires over 5450 gal. Plausible; Candidate may not use graph correctly when changing the RWST parameter and holding the other parameters constant, it can also be challenging to correctly trace the right line for RWST ppm.
- C. WRONG; This combination does not meet TRM requirements. 4593 gallons would be required in the 'A' BAST at a 3.50 w%. Plausible; Candidate may not apply the use of the graph correctly to understand the new parameters still do not meet the TRM requirements.
- D. Correct.

Technical Reference(s):

- MPS-2 Technical Requirements Manual (TRM)
- Vision Question 416707

Provided reference(s):

- TRM 3.1.2.8, Borated Water Sources – Operating
- TRM Figure 3.1-1, Minimum Volume vs Stored BAST concentration.

Learning Objective:

Source: Modified Vision 416707

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(1): Conditions and limitations in the facility license.

Comments:

The plant is operating at 100% power. Based on the following information, what is the minimum level necessary for each BAST, by itself, to conform with the requirement stated in the TRM?

- o RWST concentration = 1.09% (1900 ppm)
 - o RWST level = 95% (451,250 gal)
 - o "A" BAST concentration = 3.25% (5683 ppm)
 - o "B" BAST concentration = 3.00% (5245 ppm)
- A. 'A' BAST min level 3750 Gals 'B' BAST min level 4878 Gals
- B. 'A' BAST min level 4050 Gals 'B' BAST min level 5245 Gals
- C. 'A' BAST min level 4600 Gals 'B' BAST min level 5450 Gals
- D. 'A' BAST min level 4350 Gals 'B' BAST min level 5200 Gals

Answer: C

Answer Explanation

By referencing Tech Requirements Figure 3.1-1 the volumes necessary to be above the curve for 1900 ppm RWST concentration, given the concentration of the A and B BASTs at 3.25% and 3.0% respectively requires at least 4600 and 5450 gal respectively. REF: Tech Requirements. 3.1.2.8, FIG 3.1-1

Question 1 Info

Question Type:	Multiple Choice
Status:	Review
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
Allow multiple selections?	No
Randomize choice order?	No
System ID:	416707
Version ID:	499383
User-Defined ID:	0053345-MB
Cross Reference Number:	
Topic:	MP2*LORT*2318 [004 CVC-01-C 3112] (9/30/97) CVCS, TS 3.1.2.8, ADMIN, RWST, BAST
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	

Question #96

An activity has been proposed which has the following conditions:

- The activity is a test is not described in the FSAR.
- Performance of the test conflicts with Technical Specifications requirements.

Identify the requirements to conduct the activity.

A 10CFR50.59 evaluation 1. required.

AND

Prior approval from the NRC 2. required.

- A. 1. is
2. is NOT
- B. 1. is
2. is
- C. 1. is NOT
2. is NOT
- D. 1. is NOT
2. is

Question #96

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☒ SRO

Tier # 3

Group #

K/A # G2.2.5

Importance Rating:3.2

K/A Statement: EQUIPMENT CONTROL: Knowledge of the process for making design or operating changes to the facility, such as 10 CFR 50.59, “Changes, Tests and Experiments,” screening and evaluation processes, administrative processes for temporary modifications, disabling annunciators, or installation of temporary equipment

Proposed Answer: B

Justification: The screening has determined that the activity constitutes a test not described in the FSAR so it must be further evaluated against the eight criteria of 10 CFR 50.59 c.2, and since there is a TS conflict a license amendment request must be submitted and approval obtained prior to conducting the activity.

Plausibility:

- A. Wrong, the license amendment must be submitted and approved prior to conducting the activity. Plausible: a student unfamiliar with the process may believe that an amendment request which demonstrates a lack of significant hazards may not require prior approval.
- B. Correct.
- C. Wrong, the activity is a test not described in the FSAR, so the 10 CFR 50.59 evaluation is required. Additionally, because of the TS conflict, the license amendment must be submitted and approved prior to conducting the activity. Plausible: a student without detailed knowledge of the process may conclude that an amendment request which demonstrates a lack of significant hazards may not require prior approval, and additionally, since a TS change is already being processed, a 10 CFR 50.59 evaluation is not required.
- D. Wrong, the activity is a test not described in the FSAR, so the 10 CFR 50.59 evaluation is required. Plausible: student may believe that since a TS change is already being processed, a 10 CFR 50.59 evaluation is not required (this would be true if a TS change was the only relevant attribute to this activity).

Technical Reference(s):

- CM-AA-400, “10 CFR 50.59 and 10 CFR 72.48—Changes, Tests, and Experiments”
- North Anna 2009 NRC SRO exam Question 96

Provided reference(s): None

Learning Objective: OUTLINE the process for development of a new document (251838).

Source: Bank

Cognitive Level: Memory

10CFR55.43(b)(3): Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments: CM-AA-400 is the successor procedure for DNAP 3004, the procedure used on the original North Anna question. Changed source from New to Bank to consistently apply philosophy that industry bank questions count as Bank. SRM 7/26.

Question #97

The plant is in MODE 1.

- It has been identified that SR 4.5.1.e, “Verify that the closing coil in the valve breaker cubicle is removed at the frequency specified in the Surveillance Frequency Control Program,” was last performed 40 days ago for the ‘A’ SIT.
- The Surveillance Frequency Control Program specifies an interval of 31 days for SR 4.5.1.e.

Which of the following describes the operability of the ‘A’ SIT

1. The ‘A’ Safety Injection Tank is 1..

AND

2. Why?

- A.
 1. OPERABLE
 2. The surveillance must be performed within the next 31 days. If the surveillance is delayed > 24 hours, a risk evaluation is performed.
- B.
 1. OPERABLE
 2. The surveillance must be performed within the next 31 days plus a 25% grace period.
- C.
 1. OPERABLE
 2. The surveillance and a risk evaluation must be performed within the next 24 hours.
- D.
 1. NOT OPERABLE
 2. The SIT remains NOT OPERABLE until the surveillance is completed successfully.

Question #97

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Tier # 3

Group #

K/A # G2.2.45

Importance Rating: 4.5

K/A Statement: EQUIPMENT CONTROL: Ability to determine and/or interpret TS with action statements of greater than 1 hour (SRO Only)

Proposed Answer: A

Justification: TS 4.0.2 allows a 25% time extension for TS surveillance requirement intervals, if the missed surveillance is discovered within that time frame. When a TS is identified as missed and it is outside the 25% extension time, TS 4.0.3 is invoked. TS 4.0.3 allows for completion of the missed TS within 24 hours or within the original TS surveillance interval (in this case 31 days) whichever is greater. If the TS 4.0.3 extension exceeds 24 hours, a risk assessment must be completed and the risk impact must be managed.

In the stem, the 31-day surveillance requirement is identified as being 40 days past due, which is outside the 25% extension time ($31 \text{ days} * 1.25 = 38.75 \text{ days}$).

Invoking 4.0.3 allows the missed surveillance to be completed within the originally allowed surveillance time of 31 days as long as a risk assessment is performed, and risk is managed.

Plausibility:

A. Correct.

B. INCORRECT: Since the missed surveillance is already outside the TS 4.0.2 allowed extension time, it cannot be applied. Plausible; Candidate may think you can apply the TS 4.0.2 extension time (25%) to any surveillance that has been identified as missed, regardless of how past-due the surveillance is.

C. INCORRECT: The surveillance must be performed but 24 hours is not the time limit. LCO 4.0.3 allows for the original 31 days for completion if any delay past 24 hours has an associated risk assessment and risk is managed. Plausible; LCO 4.0.3 mentions the 24-hour requirement regarding requiring a risk assessment and risk management actions being in place. The candidate may remember that time as the time requirement for completing the surveillance.

D. INCORRECT: The "A" SIT is still operable. When a missed surveillance is discovered outside the 25% extension time, LCO 4.0.3 allows for an additional 24 hours or the original extension time (31 days), whichever is greater before they are required to declare the "A" SIT INOPERABLE. Plausible; Candidate may believe that any failure to meet a TS Surveillance time requires immediately declaring the "A" SIT INOPERABLE.

Technical Reference(s):

- MPS-2 TECHNICAL SPECIFICATIONS, LCO 4.0.2/4.0.3 Amendment 345
- MP-PROC-000-CM-M2-STI-101 REV. 013, Millstone Unit 2 Technical Specification Surveillance Test Interval (STI) List

Provided reference(s): None

Learning Objective: 282396 – As given in Technical Specifications, state the maximum allowed extension of a surveillance interval and the principles governing the use of an extension

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(1): Conditions and limitations in the facility license.

Comments: Reworded stem to align with the wording in TSSR 4.5.1e. SRM 7/26.

Question #98

The plant is in MODE 1.

- POST INCIDENT RAD. MONITOR HI/FAILURE (C-02/3, D-10) is alarming.
- Both red and yellow lights illuminated on both RM-8240 & RM-8241, Containment High Range Radiation Monitors.
- All other Containment Area Radiation Monitors indicate normally.

What ACTIONS are taken per Technical Specifications?

- A. Per TSAS 3.3.3.1, RADIATION MONITORING: initiate the preplanned alternate monitoring method within 72 hours and restore the inoperable channel(s) to OPERABLE status within 7 days.
- B. Per TSAS 3.0.3, within 1 hour, initiate ACTION to commence a plant shutdown and subsequent cooldown. Be in at least COLD SHUTDOWN within the subsequent 36 hours.
- C. Per TSAS 3.3.3.1, RADIATION MONITORING: within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.
- D. Per TSAS 3.4.8, REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY: Within one hour, verify the specific activity of the primary coolant is $\leq 1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$.

Question #98

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Tier # 3

Group #

K/A # G2.3.5

Importance Rating: 2.9

K/A Statement: RADIATION CONTROL: Ability to use RMSs, such as fixed radiation monitors and alarms or personnel monitoring equipment

Proposed Answer: A

Justification: With both red and yellow lights energized, the radiation monitors indicate a common failure and are, therefore, inoperable. TSAS 3.3.3.1, Table 3.3-6, ACTION 17 applies. The alternate method of monitoring for EP-AA-DAM-101, Core Damage Estimation, specifies using the Main Steam Line Radiation Monitors as an alternate monitoring method.

Plausibility:

- A. Correct.
- B. Wrong, TSAS 3.0.3 does NOT apply. Even though there are 2 Containment High Range Radiation Monitors and only 1 is required to OPERABLE, the Technical Specification Action is for 1 less than the minimum required, which is none OPERABLE. Plausible: The examinee may believe that, because there are only 2 Containment Radiation Monitors, then both must be OPERABLE. If only 1 is OPERABLE, then Technical Specification LCO 3.3.3.1 must apply. If both are inoperable, then the student may believe TSAS 3.0.3 must apply.
- C. Wrong, this action is only required for an actual high radiation condition, with the Control Room Area rad monitor inoperable. Plausible: The examinee may confuse the ACTION associated with the Containment High Range Rad Monitors with the ACTION of the Control Room Radiation Monitors, which is listed in the same table.
- D. Wrong, The requirements of this TSAS ACTION do NOT apply for just this indication. Plausible: The examinee may feel that the Containment High Range detectors are indicating a possible high radiation in the Containment due to a severe crud burst.

Technical Reference(s):

- Millstone Unit 2 Technical Specifications
- Bank Vision 451893

Provided reference(s): None

Learning Objective: Given any plant Operating Mode and the operability status of any Radiation Monitoring System Instrument Channel, use information in the Unit 2 Technical Specifications or other Administrative requirements, to determine required actions (282098).

Source: Bank 451893 NRC 2011 Q97

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(4): Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments: Editorial changes to improve verbiage.

Question #99

The plant experienced a Steam Generator Tube Rupture.

1. What Time Credited Operator Action (TCOA) is performed during the event?

AND

2. What is the basis for the TCOA?

- A.
 1. Cool down the RCS to $< 515^{\circ}\text{F}$ and isolate the affected S/G within one (1) hour of the reactor trip.
 2. Prevents inadvertently opening a Main Steam Safety Valve (MSSV).
- B.
 1. Isolate Auxiliary Feedwater to the affected S/G within thirty (30) minutes of the event.
 2. Limits volume of contaminated water in the secondary systems.
- C.
 1. Cool down the RCS to $< 515^{\circ}\text{F}$ and isolate the affected S/G within one (1) hour of the reactor trip.
 2. Limits volume of contaminated water in the secondary systems.
- D.
 1. Isolate Auxiliary Feedwater to the affected S/G within thirty (30) minutes of the event.
 2. Prevents inadvertently opening a Main Steam Safety Valve (MSSV).

Question #99

☐ RO

☒ SRO

Tier # 3

Group # 4

K/A # G2.4.18

Importance Rating: 4.0

K/A Statement: EMERGENCY PROCEDURES/PLAN: Knowledge of the specific bases for emergency and abnormal operating procedures

Proposed Answer: A

Justification: EOP 2534, Steam Generator Tube Rupture, requires isolating the most affected steam generator when both RCS T_H temperatures are less than 515 °F. C OP 200.18, Time Critical Action Validation and Verification, states that this must be done within 1 hour of reactor trip. The EOP Technical guide (basis document) states that the action is necessary to prevent inadvertently opening a MSSV after the SG is isolated.

Plausibility:

- A. Correct.
- B. WRONG; this is the action for an Excess Steam Demand Event (ESDE). Plausible; Examinee may think adding less feedwater will result in less contaminated liquid to be disposed of during cleanup.
- C. WRONG; the basis for this action is to prevent a MSSV from opening. Plausible; candidate may think that lowering the RCS temperature will reduce RCS pressure, limiting amount of water being transferred to the secondary system.
- D. WRONG; the TCOA is to cool down the RCS to less than 515 °F and isolate the affected SG within one (1) hour after the reactor trip. Plausible; candidate may think that securing the AFW is a TCOA for preventing the MSSV from opening, because the SGTR will be gaining inventory from the SGTR and the Auxiliary Feedwater Pumps and could become completely full.

Technical Reference(s):

- MP-PROC-OPS-EOP 2534 REV.030, Steam Generator Tube Rupture.
- C OP 200.18 REV. 015, Time Critical Action Validation And Verification.
- EOP Technical Guides (basis)

Provided reference(s): None

Learning Objective: Outline and explain the bases for the major action steps for EOP 2534, Steam Generator Tube Rupture

Source: New

Cognitive Level: Memory

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Changed basis for B & D to limit contamination levels in the secondary systems since a SGTR would not affect CTMT pressure. DF 8/14

Question #100

The plant is in MODE 1.

- All lit annunciators go dark.
- The PPC Alarm Summary is updating.
- All Main board lights and meters are functioning.
- ANNUNCIATOR POWER SUPPLY TROUBLE (C-08, D-9) is NOT lit.

1. What procedure addresses this situation?

AND

2. What actions are taken in regard to the Emergency Plan?

- A.
 - 1. OP 2387A, Annunciator System Operation and Control
 - 2. No actions required.
- B.
 - 1. OP 2387A, Annunciator System Operation and Control
 - 2. After 15 minutes, an UNUSUAL EVENT is declared.
- C.
 - 1. AOP 2519, Loss of Annunciators
 - 2. No actions required.
- D.
 - 1. AOP 2519, Loss of Annunciators
 - 2. After 15 minutes, an UNUSUAL EVENT is declared.

Question #100

☐ RO

☒ SRO

Tier # 3

Group #

K/A # G2.4.32

Importance Rating: 4.0

K/A Statement: EMERGENCY PROCEDURES/PLAN: Knowledge of operator response to loss of annunciators

Proposed Answer: C

Justification: The given conditions are indicative of a loss of all annunciators. AOP 2519, Loss of Annunciators would be entered. No Emergency Plan actions are taken.

Plausibility:

- A. The examinee might not recognize a loss all annunciators, thinking that ANNUNCIATOR POWER SUPPLY TROUBLE would be lit during a loss of annunciators. OP 2387A does not address losing multiple annunciators.
- B. The examinee might not recognize a loss all annunciators, thinking that ANNUNCIATOR POWER SUPPLY TROUBLE would be lit during a loss of annunciators. OP 2387A does not address losing multiple annunciators. There are no Emergency Plan actions required for a loss of annunciators, the issuance of an ALERT is a legacy action.
- C. Correct.
- D. AOP 2519 would be entered but there are no Emergency Plan actions required for a loss of annunciators, the issuance of an ALERT is a legacy action.

Technical Reference(s):

- AOP 2519, Loss of Annunciators
- OP 2387A, Annunciator System Operation and Control
- FAP06-002, Millstone Power Station Unit 2 EAL Matrix

Provided reference(s): Millstone Power Station Unit 2 Emergency Action Level Matrix

Learning Objective: 280944 - Describe the effects on the annunciator system for a simultaneous loss of both VR-11 and VR-21

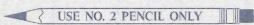
Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Changed initial conditions to MODE 1. The related EAL (which is not met) applies in MODES 1 through 4. Having the plant in MODE 1 makes it possible for the student to look at the related EAL and determine that it's not met, rather than searching and finding no related EAL. SRM 7/26.

STUDENTS



- EXAMPLE: 
- MAKE **DARK** MARKS
- ERASE **COMPLETELY** TO CHANGE
- MAKE NO STRAY MARKS

INSTRUCTORS

KEY MARKING INSTRUCTIONS

KEY: Must mark this box on Key sheet.

VERIFY: Prints correct response next to incorrect answers. If Verify is not marked, a dash will print next to incorrect answers.

RESCORE: Rescores a previously scored test. Automatically prints correct responses.

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SRO KEY

SUBJECT

DATE _____

PERIOD



SCORE

KEY 100

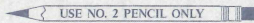
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5	5	A	B	C
6	6	A	B	C
7	7	A	B	C
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9	9	A	B	C
10	10	A	B	C
11	11	B	C	D
12	12	B	C	D
13	13	A	B	C
14	14	A	B	C
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45	45	A	B	C
46	46	A	B	C
47	47	A	B	C
48	48	A	B	C
49	49	A	B	C
50	50	A	B	C

STUDENTS



- EXAMPLE: ☐ A ☐ B ☒ C ☐ D
- MAKE **DARK** MARKS
- ERASE **COMPLETELY** TO CHANGE
- MAKE NO STRAY MARKS

INSTRUCTORS

KEY MARKING INSTRUCTIONS

KEY: Must mark this box on Key sheet.

VERIFY: Prints correct response next to incorrect answers. If Verify is not marked, a dash will print next to incorrect answers.

RESCORE: Rescores a previously scored test. Automatically prints correct responses.

NAME _____

RO KEY

SUBJECT

DATE _____

PERIOD



SCORE

KEY 75

RESCORE

KEY  VERIFY  RESCORE 

[illegible]

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References Provided to Applicants Millstone 2 2024 Exam

Question 33: Steam Tables w/ Mollier diagram

Question 73: Mollier Diagram and Steam Tables

Question 74: Steam Tables w/ Mollier diagram

Question 95: TRM 3.1.2.8, Borated Water Sources – Operating, TRM Figure 3.1-1, Minimum Volume vs Stored BAST concentration.

Question 100: Millstone Power Station Unit 2 Emergency Action Level Matrix

Attachment 3
Questions raised during the exam.

Time	Q#	Who	Question w/ response
0846	Q5	SR	Q: Is this a pump start (2 pumps running) or a pump swap? OP 2304E has directions for both. A: The question contains all the information required. Answer the question to the best of your ability.
0854	Q8	CB	Q: Am I to assume the Main Turbine is tripped or will trip? A: The question contains all the information required. Answer the question to the best of your ability.
0837	Q31	NP	Q: This question appears to have two correct answers. A: Answer the question to the best of your ability.
0840	Q39	CR	Q: Have operator actions been taken? S/G levels are stable? A: Answer the question based on the conditions given.
0841	Q7	JF	Q: Am I to combine the answers with the alarms given in the stem or do they stand independently? A: Evaluate the answers independently based on the given conditions.
0950	Q65	TH	The hidden text ““≥1720 ppm’ with ‘≥ refueling’ concentration based on validator feedback. DF 9/6” is showing. At this point, the examinees were told to stop the exam. Q65 was removed from the exam, and was replaced with a new Q65 with no hidden text showing. The exam was resumed. The chief examiner was informed of the issue and actions taken. The examinees were polled individually upon their exit from the exam room and also as a group during the exam review. All three ROs indicated that they saw the hidden text, no SROs saw the hidden text.
1005	Q36	SR	Q: Where in the plant start-up are we? Are we synched to the grid? A: The question contains all the information required. Answer the question to the best of your ability.
1008	Q4	AW	Q: Am I to assume there were no operator actions? The RMs 8123 & 8262 are normally isolated. How can they be in alarm? A: Review your copy of NUREG 1021 ES-1.2 B.8 regarding making assumptions.
1107	Q65	AW	Q: Does ‘boron concentration’ in answer B mean RCS boron A: The question contains all the information required. Answer the question to the best of your ability.
1123	Q78	JF	Q: Is #2 S/G the affected S/G? A: yes Note that the question’s stem stated a SGTR had occurred and gave the #2 S/G pressure, not the #1 S/G pressure.
1136	Q9	KB	Q: OP 2207 states to ‘REFER TO’, the answer has ‘GO TO’. Were the words in the answers written purposely? A: Given the stated given conditions, answer the question to the best of your ability.
1144	Q87	SR	Q: Was there some sort of failure between the 4 bullets and “subsequently”. A: The question contains all the information required. Answer the question to the best of your ability.

Attachment 3
Questions raised during the exam.

1150	Q8	JL	Q: Is the Main Turbine tripped? A: The question contains all the information required. Answer the question to the best of your ability.
1157	Q95	NP	Q: Is the question asking about the contents of the BASTs or the delivery mechanism? A: The question contains all the information required. Answer the question to the best of your ability.
1220	Q49	CR	Q: Is B-52 a typo? A: The question contains all the information required. Answer the question to the best of your ability.
1225	Q96	CB	Q: Does “conflict with Tech Specs” mean “violate” A: The question contains all the information required. Answer the question to the best of your ability.
1227	Q8	NP	Q: What does “WILL MITIGATE” mean A: A dictionary is available to define words.
1230	Q8	SR	Q: What does “WILL MITIGATE” mean A: A dictionary is available to define words.
1241	Q87	JF	Q: Is the Low VCT level alarm real? Normally, we would validate the alarm with a level. A: The question contains all the information required. Answer the question to the best of your ability.
1324	Q97	NP	Q: Does “it has been identified” mean “just now”? A: The question contains all the information required. Answer the question to the best of your ability.
1400	Q96	JL	Q: What does “conflicts with” mean? This was the question raised by CB at 1225. At this point a pen-and-ink change request was made to the chief examiner to replace “conflicts with” with “violates”. The change request was approved. The examinees were notified of the change and a note describing the change was written on the room’s white board.
1417	Q82	CB	Q: What is the sequence of events in the bulletized items? A: A bulletized list indicates all the items happen at the same time.
1423	Q90	SR	Q: I’m having issues understanding what ‘met’ means. A: The question contains all the information required. Answer the question to the best of your ability.
1425	Q15	JL	Q: Can I assume the ‘A’ EDG picked up bus 24C? A: Review your copy of NUREG 1021 ES-1.2 B.8 regarding making assumptions.
1440	Q96	CB	Q: Assuming the activity is a test, and the test is not in the FSAR, but does the activity concerns components contained within the FSAR since Tech Specs are involved? A: The question contains all the information required. Answer the question to the best of your ability.
1503	Q87	SR	Q: I’m still having issues with “subsequently”. A: The question contains all the information required. Answer the question to the best of your ability.

Attachment 3
Questions raised during the exam.

1606	Q9	KB	Q: OP 2207 Conditional Actions states to 'REFER TO', the answer has 'GO TO'. Were the words in the answers written purposely? A: Given the stated given conditions, answer the question to the best of your ability.
1613	Q15	KB	Q: When you ask if the discharge rate is higher, do you mean even slightly, because the 24C loses power momentarily. A: Given the stated given conditions, answer the question to the best of your ability.
1637	Q79	CR	Q: is this under normal Control Room staffing? A: Review your copy of NUREG 1021 ES-1.2 B.8 regarding making assumptions.
1645	Q96	CR	Q: Is this a necessary activity? A: The question contains all the information required. Answer the question to the best of your ability.
1653	Q59	KB	Q: Normally SDC would be the first choice to cool the SFP but that is omitted as an answer, is there a reason for that A: Given the stated given conditions, answer the question to the best of your ability.

Attachment 4

Questions comment, corrections or enhancements that will be made before reuse:

Question	Comment
8	Correction: Will add information to the stem to clarify that RPS trips will not function to make answer "B" unambiguously wrong. Justification submitted in a separate document for accepting both "A" and "B" as correct.
16	Enhancement: OP 2326C, Off-Normal Service Water System Operations, is an infrequently used procedure and a minimally competent RO candidate may not be aware that it only applies in MODES 1 through 4. Consider using a different distractor in future versions.
22	Enhancement: Include the words "CORE ALTERATIONS" in the correct answer "C."
25	Enhancement: SSP 16.1/16.2 closing on the opposite facility CIAS from what is expected from their component numbering confused several students and the distinction is arguably minutia. Consider using more straightforward components in future versions of this question.
30	Correction: Detailed knowledge of the refueling requirements of OP 2209A exceeds the level of knowledge expected for a minimally competent RO candidate and the question should be marked as SRO for future use, based on NUREG 1021 Figure 4.2-3, question requires "assessment of plant conditions (normal , abnormal, or emergency) and then prescribing a procedure or section of a procedure to mitigate or recover or with which to proceed ."
36	Enhancement: Make power 12.5% in the stem of the question for better operational validity (the turbine is normally started up with a power band of 12%-13%). Consider mentioning in the stem that the generator is not online.
38	Enhancement: Consider writing the second part to be specific to components with NO lights lit on C01X, because if a white light is lit, it would lead operators to believe they would not be successful realigning the component from the Control Room.
45	Taking manual control of the affected SGFP is an immediate operator action for the abnormal SGWL which would result from the rising SGFP speed. The stem gives no indication that taking manual control would NOT be successful. Justification submitted in a separate document for changing the key such that "A" is correct. Additionally, several students were steered away from "C" due to the mention of CNM-2.
58	Enhancement: Clarify lineup in stem such that EB purge fan F-23 is NOT expected to trip on high EB D/P.
80	Enhancement: Clarify for design function of 201D that it's Tech Spec design (as opposed to overall design).
86	Enhancement: Consider reversing part 1 and part 2, since the reactor would be tripped prior to tripping the affected RCP.

Attachment 6:

Performance Analysis (Questions missed by > 50% of candidates)

Question	Comment
9	7/11 Miss; Question was on the actions to take if a tube rupture occurs during plant cooldown, specifically on SDC at 190°F RCS temperature and 150 psia RCS pressure. Incorrect answers ranged across all of the distractors, the knowledge gap was closed during the exam review.
29	7/11 Miss; Question was on the maximum allowed temperature for SDC and the maximum RCS to SDC temperature difference allowed to initiate flow. 6 of the 7 misses involved the student selecting 265°F as the maximum temperature (correct temperature is 300°F), from unit confusion (because the maximum <i>pressure</i> for SDC is 265 psia). The remaining student selected the answer with the correct temperature, but with the differential temperature of 48°F (correct one is 20°F). Knowledge gap was closed during exam review.
45	8/11 Miss; Question was on the required actions when the 'A' SGFP Turbine Trip annunciator is lit, speed is 5550 RPM (above the electrical overspeed trip setpoint of 5400 RPM) and the pump is not tripped. 4 students selected tripping the pump and the reactor, this is not correct because only one SGFP was required in the stated conditions (55% power) and the 'B' pump remained running. This knowledge gap was closed during exam review. 4 students selected placing 'A' SGFP in Manual and ensuring FRV D/P between 40 and 150 psig. Documentation is being submitted to change the key so that this answer is the correct one, as the original proposed correct answer involved tripping the 'A' SGFP and opening CNM-2, and review of the procedures including the ARP, as written, have no path to those actions (when "IF" and "GO TO" statements are factored in).
58	6/11 Miss; Question was on what would happen to Containment Purge Valves and EB Purge Fan F-23 if purge is in progress with no temp mods installed (i.e. CPVIS not defeated) and RM-8262B CTMT Gaseous Rad Monitor alarms. All 6 students who missed it picked distractors that said that F-23 would trip, and 5 out of 6 of those misses picked the one where the first part correctly stated that the valves would close (i.e. only one student picked an answer where the purge valves did not close), that knowledge gap was closed at the exam review. Students who missed this question believed that F-23 would trip on Enclosure Building D/P due to the perturbation from the purge valves going closed. Future versions of this question will have clarification of the lineup such that the EB is not expected to build up sufficient D/P to trip F-23.
59	6/11 Miss; Question was about the preferred method to cool the SFP in MODE 5 with the RWST unavailable as a source of cooling and the only available SFP cooling pump trips on overload. The answers were split between makeup to the SFP when level lowers from evaporation (wrong, because replacing evaporated water is not expected to be sufficient cooling and not supported by the AOP) and aligning Containment Spray to cool the SFP (wrong because this option is not allowed in MODE 5). Knowledge gap closed during exam review.
90	4/8 Miss; Question was about the effect of a loss of Non-Vital panel VR-21 on the ability to meet RCS leakage spec TSAS 3.4.6.1. All misses chose answers that the TSAS is met, split between having no LCO required equipment affected and having one of two CTMT rad monitors affected (CTMT particulate and gaseous rad monitors are vitally powered). Knowledge gaps closed during exam review.