Form 4.1-PWR Pressurized-Water Reactor Examination Outline

Facility: Calvert Cliffs **K/A Catalog Rev. 3** **Rev.** DRAFT **Date of Exam:** 6/5/2023

Tier	Group	RO K/A Category Points: K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	SRO-Only Points: A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1	3	3	3				3	3			3	18	3	3	6
	2	2	1	1				2	1			1	8	2	2	4
	Tier Totals	5	4	4				5	4			4	26	5	5	10
2. Plant Systems	1	2	2	3	3	2	2	3	3	3	2	3	28	3	2	5
	2	1	1	1	1	1	1	1	1	0	1	0	9	0 2	1	3
	Tier Totals	3	3	4	4	3	3	4	4	3	3	3	37	5	3	8

Tier	CO	EC	RC	EM	Total	SRO CO	SRO EC	SRO RC	SRO EM	SRO Total
3. Generic Knowledge and Abilities Categories	2	2	1	1	6	2	2	1	2	7

Tier	Reactor Theory	Thermodynamics	Total
4. Theory	3	3	6

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 (Calvert Cliffs)
	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	(000007) (EPE 7; BW E02 & E10; CE E02) Reactor Trip, Stabilization, Recovery					X		(CE02EA2.06) Ability to determine and/or interpret the following as they apply to (CE E02) STANDARD POST-TRIP ACTIONS AND REACTOR TRIP RECOVERY (CFR: 41.10 / 43.5 / 45.13): PZR level and pressure	3.8	48
2	(000008) (APE 8) Pressurizer Vapor Space Accident		X					(000008AK2.11) Knowledge of the relationship between (APE 8) PRESSURIZER VAPOR Space Accident and the following systems or components (CFR: 41.8 / 41.10 / 45.3): PZR safeties	4.0	7
3	(000015) (APE 15) Reactor Coolant Pump Malfunctions	X						(000015AK1.02) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 15) REACTOR COOLANT Pump Malfunctions (CFR: 41.5 / 41.7 / 45.7 / 45.8): Consequences of an RCP failure	3.9	17
4	(000022) (APE 22) Loss of Reactor Coolant Makeup				X			(000022AA1.01) Ability to operate and/or monitor the following as they apply to (APE 22) LOSS OF REACTOR Coolant Makeup (CFR: 41.7 / 45.5 / 45.6): CVCS	3.8	3
5	(000025) (APE 25) Loss of Residual Heat Removal System						X	(000025) (APE 25) Loss of Residual Heat Removal System (G2.1.20) CONDUCT OF OPERATIONS: Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12)	4.6	59
6	(000026) (APE 26) Loss of Component Cooling Water	X						(000026AK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 26) LOSS OF Component Cooling Water (CFR: 41.5 / 41.7 / 45.7 / 45.8): Leakage into or out of the CCWS	3.6	57
7	(000027) (APE 27) Pressurizer Pressure Control System Malfunction				X			(000027AA1.06) Ability to operate and/or monitor the following as they apply to (APE 27) PRESSURIZER PRESSURE Control System Malfunction (CFR: 41.5 / 41.7 / 45.5 to 45.8): Operable control channel	3.6	49
8	(000029) (EPE 29) Anticipated Transient Without Scram					X		(000029EA2.12) Ability to determine and/or interpret the following as they apply to (EPE 29) ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) (CFR: 41.10 / 43.5 / 45.13): AFW flow	3.5	45
9	(000038) (EPE 38) Steam Generator Tube Rupture		X					(000038EK2.13) Knowledge of the relationship between (EPE 38) STEAM GENERATOR Tube Rupture and the following systems or components (CFR: 41.8 / 41.10 / 45.3): Main steam system	3.6	65
10	(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer			X				(CE05EK3.11) Knowledge of the reasons for the following responses and/or actions as they apply to (CE E05) EXCESS STEAM DEMAND (CFR: 41.5 / 41.10 / 45.6 / 45.13): Restoring S/G levels	3.3	72
11	(000054) (APE 54; CE E06) Loss of Main Feedwater					X		(000054AA2.08) Ability to determine and/or interpret the following as they apply to (APE 54) LOSS OF Main Feedwater (CFR: 41.10 / 43.5 / 45.13): Steam flow and/or MFW flow	3.4	22
12	(000055) (EPE 55) Station Blackout		X					(000055EK2.01) Knowledge of the relationship between (EPE 55) Station Blackout and the following systems or components (CFR: 41.8 / 41.10 / 45.3): Letdown isolation, RCP seal return, PZR PORVs, or secondary PORVs (atmospheric relief valves)	3.9	14

13	(000056) (APE 56) Loss of Offsite Power						X	(000056) (APE 56) Loss of Offsite Power (G2.1.45) CONDUCT OF OPERATIONS: Ability to identify and interpret diverse indications to validate the response of another indication. (CFR: 41.7 / 43.5 / 45.4)	4.3	58
14	(000057) (APE 57) Loss of Vital AC Instrument Bus				X			(000057AA1.02) Ability to operate and/or monitor the following as they apply to (APE 57) LOSS OF VITAL AC ELECTRICALINSTRUMENT BUS (CFR: 41.5 / 41.7 / 45.5 to 45.8): Manual control of PZR level	3.8	8
15	(000058) (APE 58) Loss of DC Power			X				(000058AK3.01) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 58) LOSS OF DC Power (CFR: 41.5 / 41.10 / 45.6 / 45.13): Operation of the EDGs	4.0	68
16	(000062) (APE 62) Loss of Nuclear Service Water			X				(000062AK3.01) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 62) LOSS OF SERVICE WATER (CFR: 41.5 / 41.10 / 45.6 / 45.13): The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the service water coolers	3.5	51
17	(000065) (APE 65) Loss of Instrument Air						X	(000065) (APE 65) Loss of Instrument Air (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)	3.8	69
18	(000077) (APE 77) Generator Voltage and Electric Grid Disturbances	X						(000077AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 77) GENERATOR VOLTAGE AND ELECTRIC Grid Disturbances (CFR: 41.5 / 41.7 / 45.7 / 45.8): Underexcitation	3.3	24
19	**(000009) (EPE 9) Small Break LOCA**						**X**	**(000009) (EPE 9) Small Break LOCA (G2.4.4) EMERGENCY PROCEDURES/PLAN: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures (CFR: 41.10 / 43.2 / 45.6)**	**4.5**	**94**
20	**(000011) (EPE 11) Large Break LOCA**					**X**		**(000011EA2.05) Ability to determine and/or interpret the following as they apply to (EPE 11) LARGE-Break LOCA (CFR: 41.10 / 43.5 / 45.13): Significance of ECCS pump operation**	**4.1**	**82**
21	**(000027) (APE 27) Pressurizer Pressure Control System Malfunction**						**X**	**(000027) (APE 27) Pressurizer Pressure Control System Malfunction (G2.4.46) EMERGENCY PROCEDURES/PLAN: Ability to verify that the alarms are consistent with the plant conditions (CFR: 41.10 / 43.5 / 45.3 / 45.12)**	**4.2**	**84**
22	**(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer**						**X**	**(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer (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)**	**4.0**	**93**
23	**(000055) (EPE 55) Station Blackout**					**X**		**(000055EA2.04) Ability to determine and/or interpret the following as they apply to (EPE 55) Station Blackout (CFR: 41.10 / 43.5 / 45.13): Instruments and controls operable with only DC battery power available**	**3.9**	**92**

24	**(000065) (APE 65) Loss of Instrument Air**					**X**		**(000065AA2.02) Ability to determine and/or interpret the following as they apply to (APE 65) LOSS OF Instrument Air (CFR: 41.10 / 43.5 / 45.13): Airflow readings**	**2.7**	**83**
	(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

ES-4.1-PWR PWR Examination Outline (Calvert Cliffs)

Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/**SRO**)

Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
25	(000028) (APE 28) Pressurizer (PZR) Level Control Malfunction				X			(000028AA1.09) Ability to operate and/or monitor the following as they apply to (APE 28) PRESSURIZER (PZR) Level Control Malfunction (CFR: 41.5 / 41.7 / 45.5 to 45.8): Auto/manual control of PZR level	3.7	53
26	(000033) (APE 33) Loss of Intermediate Range Nuclear Instrumentation	X						(000033AK1.02) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 33) LOSS OF INTERMEDIATE RANGE Nuclear Instrumentation (CFR: 41.5 / 41.7 / 45.7 / 45.8): Equivalency and/or overlap among source range, intermediate range, and power range channel readings	3.5	12
27	(000036) (APE 36; BW/A08) Fuel-Handling Incidents			X				(000036AK3.04) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 36) FUEL HANDLING INCIDENTS (CFR: 41.5 / 41.10 / 45.6 / 45.13): Establishing containment isolation or closure	3.9	38
28	(000068) (APE 68; BW A06) Control Room Evacuation		X					(000068AK2.11) Knowledge of the relationship between (APE 68) Control Room Evacuation and the following systems or components (CFR: 41.8 / 41.10 / 45.3): AFW	4.0	50
29	(000076) (APE 76) High Reactor Coolant Activity				X			(000076AA1.05) Ability to operate and/or monitor the following as they apply to (APE 76) HIGH REACTOR COOLANT ACTIVITY (CFR: 41.5 / 41.7 / 45.5 to 45.8): PRM	3.2	4
30	(CE A16) Excess RCS Leakage					X		(CA16AA2.04) Ability to determine and/or interpret the following as they apply to (CE A16) EXCESS RCS Leakage (CFR: 41.10 / 43.5 / 45.13): PZR level and pressure	3.3	31
31	(CE E09) Functional Recovery	X						(CE09EK1.07) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (CE E09) FUNCTIONAL RECOVERY (CFR: 41.5 / 41.7 / 45.7 / 45.8): Evaluating the RCS pressure control safety function and implementing the correct success path for plant conditions	4.3	44
32	(CE E13*) Loss of Forced Circulation/LOOP/Blackout						X	(CE E13*) Loss of Forced Circulation/LOOP/Blackout (G2.2.36) EQUIPMENT CONTROL: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation (CFR: 41.10 / 43.2 / 45.13)	3.1	19
33	**(000005) (APE 5) Inoperable/Stuck Control Rod**						**X**	**(000005) (APE 5) Inoperable/Stuck Control Rod (G2.2.23) EQUIPMENT CONTROL: Ability to track TS limiting conditions for operation (CFR: 41.10 / 43.2 / 45.13)**	**4.6**	**78**
34	**(000067) (APE 67) Plant Fire On Site**					**X**		**(000067AA2.17) Ability to determine and/or interpret the following as they apply to (APE 67) PLANT Fire On Site (CFR: 41.10 / 43.5 / 45.13): Systems that may be affected by the fire**	**3.5**	**76**
35	**(000069) (APE 69; W E14) Loss of Containment Integrity**					**X**		**(000069AA2.03) Ability to determine and/or interpret the following as they apply to (APE 69) LOSS OF Containment Integrity (CFR: 41.10 / 43.5 / 45.13): Containment pressure**	**3.8**	**80**
36	**(000074) (EPE 74; W E06 & E07) Inadequate Core Cooling**						**X**	**(000074) (EPE 74; W E06 & E07) Inadequate Core Cooling (G2.4.40) EMERGENCY PROCEDURES/PLAN: Knowledge of SRO responsibilities in emergency plan implementing procedures (SRO Only) (CFR: 43.5 / 45.11)**	**4.5**	**99**

	000001 (APE 1) Continuous Rod Withdrawal / 1										
	000003 (APE 3) Dropped Control Rod / 1										
	000024 (APE 24) Emergency Boration / 1										
	000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7										
	000037 (APE 37) Steam Generator Tube Leak / 3										
	000051 (APE 51) Loss of Condenser Vacuum / 4										
	000059 (APE 59) Accidental Liquid Radwaste Release / 9										
	000060 (APE 60) Accidental Gaseous Radwaste Release / 9										
	000061 (APE 61) Area Radiation Monitoring System Alarms / 7										
	000078 (APE 78*) RCS Leak / 3										
	(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									
K/A Category Totals:		2	1	1	2	3	3	Group Point Total:		12

ES-4.1-PWR	PWR Examination Outline (Calvert Cliffs) 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									(003K3.02) Knowledge of the effect that a loss or malfunction of the (SF4P RCP) REACTOR COOLANT PUMP SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): S/G	3.9	67
38	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM		X										(004K2.02) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM Pumps used to makeup to CVCS	3.0	26
39	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM			X									(005K3.05) Knowledge of the effect that a loss or malfunction of the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): ECCS	4.3	2
40	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM				X								(005K4.11) Knowledge of (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Lineup for low head recirculation mode (external and internal)	4.0	28
41	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM									X			(006A3.02) Ability to monitor automatic features of the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM, including (CFR: 41.7 / 45.7): Pumps	4.2	52
42	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM			X									(006K3.03) Knowledge of the effect that a loss or malfunction of the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): Containment Spray System	3.8	46
43	(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM									X			(007A3.01) Ability to monitor automatic features of the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM, including (CFR: 41.7 / 45.7): Components that discharge to the PRT/quench tank	3.4	21

44	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM									X			(008A3.01) Ability to monitor automatic features of the (SF8 CCW) COMPONENT COOLING WATER SYSTEM, including (CFR: 41.7 / 45.7): Setpoints for normal operations, warnings, and trips that are applicable to the CCWS	3.6	18
45	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM											X	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM (G2.2.18) EQUIPMENT CONTROL: Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments and work prioritization (CFR: 41.10 / 43.5 / 45.13)	2.6	41
46	(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM						X						(010K6.13) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM (CFR: 41.7 / 45.7): ESFAS	4.0	43
47	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM		X										(012K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF7 RPS) REACTOR PROTECTION SYSTEM RPS channels, components, and interconnections	4.0	75
48	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM					X							(012K5.01) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF7 RPS) REACTOR PROTECTION SYSTEM (CFR: 41.5 / 45.3): Departure from nucleate boiling	3.9	20
49	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM					X							(013K5.23) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (CFR: 41.5 / 45.3): Inadequate core cooling	4.2	25
50	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM											X	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM (191002K1.14) SENSORS AND DETECTORS (CFR: 41.7): (TEMPERATURE) Failure modes of thermocouple, RTD, and/or thermometers	2.9	30
51	(026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM						X						(026K6.08) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF5 CSS) CONTAINMENT SPRAY SYSTEM (CFR: 41.7 / 45.7): RHRS	3.5	35

52	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM							X					(039A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM, including (CFR: 41.5 / 45.5): RCS T-ave	3.9	60
53	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM								X				(039A2.05) Ability to (a) predict the impacts of the following on the (SF4S MSS) MAIN AND REHEAT STEAM 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): Increasing steam demand and its relationship to increases in reactor power	4.5	47
54	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM							X					(059A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4S MFW) MAIN FEEDWATER SYSTEM, including (CFR: 41.5 / 45.5): S/G level and comparison with normal values	3.5	39
55	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM											X	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM (G2.2.2) EQUIPMENT CONTROL: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels (CFR: 41.6 / 41.7 / 45.2)	4.6	16
56	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM							X					(061A1.02) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM, including (CFR: 41.5 / 45.5): S/G Pressure	3.8	23
57	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM								X				(062A2.20) Ability to (a) predict the impacts of the following on the (SF6 ED AC) AC ELECTRICAL DISTRIBUTION 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): Loss of offsite power sources	3.8	62
58	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM				X								(062K4.11) Knowledge of (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Load shedding	3.7	29

59	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM									X			(063A4.03) Ability to manually operate and/or monitor the (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Battery discharge rate	3.5	36
60	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM								X				(064A2.07) Ability to (a) predict the impacts of the following on the (SF6 EDG) EMERGENCY DIESEL GENERATOR 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): Overexcitation or underexcitation	3.2	42
61	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM	X											(073K1.08) Knowledge of the physical connections and/or cause and effect relationships between the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): SWS	3.1	32
62	(076) (SF4S SW) SERVICE WATER SYSTEM										X		(076A4.01) Ability to manually operate and/or monitor the (SF4S SW) SERVICE WATER SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): SWS pumps	3.9	63
63	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM	X											(078K1.03) Knowledge of the physical connections and/or cause and effect relationships between the (SF8 IAS) INSTRUMENT AIR SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): Containment air	3.4	56
64	(103) (SF5 CNT) CONTAINMENT SYSTEM			X									(103K4.04) Knowledge of (SF5 CNT) CONTAINMENT SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Personnel access hatch and emergency access hatch	3.3	10
65	**(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM**											**X**	**(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM (G2.4.18) EMERGENCY PROCEDURES/PLAN: Knowledge of the specific bases for emergency and abnormal operating procedures (CFR: 41.10 / 43.1 / 45.13)**	**4.0**	**95**

66	**(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM**								**X**				**(010A2.08) Ability to (a) predict the impacts of the following on the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL 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): Safety valves failure to reseat**	**3.9**	**85**
67	**(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 / 45.6): Cooling water system malfunction**	**3.4**	**97**
68	**(076) (SF4S SW) SERVICE WATER SYSTEM**								**X**				**(076A2.07) Ability to (a) predict the impacts of the following on the (SF4S SW) SERVICE WATER 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): Heat exchanger and condenser failure**	**3.2**	**88**
69	**(078) (SF8 IAS) INSTRUMENT AIR SYSTEM**											**X**	**(078) (SF8 IAS) INSTRUMENT AIR SYSTEM (G2.2.5) CONDUCT OF OPERATIONS: 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)**	**3.2**	**79**
	025 (SF5 ICE) ICE CONDENSER SYSTEM														
	053 (SF1; SF4P ICS*) INTEGRATED CONTROL SYSTEM														
K/A Category Totals:		2	2	3	3	2	2	3	6	3	2	5	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	(001) (SF1 CRDS) CONTROL ROD DRIVE SYSTEM					X							(001K5.18) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF1 CRDS) CONTROL ROD DRIVE SYSTEM (CFR: 41.5 / 45.3): Anticipation of criticality at any time when adding positive reactivity during startup	4.3	15
71	(011) (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM		X										(011K2.02) Knowledge of electrical power supplies to the following (CFR: 41.7): PZR heaters	3.3	37
72	(016) (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM			X									(016K3.03) Knowledge of the effect that a loss or malfunction of the (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): Steam Dump System	3.2	9
73	(017) (SF7 ITM) IN CORE TEMPERATURE MONITOR SYSTEM								X				(017A2.02) Ability to (a) predict the impacts of the following on the (SF7 ITM) IN CORE TEMPERATURE MONITOR 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): Elevated in-core temperatures that can cause or have caused core damage	4.1	27
74	(033) (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM						X						(033K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM (CFR: 41.7 / 45.7): SFPCS pumps	3.6	6
75	(035) (SF4P SG) STEAM GENERATOR SYSTEM							X					(035A1.03) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4P SG) STEAM GENERATOR SYSTEM, including (CFR: 41.5 / 45.5): Feed flow/steam flow	4.0	70
76	(050) (SF9 CRV*) CONTROL ROOM VENTILATION										X		(050A4.03) Ability to manually operate and/or monitor the (SF9 CRV) CONTROL ROOM VENTILATION in the control room (CFR: 41.7 / 45.5 to 45.8): Dampers	3.0	74

77	(055) (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM	X											(055K1.06) Knowledge of the physical connections and/or cause and effect relationships between the (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): PRMS	3.0	66
78	(071) (SF9 WGS) WASTE GAS DISPOSAL SYSTEM				X								(071K4.04) Knowledge of (SF9 WGS) WASTE GAS DISPOSAL SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Isolation of waste gas release tanks	3.1	34
79	**(027) (SF5 CIRS) CONTAINMENT IODINE REMOVAL SYSTEM**								**X**				**(027A2.01) Ability to (a) predict the impacts of the following on the (SF5 CIRS) CONTAINMENT IODINE REMOVAL 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): High temperature in the filter system**	**2.7**	**91**
80	**(075) (SF8 CW) CIRCULATING WATER SYSTEM**											**X**	**(075) (SF8 CW) CIRCULATING WATER SYSTEM (G2.4.47) EMERGENCY PROCEDURES/PLAN: Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential) (CFR: 41.10 / 43.5 / 45.12)**	**4.2**	**89**
81	**(086) (SF8 FPS) FIRE PROTECTION SYSTEM**								**X**				**(086A2.05) Ability to (a) predict the impacts of the following on the (SF8 FPS) FIRE PROTECTION 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): Fire in the plant**	**3.8**	**86**
	002 (SF2; SF4P RCS) REACTOR COOLANT SYSTEM														
	014 (SF1 RPI) ROD POSITION INDICATION SYSTEM														
	015 (SF7 NI) NUCLEAR INSTRUMENTATION SYSTEM														

	028 (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM														
	029 (SF8 CPS) CONTAINMENT PURGE SYSTEM														
	034 (SF8 FHS) FUEL HANDLING EQUIPMENT SYSTEM														
	041 (SF4S SDS) STEAM DUMP / TURBINE BYPASS CONTROL SYSTEM														
	045 (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM														
	056 (SF4S CDS) CONDENSATE SYSTEM														
	068 (SF9 LRS) LIQUID RADWASTE SYSTEM														
	072 (SF7 ARM) AREA RADIATION MONITORING SYSTEM														
	079 (SF8 SAS**) STATION AIR SYSTEM														
K/A Category Totals:		1	1	1	1	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 (Calvert Cliffs)		
Facility:	Calvert Cliffs	Date of Exam:	6/5/2023

Generic Knowledge and Abilities Outline (Tier 3) (RO/SRO)

Category	K/A #	Topic	Item #	RO IR	RO Q#	SRO-Only IR	SRO-Only Q#
1. Conduct of Operations	G2.1.19	(G2.1.19) CONDUCT OF OPERATIONS: Ability to use available indications to evaluate system or component status (CFR: 41.10 / 45.12)	82	3.9	64		
	G2.1.28	(G2.1.28) CONDUCT OF OPERATIONS: Knowledge of the purpose and function of major system components and controls (CFR: 41.7)	83	4.1	5		
	G2.1.42	**(G2.1.42) CONDUCT OF OPERATIONS: Knowledge of new and spent fuel movement procedures (SRO Only) (CFR: 43.7 / 45.13)**	**84**			**3.4**	**81**
	G2.1.9	**(G2.1.9) CONDUCT OF OPERATIONS: Ability to direct licensed personnel activities inside the control room (SRO Only) (CFR: 43.1 / 45.5 / 45.12 / 45.13)**	**85**			**4.5**	**90**
	Subtotal			N/A	2	N/A	**2**
2. Equipment Control	G2.2.13	(G2.2.13) EQUIPMENT CONTROL: Knowledge of tagging and clearance procedures (CFR: 41.10 / 43.1 / 45.13)	86	4.1	54		
	G2.2.14	(G2.2.14) EQUIPMENT CONTROL: Knowledge of the process for controlling equipment configuration or status (CFR: 41.10 / 43.3 / 45.13)	87	3.9	40		
	G2.2.37	**(G2.2.37) EQUIPMENT CONTROL: Ability to determine operability or availability of safety-related equipment (SRO Only) (CFR: 43.2 / 43.5 / 45.12)**	**88**			**4.6**	**96**
	G2.2.38	**(G2.2.38) EQUIPMENT CONTROL: Knowledge of conditions and limitations in the facility license (CFR: 41.7 / 41.10 / 43.1 / 45.13)**	**89**			**4.5**	**98**
	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	55		
	G2.3.6	**(G2.3.6) RADIATION CONTROL: Ability to approve liquid or gaseous release permits (CFR: 41.13 / 43.4 / 45.10)**	**91**			**3.8**	**100**
	Subtotal			N/A	1	N/A	**1**
4. Emergency Procedures / Plan	G2.4.34	(G2.4.34) EMERGENCY PROCEDURES/PLAN: Knowledge of RO responsibilities outside the main control room during an emergency (CFR: 41.10 / 43.5 / 45.13)	92	4.2	33		
	G2.4.23	**(G2.4.23) EMERGENCY PROCEDURES/PLAN: Knowledge of the bases for prioritizing emergency operating procedures Implementation (CFR: 41.10 / 43.5 / 45.13)**	**93**			**4.4**	**87**
	G2.4.21	**(G2.4.21) EMERGENCY PROCEDURES/PLAN: Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions (CFR: 41.7 / 43.5 / 45.12)**	**94**			**4.6**	**77**

	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 (Calvert Cliffs)		
Facility:	Calvert Cliffs	Date of Exam:	6/5/2023

Theory (Tier 4) (RO)

Category	K/A #	Topic	Item #	RO IR	RO Q#
Reactor Theory	192006	(192006K1.10) FISSION PRODUCT POISONS (CFR: 41.1): Plot the curve and explain the reasoning for the reactivity insertion by xenon-135 versus time for the following: -- reactor startup with xenon-135 already present in the core	95	3.2	13
	192007	(192007K1.04) FUEL DEPLETION AND BURNABLE POISONS (CFR: 41.1): Describe how and why boron concentration changes over core life	96	3.4	23
	192008	(192008K1.14) REACTOR OPERATIONAL PHYSICS (CFR: 41.1): (INTERMEDIATE RANGE OPERATION) Describe reactor power and startup rate response prior to reaching the POAH	97	3.1	74
	Subtotal			N/A	3
Thermodynamics	193003	(193003K1.08) STEAM (CFR: 41.14): Define the following term: -- saturated liquid	98	2.8	66
	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	9
	193009	(193009K1.07) CORE THERMAL LIMITS (CFR: 41.14): Describe factors that affect peaking and hot channel factors	100	3.3	27
	Subtotal			N/A	3
Tier 4 Point Total				**N/A**	**6**

Exelon Generation

Exam Material

VISION Report

(Test Key) 2023 ILT NRC RO EXAM

CCNPP Operations NRC Examinations

December 15, 2022

Test	2023 ILT NRC RO EXAM
VISION ID	367277
Status	

EXAMINATION COVER SHEET

Exam Title (ID)	2023 ILT NRC RO EXAM (367277)				
Training Program	CCNPP Operations NRC Examinations				
LMS Component ID		**Total Points**	75.00	**Pass Criteria** = 80 %	
Trainee Name		**Employee ID**			
Graded By / Date		**Grade**	____ / 75.00 = _______ %		

Review and Approval			
Instructor		**Date**	
Technical Review		**Date**	
Training Supv		**Date**	

Examination Rules

1. References may NOT be used during this exam, unless otherwise stated.
2. Read each question carefully before answering. If you have any questions or need clarification during the exam, contact the exam proctor.
3. Conversation with other trainees during the exam is prohibited.
4. Partial credit will NOT be considered, unless otherwise stated. Show **all** work and state **all** assumptions when partial credit may be given.
5. Restroom trips are limited and only one examinee at a time may leave.
6. For exams with time limits, you have ____ minutes to complete the exam.
7. The examinee agrees to refrain from discussing the content of the exam until the end of the exam cycle.

Examination Integrity Statement

Cheating or compromising the exam will result in disciplinary actions up to and including termination.

"I acknowledge that I am aware of the Exam Rules stated above. Further, I have not given, received, or observed any aid or information regarding this exam prior to or during its administration that could compromise this exam."

Examinee Signature ______________________________________ Date ____________

Review Acknowledgement

"I acknowledge that the correct answers to the exam questions were indicated to me following the completion of the exam. I have had the opportunity to review the exam questions with the instructor to ensure my understanding."

Examinee Signature ______________________________________ Date ____________

Question 1	ID: 2479399	Points: 1.00

Given a Unit-1 reactor startup in progress:

- Reactor was shut down seven days ago.
- MOL.
- Operators have established a positive 0.3 DPM SUR in the intermediate range after collecting critical rod height data.

With the above conditions, Unit-1:

A. reactor power will continue to RISE at 0.3 DPM until the POAH is reached.

B. reactor power will continue to RISE at 0.3 DPM until a reactor trip occurs.

C. SUR will immediately decay to 0. Periodic additions of positive reactivity will be necessary to maintain a stable positive SUR.

D. SUR will continuously rise until the POAH is reached. Periodic insertions of rods will be necessary to maintain a stable SUR.

Answer	A

Answer Explanation

A. Correct. With a Xenon free reactor startup, once a stable SUR in the IR is established, it will remain steady until the POAH is reached. This is due to the lack of feedback from MTC and FTC until after the POAH is reached.

B. Incorrect. Reactor power will turn as soon as the POAH is reached. Intermediate range trip does not occur until 25% equivalent power. Plausible since the IR trip is designed to prevent core damage in the event of an excessive reactivity addition below the POAH.

C. Incorrect. With a Xenon free startup, SUR will remain steady once established until the POAH is reached. Plausible since this would be true if Xenon was building in following a trip.

D. Incorrect. With a Xenon free startup, SUR will not continuously rise unless additional positive reactivity is added. Power will continue to drop (not stabilize). Plausible since this would be true if Xenon was continuing to decay following the trip.

Question Information

Topic	Q1 - Describe reactor power and startup rate response prior to reaching the POAH				
User ID	Q2479399			**System ID**	2479399
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	OP-2-1 Rev 05400		
Training Objective	Describe reactor power response prior to reaching the point of adding heat.		
Previous NRC Exam Use	None		

K/A Reference(s)

192008.K1.14	Safety Function	Tier	Group	RO Imp: 3.1	SRO Imp:
Describe reactor power and startup rate response prior to reaching the POAH					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 2	ID: 2479601	Points: 1.00

Unit-2 is in Mode 5.

- SDC is in service with 21 LPSI pump running
- RCS temperature is 145˚F
- RCS level is 39 feet
- RCS water level starts to lower slowly

Which of the following describes the INITIAL required action if RCS water level reaches 37.6 feet, and the reason for the action?

A. Start a HPSI Pump to refill the RCS

B. Reduce SDC flow to protect the LPSI pumps

C. Stop Charging Pumps to minimize radiation levels in the Aux Building

D. Stop 21 LPSI pump and place BOTH LPSI pump handswitches in Pull-To-Lock to prevent pump cavitation

Answer	B

Answer Explanation

A. Incorrect. Plausible since RCS level is decreasing, however there is no provision to start the HPSI pump and the first concern is to protect the LPSI pumps, then restore level.

B. Correct. Per AOP-3B section V Block step A. Protect the LPSI pumps, at 37.6 feet RCS flow is reduced to 1800 gpm.

C. Incorrect. Plausible if the operator misinterprets which pumps may be impacted by cavitation due to the lower RCS level. Plausible since the loss of RCS inventory could affect Containment and Aux Building radiation levels.

D. Incorrect. Plausible since this is the guidance provided if RCS level reached 36.8 feet.

Question Information

Topic	Q2 - AOP-3B actions required for an RCS Leak while on SDC.				
User ID	Q2479601			**System ID**	2479601
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-3B, Rev 02900		
Training Objective	Given the plant in Modes 4-6, respond to a loss of SDC with pressurization of the RCS possible per AOP-3B.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.005.K3.05	Safety Function	Tier 2	Group	RO Imp: 4.3	SRO Imp:
Knowledge of the effect that a loss or malfunction of the Residual Heat Removal System will have on the following systems or system parameters: (CFR: 41.7 / 45.6) ECCS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 3	ID: 2479444	Points: 1.00

Unit-1 is at 100% power, with 11 Charging Pump in operation.

At 1000 the following annunciators are received:
- 1C07 F-41, 11 CHG PP SIAS BLOCKED/AUTO START
- 1C07 F-45, CHG HDR FLOW LO/PRESS LO
- 1C19, R-04, U-1 480V ESF U/V TRIP

At 1001, The RO reports that Charging Header Flow has lowered to zero GPM, and PZR Level is slowly lowering.

(1) What procedure will the RO perform to resolve this issue?

(2) What action is required?

A. (1) 1C07 Alarm Manual
(2) Isolate Letdown

B. (1) AOP-2B Loss of All Charging
(2) Start 12 or 13 Charging Pump

C. (1) AOP-2B, Loss of All Charging
(2) Isolate Letdown

D. (1) 1C07 Alarm Manual
(2) Start 12 or 13 Charging Pump

Answer	D

Answer Explanation

A. Incorrect. (1) Correct as stated below. (2) Incorrect since isolating letdown is only performed if Charging Flow cannot be restored.

B. Incorrect. (1) Incorrect as stated below. (2) Correct as stated below.

C. Incorrect. (1) Incorrect since AOP-2B is not the correct procedure for a trip of a single charging pump. Plausible since AOP-2B handles conditions involving no Charging Pumps in operation. (2) Incorrect. The alarm at 1C19 indicates that this is not a common mode failure. Starting 12 or 13 CHG PP is the correct action, but this is only directed by AOP-2B if an individual CHG PP is in operation and showing signs of gas binding. Isolating letdown is only performed if Charging cannot be restored. Plausible since the operator may recognize that Letdown temperature will continue to rise until an automatic isolation signal is reached and may conclude that isolating Letdown is the correct action.

D. Correct. (1) Correct since the 1C07 alarm manual provides the direction to mitigate a single Charging Pump trip. (2) Correct since starting a backup Charging Pump is the correct action directed by the 1C07 alarm manual.

Question Information

Topic	Q3 - 1C07 or AOP-2B				
User ID	Q2479444			**System ID**	2479444
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	1C07 Alarm Manual Rev 03708		
Training Objective	Take action in accordance with the alarm manuals to correct, control, or mitigate the consequences of a given failure.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.022.AA1.01	Safety Function	Tier 1	Group	RO Imp: 3.8	SRO Imp:
Ability to operate and/or monitor the following as they apply to Loss of Reactor Coolant Makeup: (CFR: 41.7 / 45.5 / 45.6) CVCS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Question 4	ID: 2479410	Points: 1.00

Unit-1 is operating at 100% power when the following transient occurs:

- RAD MON LVL HI alarms on 1C07
- High RMS reading on Process Rad Monitor, RI-202, is valid

Which of the following is a required action?

A. Bypass the CVCS IXs.

B. Maximize CVCS purification flow.

C. Verify ONLY Letdown Isolation CV, 1-CVC-515 (L/D STOP), automatically shut.

D. Verify BOTH Letdown Isolation CVs, 1-CVC-515 (L/D STOP) and 1-CVC-516 (L/D CNTMT ISOL) automatically shut.

Answer	B

Answer Explanation

A. Incorrect. Bypassing IX is plausible since these are actions for high IX D/P and temperature. However, with high RCS activity, maximum purification is desired per AOP-6A.

B. Correct. Per the 1C07 alarm manual for RAD MON LVL HI, there are no automatic CVCS actions associated with high RCS activity. A valid RMS alarm requires entry into AOP-6A which directs maximizing letdown and purification flow.

C. Incorrect. Only CV-515 automatically shutting is plausible since only this CV shuts on high letdown temperature. However, there is no automatic signal for a high radiation condition.

D. Incorrect. Both Letdown CVs isolating is plausible since both valves will automatically isolate on a CVCIS. However, the CVCIS is caused by pressure in penetration rooms, not high radiation conditions.

Question Information

Topic	Q4 - RCS High Activity				
User ID	Q2479410			**System ID**	2479410
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	2
Technical Reference and Revision #	1C07-ALM, Rev 03708		
Training Objective	Given an abnormal chemistry condition in the Primary or Secondary, the trainee will be able to identify the event, take appropriate actions per procedures, to mitigate the event, and understand the basis for those actions.		
Previous NRC Exam Use	2018 NRC Exam		

K/A Reference(s)

APE.076.AA1.05	Safety Function	Tier 1	Group	RO Imp: 3.2	SRO Imp:
Ability to operate and/or monitor the following as they apply to High Reactor Coolant Activity: (CFR: 41.7 / 45.5 / 45.6) PRM					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 5	ID: 2479959	Points: 1.00

Which of the following is a purpose of the Reactor Coolant System?

A. Provide sufficient flow rate through the core to ensure that the Reactor Vessel Head is not voided in a PZR Vapor Space incident.

B. Piping serves as one of the three barriers preventing the release of fission products to the environment.

C. Provide sufficient quench tank capacity to accept a continuous discharge of the PORVs due to a loss of load accident, without rupturing the quench tank rupture disk.

D. Provide sufficient volume of water to prevent core uncovery at any time during any design basis event.

Answer	B

Answer Explanation

A. Incorrect. Plausible since a loss of Reactor Coolant or loss of subcooling may create a void in the RX Vessel Head. Preventing a void in the RX Vessel Head is not a function of the RCS per the UFSAR.

B. Correct. Per the UFSAR, the 3 fission barriers are RCS, Fuel Clad, and Containment.

C. Incorrect. Plausible since the RCS will be directed to the Quench Tank if the PORVs open during a high pressure event. The Quench Tank does not have the capacity for continuous PORV discharge.

D. Incorrect. Plausible since the RCS inventory acts to keep to the reactor core covered under normal conditions. This would be a function of Safety Injection, not RCS.

Question Information

Topic	Q5 - RCS Purpose				
User ID	Q2479959			**System ID**	2479959
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	UFSAR Rev 5200		
Training Objective	Recall the purpose and design features of the Reactor Coolant System.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.1.28	Safety Function	Tier 3	Group	RO Imp: 4.1	SRO Imp: 4.1
Knowledge of the purpose and function of major system components and controls (CFR: 41.7)					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 6	ID: 2479719	Points: 1.00

Both Units are at 100% power.

- Spent fuel pool cooling alignment: 11 SFP Pump - 11 SFP Cooler - Suction from 11 SFP and Discharging to 21 SFP
- SFP Time to 200°F is 36 hours

At 1000

- 11 SFP Trips

(1) What other pumps are available for SFP Cooling in the current mode?
(2) If no pumps can be started, when is the earliest that 200°F be reached?

A. (1) 12 SFP Pump ONLY
(2) Tomorrow at 2200

B. (1) 12 SFP Pump ONLY
(2) Tomorrow at 2000

C. (1) 12 SFP Pump, 11 LPSI, and 12 LPSI
(2) Tomorrow at 2200

D. (1) 12 SFP Pump, 11 LPSI, and 12 LPSI
(2) Tomorrow at 2000

Answer	A

Answer Explanation

A. Correct. (1) Per OI-3B, LPSI pumps are only available if the unit is defueled. (2) 36 hours from 1000 is 2200 tomorrow.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible if the operator misinterprets that when 36 hours from 1000 is. (2) Plausible if the operator is aware that LPSI pumps can be used to cool the SFP but misinterprets when they are allowed to be used.

Question Information

Topic	Q6 - Loss of SFP Pump effect on SFPCS				
User ID	Q2479719			**System ID**	2479719
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-6F, Rev 00800 OI-3B, Rev 03400.00		
Training Objective	Given the Spent Fuel Pool Cooling System and parameters in any mode of operation, respond and operate the system in accordance with: OI-24(A-H).		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.033.K6.01	Safety Function	Tier 2	Group	RO Imp: 3.6	SRO Imp:
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Spent Fuel Pool Cooling System: (CFR: 41.7 / 45.7) SFPCS pumps					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 7	ID: 2479440	Points: 1.00

Which of the following describes the response of the Pressurizer Code Safety Valves on an overpressure event?

A. The first safety lifts at 2450 PSIA and the second safety lifts at 2525 PSIA.

B. The first safety lifts at 2450 PSIA and the second safety lifts at 2700 PSIA.

C. The first safety lifts at 2500 PSIA and the second safety lifts at 2525 PSIA.

D. The first safety lifts at 2500 PSIA and the second safety lifts at 2700 PSIA.

Answer	C

Answer Explanation

A. Incorrect. The first part is incorrect but plausible since it matches the setpoint of DSS. The second part is the correct setpoint as listed in the UFSAR.

B. Incorrect. The first part is incorrect as stated above. The second part is incorrect but plausible if the operator misinterprets the Technical Specification Safety limit with the actual setpoint numbers of the safety valve.

C. Correct. The first and second parts are both correct as stated in the UFSAR revision 52 table 4-19.

D. Incorrect. The first part is correct as stated in the UFSAR. The second part is incorrect as stated above.

Question Information

Topic	Q7 - Identify when Safety Valves will lift				
User ID	Q2479440			System ID	2479440
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	CLOSED	Cognitive Level	LOW
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.8 Components, capacity, and functions of emergency systems.		

References Provided	None

K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	UFSAR Rev 52		
Training Objective	Recall the operation and basis of the following in relation to the RCS: PORVs and PZR Code Safeties.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.008.AK2.11	Safety Function	Tier 1	Group	RO Imp: 4.0	SRO Imp:
Knowledge of the relationship between a Pressurizer Vapor Space Accident and the following systems or components: (CFR: 41.7 / 45.7) PZR safeties					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 8	ID: 2479454	Points: 1.00

Unit-2 is at 100% power.

At time 0100

- Pressurizer Level Control instrumentation is selected to Channel X
- Pressurizer Pressure Control instrumentation is selected to Channel X
- Reactor Reg is selected to Channel X

At time 0105

- Multiple alarms are reported by the RO due to a loss of 2Y02 AC Instrument Bus

What action is required?

The Crew will ______.

A. Shift Letdown Controller, 2-PIC-201, to MANUAL and control Letdown backpressure at 440 to 480 PSIG.

B. Shift Pressurizer Pressure and Level instrumentation to Channel Y and shift PZR HTR LO LVL CUTOFF SEL switch to the Y position.

C. Verify LETDOWN THROTTLE VLV CONTROLLER, 2-HIC-110, in MANUAL until RCS loop 22 instruments have been isolated from RRS Channel X.

D. Place VCT OUT, 2-CVC-501-MOV in OPEN, place CHG PP SUCT, 2-CVC-504-MOV in CLOSE, and adjust turbine load as necessary to maintain Tcold on program.

Answer	C

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets the impact from 2Y02 to be the impact from 2Y09 which would impact the letdown system.

B. Incorrect. Plausible if the operator misinterprets the impact from 2Y02 to be the impact from 2Y01 which would impact the pressurizer controllers.

C. Correct. Per AOP-7J, channel B RPS deenergized in indicative of a loss of 2Y02 and these are the actions required.

D. Incorrect. Plausible if the operator misinterprets the impact from 2Y02 to be the impact from 2Y10 which would impact the letdown system.

Question Information

Topic	Q8 - Evaluate the effect of a Loss of 120VAC bus 2Y02				
User ID	Q2479454			**System ID**	2479454
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-7J, Rev 01900		
Training Objective	Given a loss of a Vital AC or Vital DC bus, diagnose the event and take the appropriate action per AOP-7J.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Reference(s)

APE.057.AA1.02	Safety Function	Tier 1	Group	RO Imp: 3.8	SRO Imp:
Ability to operate and/or monitor the following as they apply to Loss of Vital AC Electrical Instrument Bus: (CFR: 41.7 / 45.5 / 45.6) Manual control of PZR level					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 9	ID: 2479431	Points: 1.00

RCS pressure is 1000 PSIA with a steam bubble in the pressurizer.
Pressurizer power-operated relief valve (PORV) tailpipe temperature has been steadily rising.
The pressurizer vapor space contains 100.0% quality saturated steam.
PORV downstream pressure is 40 PSIA.

Assuming PORV leakage is an ideal throttling process:

(1) Which of the following will be the approximate PORV tailpipe temperature?

(2) What is the phase of escaping fluid if a PORV is leaking by?

A. (1) 267°F
(2) saturated

B. (1) 267°F
(2) superheated

C. (1) 312°F
(2) saturated

D. (1) 312°F
(2) superheated

Answer	D

Answer Explanation

A. Incorrect. (1) Incorrect but plausible if the operator misinterprets that the leaking fluid reaches atmospheric pressure and will then conclude that 267°F is the downstream temperature. (2) Incorrect but plausible since the fluid in the PZR is saturated and the operator may conclude that the fluid remains in a saturated condition downstream.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

D. Correct. (1) Correct as shown using the Steam Tables and the values given resulting in a downstream temperature of 312°F. (2) Correct as shown using the Steam Tables which confirms the resulting downstream fluid temperature and pressure is superheated.

Question Information

Topic	Q9 - PORV tailpipe temp and phase of escaping fluid if PORV leaks				
User ID	Q2479431			**System ID**	2479431
Status	Active	**Point Value**	1.00	**Time (min)**	4

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.14 Principles of heat transfer thermodynamics and fluid mechanics.		

References Provided	Steam Tables
K/A Justification	No additional information
SRO-Only Justification	Not Applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	Steam Tables		
Training Objective	Determine the exit conditions for a throttling process based on the use of steam and/or water.		
Previous NRC Exam Use	None		

K/A Reference(s)

193004.K1.15	Safety Function	Tier	Group	RO Imp: 2.8	SRO Imp:
Determine the exit conditions for a throttling process based on the use of steam and/or water					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 10	**ID: 2479715**	**Points: 1.00**

Unit-1 is at 100% power.

The following alarm occurs:

U-1 EMERGENCY AIRLOCK

(1) Which watchstander (on their normal area of ownership) would be sent to investigate?
(2) What is a possible cause of this alarm?

A. (1) Auxiliary Building Operator (ABO)
(2) The exterior airlock door is open

B. (1) Auxiliary Building Operator (ABO)
(2) The exterior airlock door equalizing valve is shut

C. (1) Outside Operator (OSO)
(2) The exterior airlock door is open

D. (1) Outside Operator (OSO)
(2) The exterior airlock door equalizing valve is shut

Answer	**C**

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Incorrect. (1) Plausible if the operator misinterprets which airlock is in the aux building. (2) Plausible if the operator misinterprets the normal position of the equalizing valves and determines it is normally open to allow air to equalize.

C. Correct. (1) The emergency airlock door is located on the OSO watch area. (2) Per 1C33, a possible cause is the exterior airlock door being open.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Q10 - Emergency Airlock alarm				
User ID	Q2479715			**System ID**	2479715
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.9 Shielding, isolation, and containment design features, including access limitations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	1C33-ALM, Rev 03700		
Training Objective	Given a control room alarm associated with 1C33, determine its cause and impact on the systems and required actions.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.103.K4.04	Safety Function	Tier 2	Group	RO Imp: 3.3	SRO Imp:
Knowledge of Containment System design features and/or interlocks that provide for the following: (CFR: 41.7) Personnel access hatch and emergency access hatch					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 11	ID: 2479761	Points: 1.00

Unit-1 was at 100% power when a transient occurred, resulting in a reactor trip.

- Main Feedwater has been lost and AFW flow has been initiated to both S/Gs
- 11 S/G pressure is 715 PSIA
- 12 S/G pressure is 805 PSIA

What is the status of AFW?

A. Supplying flow to 11 and 12 S/Gs.

B. Supplying flow to 12 S/G, blocked to 11 S/G.

C. Supplying flow to 11 S/G, blocked to 12 S/G.

D. Blocked to both 11 and 12 S/Gs.

Answer	A

Answer Explanation

A. Correct. Reference 1C03 alarm manual, AFAS block setpoint is 115 PSID between the SGs. The stem shows only a 90 PSID difference in SG level, therefore no AFAS Block has initiated, all AFW Block Valves should be open.

B. Incorrect. Plausible if the operator misinterprets the setpoint of AFAS Block and incorrectly concludes that AFW flow is blocked to 11 SG. Incorrect since Pressures given are above the AFAS Block setpoint.

C. Incorrect. Plausible if the operator misinterprets the setpoint of AFAS Block and incorrectly concludes that AFW flow is blocked to 12 SG. Even if pressures were below the AFAS Block setpoint, the system blocks the LOWER SG pressure, which would be 11 SG.

D. Incorrect. Plausible if the operator misinterprets the setpoint of AFAS Block and incorrectly concludes that AFW flow is blocked to both SGs. AFAS Block does not isolate both SGs.

Question Information

Topic	Q11 - Given SG pressures and plant conditions, determine the status of AFW				
User ID	Q2479761			**System ID**	2479761
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	1C03 Alarm Manual Rev 06300		
Training Objective	Determine the status of AFW Flow to each S/G given plant conditions and S/G pressures.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.061.A1.02	Safety Function	Tier 2	Group	RO Imp: 3.8	SRO Imp:
Ability to predict and/or monitor changes in parameters associated with operation of the Auxiliary/Emergency Feedwater System, including: (CFR: 41.5 / 45.5) S/G pressure					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 12	ID: 2479556	Points: 1.00

Assuming NIs have become decalibrated, what redundant power indication would provide the most accurate indication of reactor power?

A. Delta-T Power.

B. Main Generator MWe.

C. Main Turbine exhaust pressure.

D. Comparison of CETs to Thot digital readouts on panel 1(2)C06.

Answer	A

Answer Explanation

A. Correct. IAW OP-3 when changing power "RPS Delta-T Power should be used as the primary power indication, until the LRNIs are calibrated at the new steady state RCS temperature"

B. Incorrect. Plausible since it can be used as a rough estimate of Reactor Power but is influenced by many factors such as bay temperature that changes turbine efficiency.

C. Incorrect. Plausible if the operator misinterprets this indication for steam header pressure which does have a correlation to Reactor Power.

D. Incorrect. Plausible if the operator misinterprets that comparing Thot to CET's would reflect the delta T across the core and correctly estimate a power level.

Question Information

Topic	Q12 - NIs have become decalibrated, what redundant power indication is used				
User ID	Q2479556			**System ID**	2479556
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.2 General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.		

References Provided	None

K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	OP-3, Rev 07700		
Training Objective	Recall the redundant power indication that should be monitored to ensure NIs are calibrated properly.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.033.AK1.02	Safety Function	Tier 1	Group	RO Imp: 3.5	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Intermediate Range Nuclear Instrumentation: (CFR: 41.8 / 41.10 / 45.3) Equivalency and/or overlap among source range, intermediate range, and power range channel readings					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

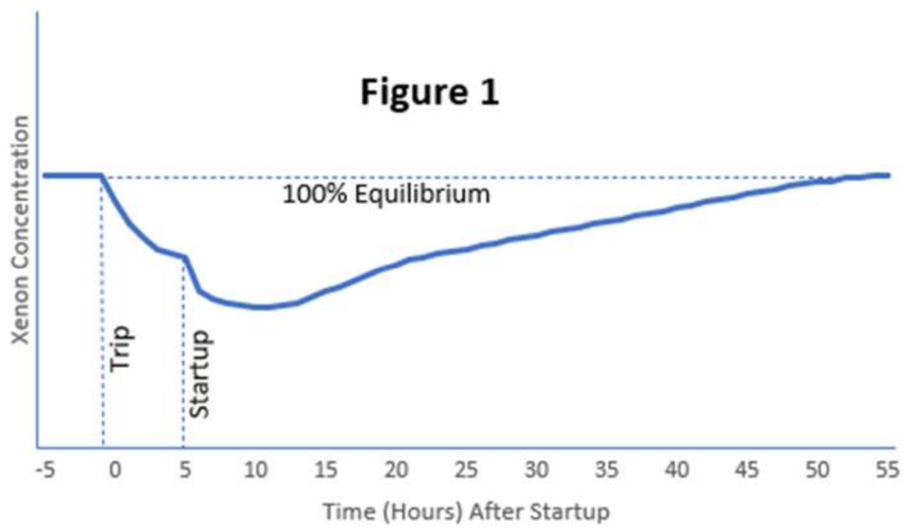
None

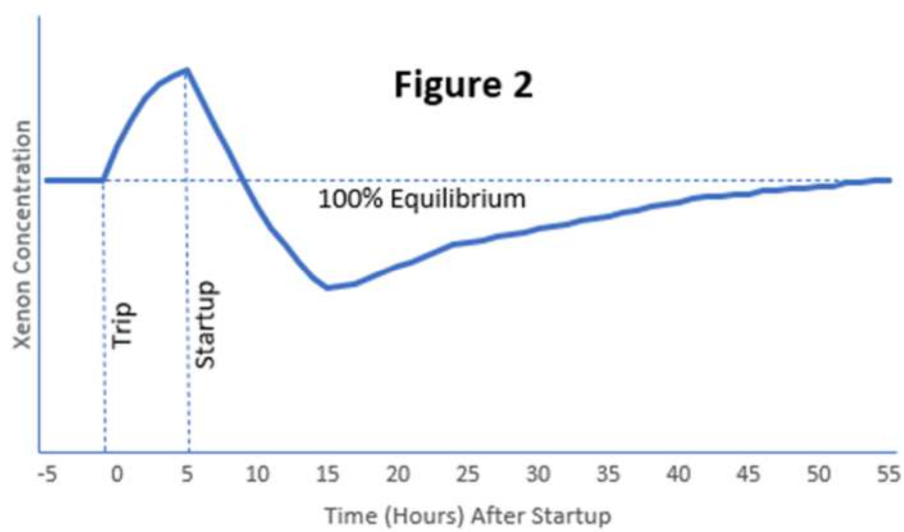
Question 13	ID: 2479416	Points: 1.00

The Unit is at steady state 100% power when a reactor trip occurs.

(1) Which of the following figures shows the trend of Xenon concentration from the time the reactor trips until the startup is performed?

The trend of Xenon concentration from the reactor trip until the startup is because the decay rate of Iodine-135 to Xenon-135 is ___(2)___ than the neutron absorption rate of Xenon-135.





A. (1) Figure 1
(2) higher

B. (1) Figure 1
(2) lower

C. (1) Figure 2
(2) lower

D. (1) Figure 2
(2) higher

Answer	D

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the operator will recall that the Xenon concentration trends in the direction of reactor power and will conclude that the concentration starts to lower upon the reactor trip. (2) Correct as stated below.

B. Incorrect. (1) Incorrect as stated above. (2) Incorrect but plausible since the operator may misinterpret the magnitude between the creation of Xenon and the removal of Xenon through neutron absorption.

C. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

D. Correct. (1) Correct since Figure 2 shows the Xenon concentration rising after the reactor trip which is the correct trend. Then, upon the startup, the Xenon concentration will quickly lower and eventually return to the 100% equilibrium value. (2) Correct since the creation of Xenon through decay is higher after the reactor trip for the first few hours compared to the removal of Xenon through neutron absorption.

Question Information

Topic	Q13 - Reactor Startup with Xenon				
User ID	Q2479416			**System ID**	2479416
Status	Active	**Point Value**	1.00	**Time (min)**	4

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.		

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	UFSAR, Rev 51		
Training Objective	Plot the curve and explain the reason for the reactivity insertion by xenon-135 versus time for the following: reactor startup with xenon-135 already present in the core.		
Previous NRC Exam Use	None		

K/A Reference(s)

192006.K1.10	Safety Function	Tier	Group	RO Imp: 3.2	SRO Imp:
Plot the curve and explain the reasoning for the reactivity insertion by xenon-135 versus time for the following: reactor startup with xenon-135 already present in the core					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 14	ID: 2479448	Points: 1.00

During an extended station blackout event what are the fail positions of the following valves:

(1) Containment Spray CVs
(2) Letdown Isolation CVs

A. (1) Open
(2) Closed

B. (1) Open
(2) Open

C. (1) Closed
(2) Open

D. (1) Closed
(2) Closed

Answer	A

Answer Explanation

A. Correct. (1) Fail position of the Containment Spray CVs is open to ensure spray flow on loss of air. (2) Fail position of the Letdown CVs is closed to minimize loss of inventory during an accident concurrent with loss of air.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) Plausible if the operator misinterprets that the normal position is closed to prevent spraying water into the containment and since during a blackout no other casualties are happening the operator determines that spray should fail closed preventing spraying down the containment. (2) Plausible if the operator misinterprets that the letdown valves would fail to the normal position of open.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Q14 - Extended SBO Instrument Air Effects				
User ID	Q2479448			**System ID**	2479448
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	Operators are not expected to memorize the fail position of all air operated valves, therefore this question requires the students to process the purpose of each of these sets of valves and determine which way they should fail to keep plant safe that is reason behind the higher cognitive classification of this question.

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	60730SH0002, Rev 0079 60731SH0003, Rev 0033		
Training Objective	Given an extended Station Blackout with a loss of Instrument Air, determine the effect on the following: Containment Spray system and Charging and Letdown system (CVCS).		
Previous NRC Exam Use	2014 NRC		

K/A Reference(s)

EPE.055.EK2.01	Safety Function	Tier 1	Group	RO Imp: 3.9	SRO Imp:
Knowledge of the relationship between a Station Blackout and the following systems or components: (CFR: 41.7 / 45.7) Letdown isolation, RCP seal return, PZR PORVs, or secondary PORVs (atmospheric relief valves)					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Question 15	ID: 2479921	Points: 1.00

- A reactor startup is in progress with power at 2 x 10^{-6}%.
- Group 4 regulating CEA withdrawal is being performed.
- RCS pressure is 2250 PSIA and Tcold is 532°F.

Which of the following is the operational concern as criticality is approached?

A. Each Group 4 CEA must be manually withdrawn to the upper electrical limit (UEL) as each group reaches the upper group stop.

B. The maximum allowed sustained startup rate is limited to 1 decade/minute (DPM).

C. The Shutdown Monitor setpoint must be reset prior to causing an alarm.

D. The 1/M base count rate must be updated after each subsequent CEA withdrawal once Reg Group 3 is withdrawn to 90 inches.

Answer	B

Answer Explanation

A. Incorrect – while taking the reactor critical the subsequent CEA banks are NOT to be withdrawn to the UEL until AFTER the reactor is critical.

B. Correct – Cautions while approaching criticality per OP-2: CAUTION ● Do NOT exceed a 1 DPM sustained startup rate. ● Criticality shall be anticipated when CEAs are being withdrawn. [B0032] ●Withdraw CEAs only in a deliberate and carefully controlled manner. During withdrawal, constantly monitor nuclear instrumentation and redundant indications of reactor power level and neutron flux.

C. Incorrect – the SDM is reset AFTER the alarms to ensure the 1.5x count rise has been received.

D. Incorrect – the 1/M base count is reset prior to the approach to criticality and then left alone.

Question Information

Topic	Q15 - Approach to Criticality Operational Implication				
User ID	Q2479921			System ID	2479921
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	CLOSED	Cognitive Level	LOW
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	OP-2-1 Rev 05400		
Training Objective	Recall the requirements of OP-2 during the approach to criticality.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.001.K5.18	Safety Function	Tier 2	Group	RO Imp: 4.3	SRO Imp:
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Control Rod Drive System: (CFR: 41.1 / 41.2 / 41.5 / 41.6 / 45.7) Anticipation of criticality at any time when adding positive reactivity during startup					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 16	ID: 2479640	Points: 1.00

Unit-1 is operating at 100% power:

- 11 SGFP Trips due to high vibrations
- 11 SGFP Mini-Flow valve (1-FW-4484) is shut
- AOP-3G, Malfunction of the Main Feedwater System is implemented

What is the required operator action?

A. Reset 11 SGFP.

B. Immediately trip the reactor.

C. Reduce power to 88% or less.

D. Ensure 13 AFW Pump is operating.

Answer	C

Answer Explanation

A. Incorrect. Plausible since this is an action described in AOP-3G but with the mini-flow shut 11 SGFP will not reset.

B. Incorrect. Plausible since this was the correct action if 13 SGFP is tagged out.

C. Correct. Per AOP-3G, a downpower to 88% is required.

D. Incorrect. Plausible if the operator misinterprets which 13 feed pump is required to be in service.

Question Information

Topic	Q16 - SGFP Trip actions required (after 13 SGFP installation) -100% power				
User ID	Q2479640			System ID	2479640
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	CLOSED	Cognitive Level	HIGH
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-3G, Rev 01600		
Training Objective	Given a trip of a Turbine Driven Feed Pump at 100% power, evaluate the response of the Main Feedwater system.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.2.02	Safety Function	Tier 3	Group	RO Imp: 4.6	SRO Imp: 4.1
Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels (CFR: 41.6 / 41.7 / 45.2)					
G.SYS.059	Safety Function	Tier	Group	RO Imp:	SRO Imp:
MFW Main Feedwater System					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 17	ID: 2479442	Points: 1.00

Unit-1 is at 100% power when the following occurs:

- 11A RCP trips
- On the trip 11A RCP develops a seal leak
- EOP-5, Loss of Coolant Accident, is implemented

(1) What is the Low Flow RPS trip Setpoint?

(2) What method is directed in EOP-5 to control RCS pressure?

A. (1) 92% flow
(2) Aux Spray

B. (1) 92% flow
(2) Main Spray

C. (1) 96% flow
(2) Main Spray

D. (1) 96% flow
(2) Aux Spray

Answer	A

Answer Explanation

A. Correct. (1) Per 1C05-ALM, the RCS low flow trip setpoint is less than 92% flow. (2) Per EOP-5, Main Spray is only used with all RCPs operating.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) Plausible if the operator incorrectly recalls the RPS trip setpoint to be 96% rather than the correct value of 92%. The operator may recall that the pre-trip setpoint is 94% with a 2% margin from the actual trip setpoint. The operator may then incorrectly interpret the actual setpoint to be 96%, or 2% higher than the pre-trip setpoint rather than 2% lower than the pre-trip setpoint. (2) Plausible since with only 1 RCP tripped there could be adequate flow to use main spray but this is not per procedure.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Q17 - Loss of RCPs on RCS				
User ID	Q2479442			**System ID**	2479442
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	1C05-ALM, Rev 04101		
Training Objective	Given a change in plant conditions, correctly determine the effect on the Reactor Coolant System (RCS).		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.015.AK1.02	Safety Function	Tier 1	Group	RO Imp: 3.9	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Reactor Coolant Pump Malfunctions: (CFR: 41.8 / 41.10 / 45.3) Consequences of an RCP failure					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 18	ID: 2479610	Points: 1.00

Unit-1 is operating at 100% power the following alarm is received:

11/12 CC HX CC OUT TEMP HI

(1) What is the impact on the plant?

(2) What action should be taken per the alarm manual to mitigate this impact?

A. (1) Reactor power rises
(2) Verify 11 and 12 CC Heat Exchanger Temperature Control Bypass Valves are open.

B. (1) Reactor power rises
(2) Verify CVCS Ion Exchanger Bypass Valve in bypass.

C. (1) Reactor power lowers
(2) Verify 11 and 12 CC Heat Exchanger Temperature Control Bypass Valves are open.

D. (1) Reactor power lowers
(2) Verify CVCS Ion Exchanger Bypass Valve in bypass.

Answer	D

Answer Explanation

A. Incorrect. Plausible if the impact of high CC temperature on letdown temperature and boron on the IX is misinterpreted as causing power to rise. Plausible since the alarm manual says to verify these valves are shut.

B. Incorrect. Plausible if the impact of high CC temperature on letdown temperature and boron on the IX is misinterpreted as causing power to rise.

C. Incorrect. Plausible since the alarm manual says to verify these valves are shut.

D. Correct. Per 1C13, this alarm will cause letdown temperatures to increase causing power to lower. Action is to ensure the Ion Exchangers bypass valve opens to protect the resin from high temperatures.

Question Information

Topic	Q18 - Impact of high CCW temperature on the plant				
User ID	Q2479610			**System ID**	2479610
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-7C, Rev 00500 1C13, Rev 05700		
Training Objective	Given the Component Cooling Water System and parameters in any mode of operation, respond and operate the system in accordance with OI-16, OI-1A, UFSAR, 1C13 alarm manual, technical specifications, AOP-7C and drawings 60710/62710.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.008.A3.01	Safety Function	Tier 2	Group	RO Imp: 3.6	SRO Imp:
Ability to monitor automatic features of the Component Cooling Water System, including: (CFR: 41.7 / 45.5) Setpoints for normal operations, warnings, and trips that are applicable to the CCWS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 19	ID: 2480024	Points: 1.00

Both Units are at 75% power.

- P-13000-3 transformer is tagged out for maintenance.
- The 500KV Black Bus is lost.

(1) What action will automatically occur?

(2) What RPS trip unit will cause the reactor trip based on its Tech Spec allowable value being violated?

A. (1) ONLY Unit-1 will trip.
(2) Trip Unit 3, Reactor Coolant Flow-Low

B. (1) ONLY Unit-1 will trip.
(2) Trip Unit 4, Pressurizer Pressure-High

C. (1) ONLY Unit-2 will trip.
(2) Trip Unit 4, Pressurizer Pressure-High

D. (1) ONLY Unit-2 will trip.
(2) Trip Unit 3, Reactor Coolant Flow-Low

Answer	A

Answer Explanation

A. Correct. (1) Correct since a loss of the Black Bus will deenergize the Unit-1 RCP bus. (2) Correct since the Trip Unit 3 allowable band of greater than or equal to 92% of design flow will occur immediately and trip the reactor.

B. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible since the loss of the Black Bus will cause almost all electrical buses on Unit-1 to be lost causing several actions that will contribute to a high RCS temperature and pressure condition. These actions include a loss of Main Feedwater, a loss of forced RCP circulation, and a lowering of S/G level.

C. Incorrect. (1) Incorrect but plausible if the operator misinterprets what electrical supply buses are powered from the Black Bus. (2) Incorrect as stated above.

D. Incorrect. (1) Incorrect as stated above. (2) Correct as stated above.

Question Information

Topic	Q19 - Plant effects due to loss of 500KV Black Bus				
User ID	Q2480024			**System ID**	2480024
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	TS Table 3.3.1-1 Rev 232/208 61001SH0001 Rev 54		
Training Objective	Explain the design features and/or interlocks of the 500KV system that provide automatic breaker trips.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.2.36	Safety Function	Tier 3	Group	RO Imp: 3.1	SRO Imp: 4.2
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation (CFR: 41.10 / 43.2 / 45.13)					
G.CE.E13	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Loss of Forced Circulation and/or LOOP and/or a Blackout					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 20	ID: 2479629	Points: 1.00

(1) As power is lowered, RPS Trip Unit #3, RCS Low Flow Trip, __________ below $1x10^{-4}$% reactor power.

(2) The basis for the RCS Low Flow Trip is to prevent __________.

A. (1) is automatically bypassed
(2) high RCS pressure conditions

B. (1) is automatically bypassed
(2) challenging DNBR limits

C. (1) is **NOT** automatically bypassed
(2) high RCS pressure conditions

D. (1) is **NOT** automatically bypassed
(2) challenging DNBR limits

Answer	D

Answer Explanation

A. Incorrect. Plausible since the operator will recall the Zero Power Mode bypass is automatically removed when power is 10E-4% or greater but must be manually bypassed below 10E-4% power with Zero Power Mode Bypass switch. Plausible since on a loss of all RCPs, RCS Pressure will go up but this is not the basis of the low flow trip.

B. Incorrect. Plausible since the operator will recall the Zero Power Mode bypass is automatically removed when power is 10E-4% or greater but must be manually bypassed below 10E-4% power with Zero Power Mode Bypass switch. Per Tech Spec Bases this trip prevents challenging DNBR limits.

C. Incorrect. Plausible since on a loss of all RCPs, RCS Pressure will go up but this is not the basis of the low flow trip.

D. Correct. This trip unit must be manually bypassed below 10E-4% power with Zero Power Mode Bypass switch if required. Per Tech Spec Bases this trip prevents challenging DNBR limits.

Question Information

Topic	Q20 - RPS Low Flow Trip and basis				
User ID	Q2479629			System ID	2479629
Status	Active	Point Value	1.00	Time (min)	2

Open or Closed Reference	CLOSED	Cognitive Level	HIGH
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.8 Components, capacity, and functions of emergency systems.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	TS 3.3.1 Basis Rev 2		
Training Objective	From memory, recall the automatic RPS trips, including their basis, setpoints, indications, and any associated interlocks, in accordance with the UFSAR, Tech Specs, and 1C05 Alarm Manual.		
Previous NRC Exam Use	2019 NRC		

K/A Reference(s)

SYS.012.K5.01	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp:
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Reactor Protection System: (CFR: 41.5 / 45.7) Departure from nucleate boiling					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 21	ID: 2479698	Points: 1.00

Unit-1 is in Mode 4 with Shutdown Cooling in service.

- SINGLE MPT protection enabled

(1) What is the lowest RCS pressure that will automatically open the PORVs?

(2) What action can be taken to isolate a PORV with a high pressure condition due to a failed transmitter?

A. (1) 2450 PSIA
(2) Place the PORV Override handswitch, 1-HS-1402(1404) in OVERRIDE TO CLOSE

B. (1) 2450 PSIA
(2) Place the PORV Block handswitch, 1-HS-403(405) in CLOSE

C. (1) 410 PSIA
(2) Place the PORV Override handswitch, 1-HS-1402(1404) in OVERRIDE TO CLOSE

D. (1) 410 PSIA
(2) Place the PORV Block handswitch, 1-HS-403(405) in CLOSE

Answer	D

Answer Explanation

A. Incorrect. (1) Incorrect but plausible if the operator misinterprets the setpoint from Mode 1. (2) Incorrect as stated below.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Incorrect. (1) Correct as stated below. (2) Incorrect but plausible if the operator misinterprets that the PORV override handswitch can override an active high pressure condition.

D. Correct. (1) Correct as stated in the 1C06 Alarm Manual for Window E-21 PORV Energized when in Single MPT Enable mode. In Mode 4 with SDC in service, the plant conditions will be less than 369°F when MPT is placed in Single MPT Enable. (2) Correct as stated in the 1C06 Alarm Manual for Window E-21 PORV Energized when in Single MPT Enable mode.

Question Information

Topic	Q21 - PORV Logic and Circuitry				
User ID	Q2479698			**System ID**	2479698
Status	Active	**Point Value**	1.00	**Time (min)**	4

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	4
Technical Reference and Revision #	1C06 Alarm Manual Rev 05400 60616SH0056 Rev 8 60616BSH0057 Rev 4		
Training Objective	Given plant conditions, determine the operating characteristics of the following: PORVs.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.007.A3.01	Safety Function	Tier 2	Group	RO Imp: 3.4	SRO Imp:
Ability to monitor automatic features of the Pressurizer Relief Tank/Quench Tank System, including: (CFR: 41.5 / 41.7 / 45.5) Components that discharge to the PRT/quench tank					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 22	ID: 2479439	Points: 1.00

Unit-2 is at 100% power.

- Both Steam Generator Levels are rising at 10"/min
- Feed flow is greater than steam flow
- AOP-3G, Malfunction of the Main Feedwater, is implemented

(1) What action per AOP-3G is taken to minimize feed flow?

(2) What is the MINIMUM level that a manual reactor trip would be required?

A. (1) Shut both SG FW Isolation MOVs
(2) +55 inches

B. (1) Shut both SG FW Isolation MOVs
(2) +50 inches

C. (1) Shift both SGFP HICs to Manual and lower SGFP speed
(2) +55 inches

D. (1) Shift both SGFP HICs to Manual and lower SGFP speed
(2) +50 inches

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible since this is the setpoint for when the MSIVs are shut on high level. (2) Plausible if the operator misinterprets that the high level trip is the same number as the high end of the SG level band.

B. Incorrect. (1) See above. (2) Correct.

C. Incorrect. (1) Correct. (2) See above.

D. Correct. (1) Per AOP-3G, this is the first action taken to try and minimize feed flow. (2) Per AOP-3G at +50 inches a manual reactor trip is required.

Question Information

Topic	Q22 - Actions for a overfeed condition				
User ID	Q2479439			**System ID**	2479439
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	AOP-3G, Rev 01700		
Training Objective	Given plant conditions, diagnose a Malfunction of the Main Feedwater System and respond by directing and/or implementing the proper operator actions in accordance with the following procedures: –AOP-3G.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.054.AA2.08	Safety Function	Tier 1	Group	RO Imp: 3.4	SRO Imp: 3.4
Ability to determine and/or interpret the following as they apply to Loss of Main Feedwater: (CFR: 43.5 / 45.13) Steam flow and/or MFW flow					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 23	ID: 2479417	Points: 1.00

During the continuous full power reactor operation in the final several months of a core cycle, the reactor coolant boron concentration must be ___(1)___ periodically to compensate for ___(2)___.

A. (1) raised
(2) decreasing control rod worth and burnable poison burnout

B. (1) raised
(2) fuel depletion and buildup of fission product poisons

C. (1) lowered
(2) fuel depletion and buildup of fission product poisons

D. (1) lowered
(2) decreasing control rod worth and burnable poison burnout

Answer	C

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the RCS boron concentration is raised during the first few months of a core cycle and the operator may misinterpret the timing of when this occurs. (2) Incorrect but plausible since the control rod worth does change over the core cycle and the burnable poisons do burnout quickly in the first few months of the core cycle.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Correct. (1) Correct since RCS boron concentration must be lowered in the final several months of the core cycle. (2) Correct since both reasons contribute to why the RCS boron concentration must be lowered.

D. Incorrect. (1) Correct as stated above. (2) Incorrect as stated above.

Question Information

Topic	Q23 - Boron Changes Over Core Life				
User ID	Q2479417			**System ID**	2479417
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	COLR Unit 1 Cycle 26		
Training Objective	Describe how and why boron concentration changes over core life.		
Previous NRC Exam Use	None		

K/A Reference(s)

192007.K1.04	Safety Function	Tier	Group	RO Imp: 3.4	SRO Imp:
Describe how and why boron concentration changes over core life					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

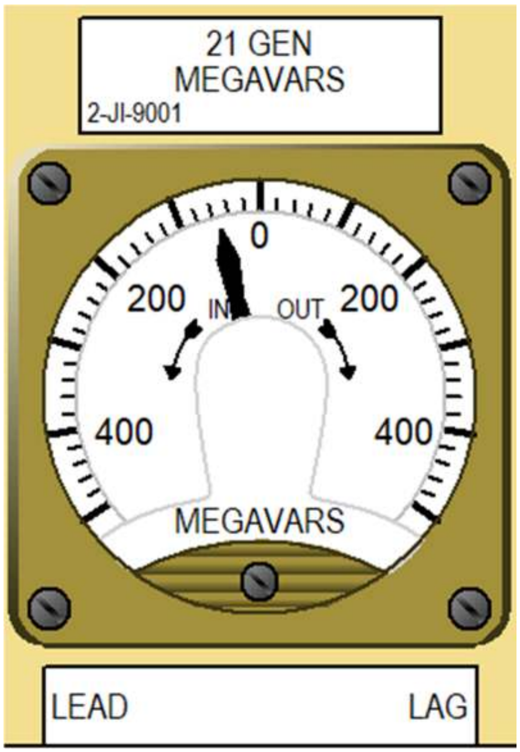
Cross Reference Links

None

Question 24	ID: 2479458	Points: 1.00

Using provided reference:

Unit-2 is at 600 MWe.
- Grid disturbances are occurring making reactive power swing
- Reactive power is now as shown below



(1) What is the state of the generator?
(2) What is the Excitation Limit (in MVAr) for 600 MWe for the current state of the generator?

A. (1) Overexcited
(2) -350

B. (1) Overexcited
(2) 300

C. (1) Underexcited
(2) -350

D. (1) Underexcited
(2) 300

Answer	C

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Incorrect. (1) Plausible if the operator misinterprets which direction on the graph is the lead direction. (2) Plausible if the operator determines the generator is overexcited and this is the power factor limit.

C. Correct. (1) Per OI-43, a generator operating in the lead direction is underexcited. (2) Per OI-43, the UEL is -350.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Q24 - Reference - Underexcited Generator and limits				
User ID	Q2479458			System ID	2479458
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	OI-43A-2 PAGE 88 ONLY	Cognitive Level	HIGH
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	OI-43A-2, Figure 6 - page 88 ONLY Embedded Reference
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	4
Technical Reference and Revision #	OI-43A-2, Rev 04400		
Training Objective	Given the U2 Main Turbine, MSR and Auxiliary Systems and parameters in any mode of operation, respond and operate the systems in accordance with OI-43A-2.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.077.AK1.03	Safety Function	Tier 1	Group	RO Imp: 3.3	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.4 / 41.5 / 41.7 / 41.10 / 45.8) Underexcitation					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 25	ID: 2479718	Points: 1.00

What ESFAS signals are locally initiated after commencing OTCC in EOP-3, Loss of All Feedwater, to provide additional flow through the reactor core?

A. SIAS A6 and B6

B. RAS A1 and B1

C. CSAS A2 and B2

D. AFAS A4 and B4

Answer	A

Answer Explanation

A. Correct. In Block Step J, SIAS A6 and B6 are locally initiated to start the 2 backup Charging Pumps which would normally turn off automatically based on high PZR level deviation.

B. Incorrect. Plausible since Block Steps M and N direct the crew to perform actions in preparation of RAS and then again after the automatic RAS actuation.

C. Incorrect. Plausible since Block Step P directs the crew to perform a local reset of CSAS when containment pressure lowers.

D. Incorrect. Plausible since EOP-3 contains numerous steps to direct the crew to restore AFW flow if a pump or AFW source becomes available.

Question Information

Topic	Q25 - OTCC and ESFAS Actuations				
User ID	Q2479718			System ID	2479718
Status	Active	Point Value	1.00	Time (min)	2

Open or Closed Reference	CLOSED	Cognitive Level	LOW
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	EOP-3-1 Rev 02400		
Training Objective	Recall the operator actions taken during OTCC in EOP-3.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.013.K5.23	Safety Function	Tier 2	Group	RO Imp: 4.2	SRO Imp:
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Engineered Safety Features Actuation System: (CFR: 41.3 / 41.4 / 41.5 / 45.7) Inadequate core cooling					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 26	ID: 2479598	Points: 1.00

On a loss of 11 4KV Bus, which one of the below listed boration flow paths would be available?

A. RWT outlet valve (CVC-504)

B. 12 BA pump and BA direct M/U valve (CVC-514)

C. 11 or 12 BAST gravity drain valves (CVC-508 and 509)

D. 11 BA pump, BA flow control valve (CVC-210Y), VCT M/U valve (CVC-512), and VCT outlet valve (CVC-501)

Answer	B

Answer Explanation

A. Incorrect. Plausible since this would be an available path on a loss of MCC-104 and the operator misinterprets the failures.

B. Correct. Per AOP-7I, this is the correct boration flow path for the loss of MCC-114.

C. Incorrect. Plausible since this would be an available path on a loss of MCC-104 and the operator misinterprets the failures.

D. Incorrect. Plausible if the operator misinterprets that MCC-114 is B train since it has an even number (normal numbering/naming scheme) and the flow path contains A train components, but MCC-114 is A train.

Question Information

Topic	Q26 - Loss of MCC-114 impact on CVCS				
User ID	Q2479598			**System ID**	2479598
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-07I, Rev 03700		
Training Objective	From memory, describe the operation of the following in accordance with the Operating Instructions, Alarm Manuals, EOPs and AOPs: Boric Acid Storage Tanks (BASTs) and Boric Acid Pumps.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.004.K2.02	Safety Function	Tier 2	Group	RO Imp: 3.0	SRO Imp:
Knowledge of electrical power supplies to the following: (CFR: 41.6 / 41.7) Pumps used to makeup to CVCS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 27	ID: 2479438	Points: 1.00

Which of the following describes the fuel-to-coolant thermal conductivity for a fuel rod at the beginning of a fuel cycle (BOC) compared to the end of a fuel cycle (EOC)?

A. Greater at BOC, due to a higher fuel pellet density.

B. Greater at BOC, due to lower contamination of fuel rod fill gas with fission product gases.

C. Smaller at BOC, due to a larger gap between the fuel pellets and cladding.

D. Smaller at BOC, due to a smaller corrosion film on the surface of the fuel rods.

Answer	C

Answer Explanation

A. Larger at BOC (FALSE) due to a higher fuel pellet density (FALSE)
Incorrect - Fuel-to-coolant thermal conductivity INCREASES over core life. Initially following core exposure, the fuel pellet density increases as the ceramic UO_2 is sintered. At the end of core life fuel pellet density has become less dense, due to swelling of the pellet due to the buildup of fission product gases. However, the change in density is overcome by the pellet swelling and cladding creep, which reduces the pellet-to-clad gap and an overall increase thermal conductivity.

B. Larger at BOC (FALSE) due to lower contamination of fuel rod fill gas with fission product gases (TRUE)
Incorrect - Fuel-to-coolant thermal conductivity INCREASES over core life. Although fission product gases accumulate in the fuel pellets causing pellet swell, which in combination with clad creep reduces the pellet to clad gap. The buildup of fission product gases does reduce thermal conductivity; however, this effect is minimized by the combination of clad creep and pellet swell which reduces the gap and overall increase thermal conductivity.

C. Smaller at BOC (TRUE) due to a larger gap between the fuel pellets and clad (TRUE)
CORRECT - New fuel compacts slightly when it is irradiated in the reactor. This is called densification. Densification occurs as a result of the elimination of small pores in the fuel pellet. The pellets get shorter and smaller in diameter, increasing the gap between the fuel and the cladding, and decreasing thermal conductance.

After the initial increase in the pellet-to-clad gap, as the core ages and the fuel burns out, fission gases form causing a gradual swelling of the fuel pellets. Also, fission products are less dense than the fuel. This has the effect of reducing the size of the gap between the fuel and the clad. This results in a smaller DT_{gap} and thus, a smaller overall DT and centerline temperature.

As the core is operated, the clad tends to creep inward towards the fuel pellets due to the high temperatures and pressures in the core. Together, both of these factors lead to a decreased fuel to clad gap, resulting in a decrease in DT_{gap}, in a decrease overall DT, and a decrease in fuel pellet centerline temperature.

The combined effects of clad creep and fuel pellet swelling can result in a reduction in the pellet-to clad gap and even in the fuel pellets coming in contact with the cladding. This would result in the gap at that point being reduced to zero, resulting in a reduction of the overall DT and a reduction in the fuel pellet centerline temperature being reduced.

D. Smaller at BOC (TRUE) due to a smaller corrosion film on the surface of the fuel rods (FALSE)
Incorrect - Although there is a smaller corrosion film on the surface of the fuel rods at BOL, the increase in corrosion layer of core life is minor in comparison to the reduction in the pellet-to-clad gap resulting from the pellet swell and cladding creep. An increase in corrosion layer on the cladding surface over core life would INCREASE the fuel-to-coolant thermal conductivity.

Question Information

Topic	Q27 - Thermal conductivity and fuel centerline temperature				
User ID	Q2479438			**System ID**	2479438
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.14 Principles of heat transfer thermodynamics and fluid mechanics.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	

Technical Reference and Revision #	
Training Objective	Describe factors that affect peaking and hot channel factors.
Previous NRC Exam Use	None

K/A Reference(s)

193009.K1.07	Safety Function	Tier	Group	RO Imp: 3.3	SRO Imp:
Describe factors that affect peaking and hot channel factors					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 28	ID: 2479603	Points: 1.00

Unit-1 is on SDC (Shutdown Cooling) with one LPSI pump running.

- An inadvertent RAS occurs.

What effect does this have on the LPSI pump being used for SDC and the idle LPSI pump?

A. Running pump remains operating; idle pump starts.

B. Running pump remains operating; idle pump is available.

C. Running pump will trip; idle pump is available.

D. Running pump will trip; idle pump starts.

Answer	B

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets what pumps start on a RAS and determines LPSIs start.

B. Correct. While aligning shutdown cooling per OI-3B section 6.1 - BOTH LPSI RAS Override Key switches are in RAS OVERRIDE prior to SDC initiation

C. Incorrect. Plausible since this is the effect on a normal running LPSI pump not in RAS OVERRIDE.

D. Incorrect. Plausible since this is the effect on a normal running LPSI pump not in RAS OVERRIDE. Plausible if the operator misinterprets what pumps start on a RAS and determines LPSIs start.

Question Information

Topic	Q28 - Identify effect of RAS on operating LPSI pump.				
User ID	Q2479603			**System ID**	2479603
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	OI-3B, Rev 03400		
Training Objective	Given that shutdown cooling is operating or secured apply the PRECAUTIONS, INITIAL CONDITIONS, and CAUTIONS of OI-3B.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.005.K4.11	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp:
Knowledge of Residual Heat Removal System design features and/or interlocks that provide for the following: (CFR: 41.7) Lineup for low head recirculation mode (external and internal)					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 29	ID: 2479778	Points: 1.00

While touring the 45 foot Switchgear Room, the TBO reports:

- White Load Shed Verification Light is lit on 4KV Bus 14

(1) What is the status of the Load Shed Verification Relays?

(2) What is the reason for the status?

A. (1) Load Shed Verification Relays will NOT function correctly
(2) Either the normal or alternate feeders to 4KV Bus 14 are closed

B. (1) Load Shed Verification Relays will NOT function correctly
(2) Both the normal and alternate feeders to 4KV Bus 14 are open

C. (1) Load Shed Verification Relays will function correctly
(2) Either the normal or alternate feeders to 4KV Bus 14 are closed

D. (1) Load Shed Verification Relays will function correctly
(2) Both the normal and alternate feeders to 4KV Bus 14 are open

Answer	C

Answer Explanation

A. Incorrect. White lights are normally on, indicating that the load shed relays are deenergized. As long as either the normal or alternate feeders for 14 4KV bus are closed, this is the normal condition. Plausible if candidate misinterprets the relays in the plant are normally energized.

B. Incorrect. Relays would be energized with both feeder breakers open, but light would be out. Plausible if candidate misinterprets the relays in the plant are normally energized and that the load shed verification is lit when a load shed should have occurred due to both breakers open but did not.

C. Correct. Per drawings, light is normally lit, indicating that either the normal or alternate power supply breaker is on the bus. Relays are normally de-energized.

D. Incorrect. Both feeder breakers being open would cause the white light to be extinguished. Plausible if candidate misinterprets that the load shed verification light is lit when a load shed has occurred with both breakers to the bus open.

Question Information

Topic	Q29 - How load shed verification light works				
User ID	Q2479778			**System ID**	2479778
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	Dwgs 61071sh0007, Rev 19 Dwgs 61071sh0008, Rev 20 Dwgs 61071sh0016, Rev 17 Dwg 61005, Rev 37		
Training Objective	Describe the 4KV design features and/or interlocks, including initiating devices, which provide for the following: SR 4Kv bus UV and load shed verification relay circuit.		
Previous NRC Exam Use	2021 NRC RO Exam		

K/A Reference(s)

SYS.062.K4.11	Safety Function	Tier 2	Group	RO Imp: 3.7	SRO Imp:
Knowledge of AC Electrical Distribution System design features and/or interlocks that provide for the following: (CFR: 41.7) Load shedding					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 30	ID: 2479635	Points: 1.00

While performing Containment Environment in EOP-0, Post Trip Immediate Actions, the following indications are observed:

- Rx Cavity Temperature is 130°F
- Containment Dome Temperature is 130°F

A short circuit failure occurs on the Containment Dome Temperature Instrument, causing its output to fail low.

What action, if any, is required per EOP-0 for containment temperature?

A. Verify all CACs are operating, open their Emergency Outlet Valves AND start all Iodine Filter Fans

B. Verify all CACs are operating and open their Emergency Outlet Valves ONLY

C. Start all Iodine Filter Fans ONLY

D. No actions required.

Answer	B

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets that both CACs and the iodine filter fans would help to reduce containment temperature.

B. Correct. Per EOP-0, when temperature is greater than 120°F all CACs are started and the emergency outlet valves are opened.

C. Incorrect. Plausible if the operator misinterprets that the iodine filter fans would help to reduce containment temperature.

D. Incorrect. Plausible if the operator misinterprets that with the failure containment temperature is now below the setpoint in EOP-0 and no actions are required.

Question Information

Topic	Q30 - Containment TC short circuit				
User ID	Q2479635			**System ID**	2479635
Status	Active	**Point Value**	1.00	**Time (min)**	4

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-0, Rev 01500		
Training Objective	For each EOP-0 Safety Function, analyze and determine correct alternate actions to take.		
Previous NRC Exam Use	None		

K/A Reference(s)

191002.K1.14	Safety Function	Tier	Group	RO Imp: 2.9	SRO Imp:
Failure modes of thermocouple, RTD, and/or thermometers					
G.SYS.022	Safety Function	Tier	Group	RO Imp:	SRO Imp:
CCS Containment Cooling System					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 31	ID: 2479661	Points: 1.00

Unit-1 is at 100% power with 2 Charging Pumps in operation.

- A 45 GPM RCS leak occurs
- The crew implements AOP-2A, Excessive Reactor Coolant Leakage
- The US directs the implementation of Block Step E, Attempt to Isolate the Leak
- The operator performs the first several steps of Block Step E and determines the leak is not yet isolated
- The operator establishes the appropriate number of Charging Pumps running to determine if the leak is on the Charging header

(1) When checking indications for a possible Charging header leak, what is the trend of Pressurizer level?

(2) If the leak is on the Charging header, how can it be positively identified?

A. (1) PZR level is rising.
(2) Lowering Pressurizer level with minimum Letdown flow.

B. (1) PZR level is rising.
(2) Charging header pressure less than Pressurizer pressure.

C. (1) PZR level is lowering.
(2) Lowering Pressurizer level with minimum Letdown flow.

D. (1) PZR level is lowering.
(2) Charging header pressure less than Pressurizer pressure.

Answer	D

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since 2 Charging Pumps were initially in operation and the leak rate is only 45 GPM. The operator may misinterpret which steps have already been performed and conclude that PZR level is rising. (2) Incorrect but plausible since a 45 GPM Charging header leak will cause lowering PZR level when Letdown flow is at minimum.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

D. Correct. (1) Correct since only one Charging Pump will be running when this action is taken. A 45 GPM leak will cause PZR level to lower. (2) Correct as stated in the AOP-2A Note prior to checking for a Charging header leak.

Question Information

Topic	Q31 - Charging header leak identification and PZR Level determination				
User ID	Q2479661			**System ID**	2479661
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	AOP-2A-1 Rev 03800		
Training Objective	Given plant conditions, diagnose Excessive Reactor Coolant Leakage and respond by directing and/or implementing the proper operator actions in accordance with the following procedures: AOP-2A.		
Previous NRC Exam Use	None		

K/A Reference(s)

CE.A16.AA2.04	Safety Function	Tier 1	Group	RO Imp: 3.3	SRO Imp: 3.9
Ability to determine and/or interpret the following as they apply to Excess RCS Leakage: (CFR: 41.10 / 43.5 / 45.13) PZR level and pressure					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 32	ID: 2479704	Points: 1.00

The SRW Rad monitor, RMS-1595, has alarmed and SRW head tank level is slowly rising.

Which of the following could be the source of the problem?

A. CVCS Letdown HX

B. 11 SG Blowdown HX

C. 11 Containment Air Cooler

D. 11 Spent Fuel Pool Cooling System

Answer	D

Answer Explanation

A. Incorrect. Plausible if candidate misinterprets the cooling medium to letdown and determines that due to a pressure differential the leak from letdown would be into the SRW system.

B. Incorrect. Plausible since a leak from a blowdown HX could cause an RMS alarm but SRW does not cool 11 BDHX (only 12 BDHX).

C. Incorrect. Plausible since CACs are cooled by SRW but a leak would be into the containment vice into SRW due to the pressure differential.

D. Correct. Per 1C22-ALM, SFP coolers are cooled by SRW. SRW is throttled such that SFP water is at a higher pressure that SRW. Therefore, a leak would be into the SRW system.

Question Information

Topic	Q32 - SRW Rad monitor has alarmed what is most likely source?				
User ID	Q2479704			**System ID**	2479704
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.		

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	1C22-ALM, Rev 04500		
Training Objective	Identify the feature of the SRW system that provides the capability to monitor for radioactivity, and the alarms associated with that feature.		
Previous NRC Exam Use	2020 NRC		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Reference(s)

SYS.073.K1.08	Safety Function	Tier 2	Group	RO Imp: 3.1	SRO Imp:
Knowledge of the physical connections and/or cause and effect relationships between the Process Radiation Monitoring System and the following systems: (CFR: 41.7 to 41.9 / 41.11 / 45.8 / 45.9) SWS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 33	ID: 2480002	Points: 1.00

Per OP-CA-101-111-0200, Shift Complement and Responsibilities:

(1) Can an extra licensed Reactor Operator, who is not assigned another required position, be designated the Operations Technical Advisor (OTA) when the 5 person fire brigade does not have the full qualifications required?

(2) Can an extra licensed Reactor Operator, who is not assigned another required position, fill the role of an Equipment Operator, for the purpose of safe shutdown and equipment rounds, that they were previously qualified as?

A. (1) No
(1) No

B. (1) No
(1) Yes

C. (1) Yes
(2) No

D. (1) Yes
(2) Yes

Answer	D

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since all 5 members of the fire brigade are required to maintain fire training qualification per OP-AA-201-005 which the licensed Reactor Operators do not possess. In addition, the licensed Reactor Operators do not possess the fire brigade qualifications in their LMS matrix. (2) Incorrect but plausible since the licensed Reactor Operators do not possess the Equipment Operator qualifications in their LMS matrix. Those qualifications have been unassigned to them in the LMS matrix.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

D. Correct. (1) Correct as stated in OP-CA-101-111-0200 Section 4.1.8. (2) Correct as stated in OP-CA-101-111-0200 Section 4.1.9.

Question Information

Topic	Q33 - RO Responsibilities Outside the Control Room				
User ID	Q2480002			**System ID**	2480002
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	OP-CA-101-111-0200 Rev 002		
Training Objective	Recall the responsibilities of a licensed operator per OP-CA-101-111-0200.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.4.34	Safety Function	Tier 3	Group	RO Imp: 4.2	SRO Imp: N/A
Knowledge of RO responsibilities outside the main control room during an emergency (CFR: 41.10 / 43.5 / 45.13)					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 34	ID: 2479880	Points: 1.00

A Waste Gas Release per OI-17B of 11 Waste Gas Decay Tank to the Unit-2 Plant Vent Stack is in progress when the following occurs:

- The Gaseous Waste Discharge Radiation Monitor 0-RI-2191 indication exceeds the High Alarm setpoint

(1) What automatic actions will occur to secure the Waste Gas Discharge?

(2) What action is required to prevent lifting the Waste Gas Discharge Header Relief Valve from going to the Waste Gas Surge Tank?

A. (1) The Waste Gas Discharge valves, 0-WGS-2191-CV and 0-WGS-2192-CV, will shut.
(2) Shut Waste Gas Discharge To U-2 Plant Vent, 0-WGS-684.

B. (1) Both 11 and 12 Waste Gas Compressors will trip.
(2) Shut the Waste Gas Discharge Header Flow Control Valve, 0-WGS-2191-PCV.

C. (1) The Waste Gas Discharge valves, 0-WGS-2191-CV and 0-WGS-2192-CV, will shut.
(2) Shut the Waste Gas Discharge Header Flow Control Valve, 0-WGS-2191-PCV.

D. (1) Both 11 and 12 Waste Gas Compressors will trip.
(2) Shut Waste Gas Discharge To U-2 Plant Vent, 0-WGS-684.

Answer	C

Answer Explanation

A. Incorrect. A High Alarm condition on 0-RI-2191 will automatically shut both Waste Gas Discharge Valves; Plausible when the candidate recalls that 0-WGS-684 is the final valve in the lineup to isolate the release to the environment through the Main Plant Vent Stack. But this action is not directly per the procedure or alarm manual in response to a high alarm condition on 0-RI-2191 and will not prevent lifting the referenced Relief Valve.

B. Incorrect. Plausible when the candidate recalls that a Waste Gas Compressor Leak will automatically trip the Waste Gas Compressor. Also, plausible if the candidate associates the Waste Gas Compressors as the motive force used during the Waste Gas

Decay Tank release; The second part is the correct action per OI-17B and the 1C22 Alarm Manual in response to a high alarm condition and for the exact reason stated.

C. Correct. A High Alarm condition on 0-RI-2191 will automatically shut both Waste Gas Discharge Valves; The second part is the correct action per OI-17B and the 1C22 Alarm Manual in response to a high alarm condition and for the exact reason stated.

D. Incorrect. Plausible when the candidate recalls that a Waste Gas Compressor Leak will automatically trip the Waste Gas Compressor. Also, plausible if the candidate associates the Waste Gas Compressors as the motive force used during the Waste Gas Decay Tank release; Plausible when the candidate recalls that 0-WGS-684 is the final valve in the lineup to isolate the release to the environment through the Main Plant Vent Stack. But this action is not directly per the procedure or alarm manual in response to a high alarm condition on 0-RI-2191 and will not prevent lifting the referenced Relief Valve.

Question Information

Topic	Q34 - Accidental Gaseous Release valve lineup				
User ID	Q2479880			**System ID**	2479880
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	OI-17B, Rev 02600 60735SH0001, Rev 0049 60735SH0002, Rev 0047		
Training Objective	Given conditions, determine if the waste gas system is operating properly.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.071.K4.04	Safety Function	Tier 2	Group	RO Imp: 3.1	SRO Imp:
Knowledge of Waste Gas Disposal System design features and/or interlocks that provide for the following: (CFR: 41.7) Isolation of waste gas release tanks					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 35	ID: 2479720	Points: 1.00

Unit-2 is in Mode 5 with 21 LPSI pump running for SDC.

- RCS pressure at 200 psia
- RCS temperature at 140°F
- The plant has been shutdown for 3 days
- 22 LPSI pump is out of service
- 21 LPSI pump breaker trips on overcurrent and cannot be restarted

Which of the following actions would be taken **FIRST** to restore core cooling?

A. Establish core cooling by bleeding steam from the steam generators.

B. Reduce RCS pressure and line up a Containment Spray Pump to provide shutdown cooling.

C. Start a HPSI pump and open the PORVs to provide core cooling via RCS blowdown to containment.

D. Start all Charging Pumps and open the PORVs to provide core cooling via RCS blowdown to containment.

Answer	B

Answer Explanation

A. Incorrect. Incorrect since heatup to a higher mode should be avoided, and AOP-3B directs other actions first. Plausible since the operator may recall that Section VI is used when pressurization of the RCS is possible, and Steam Generators can be used for heat removal.

B. Correct. Per AOP-3B Preliminary Step C, RCS pressure is reduced below 170 psia and then a Containment Spray pump is aligned to provide shutdown cooling flow.

C. Incorrect. Plausible since the operator may recall that Section VI Block Step H directs the use of Once Through Cooling when the Steam Generators are not available, and a common mode failure of shutdown cooling exists. In this step, a HPSI is used to control RCS level.

D. Incorrect. Plausible since the operator may recall that Section VI Block Step H directs the use of Once Through Cooling when the Steam Generators are not available, and a common mode failure of shutdown cooling exists. In this step, all Charging Pumps are used to control RCS level if a HPSI pump is not available.

Question Information

Topic	Q35 - Loss of SDC with SDC flowpath available				
User ID	Q2479720			**System ID**	2479720
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	AOP-3B-2 Rev 02900		
Training Objective	Recall the steps per AOP-3B for restoring SDC flow when neither LPSI pump is available.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Reference(s)

SYS.026.K6.08	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Containment Spray System: (CFR: 41.7 / 45.7) RHRS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 36	ID: 2479691	Points: 1.00

Unit-2 was at 100% power when a Station Blackout occurred.

- 11, 12, 21, and 22 DC Bus voltages are 129 VDC and lowering
- No ELAP actions have begun

(1) What is the design coping time before the batteries reach minimum voltage?

(2) What component(s) would be affected as the minimum bus voltage is reached?

A. (1) 4 hours
(2) AFW Flow Control Valves start to fail open at 1C04.

B. (1) 4 hours
(2) Secondary CEA display goes dark at 1C05.

C. (1) 7 hours
(2) AFW Flow Control Valves start to fail open at 1C04.

D. (1) 7 hours
(2) Secondary CEA display goes dark at 1C05.

Answer	A

Answer Explanation

A. Correct. (1) Per EOP-7 Bases and FSG-4 Bases the batteries are designed to last 4 hours in SBO before minimum voltage of 105 VDC is reached and no longer is sufficient for supporting instrumentation and controls. ELAP actions to minimize loading extends batteries to at least 7 hours. (2) One of those controls lost will be the AFW Flow Control Valves, which will begin to fail open, regardless of air accumulator status.

B. Incorrect. (1) Correct. (2) CEA indications is plausible since all CEA indications on 1C05 and inputs to the PPC are lost during a SBO. However, the CEA indications are lost due to a loss of vital 4KV busses, not vital DC.

C. Incorrect. (1) Plausible since ELAP actions to minimize loading extends batteries to at least 7 hours. (2) Correct.

D. Incorrect. (1) See above. (2) See above.

Question Information

Topic	Q36 - SBO Battery limits				
User ID	Q2479691			**System ID**	2479691
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

<<References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	**3**
Technical Reference and Revision #	EOP-07-02 Rev 02200 EOP-07-TB Rev 02400 FSG-4-TB Rev 001		
Training Objective	RECALL from memory the operation of the following components during station blackout conditions per EOP-7: • Atmospheric Dump Valves (ADVs) • AFW Steam Supply Valves • AFW Flow Control Valves		
Previous NRC Exam Use	2019 NRC RO Exam		

K/A Reference(s)

SYS.063.A4.03	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Battery discharge rate					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 37	ID: 2479939	Points: 1.00

What is the power supply to the following components?

(1) 22 Proportional Heater Bank

(2) 23 Backup Heater Bank

A. (1) 21B 480V Load Center
(2) 24B 480V Load Center

B. (1) 24A 480V Load Center
(2) 24B 480V Load Center

C. (1) 24A 480V Load Center
(2) 23B 480V Load Center

D. (1) 21B 480V Load Center
(2) 23B 480V Load Center

Answer	B

Answer Explanation

A. Incorrect - 1st part is incorrect as stated below. 2nd part is correct as stated below.

B. Correct - 1st and 2nd parts are correct as stated in OI-27D-2 for each power supply.

C. Incorrect - 1st part is correct as stated above. 2nd part is incorrect as stated below.

D. Incorrect - 1st part is incorrect but plausible since the operator may interpret the power supply to the Proportional Heaters to be safety related and powered from the B side of 21 480V Load Center. The Proportional Heaters are not Tech Spec equipment, and the operator may conclude that the power supply is from a non Tech Spec credited source. 2nd part is incorrect but plausible since the operator may recall that 21 and 24 are the Backup Heaters credited per the Tech Specs which would coincide with the 21 and 24 4KV Buses. The operator may easily misinterpret the numbering scheme of Backup Heaters which are different than many other plant systems.

Question Information

Topic	Q37 - PZR Heater Power Supply				
User ID	Q2479939			**System ID**	2479939
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	OI-27D-2 Rev 02500		
Training Objective	Determine the effect on plant operation upon a loss of any 480 VAC power supplies.		
Previous NRC Exam Use	2019 NRC RO Exam		

K/A Reference(s)

SYS.011.K2.02	Safety Function	Tier 2	Group	RO Imp: 3.3	SRO Imp:
Knowledge of electrical power supplies to the following: (CFR: 41.7) PZR heaters					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 38	ID: 2479559	Points: 1.00

Which of the following radiation monitors must be operable to ensure the Containment Purge System will be automatically secured should a fuel handling incident occur inside the Containment?

Why is containment purge secured on a fuel handling incident in containment?

A. Containment Area Radiation Monitors (RE-5316 A thru D)
To reduce the release of activity from the containment.

B. Containment Area Radiation Monitors (RE-5316 A thru D)
To maintain a positive pressure in the containment.

C. Containment High Range Monitors (RE-5317 A/B)
To maintain a positive pressure in the containment.

D. Containment High Range Monitors (RE-5317 A/B)
To reduce the release of activity from the containment.

Answer	A

Answer Explanation

A. Correct.(1) Per OI-36, Containment Purge System: IF moving irradiated fuel assemblies within the containment, THEN all four channels of Containment Area Radiation Monitors RI-5316A, B, C, and D are operable on the unit to be purged. (1) Per AOP-6D, containment purge is isolated to reduce activity released.

B. Incorrect. (1) Correct. (2) See below

C. Incorrect. (1) Plausible if the operator misinterprets which containment radiation monitors control purge. (2) Plausible if the operator misinterprets which type of pressure control is maintained within the aux building.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Q38 - Which instrument ensures that Cntmt Purge will be secured on a Fuel Handling Incident?				
User ID	Q2479559			**System ID**	2479559
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.13 Procedures and equipment available for handling and disposal of radioactive materials and effluents.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-6D-TB, Rev 01500 TS 3.3.7, Rev 279		
Training Objective	Given plant conditions, diagnose a Fuel Handling Incident and respond by directing and/or implementing the proper operator actions in accordance with the following procedures:AOP-6D and AOP-6D Basis.		
Previous NRC Exam Use	2008 NRC RO Exam		

K/A Reference(s)

APE.036.AK3.04	Safety Function	Tier 1	Group	RO Imp: 3.9	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to Fuel Handling Incidents: (CFR: 41.10 / 45.6 / 45.13) Establishing containment isolation or closure					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 39	ID: 2479759	Points: 1.00

Using provided references:

Initial S/G Pressure is 900 PSIA
Initial Indicated Level is 0 inches

Final S/G Pressure is 100 PSIA
Final Indicated Level is +10 Inches

What is the change in ACTUAL S/G level?

A. Raises by 2 inches

B. Lowers by 10 inches

C. Raises by 10 inches

D. Lowers by 18 inches

Answer	B

Answer Explanation

A. Incorrect. Plausible if the operator uses a change in indicated level of 10 inches and a change in S/G pressure to 500 PSIA.

B. Correct. Per OI-12B-1 Figure 1, an indicated level of +10 inches at a S/G pressure of 100 PSIA, the actual level equals -10 inches. So, the overall change is actual S/G level lowers by 10 inches.

C. Incorrect. Plausible if the operator assumes that indicated level tracks directed with a change in actual level or if the operator determines the actual level using the line for a S/G pressure of 900 PSIA.

D. Incorrect. Plausible if the operator uses a change in indicated level of 10 inches and a change in S/G pressure to 14.7 PSIA.

Question Information

Topic	Q39 - Reference - SG Actual vs Indicated Levels				
User ID	Q2479759			**System ID**	2479759
Status	Active	**Point Value**	1.00	**Time (min)**	5

Open or Closed Reference	OI-12B-1 FIG 1	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	OI-12B-1 Figure 1, page 121
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	OI-12B-1 Rev 01700		
Training Objective	Using provided references, determine the actual S/G level compared to indicated level.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.059.A1.05	Safety Function	Tier 2	Group	RO Imp: 3.5	SRO Imp:
Ability to predict and/or monitor changes in parameters associated with operation of the Main Feedwater System, including: (CFR: 41.5 / 45.5) S/G level and comparison with normal values					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 40	ID: 2479882	Points: 1.00

(1) What is the MINIMUM distance required to be maintained from a MCC breaker to prevent inadvertent bumping/mispositioning of the breaker without the need for additional briefings per HU-AA-101, Human Performance Tools and Verification Practices?

(2) If a MCC breaker was bumped and mispositioned, what tag would be hung on the breaker to regain configuration control?

A. (1) 9 inches
(2) Issue Report Equipment Deficiency Tag

B. (1) 2 Feet
(2) Issue Report Equipment Deficiency Tag

C. (1) 9 inches
(2) Equipment Status Tag

D. (1) 2 Feet
(2) Equipment Status Tag

Answer	D

Answer Explanation

A. Incorrect. (1) 9" is plausible as this is Flash Protection boundary in SA-CA-129-1001 for a 51-239V breaker. However, per HU-AA-101, 2 feet is the minimum distance maintained to prevent inadvertent bumping or mispositioning of a component without the need for additional briefings or a 2-minute drill. (2) An Equipment Deficiency Tag is plausible since an Issue Report would be written for the condition and EDTs are hung in the field to help communicate equipment deficiencies. However, a mispositioned component would not be considered an equipment deficiency. Per OP-AA-108-101, an EST is used to identify temporary abnormal equipment position to ensure configuration control.

B. Incorrect. (1) Correct. (2) See above.

C. Incorrect. (1) See above. (2) Correct.

D. Correct. (1) Per HU-AA-101, 2 feet is the minimum distance maintained to prevent inadvertent bumping or mispositioning of a component without the need for additional briefings or a 2-minute drill. (2) Per OP-AA-108-101, an EST is used to identify temporary abnormal equipment position to ensure configuration control.

Question Information

Topic	Q40 - Configuration Control				
User ID	Q2479882			**System ID**	2479882
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	OP-AA-108-101, Rev 018 HU-AA-101, Rev 013		
Training Objective	Recognize and model proper HU tool usage IAW Constellation HU Procedures.		
Previous NRC Exam Use	2018 NRC RO Exam		

K/A Reference(s)

P2.2.14	Safety Function	Tier 3	Group	RO Imp: 3.9	SRO Imp: 4.3
Knowledge of the process for controlling equipment configuration or status (CFR: 41.10 / 43.3 / 45.13)					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 41	ID: 2480023	Points: 1.00

Unit-1 is at 100% power.

- 11 Component Cooling Pump is running
- 12 Component Cooling Heat Exchanger outlet, 1-CC-3826-CV is being shut for maintenance
- A dedicated operator is NOT being assigned to reopen it in the event SIAS actuates

What action is required?

A. Place only 12 Component Cooling Pump in Pull-to-Lock

B. Place only 13 Component Cooling Pump in Pull-to-Lock

C. Place 12 and 13 Component Cooling Pumps in Pull-To-Lock

D. Place 12 Component Cooling Pump in Pull-to-Lock and shift 13 Component Cooling Pump power supply to 14B bus

Answer	C

Answer Explanation

A. Incorrect. Plausible if the operator determines that since 13 CC pump is the swing pump that only 12 CC pump has auto start signals on SIAS and that it is the only pump that needs to be placed in PTL.

B. Incorrect. Plausible if the operator misinterprets the caution a determines that only 2 CC pumps shall be operated with only 1 CCHX so only 1 pump needs to be placed in PTL.

C. Correct. Per OI-16 caution, 2 CC pumps shall not be operated with only 1 CCHX aligned for service. So, 2 CC pumps need to be placed in PTL.

D. Incorrect. Plausible if the operator misinterprets the caution for why 3 CC pumps should not be run at the same time and determines similar actions are required for single CCHX operation.

Question Information

Topic	Q41 - Single CCW HX operation				
User ID	Q2480023			**System ID**	2480023
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	2
Technical Reference and Revision #	OI-16, Rev 04200		
Training Objective	Shift to single Component Cooling Heat Exchanger operation and take actions to prevent 2 Component Cooling Pump operation on a single Heat Exchanger per OI-16 and the plaque on 1(2)C13.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.2.18	Safety Function	Tier 3	Group	RO Imp: 2.6	SRO Imp: 3.9
Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments and work prioritization (CFR: 41.10 / 43.5 / 45.13)					
G.SYS.008	Safety Function	Tier	Group	RO Imp:	SRO Imp:
CCW Component Cooling Water System					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 42	ID: 2479702	Points: 1.00

Unit-1 is operating at 100% power.

- The 1A EDG is paralleled to the 11 4KV bus per STP O-008A-1 Test of 1A DG and 11 4KV BUS UV

Then, the following local alarms are received for the 1A EDG:

LOSS OF EXCITATION [40]

GENERATOR OVERVOLTAGE [59/2]

GENERATOR BRG TEMP "HI-HI"

How will the EDG system respond?

A. The 1A EDG will continue to run and the EDG output breaker remains closed

B. The 1A EDG will continue to run and the EDG output breaker will open

C. 1A EDG trips and the EDG output breaker will open

D. 1A EDG trips and the EDG output breaker remains closed

Answer	C

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets that an emergency start signal is present.

B. Incorrect. Plausible if the operator misinterprets what the Overvoltage alarm does and determines that the EDG will not trip.

C. Correct. Per 1C188-ALM, these alarms will trip both the EDG and the output breaker.

D. Incorrect. Plausible if the operator misinterprets what the loss of excitation alarm does and determines the output breaker will remain closed.

Question Information

Topic	Q42 - Loss of Excitation alarm and impact on 1A EDG during a normal run				
User ID	Q2479702			**System ID**	2479702
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	1C188-ALM, Rev 00900 OI-21A-1, Rev 02700		
Training Objective	Given 1A Diesel Generator load, power factor, voltage, and frequency, determine if parameters are within limits IAW station procedures.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.064.A2.07	Safety Function	Tier 2	Group	RO Imp: 3.2	SRO Imp: 3.5
Ability to (a) predict the impacts of the following on the Emergency Diesel Generators 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) Overexcitation or underexcitation					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 43	**ID: 2479703**	**Points: 1.00**

Given a SIAS actuation signal and Pressurizer level has been restored from 50 inches to 120 inches:

Which of the following sets of Pressurizer Heaters will automatically reenergize?

A. Proportional heaters and two banks of backup heaters.

B. Proportional heaters ONLY.

C. All backup heater banks.

D. Two banks of backup heaters ONLY.

Answer	**D**

Answer Explanation

A. Incorrect - Plausible since PZR level has been restored to above the low level heater cutoff. Incorrect since the Proportional Heaters must be reset by placing their handswitches to off and then back to auto; however, two banks of backup heaters are energized as remaining two banks of Backup Heaters are de-energized due to SIAS.

B. Incorrect - Plausible since the operator may misinterpret which heaters remain de-energized with a SIAS signal present. The Proportional Heaters must be reset by placing their handswitches to OFF and then back to AUTO.

C. Incorrect - Plausible since the operator may misinterpret which heaters remain de-energized with a SIAS signal present. Two banks of Backup Heaters are de-energized due to SIAS.

D. Correct - Two banks of Backup Heaters are de-energized due to the SIAS, the remaining two banks would return to the auto operation mode when the Pressurizer Low Level Heater cutout cleared at ~ 101 inches and rising in the Pressurizer.

Question Information

Topic	Q43 - Pressurizer Heater operation and SIAS				
User ID	Q2479703			**System ID**	2479703
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	63058ASH0001 Rev 59		
Training Objective	Determine the status and operation of the following plant systems upon a SIAS actuation: PZR heaters.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.010.K6.13	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp:
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Pressurizer Pressure Control System: (CFR: 41.7 / 45.7) ESFAS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 44	ID: 2479561	Points: 1.00

Using provided references:

Unit 1 is at 100% power.

- A 100 GPM SG Tube leak occurs
- AOP-2A, excessive Reactor Coolant Leakage is implemented
- Unit-1 is manually tripped at 537°F Tave
- EOP-0, Post Trip Immediate Actions, is implemented
- A Loss of Offsite Power and a LOCA in containment occur
- SIAS failed to automatically actuate
- The manual pushbuttons work to actuate SIAS
- EOP-8, Functional Recovery, is implemented

What is the success path for Pressure and Inventory Control?

A. PIC-1

B. PIC-2

C. PIC-3

D. PIC-4

Answer	D

Answer Explanation

A. Incorrect. Plausible if the operator determines that since SIAS failed to operate then no SIAS exists.

B. Incorrect. Plausible if the operator determines that aux spray is not available due to the LOOP and that PORVs are needed to lower RCS pressure.

C. Incorrect. Plausible since PIC-3 is titled LOOP and the operator determines that PIC-3 is required anytime there is a LOOP. Plausible if the operator determines that since SIAS failed to operate then no SIAS exists.

D. Correct. SIAS may not have actuated, but it needed to be actuated, and HPSI injection is needed. Per EOP-8, PIC-4 is the correct success path.

Question Information

Topic	Q44 - Reference - PIC with failure of SIAS				
User ID	Q2479561			System ID	2479561
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	EOP-8-1, PAGES 32-33	Cognitive Level	HIGH
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	PIC RAT, pages 32-33
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	EOP-8-1, Rev 04400		
Training Objective	Given a set of plant conditions, identify the success paths for all safety functions, including the order of priority, in accordance with EOP-8.		
Previous NRC Exam Use	None		

K/A Reference(s)

CE.E09.EK1.07	Safety Function	Tier 1	Group	RO Imp: 4.3	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Functional Recovery: (CFR: 41.5 / 41.7 / 45.7 / 45.8 / 45.9) Evaluating the RCS pressure control safety function and implementing the correct success path for plant conditions					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 45	ID: 2479491	Points: 1.00

When implementing EOP-0 Alternate Actions for an ATWS, which of the following lists the parameters/indications specified in EOP-0 to check the reactor has tripped?

A. NI power ONLY

B. Startup Rate and Delta-T power

C. NI power and Delta-T power

D. NI power and Startup Rate

Answer	D

Answer Explanation

A. Incorrect - NI Power is correct as stated in EOP-0. Plausible if operator misinterprets that ONLY NI power needs to be looked at to verify that the reactor is tripped since 0% power could indicate a reactor trip. EOP-0 ensures redundant indications are used to validate a reactor trip.

B. Incorrect - Startup Rate is correct as stated in EOP-0 and its Technical Basis. Delta-T power is plausible since the operator will recall the use of redundant indications to validate plant parameters. If a reactor trip is not readily apparent, the operator may use Delta-T power as a redundant indication. But EOP-0 Technical Basis states that NI Power indication was chosen so that RCS temperature impacts do not interfere with the verification of reactor power.

C. Incorrect - Delta-T power is plausible since the operator will recall the need to use redundant indications to validate plant parameters. If a reactor trip is not readily apparent, the operator may use Delta-T power as a redundant indication. But EOP-0 Technical Basis states that NI Power indication was chosen so that RCS temperature impacts do not interfere with the verification of reactor power.

D. Correct - Per EOP-0, a prompt drop in NI Power and a negative startup rate are used to verify the reactor is tripped for a normal trip and for an ATWS.

Question Information

Topic	Q45 - Indications used to verify a reactor trip has occurred				
User ID	Q2479491			System ID	2479491
Status	Active	Point Value	1.00	Time (min)	2

Open or Closed Reference	CLOSED	Cognitive Level	LOW
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.5 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.		

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	EOP-0-1 Rev 01500		
Training Objective	Determine if a reactor trip has occurred or should have occurred.		
Previous NRC Exam Use	2019 NRC RO Exam		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Reference(s)

EPE.029.EA2.14	Safety Function	Tier 1	Group	RO Imp: 4.6	SRO Imp: 4.3
Ability to determine and/or interpret the following as they apply to an Anticipated Transient Without Scram: (CFR: 43.5 / 45.13) Occurrence of a reactor trip					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 46	ID: 2479607	Points: 1.00

Unit-2 is at 100% power when the following alarm is received:

WEST ECCS PP RM LVL HI

- The ABO looking through the Component Cooling Room access hatch observes a significant amount of water in the room and continuing to rise
- The CRO observes 21 Refueling Water Tank (RWT) level is 462 inches and lowering

(1) Which Containment Spray pump must be placed in PTL?

(2) If unable to place the CS pump in PTL, where could the breaker be locally operated?

A. (1) 21 CS Pump
(2) 22 4KV Bus

B. (1) 21 CS Pump
(2) 24 4KV Bus

C. (1) 22 CS Pump
(2) 24 4KV Bus

D. (1) 22 CS Pump
(2) 22 4KV Bus

Answer	C

Answer Explanation

A. Incorrect. (1) Plausible if the operator misinterprets which pump is in the west ECCS pump room. (2) Plausible if the operator misinterprets that the normal power numbering scheme exists for this pump.

B. Incorrect. (1) See above. (2) Correct.

C. Correct. (1) Per OI-3A, 22 CS Pump is in the West ECCS Pump Room. (2) Per AOP-7I, 22 CS Pump is powered from 24 4KV Bus.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Q46 - ECCS header rupture impact on Containment Spray				
User ID	Q2479607			**System ID**	2479607
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	AOP-7I, Rev 03600 OI-3A, Rev 04100		
Training Objective	Given an alarm, demonstrate proper alarm response and status control per applicable procedure.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.006.K3.03	Safety Function	Tier 2	Group	RO Imp: 3.8	SRO Imp:
Knowledge of the effect that a loss or malfunction of the Emergency Core Cooling System will have on the following systems or system parameters: (CFR: 41.7 / 45.3 / 45.4 / 45.6) CSS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 47	ID: 2479745	Points: 1.00

Unit-2 is operating at 100% when 22 ADV fails full open

(1) What is the impact on the Main Steam system?

(2) What actions in AOP-7K, Overcooling Event in Mode 1 or 2, should be taken first?

A. (1) Main Steam header flow rises ~2.5%.
(2) Trip the Reactor.

B. (1) Main Steam header flow rises ~2.5%.
(2) Insert CEAs to restore power.

C. (1) Main Steam header flow rises ~10%.
(2) Trip the Reactor.

D. (1) Main Steam header flow rises ~10%.
(2) Insert CEAs to restore power.

Answer	B

Answer Explanation

A. Incorrect. (1) Correct as stated below. (2) Incorrect but plausible since tripping the reactor is the AOP-7K action if power cannot be controlled and a reactor trip is imminent. The RO will be able to control power < the trip setpoint since one ADV will only raise steam flow by 2.5%.

B. Correct. (1) Correct. ADVs combined capacity equates to 5% total steam flow, with 1 ADV passing a max of 2.5% flow. (2) Correct. Per AOP-7K, initial steps would be to lower power by inserting CEAs or borating.

C. Incorrect. (1) Incorrect. 10% rise in steam flow is plausible since this is the main steam flow increase from a TBV. (2) Incorrect as stated above.

D. Incorrect. (1) Incorrect. 10% rise in steam flow is plausible since this is flow increase from a TBV. ADVs combined capacity only equate to 5% total steam flow, with 1 ADV passing a max of 2.5% flow. (2) Correct as stated above.

Question Information

Topic	Q47 - ADV Failure and Response				
User ID	Q2479745			System ID	2479745
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	CLOSED	Cognitive Level	LOW
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	AOP-7K-2 Rev 0300		
Training Objective	Given plant conditions, an event in progress, and several alternate actions, determine the most appropriate actions in accordance with AOP-7K.		
Previous NRC Exam Use	2018 NRC RO Exam		

K/A Reference(s)

SYS.039.A2.05	Safety Function	Tier 2	Group	RO Imp: 4.5	SRO Imp: 4.0
Ability to (a) predict the impacts of the following on the Main and Reheat Steam 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) Increasing steam demand and its relationship to increases in reactor power					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 48	ID: 2479414	Points: 1.00

A reactor trip from full power has occurred, and the following conditions exist:

- Pressurizer level is 140 inches and stable
- One Charging Pump is available
- Pressurizer pressure is 1900 PSIA and slowly rising
- RCS Temperature is at 532°F and being maintained by TBV operation
- RCS Subcooling is 65°F and slowly rising
- All immediate actions for PIC safety function have been completed
- The Reactor Operator reports "Pressure and Inventory Control is NOT met" to the Unit Supervisor

Which of the following conditions supports the Reactor Operator safety function status report?

A. Letdown has been isolated.

B. All Charging Pumps are NOT in operation.

C. RCS subcooling should be higher based on RCS pressure.

D. Pressurizer level is not trending toward T_{AVG} programmed level setpoint.

Answer	D

Answer Explanation

A. Incorrect. Plausible if the operator determines that letdown in service is needed for PIC to be met since this is the normal operating condition for CVCS.

B. Incorrect. Plausible if the operator determines that all 3 charging pumps have to be available for PIC to be met since this the normal operating condition for CVCS.

C. Incorrect. Plausible if the operator determines that subcooling should be higher it will still be in the normal band to call PIC met based on subcooling.

D. Correct. per EOP-0, Pressurizer level should be between 80 - 180" and TRENDING towards 160".

Question Information

Topic	Q48 - Evaluation of PIC Safety Function status				
User ID	Q2479414			**System ID**	2479414
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-0, Rev 01500		
Training Objective	Given a plant condition requiring a reactor trip the license candidate will demonstrate an understanding of the strategy, basis and operator actions of EOP-0 to direct or implement the procedural steps, including warnings, notes, and cautions.		
Previous NRC Exam Use	2004 NRC RO Exam		

K/A Reference(s)

CE.E02.EA2.06	Safety Function	Tier 1	Group	RO Imp: 3.8	SRO Imp: 3.9
Ability to determine and/or interpret the following as they apply to Standard Post-Trip Actions and Reactor Trip Recovery: (CFR: 43.5 / 45.13) PZR level and pressure					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 49	ID: 2479489	Points: 1.00

Unit-1 is at 100% power when the following occurs:

- 11 4KV Bus Normal Feeder breaker trips open
- The 11 4KV Bus is repowered with its associated EDG
- No operator actions are performed

Then, the in-service Pressurizer Pressure Controller (PIC-100) output signal fails to 75%, resulting in:

- Main Spray Valves opening

Which of the following describes the Proportional Heaters Response to this event without operator action?

A. Bank 1 auto-energizes
Bank 2 output goes to minimum

B. Bank 1 remains deenergized
Bank 2 output goes to minimum

C. Bank 1 auto-energizes
Bank 2 output goes to maximum

D. Bank 1 remains deenergized
Bank 2 output goes to maximum

Answer	B

Answer Explanation

A. Incorrect. (1) See below (2) Correct.

B. Correct. (1) #11 Proportional Heater will remain de-energized due to the momentary loss of 11 4Kv Bus, until its handswitch is manually reset. (2) As PIC output goes to 75%, #12 Proportional Heater output will go to zero since its handswitch has remained in its normal position of Auto.

C. Incorrect. (1) Plausible if the proportional heater control circuit is thought to auto start when heaters are required due to lowering pressurizer pressure. When the 4KV bus is re-powered from the EDG the handswitch for the proportional heater has to be manually reset. (2) Plausible since normally when pressurizer pressure lowers the proportional heaters will go to maximum output but not for this failure mode.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Q49 - Proportional Heater Response				
User ID	Q2479489			System ID	2479489
Status	Active	Point Value	1.00	Time (min)	2

Open or Closed Reference	CLOSED	Cognitive Level	HIGH
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	USFAR, Chapter 7 (figure 7-13), Rev 52 AOP-7I-1, Rev 03700		
Training Objective	From memory, recall the operation and basis of the following in relation to the RCS in accordance with the Tech Specs, UFSAR and Operating Procedures: PZR Level/Pressure Control.		
Previous NRC Exam Use	2020 NRC RO Exam		

K/A Reference(s)

APE.027.AA1.06	Safety Function	Tier 1	Group	RO Imp: 3.6	SRO Imp:
Ability to operate and/or monitor the following as they apply to a Pressurizer Pressure Control System Malfunction: (CFR: 41.7 / 45.5 / 45.6) Operable control channel					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 50	ID: 2479560	Points: 1.00

AOP-9A allows the Turbine Building Operator to control AFW discharge pressure locally if required. The operator at the remote shutdown panel, 1C43, can verify adequate AFW pump speed by:

A. Comparing local (AFW pump room) AFW pump discharge pressure as reported by the TBO, with 11 and 12 S/G pressure indications at 1C43.

B. Comparing AFW pump discharge pressure at 1C43, with local (AFW pump room) 11 and 12 S/G pressure indications as reported by the TBO.

C. Comparing local (AFW pump room) AFW pump speed indications as reported by the TBO, with AFW pump flow indications at 1C43.

D. Comparing local (AFW pump room) AFW pump flow as reported by the TBO, with Main Steam header flow indications at 1C43.

Answer	A

Answer Explanation

A. Correct. The indications are available as listed.

B. Incorrect. Plausible if the operator misinterprets that SG pressure can be monitored in the AFW Pump room.

C. Incorrect. Plausible if the operator misinterprets where pump speed and pump flow can be monitored.

D. Incorrect. Plausible if the operator misinterprets that AFW flow can be monitored in the AFW pump room.

Question Information

Topic	Q50 - How does operator control AFW pump speed				
User ID	Q2479560			**System ID**	2479560
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.		
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-9A, Rev 02000		
Training Objective	Operate the AFW system components as required to maintain RCS temperatures and SG levels to support AOP-9A plant shutdown.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.068.AK2.11	Safety Function	Tier 1	Group	RO Imp: 4.0	SRO Imp:
Knowledge of the relationship between Control Room Evacuation and the following systems or components: (CFR: 41.7 / 45.7) AFW					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 51	ID: 2479492	Points: 1.00

Unit-1 experienced a SW rupture which required use of the emergency return discharge header.

Why does AOP-7A, Loss of Saltwater Cooling, Attachment 2 contain a Caution regarding how 12 CC HX temperature is controlled when the SW emergency return discharge header is in service?

A. 12 Component Cooling HX temperature is controlled by locally adjusting 12 CC Heat Exchanger Outlet, 1-SW-5206-CV.

B. 12 Component Cooling HX temperature will continue to be controlled automatically.

C. 12 Component Cooling HX temperature is controlled by locally throttling the emergency discharge CV, 1-SW-5149-CV.

D. 12 Component Cooling HX has no temperature control in this mode.

Answer	D

Answer Explanation

A. Incorrect. Plausible since this SW valve has a local flow indicating controller to control its position and the SW flow to the CC HXs.

B. Incorrect. Plausible since the operator will recall that the 11 SW header heat exchangers are removed from service and may conclude that AOP-7A provides a caution stating that normal automatic temperature control remains.

C. Incorrect. Plausible since the operator may recall that several SW valves have local flow indicating controllers and 1-SW-5149-CV has a local air bottle attachment to manually control its valve position when air or control power is lost.

D. Correct. Per AOP-7A Attachment 2. When using the Emergency Discharge Return Header, there is NO temperature control of 12 CC HX.

Question Information

Topic	Q51 - SW system rupture required using the emergency return header				
User ID	Q2479492			**System ID**	2479492
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	AOP-7A-1 Rev 01600		
Training Objective	Recall the strategy and basis for the major actions in AOP-7A, Loss of Saltwater Cooling.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.062.AK3.01	Safety Function	Tier 1	Group	RO Imp: 3.5	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to Loss of Service Water: (CFR: 41.4 / 41.8 / 45.7) The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the service water coolers					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 52	ID: 2479606	Points: 1.00

Which condition causes the HPSI pumps to start automatically?

A. Pressurizer level lowers to 101".

B. RWT level lowers to 45 inches.

C. Containment pressure rises to 2.8 psig.

D. Pressurizer pressure lowers to 1785 psia.

Answer	C

Answer Explanation

A. Incorrect. Plausible since this is the AOP-2A manual trip level and the operator misinterprets that HPSI pumps also start at this level.

B. Incorrect. Plausible since this is the RAS setpoint and the operator misinterprets that HPSI pumps are started on a RAS.

C. Correct. Per 1C08-ALM, HPSI pumps will start on a SIAS signal: 1740 psia RCS pressure, or 2.8 psig containment pressure

D. Incorrect. Plausible since this is the pressure for the Pressurizer Pressure Block Permitted signal and the operator misinterprets that HPSI pumps are started at this pressure in preparation for SIAS.

Question Information

Topic	Q52 - The HPSI pumps start when which plant parameter is reached?				
User ID	Q2479606			System ID	2479606
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	CLOSED	Cognitive Level	LOW
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None

K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	EOP-ATT, Rev 02400 1C08-ALM, 03800		
Training Objective	Given the ECCS HPSI, LPSI and SIT systems and parameters in any mode of operation, respond and operate the system in accordance with OI-03A, OI-03B, UFSAR, 1(2)C08 & 1(2)C09 alarm manual, EOPs and system manuals.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.006.A3.02	Safety Function	Tier 2	Group	RO Imp: 4.2	SRO Imp:
Ability to monitor automatic operation of the Emergency Core Cooling System, including: (CFR: 41.7 / 45.5) Pumps					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 53	ID: 2479631	Points: 1.00

Unit-1 is at 100% power.

- Boron equalization is in progress with 2 Charging Pumps operating
- Pzr Level Control Channel LT-110X is selected

The following transient occurs:
- Process Variable for 1-LT-110X fails high
- Alarm Window E-33, PZR CH X LVL, annunciates

(1) What is the effect on the plant?

(2) What action must be taken?

A. (1) Letdown flow goes to minimum and all Charging Pumps start
(2) Shift 1-HS-110, PZR LVL CH SEL switch and 1-HS-100-3, PZR HTR LO LVL CUT-OFF SEL switch, to "Y"

B. (1) Letdown flow goes to maximum and backup Charging Pumps stop
(2) Shift 1-HS-110, PZR LVL CH SEL switch and 1-HS-100-3, PZR HTR LO LVL CUT-OFF SEL switch, to "Y"

C. (1) Letdown flow goes to minimum and all Charging Pumps start
(2) Place Letdown controller, 1-HIC-110, in manual to match output signal prior to transient and shift 1-HS-100-3, PZR HTR LO LVL CUT-OFF SEL switch, to "Y"

D. (1) Letdown flow goes to maximum and backup Charging Pumps stop
(2) Place Letdown controller, 1-HIC-110, in manual to match output signal prior to transient and shift 1-HS-100-3, PZR HTR LO LVL CUT-OFF SEL switch, to "Y"

Answer	B

Answer Explanation

A. Incorrect - Plausible if operator assumes setpoint fails high rather than process variable and letdown flow goes to minimum and backup charging pumps start. 2nd part are actions needed to regain control of letdown and charging pump operation.

B. Correct - Since process greater than setpoint, system will maximize letdown flow and stop all backup charging pumps

C. Incorrect - Plausible if operator assumes setpoint fails high rather than process variable and letdown flow goes to minimum and backup charging pumps start. Plausible if operator assumes letdown controller malfunctioning and places in manual and matching to previous output value before transient would minimize transient on letdown system, however, shifting to other level control channel restores letdown flow to normal allowing HIC-110 to remain in automatic.

D. Incorrect - Plausible if operator assumes letdown controller malfunctioning and places in manual and matching to previous output value before transient would minimize transient on letdown system, however, shifting to other level control channel restores letdown flow to normal allowing HIC-110 to remain in automatic.

Question Information

Topic	Q53 - Pzr Level Control Malfunction and effects on system operation				
User ID	Q2479631			**System ID**	2479631
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	10 CFR 55.41 RO WRITTEN EXAMINATION		
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	1C06-ALM Rev 05600		
Training Objective	Given a failure of any RCS pressure, temperature or level instrument, determine the impact to the plant.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.028.AA1.09	Safety Function	Tier 1	Group	RO Imp: 3.7	SRO Imp:
Ability to operate and/or monitor the following as they apply to a Pressurizer Level Control Malfunction: (CFR: 41.7 / 45.5 / 45.6) Auto/manual control of PZR level					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 54	ID: 2479881	Points: 1.00

Which of the following types of Tags CAN be used to establish personnel protection in accordance with OP-AA-109-101, Personnel and Equipment Tagout Process?

1. Danger Tags
2. Special Condition Tags with a locking device
3. Information/Caution Tags

A. 1 and 2 ONLY

B. 2 and 3 ONLY

C. 1 and 3 ONLY

D. 1, 2, and 3

Answer	A

Answer Explanation

A. Correct. Per OP-AA-109-101, danger tags and SCTs with a locking device can be used for personnel protection.

B. Incorrect. Plausible if the operator determines that since caution tags are used for pump handswitches that they are providing the protection against a pump start for personnel protection. Plausible if the operator determines that danger tags are used to isolate the system but not for personnel protection.

C. Incorrect. Plausible if the operator determines that since caution tags are used for pump handswitches that they are providing the protection against a pump start for personnel protection. Plausible since SCT tags alone are not able to be used for personnel protection.

D. Incorrect. Plausible if the operator determines that since caution tags are used for pump handswitches that they are providing the protection against a pump start for personnel protection.

Question Information

Topic	Q54 - What types of tags can be used for personnel protection				
User ID	Q2479881			**System ID**	2479881
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	OP-AA-109-101, Rev 018		
Training Objective	Explain the requirements and process used for Clearance and Safety Tagging in accordance with OP-AA-109-101.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.2.13	Safety Function	Tier 3	Group	RO Imp: 4.1	SRO Imp: 4.3
Knowledge of tagging and clearance procedures (CFR: 41.10 / 43.1 / 45.13)					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 55	ID: 2481824	Points: 1.00

Which of the following components could be eligible to have the independent verification waived?

A. A valve 6 feet off the ground

B. A valve in a high radiation area

C. A valve inside a configuration control zone

D. A valve that was peer checked during manipulation

Answer	B

Answer Explanation

A. Incorrect. Plausible since working at heights has extra requirements and dangers and this danger warrants waiving the IV.

B. Correct. Per HU-AA-101, The shift manager may waive verification requirements for ALARA concerns.

C. Incorrect. Plausible since configuration control zones have extra requirements and limits on access.

D. Incorrect. Plausible if peer check and concurrent verification are misinterpreted to not need an IV.

Question Information

Topic	Q55 - IV waived for ALARA				
User ID	Q2481824			**System ID**	2481824
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	10 CFR 55.41 RO WRITTEN EXAMINATION		

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	HU-AA-101, Rev 013		
Training Objective	Apply HU performance tools at CCNPP.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.3.12	Safety Function	Tier 3	Group	RO Imp: 3.2	SRO Imp: 3.7
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)					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 56	ID: 2479714	Points: 1.00

(1) Which valve will secure ALL instrument air to containment when CLOSED?
(2) What pressure will this valve CLOSE at?

A. (1) 1-CV-2085, Containment IA Supply CV
(2) 2.8 psig containment pressure

B. (1) 1-CV-2085, Containment IA Supply CV
(2) 75 psig instrument air pressure

C. (1) 1-MOV-2080, IA CNTMT ISOL
(2) 2.8 psig containment pressure

D. (1) 1-MOV-2080, IA CNTMT ISOL
(2) 75 psig instrument air pressure

Answer	C

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Incorrect. (1) Plausible if the operator misinterprets which valve is located upstream and isolates all of instrument air, 1-CV-2085 isolates air to many valves in containment but not all of them. (2) Plausible since this is the pressure that 1-CV-2085 would shut isolating instrument air to many valve in containment.

C. Correct. (1) Per 60712SH0003, when 1-MOV-2080 is shut all instrument air to containment is secured. (2) Per EOP-ATT, this valve is shut on a CIS signal at 2.8 psig containment pressure.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Q56 - IA-2080 vs IA-2085				
User ID	Q2479714			**System ID**	2479714
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	60712SH0003, Rev 0116 60712SH0007, Rev 0005 EOP-ATT, Rev 02400		
Training Objective	Describe the automatic functions of the following components, and their functional relationship to a loss of instrument air: CV-2085 and CV-2080		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.078.K1.03	Safety Function	Tier 2	Group	RO Imp: 3.4	SRO Imp:
Knowledge of the physical connections and/or cause and effect relationships between the Instrument Air System and the following systems: (CFR: 41.3 to 41.8 / 45.7 / 45.8) Containment air					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 57	ID: 2479449	Points: 1.00

Unit-1 is at 100% power.

- A leak develops in the Component Cooling Water (CCW) System
- CCW Head Tank is 46" and rising
- CCW RMS, RI-3819, is reading 800 CPM and rising

Which of the following components is the source of the CCW leak?

A. CEDM Cooler.

B. 11B RCP Seal Cooler.

C. RCS Sample Penetration Cooler.

D. Reactor Coolant Drain Tank Heat Exchanger.

Answer	B

Answer Explanation

A. Incorrect. CEDM cooler is plausible if leak in CEDM cooler is believed to be RCS leak into the CEDM cooler since RCS is on the pressure boundary side of the CEDM motor housing. However, a leak in the CEDM cooler would cause CC Head Tank levels to lower and would not affect the RMS reading.

B. Correct. Per the 1C13 alarm manual for CC Heat Tank level, a high level can be caused by leakage in the CCW system from the RCP seal cooler. Per the 1C22 alarm manual, CC RMS will rise due to a leak from the RCP seal coolers. Both results are validated by the simulator response.

C. Incorrect. RCS Sample Penetration Cooler is plausible if leak on cooler is believed to be RCS leak from RCS sample line into the cooler. However, the cooler only cools the concrete around the penetration, not the RCS sample line. A leak in the cooler would cause a leak from the CCW system, causing the head tank to lower.

D. Incorrect. RCDT HX is plausible if leak in RC Drain Tank allowed contents to contaminate the CCW system and cause RMS readings to rise. However, CCW pressure is > RCDT pressure and a leak in the HX will cause CC Head Tank to lower, not rise.

Question Information

Topic	Q57 - CCW Leak Indications				
User ID	Q2479449			**System ID**	2479449
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	AOP-7C-1 Rev 00500		
Training Objective	Given a transient in the CCW System, identify the cause and/or the corrective actions.		
Previous NRC Exam Use	2018 NRC RO Exam		

K/A Reference(s)

APE.026.AK1.01	Safety Function	Tier 1	Group	RO Imp: 3.6	SRO Imp:
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Component Cooling Water: (CFR: 41.5 / 41.7 / 45.7 / 45.8) Leakage into or out of the CCWS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 58	ID: 2479452	Points: 1.00

Unit-1 is at 100% power when the following occurs:

- Loss of Offsite Power
- RCS Press is 2200 PSIA
- T_{COLD} is 534°F
- T_{CET} is 559°F

(1) What is the current Subcooling value?

(2) What is this Subcooling value validating?

A. (1) Approximately 90°F
(2) That adequate shutdown margin exists.

B. (1) Approximately 90°F
(2) That pressurizer level indication is representative of total RCS inventory.

C. (1) Approximately 115°F
(2) That adequate shutdown margin exists.

D. (1) Approximately 115°F
(2) That pressurizer level indication is representative of total RCS inventory.

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See Below.

B. Correct. (1) Since RCPs are not running after a loss of offsite power Tcet is used to calculate subcooling using the steam tables. Tsat for 2200 psia is 649F. 649-559=90F. (2) Per EOP-2-TB, subcooling values within the band is the primary parameter used to validate pressurizer level indication as representative of total RCS inventory.

C. Incorrect. (1) Plausible if the Tcold value is used for the subcooling calculations since Tcold value would be used in many other situations. (2) Plausible since temperature can impact the ability to meet shutdown margin but that is not what adequate subcooling validates.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Q58 - EOP-2 subcooling operational indications				
User ID	Q2479452			**System ID**	2479452
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	EOP-2-TB, Rev 01900		
Training Objective	For a loss of forced circulation, from memory recall core and plant parameters responses.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.1.45	Safety Function	Tier 3	Group	RO Imp: 4.3	SRO Imp: 4.3
Ability to identify and interpret diverse indications to validate the response of another indication. (CFR: 41.7 / 43.5 / 45.4)					
G.APE.056	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Loss of Offsite Power					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 59	ID: 2479446	Points: 1.00

Unit-1 is in MODE 6.

- Refueling Pool level is 61 feet in preparation for UGS removal
- SFP level is normal at 67 feet
- A control circuit malfunction causes 1-SI-652-MOV (SDC HDR Return ISOL) to shut
- ALL attempts to reopen 1-SI-652-MOV are unsuccessful
- The US enters AOP-3B, Abnormal Shutdown Cooling Conditions

What now is the OPTIMAL strategy to restore core decay heat removal capability?

A. Use SFP Cooling to cool the Refueling Pool

B. Use a HPSI Pump and makeup inventory lost to boil-off

C. Start Once-Thru-Cooling with HPSI or LPSI Pump

D. Open the SFP Transfer Gate Valve and use SFP Cooling

Answer	A

Answer Explanation

A. Correct - Per AOP-3B section for common mode loss of SDC with the RFP that is or can be filled.

B. Incorrect - Plausible since Boil-off of the Refueling Pool inventory is an option but would not be the optimal method since the boil-off would cause Containment to be uninhabitable and has the potential for offsite dose release from open penetrations.

C. Incorrect - Plausible since this is an option used in the AOP, however these actions are when the RFP is NOT available.

D. Incorrect - Plausible since the SFP and Refuel pool can be connected through the transfer gate, however the SFP Transfer Gate is verified shut in AOP-3B eliminating that possible cooling path and the levels provided would cause SFP level to sluice to the RFP lowering level in the SFP, which is not optimal.

Question Information

Topic	Q59 - A complete loss of SDC occurs w/RFP level at 63 Feet				
User ID	Q2479446			**System ID**	2479446
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	AOP-3B-1 Rev 03200		
Training Objective	Given plant conditions resulting in a loss of shutdown cooling, determine the required actions to maintain plant parameters within desired limits IAW with AOP-3B.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.1.20	Safety Function	Tier 3	Group	RO Imp: 4.6	SRO Imp: 4.6
Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12)					
G.APE.025	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Loss of Residual Heat Removal System					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 60	ID: 2479639	Points: 1.00

A rapid downpower is in progress on Unit-1.

- RCS Tcold is being maintained within 0.5°F of program.
- At approximately 65% power, with no adjustment in turbine load, RCS Tcold rises approximately 2.0°F.

Which of the following is the cause of the RCS temperature change?

A. Closure of MSR Second Stage High Load MOVs.

B. Closure of Turbine Bypass valves during the downpower.

C. Opening of the MSR 2nd Stage Drain Tank High Level Dump Valves.

D. Lowering of S/G level during High Power to Low Power transfer of Ovation.

Answer	A

Answer Explanation

A. Correct. At approximately 65% power the 2nd stage high load MOVs SHUT (OP-3 section 6.4.C)

B. Incorrect. Plausible if the operator misinterprets that the temperature setpoint that takes away the quick open TBV function actually closes the TBVs.

C. Incorrect. Plausible if the operator misinterprets when in a downpower the 2nd Stage Drn Tk HLDCVs are opened per procedure.

D. Incorrect. Plausible if the operator misinterprets when the Ovation high to low power mode transfer occurs.

Question Information

Topic	Q60 - Rapid Down Power, reach 65% power, T-cold rises 2°F: reason is MSR 2nd stg HL MOVs closure				
User ID	Q2479639			**System ID**	2479639
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	OP-03, Rev 07700		
Training Objective	Given the unit is operating at 100% power and a plant transient occurs, perform a rapid downpower and stabilize the plant while performing the function as the Unit Reactor Operator (RO) or Control Room Operator (CRO) IAW OP-3.		
Previous NRC Exam Use	2008 NRC RO Exam		

K/A Reference(s)

SYS.039.A1.05	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp:
Ability to predict and/or monitor changes in parameters associated with operation of the Main and Reheat Steam System, including: (CFR: 41.5 / 45.5) RCS T-ave.					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 61	ID: 2479941	Points: 1.00

Unit-1 is at 40% power, MOC, when the following occurs:

- 1-PT-4056, MS Turb Bypass VLV Controller process variable, fails to 950 PSIA
- NO operator action is taken

What is the impact on Unit-1?

A. Reactor power lowers and stabilizes at a lower value.

B. Reactor power lowers and then the reactor automatically trips.

C. Reactor power rises and then the reactor automatically trips.

D. Reactor power rises and stabilizes at a higher value.

Answer	C

Answer Explanation

A. Incorrect. Plausible if MTC is believed to be positive at low power levels and the TBVs opening will cause RCS Temperature to lower which will cause reactor power to lower.

B. Incorrect. Plausible if MTC is believed to be positive at low power levels and the TBVs opening will cause RCS Temperature to lower which will cause reactor power to lower and continue to lower causing a reactor trip on TM/LP.

C. Correct. The failure of the PT will cause the Process Variable to fail to 950 PSIA which is above the setpoint of 900 causing the TBVs to Open. Since the PV is failed the TBVs will continue to open until the controller is at 100% (all 4 TBVs open). The TBVs failing open will lower RCS Temperature which will cause Reactor Power to Rise ~40%, which is above the VOPT setpoint of 8% above the current power level.

D. Incorrect. TBVs will raise power approximately 40% which is above the VOPT setpoint of ~8% above current power. Plausible if VOPT trip is thought to only occur at higher power levels (VOPT is activated for power greater than 30%).

Question Information

Topic	Q61 - PT-4056 Failure and impact on RCS				
User ID	Q2479941			**System ID**	2479941
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-7K-1, Rev 3 60911SH0001, Rev 14		
Training Objective	Given plant conditions, indications, and events in progress, evaluate the impact of the event on the following: Reactor Power.		
Previous NRC Exam Use	2021 NRC RO Exam		

K/A Reference(s)

SYS.016.K3.03	Safety Function	Tier 2	Group	RO Imp: 3.2	SRO Imp:
Knowledge of the effect that a loss or malfunction of the Nonnuclear Instrumentation System will have on the following systems or system parameters: (CFR: 41.7 / 45.6) SDS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 62	ID: 2479684	Points: 1.00

Unit-1 is at 100% power.

- P-13000-3 is tagged out for maintenance
- A loss of the 500KV Red Bus occurs

On Unit-1, what procedure should be entered and what action should be taken?

A. (1) AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power
(2) Place all Charging pumps in PTL and realign charging pump suction to the VCT

B. (1) AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power
(2) Shut Both MSIVs

C. (1) EOP-0, Post-Trip Immediate Actions
(2) Place 2 Charging pumps in PTL and realign charging pump suction to the VCT

D. (1) EOP-0, Post-Trip Immediate Actions
(2) Shut Both MSIVs

Answer	A

Answer Explanation

A. Correct. (1) Per 61001SH0001, only 14 bus is lost on Unit-1. (2) Per AOP-7I, these are the required actions.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since this is the correct answer for the loss on Unit-2. (2) Plausible since this is the correct response for the loss on Unit-2.

Question Information

Topic	Q62 - 500KV Red Bus impact on Electrical Systems				
User ID	Q2479684			**System ID**	2479684
Status	Active	**Point Value**	1.00	**Time (min)**	0

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	61001SH0001, Rev 0054 AOP-07I, Rev 03700		
Training Objective	Given plant conditions associated with various Electrical malfunctions, the license operator candidate will be able to correctly recall and/or identify the proper AOP-7I response and bases to mitigate the effects of the Electrical Bus Loss.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.062.A2.20	Safety Function	Tier 2	Group	RO Imp: 3.8	SRO Imp: 4.1
Ability to (a) predict the impacts of the following on the AC Electrical Distribution 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) Loss of offsite power sources					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 63	ID: 2479779	Points: 1.00

While operating the SRW System in a reduced load lineup per OI-15 Service Water System, differential pressure for each running pump is limited to a value specified in the procedure.

(1) Would operating a pump with a differential pressure (D/P) less than specified per OI-15 satisfy the minimum flow required through the pump?

(2) What is the relationship between the D/P and the pump flow rate?

A. (1) No
(2) lower D/P equals higher flow rate through this pump

B. (1) Yes
(2) lower D/P equals higher flow rate through this pump

C. (1) Yes
(2) lower D/P equals lower flow rate through this pump

D. (1) No
(2) lower D/P equals lower flow rate through this pump

Answer	B

Answer Explanation

A. Incorrect. (1) Incorrect but plausible if the operator misinterprets the D/P requirement and concludes that the pump operating D/P is required to be greater than the procedure limit. (2) Correct as stated below.

B. Correct. (1) Correct since the procedure D/P limit represents a high limit. (2) Correct since operating less than the high limit is the required condition.

C. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible if the operator misinterprets the pump operation and concludes that a lower D/P equals a lower flow rate through the pump.

D. Incorrect. (1) Incorrect as stated above. (2) Incorrect as stated above.

Question Information

Topic	Q63 - Why is differential pressure limited during reduced load lineup per OI-15				
User ID	Q2479779			**System ID**	2479779
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	OI-15-1 Rev 05200		
Training Objective	Given plant conditions, the Operator should determine the required conditions for operation of the following: SRW Pumps.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.076.A4.01	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp:
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) SWS pumps					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 64	ID: 2479455	Points: 1.00

Which of the following directions are given in OP-CA-103-102-1001, Strategies for Successful Transient Mitigation, regarding the determination of system or component status during the implementation of EOP-0?

(1) Avoid shutting MSIVs based solely on loss of indications of Main Turbine Stop Valve positions.
RCS temperature response is a strong backup indication as to whether the main turbine has actually tripped.

(2) It is permissible to address more than one Safety Function or block step at a time, and specifically the Pressure/Inventory Control and RCS/Core Heat Removal Safety Functions should be addressed concurrently, due to their effect on one another.

A. NONE

B. 1 ONLY

C. 2 ONLY

D. 1 AND 2

Answer	D

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets how safety functions are concurrently performed. Plausible if the operator misinterprets the actions for loss of indications in turbine trip.

B. Incorrect. Plausible if the operator misinterprets how safety functions are concurrently performed.

C. Incorrect. Plausible if the operator misinterprets the actions for loss of indications in turbine trip.

D. Correct. Choices (1) and (2) are correct as stated in OP-CA-103-102-1001.

Question Information

Topic	Q64 - Strategies for Successful Transient Mitigation				
User ID	Q2479455			**System ID**	2479455
Status	Active	**Point Value**	1.00	**Time (min)**	4

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	OP-CA-103-102-1001 Rev 008		
Training Objective	Recall the direction and guidance provided in OP-CA-103-102-1001, Strategies for Successful Transient Mitigation.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.1.19	Safety Function	Tier 3	Group	RO Imp: 3.9	SRO Imp: 3.8
Ability to use available indications to evaluate system or component status (CFR: 41.10 / 45.12)					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 65	ID: 2479481	Points: 1.00

Unit-1 is at 100% power when the following transient occurs:

- Loss of Offsite Power
- Steam Generator Tube Rupture in 11 SG
- EOP-6, Steam Generator Tube Rupture is implemented
- T_{HOT} is 546°F and slowly rising

Which of following is the reason T_{HOT} must be lowered to < 515°F?

A. Minimizes the differential pressure across the break thereby reducing the leakrate.

B. Establishes natural circulation cooling as soon as possible during the event.

C. Reduces S/G pressure below the lift setpoints of the Main Steam Safety valves.

D. Prevents dilution of the RCS by maintaining S/G pressure lower than RCS pressure.

Answer	C

Answer Explanation

A. Incorrect. Plausible to the Operator since lowering of temperature is done in EOP-5 to support the depressurization of the RCS. Per the EOP-6 Technical Basis document: The initial cooldown is done prior to isolating the affected S/G. This action reduces the risk of challenging the steam generator safety valves of the affected S/G after it is isolated.

B. Incorrect. Plausible to the Operator since natural circ conditions will exist due to the LOOP. A cooldown to 515°F is not necessary to establish natural circulation conditions.

C. Correct. Per the EOP-6 Technical Basis document: The initial cooldown is done prior to isolating the affected S/G. This action reduces the risk of challenging the steam generator safety valves of the affected S/G after it is isolated.

D. Incorrect. Plausible to the Operator since uncontrolled dilution is always a concern when shutdown. EOP-6 accounts for the potential flow from the S/G to the RCS by requiring additional boron/SDM. Backflow from the S/G to the RCS is an available method for controlling affected S/G level.

Question Information

Topic	Q65 - Basis for cooldown to < 515°F prior to isolating affected S/G				
User ID	Q2479481			**System ID**	2479481
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	EOP-6 Technical Basis Rev 02200		
Training Objective	Recall the strategy and basis for the major actions in EOP-6, Steam Generator Tube Rupture.		
Previous NRC Exam Use	2018 NRC RO Exam		

K/A Reference(s)

EPE.038.EK2.13	Safety Function	Tier 1	Group	RO Imp: 3.6	SRO Imp:
Knowledge of the relationship between a Steam Generator Tube Rupture and the following systems or components: (CFR: 41.7 / 41.8 / 45.4 / 45.7 / 45.8) Main steam system					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 66	ID: 2479878	Points: 1.00

Unit-1 is at 100% power.

- 11, 12, and 14 Condenser Air Removal Pumps are running
- 13 Condenser Air Removal Pump is tagged out for maintenance
- The following alarm is received:

UNIT 1 CNDSR OFF-GAS 11-14 CAR PPS

(1) How many Condenser Off-Gas Radiation Monitors shall be used to validate the alarm in the control room?

(2) What could be the cause of this alarm?

A. (1) 3
(2) Steam Generator Tube Leak

B. (1) 3
(2) Condenser Tube Leak

C. (1) 4
(2) Condenser Tube Leak

D. (1) 4
(2) Steam Generator Tube Leak

Answer	A

Answer Explanation

A. Correct. (1) Per OI-35, the COGs monitor on the suction side of the CARs. (2) Per 1C22-ALM, a possible cause of the alarm is a SG tube leak.

B. Incorrect. (1) Correct. (1) See below.

C. Incorrect. (1) Plausible if the operator misinterprets the flowpath of the CARs. (2) Plausible since this RMS is used to monitor for primary to secondary leaks at all power levels and the operator misinterprets that it is TRM equipment.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Q66 - COG alarm with 3 CARs running				
User ID	Q2479878			**System ID**	2479878
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	OI-35, Rev 05100 1C22-ALM, Rev 04500		
Training Objective	Demonstrate understanding of subsystems and administrative requirements dealing with RMSs.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.055.K1.06	Safety Function	Tier 2	Group	RO Imp: 3.0	SRO Imp:
Knowledge of the physical connections and/or cause and effect relationships between the Condenser Air Removal System and the following systems: (CFR: 41.4 to 41.7 / 45.7 / 45.8) PRMS					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 67	ID: 2479685	Points: 1.00

Using provided references:

- Unit-1 RCS pressure is being maintained steady at 950 PSIA for shutdown surveillance testing
- All 4 RCPs are running
- Then, 11B RCP trips and cannot be restarted

With RCS pressure constant, what is the lower and upper band of Steam Generator pressure required to maintain the RCS Pressure/Temperature Limits of OP-5?

A. 96 PSIA to 848 PSIA

B. 96 PSIA to 680 PSIA

C. 135 PSIA to 848 PSIA

D. 135 PSIA to 680 PSIA

Answer	D

Answer Explanation

A. Incorrect - 96 PSIA represents 325F which is plausible since it is the Max Operating Pressure Curve line on Figure 5 for 3 RCPs running. 848 PSIA is incorrect as stated below.

B. Incorrect - 96 PSIA is incorrect as stated above. 680 PSIA is correct as stated below.

C. Incorrect - 135 PSIA is correct as stated below. 848 PSIA represents 525F which is plausible if the operator uses Figure 1 for when all 4 RCPS were in operation.

D. Correct - Both numbers are correct. 135 PSIA represents 350F which is the minimum temperature for 3 RCPs running. 680 PSIA represents 500F which is the upper band of the curve on OP-5-1 Figure 5.

Question Information

Topic	Q67 - Reference - RCP Pump Curves and SG Pressure				
User ID	Q2479685			System ID	2479685
Status	Active	Point Value	1.00	Time (min)	5

Open or Closed Reference	OP-5-1 FIG 1,2,5	Cognitive Level	HIGH
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	10 CFR 55.41 RO WRITTEN EXAMINATION		
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	OP-5-1, Figures 1,2,5 (pages 56, 57, & 60)
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	
Technical Reference and Revision #	OP-5-1 Rev 06600		
Training Objective	Given plant conditions, determine the required temperature and/or pressure limits for the RCPs in operation.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.003.K3.02	Safety Function	Tier 2	Group	RO Imp: 3.9	SRO Imp:
Knowledge of the effect that a loss or malfunction of the Reactor Coolant Pump System will have on the following systems or system parameters: (CFR: 41.7 / 45.6) S/G					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 68	ID: 2479456	Points: 1.00

Unit-2 is at 100% power.

- A loss of 11 DC Bus occurs

(1) What EDG is now unavailable?
(2) Why?

A. (1) 2A EDG
(2) the start solenoids failed open

B. (1) 2A EDG
(2) the field flash and control power lost power

C. (1) 2B EDG
(2) the start solenoids failed open

D. (1) 2B EDG
(2) the field flash and control power lost power

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) Per AOP-7J the 2A EDG is unavailable. (2) Per AOP-7J, the field flash and control power lose power on a loss of 11 DC Bus.

C. Incorrect. (1) Plausible if the operator misinterprets which DC buses impact the EDGs. 2B is impacted by a loss of 21 DC Bus. (2) Plausible since the loss of 11 DC bus does impact the start solenoids but it fails them closed.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Q68 - Loss of 11 DC Bus EDG impact and why				
User ID	Q2479456			**System ID**	2479456
Status	Active	**Point Value**	1.00	**Time (min)**	0

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No addition information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	AOP-7J-2, Rev 01900		
Training Objective	Evaluate the effect that a loss of a 120 Volt Vital AC bus will have on plant equipment.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.058.AK3.01	Safety Function	Tier 1	Group	RO Imp: 4.0	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to Loss of DC Power: (CFR: 41.5 / 41.10 / 45.6 / 45.1) Operation of the EDGs					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 69	ID: 2479505	Points: 1.00

Unit-1 is at 100% power.

- A loss of Instrument Air occurs
- Instrument Air pressure is 55 PSIG and lowering at a rapid and continuous rate

(1) When is the earliest the Unit Supervisor will order a reactor trip?

(2) What is the reason as stated in the AOP Caution?

A. (1) At 40 PSIG and lowering
(2) The TBVs will not quick-open fully below 40 PSIG

B. (1) At 40 PSIG and lowering
(2) The TBVs will fail full open below 40 PSIG

C. (1) At 50 PSIG and lowering
(2) The TBVs will not quick-open fully below 50 PSIG

D. (1) At 50 PSIG and lowering
(2) The TBVs will fail full open below 50 PSIG

Answer	C

Answer Explanation

A. Incorrect - (1) See below. (2) Correct as stated below.

B. Incorrect - (1) Plausible since the operator will recall that the Main Feed Reg Valves fail as is at 40 psig and may incorrectly conclude that the reactor trip criteria is at the same pressure. (2) Plausible since the operator will recall that the TBVs require 20 psig instrument air pressure to ensure shut per AOP-7D Attachment 1 and may incorrectly recall that actual setting to be 40 psig and will conclude the TBVs will fail open with air pressure less than 40 psig.

C. Correct - (1) AOP-7D specifies a reactor trip at 50 PSIG I/A Header pressure. (2) Correct as stated in the AOP Caution. The TBVs will not quick-open fully below 50 PSIG.

D. Incorrect - (1) Correct as stated above. (2) Plausible since the operator will recall that the TBVs require 20 psig instrument air pressure to ensure shut per AOP-7D Attachment 1 and may incorrectly recall that actual setting to be 50 psig and will conclude the TBVs will fail open with air pressure less than 50 psig.

Question Information

Topic	Q69 - Loss of IA effects on TBVs and crew response				
User ID	Q2479505			**System ID**	2479505
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	AOP-7D-1 Rev 01600 AOP-7D-1-TB Basis Rev 00800		
Training Objective	Given plant conditions, the Operator should determine the correct actions for a Loss of Instrument Air in the following situations: Modes 1 & 2.		
Previous NRC Exam Use	2006 NRC RO Exam 2012 NRC SRO Exam		

K/A Reference(s)

P2.4.20	Safety Function	Tier 3	Group	RO Imp: 3.8	SRO Imp: 4.3
Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions, and notes (CFR: 41.10 / 43.5 / 45.13)					
G.APE.065	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Loss of Instrument Air					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Question 70	ID: 2479875	Points: 1.00

Both Units are at 100% power.

- A loss of offsite power occurs
- AFW flow and S/G levels are:
  - 11 S/G--290 GPM, -110" rising
  - 12 S/G--340 GPM, -80" rising
  - 21 S/G--350 GPM, -60" rising
  - 22 S/G--280 GPM, -80" rising

What actions, if any, should the operator take with respect to AFW flow?

A. No action required, all parameters are meeting requirements.

B. Lower flow to 12 S/G, lower flow to 21 S/G.

C. Raise flow to 11 S/G, raise flow to 22 S/G.

D. Lower flow to 21 S/G, raise flow to 11 S/G.

Answer	B

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets the limits for AFW flow to be higher and determines all parameters are in spec.

B. Correct. Per OI-32A, combined flow to both units must be <1200 GPM and <600 GPM per unit. Combined flow is 1260 GPM for BOTH units and U1 flow is 630 GPM and U2 flow is 630 GPM so both units need to lower flow.

C. Incorrect. Plausible if the operator misinterprets that all S/G need 300 gpm since this would be the initial flow from AFAS.

D. Incorrect. Plausible if the operator misinterprets the overall limit of <1200 gpm and determines S/G level recovering is more important.

Question Information

Topic	Q70 - Actions the operator should take with respect to AFW flow				
User ID	Q2479875			**System ID**	2479875
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	OI-32A, Rev 04500		
Training Objective	Given plant conditions & alarms, diagnose status of the AFW system.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.035.A1.03	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp:
Ability to predict and/or monitor changes in parameters associated with operation of the Steam Generator System, including: (CFR: 41.5 / 45.5) Feed flow/steam flow					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 71	ID: 2480025	Points: 1.00

Unit-1 is at 100% power.

- A LOCA occurs
- The appropriate EOP is entered
- Actions are taken to control pressurizer level in the required band

- The LOCA leakrate raises significantly

(1) What indication would confirm that Core Uncovery is occurring?

(2) What action will the crew perform to mitigate this condition?

A. (1) On the PAMS RVLMS screen, only the top two lights are RED.
(2) De-energize Pressurizer heaters.

B. (1) On the PAMS RVLMS screen, only the top two lights are RED.
(2) Raise HPSI flow.

C. (1) PAMS subcooling screen displays (-)20°F.
(2) De-energize Pressurizer heaters.

D. (1) PAMS subcooling screen displays (-)20°F.
(2) Raise HPSI flow.

Answer	D

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the RVLMS screen lights turning from green to red indicate a loss of liquid or inventory in that region of the reactor vessel. The top lights turning red are an indication that the reactor vessel head contains a void or bubble in that area. (2) Incorrect but plausible since the operator may misinterpret a void in the reactor vessel head causing PZR level to become solid. Then, the operator may conclude that de-energizing PZR heaters is desirable if the PZR level is solid.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

D. Correct. (1) Correct since core uncovery will cause elevated CET temperatures combined with low RCS pressure from the large break LOCA and superheated conditions. (2) Correct as stated in EOP-5 Block Step K for raising subcooling.

Question Information

Topic	Q71 - Core Uncovery Indication and Action				
User ID	Q2480025			**System ID**	2480025
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.5 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.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	EOP-5-1 Rev 03200		
Training Objective	Recognize indications of core uncovery and the required actions to mitigate the transient.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.017.A2.02	Safety Function	Tier 2	Group	RO Imp: 4.1	SRO Imp: 4.2
Ability to (a) predict the impacts of the following on the In-Core Temperature Monitor 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) Elevated in-core temperatures that can cause or have caused core damage					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 72	ID: 2479418	Points: 1.00

Unit-1 is at 100% power:

- A steam leak occurs on 12 Steam Generator
- The reactor is tripped
- EOP-4, Excess Steam Demand Event, is implemented
- The Unit Supervisor directs maintaining level in 11 Steam Generator

Per EOP-4, what is the preferred pump and why?

A. 11 AFW, to minimize additional load on the diesel

B. 11 SGFP, to minimize additional load on the diesel

C. 13 AFW, to minimize any additional loss of condensate inventory

D. 13 SGFP, to minimize any additional loss of condensate inventory

Answer	C

Answer Explanation

A. Incorrect. Plausible since this is the preferred pump and why for a different event.

B. Incorrect. Plausible since main feed is always preferred for feeding a steam generator, but the MSIVs are shut in EOP-4.

C. Correct. Per EOP-4, 11 AFW is the preferred pump since it will minimize any additional loss of condensate inventory.

D. Incorrect. Plausible since 13 SGFP pump can be used to feed Unit-1 Steam generators in a different event but not per EOP-4.

Question Information

Topic	Q72 - Basis for how to restore SG Level in EOP-4				
User ID	Q2479418			**System ID**	2479418
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-04, Rev 02000 EOP-04-TB, Rev 02000		
Training Objective	Recall the basis for operator actions for the following in accordance with EOP-4: Maintain the Unaffected S/G Level.		
Previous NRC Exam Use	None		

K/A Reference(s)

CE.E05.EK3.11	Safety Function	Tier 1	Group	RO Imp: 3.3	SRO Imp:
Knowledge of the reasons for the following responses and/or actions as they apply to Excess Steam Demand: (CFR: 41.5 / 41.10 / 45.6 / 45.13) Restoring S/G levels					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 73	ID: 2479401	Points: 1.00

Which of the following describes the temperature of a saturated liquid?

A. Below the boiling point

B. At the boiling point

C. Above the boiling point

D. Unrelated to the boiling point

Answer	B

Answer Explanation

A. Incorrect. If the temperature was below the boiling point the liquid would be classified as a subcooled liquid.

B. Correct. The temperature at which a liquid first begins to change into a vapor (or in the case of water evaporate into steam), while under a given pressure is known as the saturation temperature. At this temperature, the liquid is said to be “saturated liquid”. A saturated liquid is a liquid in a state such that the addition of heat will result in the formation of vapor. The designation “saturation” arises from the consideration that the water is saturated with heat, and if any more heat is supplied, the liquid evaporates into steam. Thus, 212°F is the saturation temperature corresponding to a saturation (vapor) pressure of 14.7 psia.
When heat is added to a liquid open to the atmosphere, the temperature increases until the boiling point is reached. At this point, further addition of heat does not cause the temperature to increase. Instead, it vaporizes the liquid. Vaporization occurs at a constant temperature.

C. Incorrect. If the temperature was above the boiling point the substance would not exist as a liquid and would be classified as a superheated vapor or gas.

D. Incorrect. Plausible if the operator misinterprets the definition of a saturated liquid and concludes that the boiling point is not related to the definition.

Question Information

Topic	Q73 - Define Saturated Liquid				
User ID	Q2479401			System ID	2479401
Status	Active	Point Value	1.00	Time (min)	2

Open or Closed Reference	CLOSED	Cognitive Level	LOW
Operator Discipline	LO-I	Operator Type	RO
10CFR55 Content	CFR: 41.14 Principles of heat transfer thermodynamics and fluid mechanics.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	Steam Tables		
Training Objective	Define or apply the following terms: Saturated Liquid.		
Previous NRC Exam Use	None		

K/A Reference(s)

193003.K1.08	Safety Function	Tier	Group	RO Imp: 2.8	SRO Imp:
Define the following terms: saturated liquid					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 74	ID: 2479876	Points: 1.00

Which one of the following set of actions will occur to the control room HVAC system upon receipt of a high radiation signal from the Control Room Vent RMS monitor, 0-RI-5350?

A. ONLY ONE Post-LOCI filter fan STARTS and its discharge damper OPENS. The Control Room toilet exhaust fan STARTS.

B. BOTH Post-LOCI filter fans START and their discharge dampers OPEN. The Control Room toilet exhaust fan STOPS.

C. BOTH Post-LOCI filter fans START and their discharge dampers OPEN. The Control Room toilet exhaust fan STARTS.

D. ONLY ONE Post-LOCI filter fan STARTS and its discharge damper OPENS. The Control Room toilet exhaust fan STOPS.

Answer	B

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets that only 1 train of Post-LOCI will actuate on 1 RMS alarm. Plausible if the operator determines that the toilet exhaust fan will start to remove the high radiation in the control to keep the control room operable.

B. Correct. Per 1C22-ALM, these are the actions that occur since Control Room ventilation normal lineup is in a recirculation mode.

C. Incorrect. Plausible if the operator determines that the toilet exhaust fan will start to remove the high radiation in the control to keep the control room operable.

D. Incorrect. Plausible if the operator misinterprets that only 1 train of Post-LOCI will actuate on 1 RMS alarm.

Question Information

Topic	Q74 - Which of the following actions occur to the cntrl rm HVAC system on a high radiation signal?				
User ID	Q2479876			**System ID**	2479876
Status	Active	**Point Value**	1.00	**Time (min)**	0

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	2
Technical Reference and Revision #	OI-22F, Rev 03200 1C22-ALM, Rev 04500		
Training Objective	Describe the design features that provide for the following during operation of the CR Ventilation and Chilled Water System: 100% recirculation during high radiation conditions.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.050.A4.03	Safety Function	Tier 2	Group	RO Imp: 3.0	SRO Imp:
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Dampers					

Learning Objective(s)

RO NRC Test

User (Sys) ID N/A (1545504)

Cross Reference Links

None

Question 75	ID: 2479706	Points: 1.00

What is the power supply to Unit-1 Channel C RPS?

A. 1Y01

B. 2Y02

C. 12 DC bus

D. 22 DC bus

Answer	C

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets the vital AC Bus schematic and determines 1Y01 powers channel C RPS.

B. Incorrect. Plausible if the operator misinterprets which Unit 2Y0 bus powers the opposite unit WRNI cabinets.

C. Correct. Per AOP-7J, 12 DC Bus powers Channel C RPS.

D. Incorrect. Plausible if the operator misinterprets the vital DC Bus schematic and determines that 22 DC bus powers C RPS.

Question Information

Topic	Q75 - Power supply to RPS channel C				
User ID	Q2479706			**System ID**	2479706
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	RO
10CFR55 Content	CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	Not applicable		
Additional Information	No additional information		
NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	AOP-7J-1, Rev 02400		
Training Objective	From memory, describe the interrelationship between RPS and the following systems/components without error: 125 VDC.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.012.K2.01	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp:
Knowledge of electrical power supplies to the following: (CFR: 41.7) RPS channels, components, and interconnections					

Learning Objective(s)

RO NRC Test
User (Sys) ID N/A (1545504)

Cross Reference Links

None

Exelon Generation

Exam Material

VISION Report

(Test Key) 2023 ILT NRC SRO EXAM

CCNPP Operations NRC Examinations

December 15, 2022

Test	2023 ILT NRC SRO EXAM
VISION ID	367278
Status	

EXAMINATION COVER SHEET

Exam Title (ID)	2023 ILT NRC SRO EXAM (367278)				
Training Program	CCNPP Operations NRC Examinations				
LMS Component ID		**Total Points**	25.00	**Pass Criteria** = 80 %	
Trainee Name		**Employee ID**			
Graded By / Date		**Grade**	____ / 25.00 = _______ %		
Review and Approval					
Instructor		**Date**			
Technical Review		**Date**			
Training Supv		**Date**			

Examination Rules

1. References may NOT be used during this exam, unless otherwise stated.
2. Read each question carefully before answering. If you have any questions or need clarification during the exam, contact the exam proctor.
3. Conversation with other trainees during the exam is prohibited.
4. Partial credit will NOT be considered, unless otherwise stated. Show **all** work and state **all** assumptions when partial credit may be given.
5. Restroom trips are limited and only one examinee at a time may leave.
6. For exams with time limits, you have ____ minutes to complete the exam.
7. The examinee agrees to refrain from discussing the content of the exam until the end of the exam cycle.

Examination Integrity Statement

Cheating or compromising the exam will result in disciplinary actions up to and including termination.

“I acknowledge that I am aware of the Exam Rules stated above. Further, I have not given, received, or observed any aid or information regarding this exam prior to or during its administration that could compromise this exam.”

Examinee Signature ____ Date ______________

Review Acknowledgement

“I acknowledge that the correct answers to the exam questions were indicated to me following the completion of the exam. I have had the opportunity to review the exam questions with the instructor to ensure my understanding.”

Examinee Signature ____ Date ______________

Question 76	ID: 2479587	Points: 1.00

SRO ONLY

Using provided references.

Unit-1 was operating at 100% power when the following occurs:

- 1C24B Window D&H4, U-1 45' SWGR RM, alarm received
- Fire and Safety Watch confirms a visible fire from one of the load centers in the 45' SWGR RM
- The fire has caused the loss of the load center
- The fire is NOT extinguished within 15 minutes

(1) Which load center bus is on fire?
(2) What EAL classification will Unit-1 declare?

A. (1) 14A 480V Bus
(2) Alert

B. (1) 11B 480V Bus
(2) Alert

C. (1) 14A 480V Bus
(2) Unusual Event

D. (1) 11B 480V Bus
(2) Unusual Event

Answer	C

Answer Explanation

A. Incorrect. 1st part is correct as stated below. 2nd part is incorrect as stated below.

B. Incorrect. 1st part is incorrect since the 11B 480V Load Center is located in the 27' Switchgear Room. The distractor is plausible if the operator recalls that Switchgear ventilation may allow a path for smoke in one Switchgear Room to cause a fire alarm in the Switchgear Room located directly above it. Also, plausible if the operator interprets the 11B load center to be a B Train component or believes the 11B load center is physically located on the 45' foot of the plant. 2nd part is incorrect since the condition given is a loss of the safety related load center which meets the criteria for an Alert per EAL 2.1. The distractor is plausible since prior to the most current EAL rev a fire in the aux building was an alert.

C. Correct. 1st part is correct since the Unit-1 45' SWGR RM alarm as stated in the 1C24B Alarm Manual is caused by smoke from within that room. The 13&14 4KV Buses and their 480V Load Centers are located within the 45' Switchgear Room. The 14A 480V Load Center is a safety related load center located within the room. 2nd part is correct since a UE would be declared for fire resulting in visible damage to a safety related component or Control Room indication of degraded performance of a safety related component within the Switchgear Room.

D. Incorrect. 1st part is incorrect as stated above. 2nd part is correct as stated above.

Question Information

Topic	Q76 - SRO-References: Fire Alarms and EAL				
User ID	Q2479587			**System ID**	2479587
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	EP-AA-1011	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	EP-AA-1011, Addendum 3
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	1C24B Alarm Manual Rev 04900 EP-AA-1011, Add 3, Rev 8		
Training Objective	State the appropriate response and required actions to a Fire in the plant IAW ERPIP 3.0.		
Previous NRC Exam Use	2019 NRC SRO Exam		

K/A Reference(s)

APE.067.AA2.17	Safety Function	Tier 1	Group	RO Imp: 3.3	SRO Imp: 3.5
Ability to determine and/or interpret the following as they apply to Plant Fire on Site: (CFR: 41.10 / 43.5 / 45.13) Systems that may be affected by the fire					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 77	ID: 2480019	Points: 1.00

SRO ONLY

Which of the following safety functions appears on the Safety Parameter Display System (SPDS) display screen, but is NOT part of the EOP Safety Function Status Checks?

A. Turbine Trip

B. Containment Isolation

C. Containment Environment

D. Core and RCS Heat Removal.

Answer	B

Answer Explanation

A. Incorrect. Plausible since this is not a Safety Function Status Check for EOP-0, but it is not displayed on SPDS.

B. Correct. There is no Containment Isolation in EOP-0. SPDS keyboard has a dedicated function key for Containment Isolation, located after Containment Environment and before Radiation Levels External to Containment.

C. Incorrect. Plausible if the operator misinterprets which containment SPDS display is part of the EOP-0 safety function status checks.

D. Incorrect. Plausible if the operator misinterprets what screens are available on SPDS.

Question Information

Topic	Q77 - SRO-Which safety function is listed on SPDS but is not part of EOP SFSC				
User ID	Q2480019			**System ID**	2480019
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-0, Rev 01500		
Training Objective	Given a plant transient requiring a reactor trip direct or implement EOP-0 to comply with strategy, basis and operator actions in accordance with EOP-0, its technical basis document, OP-CA-102-106 and OP-CA-103-102-1001.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.4.21	Safety Function	Tier 3	Group	RO Imp: 4.0	SRO Imp: 4.6
Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions (CFR: 41.7 / 43.5 / 45.12)					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 78	ID: 2479584	Points: 1.00

SRO ONLY

Unit-1 is at 100% power.

- The RO is performing STP-O-29, CEA Free Movement Test, when a Shutdown Group CEA cannot be withdrawn from 127.5 inches, after insertion
- Electrical Maintenance determines the CEA is mechanically stuck
- System Engineering has declared the CEA untrippable

Which of the following actions is required?

A. Perform a rapid shutdown per OP-3, Appendix B, Rapid Power Reduction; upon turbine trip, borate the RCS at ≥ 40 GPM to at least 2300 PPM.

B. Insert the remaining CEAs within the group to realign with the stuck CEA to clear the CEA Motion Inhibit (CMI) while maintaining power level, per OI-42 "CEDM System Operation".

C. If unable to realign the CEA after two hours, then trip the reactor and implement EOP-0, Post Trip Immediate Actions.

D. Shutdown and place the unit in Mode 3 within 6 hours per OP-3, Normal Power Operation.

Answer	D

Answer Explanation

A. Incorrect. Plausible since these are the actions required per AOP-1B for two or more untrippable CEAs.

B. Incorrect. Plausible if the operator determines the need to clear the CMI alarm to adhere to technical specifications and misinterprets that these actions will accomplish that. However, this does not clear the CMI. CMI remains in because Regulating Group CEAs are unable to move (MIRG) as Shutdown CEAs are less than 129 inches withdrawn.

C. Incorrect. Plausible since these are the actions required when two or more CEAs are misaligned by > 15 inches within their group.

D. Correct. For a single untrippable CEA, TS 3.1.4 directs the plant be placed in MODE 3 within 6 hours, per OP-3. Also, AOP-1B directs the unit be placed in Mode 3 within 6 hours with one or more CEAs untrippable.

Question Information

Topic	Q78 - SRO-Actions for untrippable CEA (stuck)				
User ID	Q2479584			**System ID**	2479584
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-1B, Rev 03005.00 TS 3.1.4, Amend 314		
Training Objective	Given CEA status determine the required actions per AOP-1B and Technical Specifications.		
Previous NRC Exam Use	2012 NRC SRO Exam		

K/A Reference(s)

P2.2.23	Safety Function	Tier 3	Group	RO Imp: 3.1	SRO Imp: 4.6
Ability to track TS limiting conditions for operation (CFR: 41.10 / 43.2 / 45.13)					
G.APE.005	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Inoperable/Stuck Control Rod					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 79	ID: 2479920	Points: 1.00

SRO ONLY

Unit-1 is at 100% power.

Due to issues with the instrument air system, both of the following evolutions are being performed per OI-19, Instrument Air.

- Draining IA B/U Storage Tanks
- Supplying Containment Instrument Air with Plant Air

(1) Which instrument air evolution requires a TCC (Temporary Configuration Change)?

(2) A 50.59 Review is required for a TCC to remain installed in the plant at power past _______ days?

A. (1) Draining IA B/U Storage Tanks
(2) 30 Days

B. (1) Draining IA B/U Storage Tanks
(2) 90 Days

C. (1) Supplying Containment Instrument Air with Plant Air
(2) 30 Days

D. (1) Supplying Containment Instrument Air with Plant Air
(2) 90 Days

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible if the operator misinterprets that extra drain lines will have to be added to the system to perform this evolution. (2) Plausible since the operator will recall that the requirement to perform an audit of active Component Manipulations and an audit of active Locked Valve Deviations per PE 0-102-58-O-M is monthly (30 days). The operator may then conclude that the TCC audit is performed on the same frequency.

B. Incorrect. (1) See above. (2) Correct.

C. Incorrect. (1) Correct. (2) See above.

D. Correct. (1) Per OI-19, a TCC is required for this evolution. (2) Per CC-AA-112, a 50.59 review is required to go beyond 90 days at power.

Question Information

Topic	Q79 - SRO-Temporary Configuration Changes for IA system				
User ID	Q2479920			**System ID**	2479920
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	CC-AA-112, Rev 30 OP-AA-108-101, Rev 18		
Training Objective	Apply the requirements of OP-AA-108-101, Control of Equipment and System Status.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.2.05	Safety Function	Tier 3	Group	RO Imp: 2.2	SRO Imp: 3.2
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)					
G.SYS.078	Safety Function	Tier	Group	RO Imp:	SRO Imp:
IAS Instrument Air System					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 80	ID: 2481612	Points: 1.00

SRO ONLY

Using provided references:

Unit-2 is at 100% power.

At time 1000

- Loss of Off-Site Power
- Large break LOCA

At time 1010

- RCS Pressure is 90 PSIA and steady
- CETs reading 325°F and slowly lowering
- Containment Pressure is at 28 PSIG and rapidly lowering
- The ABO hears noise coming from the PAL
- 2B DG tripped
- 0C DG failed to start

(1) What is the call for the containment barrier?

(2) Will all RVLMS lights being red cause an upgrade in event classification?

A. (1) Potential Loss
(2) No

B. (1) Potential Loss
(2) Yes

C. (1) Loss
(2) Yes

D. (1) Loss
(2) No

Answer	C

Answer Explanation

A. Incorrect. (1) Plausible if the operator misinterprets how many CACs are available and determines they have less than the design required cooling which is a potential loss. (2) Plausible if the operator misinterprets what the level is in the reactor vessel with all RVLMS lights red and determines the reactor vessel level is above 10 inches.

B. Incorrect. (1) See above. (2) Correct.

C. Correct. (1) Per EP-AA-1011, the containment barrier is lost since there is an unplanned lowering in containment pressure. (2) Per EP-AA-1011, all RVLMS lights being red is a potential loss of the fuel clad and would cause an upgrade to a GE.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Q80 - SRO-References: Containment Pressure EAL call				
User ID	Q2481612			**System ID**	2481612
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	EP-AA-1011	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	EP-AA-1011, Addendum 3
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EP-AA-1011, Add 3, Rev 008		
Training Objective	Given plant conditions, diagnose a Loss of Containment Integrity / Closure and respond by directing and/or implementing the proper operator actions in accordance with the procedures.		
Previous NRC Exam Use	None		

K/A Reference(s)

APE.069.AA2.03	Safety Function	Tier 1	Group	RO Imp: 4.1	SRO Imp: 3.8
Ability to determine and/or interpret the following as they apply to Loss of Containment Integrity: (CFR: 43.5 / 45.13) Containment pressure					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 81	ID: 2479958	Points: 1.00

SRO ONLY

The Unit is in MODE 6 with the Reactor Vessel Head removed.

- Fuel transfer activities have been stopped while a problem with the Fuel Transfer Cart is being fixed.
- The Fuel Handling SRO is attending an outage planning brief in the Outage Control Center.
- Nuclear Fuels Management asks the Control Room to direct the Refueling Bridge Operator to uncouple a CEA for weighing.

(1) The Unit Supervisor will determine that uncoupling a CEA for weighing _______.

(2) What oversight is required to perform this evolution?

A. (1) IS a CORE ALTERATION
(2) The evolution must be deferred until the Fuel Handling SRO has returned to the Refueling Bridge.

B. (1) is NOT a CORE ALTERATION
(2) The evolution may not commence until the Fuel Handling SRO has been notified.

C. (1) IS a CORE ALTERATION
(2) The evolution may not commence until the Fuel Handling SRO has been notified.

D. (1) is NOT a CORE ALTERATION
(2) The evolution must be deferred until the Fuel Handling SRO has returned to the Refueling Bridge.

Answer	A

Answer Explanation

A. Correct. (1) Correct as defined in FH-305. (2) Correct as stated in FH-305.

B. Incorrect. (1) Incorrect but plausible if thought that since the fuel bundle is not being relocated that it would not be a core alteration, however weighing CEAs is a core alt. (2) Incorrect but plausible if the operator misinterprets the FHS requirements and concludes that only notification is necessary.

C. Incorrect. (1) Correct as defined in FH-305. (2) Incorrect as stated above.

D. Incorrect. (1) Incorrect as stated above. (2) Correct as stated above.

Question Information

Topic	Q81 - SRO-Fuel Handling SRO duties				
User ID	Q2479958			**System ID**	2479958
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.7 Fuel handling facilities and procedures.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	FH-305 Rev 32		
Training Objective	Determine and apply the definition and requirements associated with core alterations.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.1.42	Safety Function	Tier 3	Group	RO Imp: N/A	SRO Imp: 3.4
Knowledge of new and spent fuel movement procedures (SRO Only) (CFR: 43.7 / 45.13)					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 82	ID: 2479496	Points: 1.00

SRO ONLY

A Loss of Coolant Accident is in progress and the following conditions exist:

- Core Exit Thermocouple (CET) subcooling is 25ºF
- Pressurizer pressure is 450 PSIA
- Containment pressure is 20 PSIG
- A Recirculation Actuation Signal has occurred
- It is desired to start core flush

What is the most preferred method of core flush and why?

A. Line up a HPSI pump to inject water in the pressurizer via the auxiliary spray line. Preferred due to the pressure restriction on the shutdown cooling piping.

B. Line up a HPSI pump to inject water in the pressurizer via the auxiliary spray line. Minimize the chances of clogging the containment sump strainer.

C. Line up a LPSI Pump via shutdown cooling header return isolation valve. Minimize the chances of clogging the containment sump strainer.

D. Line up a LPSI Pump via shutdown cooling header return isolation valve. Preferred due to the pressure restriction on the shutdown cooling piping.

Answer	A

Answer Explanation

A. Correct. Per EOP-5 this is the preferred lineup and reason.

B. Incorrect. Plausible if the operator misinterprets that the reasons for certain actions during RAS conditions are also true for the core flush actions since RAS has occurred.

C. Incorrect. Plausible if the operator misinterprets the pressure requirements for different core flush lineups. Plausible if the operator misinterprets that the reasons for certain actions during RAS conditions are also true for the core flush actions since RAS has occurred.

D. Incorrect. Plausible if the operator misinterprets the pressure requirements for different core flush lineups.

Question Information

Topic	Q82 - SRO-Large Break LOCA core flush				
User ID	Q2479496			**System ID**	2479496
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-5-TB, Rev 03500		
Training Objective	Given a loss of Reactor Coolant, identify and understand the basis and actions to mitigate the event in accordance with EOP-5.		
Previous NRC Exam Use	None		

K/A Reference(s)

EPE.011.EA2.05	Safety Function	Tier 1	Group	RO Imp: 4.4	SRO Imp: 4.1
Ability to determine and/or interpret the following as they apply to a Large-Break LOCA: (CFR: 43.5 / 45.13) Significance of ECCS pump operation					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 83	ID: 2479998	Points: 1.00

SRO ONLY

Unit-1 is at 100% power.

- Instrument Air pressure has been lowering slowly for 10 minutes.
- Instrument Air header pressure is 93 PSIG.
- Instrument Air Compressor discharge pressure is 105 PSIG.
- 12 IA Dryer Malfunction light is brightly lit.
- A small IA leak is identified on 12 IA Dryer Right Chamber.

(1) What is the status of 12 IA Dryer?

(2) What action is required?

A. (1) De-energized with flow through both chambers.
(2) Ensure the PA to IA cross-connect valve, 1-PA-2061-CV, is open per the Alarm Response Manual.

B. (1) De-energized with flow through both chambers.
(2) Bypass 12 IA Dryer IAW AOP-7D, Loss of Instrument Air and refer to OI-19, Instrument Air, to shift to the standby dryer if desired.

C. (1) Energized with both chambers isolated.
(2) Ensure the PA to IA cross-connect valve, 1-PA-2061-CV, is open per the Alarm Response Manual.

D. (1) Energized with both chambers isolated.
(2) Bypass 12 IA Dryer IAW AOP-7D, Loss of Instrument Air and refer to OI-19, Instrument Air, to shift to the standby dryer if desired.

Answer	B

Answer Explanation

A. Incorrect. 1st part is correct as stated below. 2nd part is incorrect as stated below.

B. Correct. Per AOP-7D, Section V.B.4 states the IA Dryer malfunction light will be brightly lit for the inservice IA Dryer and the dryer will de-energize with both chambers in service if IA Pressure has lowered to 93 psig. The AOP step then directs to bypass the IA Dryer that is the cause of the lowering IA pressure and then shift to the standby dryer if desired.

C. Incorrect. 1st part is incorrect but plausible because operator may assume that a malfunction will isolate the IA dryer, failing to the safe position. 2nd part is incorrect but plausible since this action is stated in AOP-7D but not until after IA header pressure lowers to 88 psig. The operator may not recall the actual setpoint that actuates the opening of the Plant Air to IA Cross Connect valve.

D. Incorrect. 1st part is incorrect but plausible because operator may assume that a malfunction will isolate the dryer, failing to the safe position. 2nd part is correct as stated above.

Question Information

Topic	Q83 - SRO-Given IA conditions, identify Dryer status and required actions in AOP				
User ID	Q2479998			**System ID**	2479998
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-7D-1 Rev 01600		
Training Objective	Given plant conditions, the Operator should determine the correct actions for a Loss of Instrument Air in the following situations: Modes 1&2.		
Previous NRC Exam Use	2019 NRC SRO Exam		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	Prior to this dryer modification, an IA Dryer malfunction threatened the operation of the plant. The knowledge of this modification is important to reactor safety.

K/A Reference(s)

APE.065.AA2.02	Safety Function	Tier 1	Group	RO Imp: 2.7	SRO Imp: 2.7
Ability to determine and/or interpret the following as they apply to Loss of Instrument Air: (CFR: 41.10 / 43.5 / 45.13) Airflow readings					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 84	ID: 2479510	Points: 1.00

SRO ONLY

Unit-1 is at 100% power.

- A leak develops at PZR Pressure Transmitter, 1PT100Y, the in-service PZR Pressure Controller, causing the indication to fail low
- The following alarms and indications are noted:
  - 1C05 - PROT CH TRIP, TM/LP CH PRE-TRIP,
  - 1C06 - PZR CH 100 PRESS, PZR CH Y LVL, PZR HTR CUTOFF
  - 1C08 - ACTUATION SYS SENSOR CH ZF TRIP
  - 1-PI-102C is failed low
  - 1-LI-110Y is failed low

(1) What procedure is used to diagnose a common tap failure?

(2) Does a common tap failure exist?

A. (1) OP-CA-108-3.0, Immediate Actions
(2) Yes

B. (1) OP-CA-103-102-100, Watch Standing Practices
(2) Yes

C. (1) OP-CA-103-102-100, Watch Standing Practices
(2) No

D. (1) OP-CA-108-3.0, Immediate Actions
(2) No

Answer	B

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Correct. (1) Common tap analysis is found in OP-CA-103-102-100. (2) Per OP-CA-103-102-100, the following alarms would be consistent with a common tap failure.

C. Incorrect. (1) Correct. (2) See below.

D. Incorrect. (1) Plausible if the operator misinterprets what immediate actions are contained in this procedure. (2) Plausible if the operator misinterprets what PT100Y feeds.

Question Information

Topic	Q84 - SRO-PZR Pressure Control Common Tap Failure				
User ID	Q2479510			**System ID**	2479510
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

<<NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	OP-CA-103-102-100, Rev 002		
Training Objective	Given conditions and/or parameters, associated with the RCS, determine if a common tap failure exists.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Reference(s)

P2.4.46	Safety Function	Tier 3	Group	RO Imp: 4.2	SRO Imp: 4.2
Ability to verify that the alarms are consistent with the plant conditions (CFR: 41.10 / 43.5 / 45.3 / 45.12)					
G.APE.027	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Pressurizer Pressure Control System Malfunction					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 85	ID: 2480020	Points: 1.00

SRO ONLY

Using Provided References.

Unit-1 is at 100% power.

- 1RV200, PZR Safety RV, is leaking
- Proportional heater banks 11 & 12 are available
- **ONLY** Backup heater banks 11 & 14 are available
- PZR level indicates 235 inches and rising
- One charging pump in service
- Letdown flow is at minimum
- PZR pressure is 2230 psia and lowering

(1) Which statement describes the PZR status?

(2) What action, if any, is required?

A. (1) Inoperable due to level and heater capacity
(2) Restore level within 6 hours and restore heater capacity within 72 hours

B. (1) Inoperable due to level and heater capacity
(2) Restore level within 72 hours and restore heater capacity within 6 hours

C. (1) Inoperable due to level **ONLY**
(2) Restore level within 6 hours

D. (1) Inoperable due to heater capacity **ONLY**
(2) Restore heater capacity within 72 hours

Answer	A

Answer Explanation

A. Correct - 1st part is correct since Tech Spec 3.4.9 Pressurizer requires Pressurizer water level greater than or equal to 133 inches and less than or equal to 225 inches. In addition, Tech Spec 3.4.9 requires two banks of Backup Heaters capable of being powered from an emergency power supply. 11&13 Backup Heater banks are the safety related heaters backed up by an Emergency Diesel Generator. A given pressurizer level of 235 inches and only 11&14 Backup Heaters available makes the PZR inoperable for both reasons. 2nd part is correct since Tech Spec LCO 3.4.9.A for Level has a 6 hour Completion Time and 3.4.9.B for Heaters has a 72 hour Completion Time.

B. Incorrect - 1st part is correct as stated above. 2nd part is incorrect but plausible since the Completion Times are those used in Tech Spec LCO 3.4.9 but swapped between 3.4.9.A and 3.4.9.B.

C. Incorrect - 1st part is incorrect since heater capacity does not meet the LCO requirement as stated above. 2nd part is correct since it contains the Completion Time of Tech Spec 3.4.9.A.

D. Incorrect - 1st part is incorrect since pressurizer level also does not meet the LCO requirement as stated above. 2nd part is correct since it contains the Completion Time of Tech Spec 3.4.9.B.

Question Information

Topic	Q85 - SRO-References: Operability of the PZR				
User ID	Q2480020			**System ID**	2480020
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	TS 3.4.9 REDACTED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.		
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	Tech Spec 3.4.9, Pages 3.4.9-1 to 3.4.9-2 **Ensure PZR level is redacted in both places in the Tech Spec pages**
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	Tech Spec 3.4.9 Rev 227/201, 314/292		
Training Objective	Given conditions and/or parameters associated with RCS instrumentation and appropriate references, identify the correct operator actions per the Technical Specifications.		

Previous NRC Exam Use	2019 NRC SRO Exam

K/A Reference(s)

SYS.010.A2.08	Safety Function	Tier 2	Group	RO Imp: 4.5	SRO Imp: 3.9
Ability to (a) predict the impacts of the following on the Pressurizer Pressure Control 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) Safety valves failure to reseat					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 86	ID: 2479944	Points: 1.00

SRO ONLY

Unit-2 is at 100% power.

- A fire alarm for 21 SGFP actuates
- The FASW reports a large fire on 21 SGFP

(1) What fire protection actuation occurred to aid in firefighting efforts at 21 SGFP?

(2) After 21 SGFP is tripped, the Unit Supervisor will direct a rapid downpower per AOP-3G, Malfunction of Main Feedwater System, to what power level?

A. (1) Halon activation
(2) 93%

B. (1) Halon activation
(2) 88%

C. (1) Deluge activation
(2) 88%

D. (1) Deluge activation
(2) 93%

Answer	C

Answer Explanation

A. Incorrect. (1) Plausible if the operator misinterprets what type of fire system is installed for the SGFPs. (2) Plausible since this is a power level that the US would direct in other casualties.

B. Incorrect. (1) See above. (2) Correct.

C. Correct. (1) Per 1C24A, the SGFPs are protected with deluge systems. (2) Per AOP-3G, a downpower to 88% is required for a trip of a single SGFP.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Q86 - SRO-Fire on 21 SGFP impacts on the plant				
User ID	Q2479944			**System ID**	2479944
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	1C24B-ALM, Rev 04900 AOP-3G, Rev 01700		
Training Objective	Given alarm manuals 1C17 and 1C24B, state the indication of and Operator actions for the alarms associated with the system.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.086.A2.05	Safety Function	Tier 2	Group	RO Imp: 4.0	SRO Imp: 3.8
Ability to (a) predict the impacts of the following on the Fire Protection 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) Fire in the plant					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 87	ID: 2480018	Points: 1.00

SRO ONLY

Which of the following correctly describes the relationship between success paths and the EOP-8 safety function status checks?

A. The safety function status check will determine which success path should be in use.

B. When the safety function status check acceptance criteria is met, the lowest numbered success path can be implemented.

C. When the safety function status check acceptance criteria is met, the highest numbered success path can be implemented.

D. The success path in use will determine what the safety function status check acceptance criteria is.

Answer	D

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets the safety function status check for the resource assessment table. The logic is reversed, success path determines which SFSC will be used.

B. Incorrect. Plausible since more than one success path may be implemented at the same time and the operator may misinterpret which acceptance criteria is required to be used. Meeting acceptance criteria does not dictate moving to another success path for that same safety function.

C. Incorrect. Plausible since more than one success path may be implemented at the same time and the operator may misinterpret which acceptance criteria is required to be used. Meeting acceptance criteria does not dictate moving to another success path for that same safety function.

D. Correct. This is the correct logic per EOP-8 Technical Basis Document, General Overview section.

Question Information

Topic	Q87 - SRO-EOP-8 and Relationship between Success Paths and the SFSC				
User ID	Q2480018			**System ID**	2480018
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	EOP-8 Technical Basis Rev 04400		
Training Objective	Recall the relationship between success paths and the SFSCs of EOP-8.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.4.23	Safety Function	Tier 3	Group	RO Imp: 3.4	SRO Imp: 4.4
Knowledge of the bases for prioritizing emergency operating procedures implementation (CFR: 41.10 / 43.5 / 45.13)					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 88	ID: 2479716	Points: 1.00

SRO ONLY

Using Provided References.

A loss of 11A and 11B Service Water Heat Exchangers has occurred.

Which of the following describes the required actions?

A. Isolate flow to 11 CAC.

B. Declare the 1A EDG out of service.

C. Place the plant in hot standby within 78 hours.

D. Place the plant in hot standby within 7 days.

Answer	C

Answer Explanation

A. Incorrect. Plausible if the TS condition of one SRW heat exchanger inoperable is interpreted to mean one header OOS.

B. Incorrect. Plausible if 11 header impacts the 1A EDG since 21 header would impact 2A EDG.

C. Correct. T.S. 3.7.6 allows an out-of-service time of 72 hours followed by a requirement to be in Hot Standby within 6 hours. Compliance with these conditions would require the plant be placed in Hot Standby within 78 hours.

D. Incorrect. Plausible if the required time to restore a SRW subsystem is also the time required to be in Mode 3 if the condition of B is not met.

Question Information

Topic	Q88 - SRO-References: SRW Hx operability				
User ID	Q2479716			**System ID**	2479716
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	TS 3.7.6	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.		

<<NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	Tech Spec 3.7.6 Rev 326/304		
Training Objective	Given plant conditions and the appropriate references, DETERMINE the required Technical Specifications actions.		
Previous NRC Exam Use	None		

References Provided	Tech Spec 3.7.6 Rev 326/304
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Reference(s)

SYS.076.A2.07	Safety Function	Tier 2	Group	RO Imp: 3.6	SRO Imp: 3.2
Ability to (a) predict the impacts of the following on the Service Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 41.1 / 43.5 / 45.3 / 45.6 / 45.13) Heat exchanger and condenser failure					

Learning Objective(s)

SRO NRC Test

User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 89	ID: 2479945	Points: 1.00

SRO ONLY

Using provided references.

Unit-1 is at 100% power.

- The Unit Supervisor is evaluating the removal of 13B Condenser Waterbox from service
- Condenser Vacuum is 27.0" Hg and steady
- Circulating Water Inlet temperature is 70.5°F and steady
- An ASI Plan is available
- JIT training is complete
- The Day Ahead LMP is $120 during the proposed maintenance period

(1) What reactor power level will the Unit Supervisor direct the crew to establish prior to removing the waterbox from service per OI-14A Appendix B?

(2) What action is required per the Planned Power Reduction Decision Tree of OI-14A Appendix B?

A. (1) 97%
(2) Stop unless Emergent Maintenance. Contact OMC.

B. (1) 97%
(2) Continue with maintenance preparation. Provide notification to M-NO and M-IWM.

C. (1) 93%
(2) Continue with maintenance preparation. Provide notification to M-NO and M-IWM.

D. (1) 93%
(2) Stop unless Emergent Maintenance. Contact OMC.

Answer	A

Answer Explanation

A. Correct. (1) Correct as shown in the reactor power table in Appendix B. (2) Correct as shown in the decision tree of Appendix B due to the Day Ahead LMP being greater than $100.

B. Incorrect. (1) Correct as shown above. (2) Incorrect but plausible since this action is required per the Appendix B decision tree if the actual power or plant parameter is within the yellow band of predicted.

C. Incorrect. (1) Incorrect as stated below. (2) Incorrect as stated above.

D. Incorrect. (1) Incorrect but plausible since 93% is another reactor power level that is shown in the table in Appendix B. (2) Correct as stated above.

Question Information

Topic	Q89 - SRO-References: CW Condenser Waterbox Cleaning				
User ID	Q2479945			**System ID**	2479945
Status	Active	**Point Value**	1.00	**Time (min)**	5

Open or Closed Reference	OI-14A-1 APP B	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	OI-14A-1 Appendix B (pages 88-89)
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	OI-14A-1 Rev 03200		
Training Objective	Determine required actions for removing a CW Condenser Waterbox from service.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.4.47	Safety Function	Tier 3	Group	RO Imp: 4.2	SRO Imp: 4.2
Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential) (CFR: 41.10 / 43.5 / 45.12)					
G.SYS.075	Safety Function	Tier	Group	RO Imp:	SRO Imp:
CW Circulating Water System					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 90	ID: 2479914	Points: 1.00

SRO ONLY

- EOP-6 (Steam Generator Tube Rupture) in progress
- All Intermediate Safety Function Status Checks are met
- Isolation of the affected SG is in progress

Then,

- A 200 GPM RCS leak from a PZR common tap occurs

The EOP-6 ISFSC no longer being met is ___(1)___.

With two events in progress, the Unit Supervisor will direct the CRO to complete the isolation of the affected SG per ___(2)___.

A. (1) Containment Environment
(2) the HR Success Path in EOP-8 (Functional Recovery Procedure) **ONLY**.

B. (1) Containment Environment
(2) either EOP-6 or the HR Success Path in EOP-8 (Functional Recovery Procedure)

C. (1) Rad Levels External to Containment
(2) the HR Success Path in EOP-8 (Functional Recovery Procedure) **ONLY**.

D. (1) Rad Levels External to Containment
(2) either EOP-6 or the HR Success Path in EOP-8 (Functional Recovery Procedure)

Answer	B

Answer Explanation

A. Incorrect. When LOCA starts, EOP-6 ISFSC for Containment Environment (CE) will no longer be met due to sump level alarm that will come in and, eventually, rising Containment Pressure. Only HR is plausible since EOP-8 is implemented when two events occur and the Optimal EOP is eventually exited. However, per OP-CA-103-102-1001, US can direct continued execution of required Optimal EOP steps while EOP-8 is implemented.

B. Correct. When LOCA starts, EOP-6 ISFSC for Containment Environment (CE) will no longer be met due to sump level alarm that will come in and, eventually, rising Containment Pressure. Per OP-CA-103-102-1001, US can direct continued execution of required Optimal EOP steps while EOP-8 is implemented, which would allow more timely isolation of the affected SG.

C. Incorrect. RLEC is plausible since EOP-5 RLEC SFSC would not be met with a SGTR in progress. However, RLEC is only not met in EOP-6 if the affected SG is being steamed with ADV or affected SG pressure is > 920 PSIA, and the stem stated that all EOP-6 ISFSC were initially met. Any potential external radiation due to the LOCA would not result in EOP-6 RLEC being declared not met. Only HR is plausible since EOP-8 is implemented when two events occur and the Optimal EOP is eventually exited. However, per OP-CA-103-102-1001, US can direct continued execution of required Optimal EOP steps while EOP-8 is implemented.

D. Incorrect. RLEC is plausible since EOP-5 RLEC SFSC would not be met with a SGTR in progress. However, RLEC is only not met in EOP-6 if the affected SG is being steamed with ADV or affected SG pressure is > 920 PSIA, and the stem stated that all EOP-6 ISFSC were initially met. Any potential external radiation due to the LOCA would not result in EOP-6 RLEC being declared not met. Per OP-CA-103-102-1001, US can direct continued execution of required Optimal EOP steps while EOP-8 is implemented.

Question Information

Topic	Q90 - SRO-EOP-6 to EOP-8 Implementation				
User ID	Q2479914			**System ID**	2479914
Status	Active	**Point Value**	1.00	**Time (min)**	4

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Examinee must assess plant conditions during an emergency and determine the Safety Function Status Checks no longer being met. At Calvert Cliffs, only SROs are responsible for assessing Safety Function Status in Optimal EOPs and given the responsibility to direct continued Optimal EOP steps while EOP-8 is implemented.
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3

Technical Reference and Revision #	EOP-6, Rev 02000 OP-CA-103-102-1001, Rev 008 EOP-8, Rev 04400
Training Objective	Given a set of plant conditions, demonstrate an understanding of the strategy and basis of EOP-8, including: EOP-8 Entry Process, identifying success paths in accordance with the Resource Assessment Table (RAT), and implementing success paths with the correct priority.
Previous NRC Exam Use	2018 NRC

K/A Reference(s)

P2.1.09	Safety Function	Tier 3	Group	RO Imp: N/A	SRO Imp: 4.5
Ability to direct licensed personnel activities inside the control room (SRO Only) (CFR: 43.1 / 45.5 / 45.12 / 45.13)					

Learning Objective(s)

SRO NRC Test

User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 91	ID: 2481825	Points: 1.00

SRO ONLY

Using Provided References.

What condition will require IRU filter testing per the Ventilation Filter Testing Program of Tech Spec 5.5.11?

A. After 18 months since 11 IRU was previously tested

B. After 360 hours since 11 IRU was previously tested

C. Fire and smoke through 11 IRU ventilation zone while running

D. Inadvertent Containment Spray flow through 11 IRU ventilation zone while running

Answer	C

Answer Explanation

A. Incorrect. Plausible since 18 months is the requirement for all other ventilation systems other than the IRUs.

B. Incorrect. Plausible if the operator misinterprets the requirement and concludes that half of the 720 hour limit requires testing to be performed.

C. Correct. As stated in Tech Spec 5.5.11 Ventilation Filter Testing Program.

D. Incorrect. Plausible if the operator concludes that the ingestion of excessive moisture through the IRU will cause inoperability by overloading the installed moisture separators.

Question Information

Topic	Q91 - SRO-References: IRU and Fire Impact				
User ID	Q2481825			**System ID**	2481825
Status	Active	**Point Value**	1.00	**Time (min)**	4

Open or Closed Reference	TS 5.5.11	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.		

References Provided	TS 5.5.11, Pgs 5.5-11 to 5.5-13
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	TS 5.5.11 Rev 336/314		
Training Objective	Determine the impact to safety related systems and the Tech Spec requirements.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.027.A2.01	Safety Function	Tier 2	Group	RO Imp: 2.2	SRO Imp: 2.7
Ability to (a) predict the impacts of the following on the Containment Iodine Removal 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) High temperature in the filter system					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 92	ID: 2479542	Points: 1.00

SRO ONLY

Both units are at 100% power.

At 1000:
- Offsite power is lost
- 24 4KV bus is faulted
- The 2A EDG fails to start

At 1005:
- The 0C DG is placed on 21 4KV Bus

At 1015:
- The 0C DG trips due to a mechanical failure
- No source of power is expected back in the next 4 hours

(1) At time 1015, what is the status of power availability to instrumentation and controls?

(2) When is an ELAP required to be declared per EOP-7, Station Blackout?

A. (1) Only 125V Vital DC Bus loads remain energized
(2) 1100

B. (1) All 120V Vital AC and 125V Vital DC Bus loads remain energized
(2) 1100

C. (1) Only 125V Vital DC Bus loads remain energized
(2) 1115

D. (1) All 120V Vital AC and 125V Vital DC Bus loads remain energized
(2) 1115

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible if the operator misinterprets what the batteries maintain power to in EOP-7. (2) Plausible if the operator misinterprets that the ELAP clock starts at the loss of offsite power.

B. Incorrect. (1) Correct. (2) See above.

C. Incorrect. (1) See above. (2) Correct.

D. Correct. (1) The 120V Vital AC Busses are supplied by the 125V DC batteries via their respective inverters. (2) Per EOP-7, an ELAP is declared 1 hours after the SBO.

Question Information

Topic	Q92 - SRO-Instruments and controls available during an ELAP				
User ID	Q2476542			**System ID**	2479542
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-7, Rev 02200		
Training Objective	Given a station blackout, demonstrate an understanding of the strategy, basis and operator actions of EOP-7.		
Previous NRC Exam Use	None		

K/A Reference(s)

EPE.055.EA2.04	Safety Function	Tier 1	Group	RO Imp: 4.1	SRO Imp: 3.9
Ability to determine and/or interpret the following as they apply to a Station Blackout: (CFR: 43.5 / 45.13) Instruments and controls operable with only DC battery power available					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 93	ID: 2479512	Points: 1.00

SRO ONLY

What condition allows the exit of EOP-4, Excess Steam Demand Event, and permits the implementation of OP-5, Plant Shutdown From Hot Standby to Cold Shutdown?

A. When a rediagnosis of the event using the EOP-0 Diagnostic Flow Chart confirms a second event in progress.

B. When blowdown of the affected steam generator is complete and RCS temperature is no longer lowering.

C. When Shutdown Cooling has been initiated per OI-3B, Shutdown Cooling.

D. When all Safety Function Status Check Final Acceptance criteria are met.

Answer	D

Answer Explanation

A. Incorrect. Plausible if the operator recognizes the requirement to transition from EOP-4 and incorrectly concludes that entry into OP-5 is allowed in this condition.

B. Incorrect. Plausible if the operator misinterprets that the casualty is over so EOP-4 is no longer needed.

C. Incorrect. Plausible since the final plant operating Block Step AC is to commence Shutdown Cooling which establishes the conditions for entry into OP-5.

D. Correct. As stated in EOP-4 Block Step AD, ensure that all Safety Function Status Check Final Acceptance Criteria are met.

Question Information

Topic	Q93 - SRO-EOP-4 Exit Conditions				
User ID	Q2479512			**System ID**	2479512
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-4-1, Rev 02000		
Training Objective	Given an ESDE, identify and understand the basis and actions to mitigate the event in accordance with EOP-4.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.4.51	Safety Function	Tier 3	Group	RO Imp: 3.0	SRO Imp: 4.0
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)					
G.APE.040	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Steamline Rupture					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 94	ID: 2479494	Points: 1.00

SRO ONLY

Unit-1 has implemented EOP-1 after an uncomplicated Reactor Trip.

Plant parameters have been stabilized at the following values:
- PZR Pressure is 2250 psia and steady
- PZR Level is 160" and steady
- 11&12 S/G Levels are 0" and steady
- 11&12 S/G Pressures are 900 psia and steady
- Containment Pressure is 0.1 psig and steady

10 minutes after a new plant transient occurred, the STA reports the following plant parameters:
- PZR Pressure is 1710 psia and lowering
- PZR Level is 60" and lowering
- 11&12 S/G Levels are 0" and steady
- 11&12 S/G Pressures are 900 psia and steady
- Containment Pressure is 1.9 psig and rising

Based on the new event, the Unit Supervisor will transition to ___(1)___ since the ___(2)___ Safety Function Status Check(s) is(are) not met in EOP-1.

A. (1) EOP-6
(2) Pressure and Inventory Control ONLY

B. (1) EOP-5
(2) Pressure and Inventory Control ONLY

C. (1) EOP-6
(2) Pressure and Inventory Control AND Containment Environment

D. (1) EOP-5
(2) Pressure and Inventory Control AND Containment Environment

Answer	D

Answer Explanation

A. Incorrect - 1st part is incorrect but plausible since a diagnosis of a SGTR could be made due to the lowering Pressurizer Pressure and Level and then implementation of

EOP-6 would be chosen. The plausibility of EOP-6 is further supported since EOP-6 Block Step I provides direction to control RCS pressure and subcooling to minimize the RCS leakage and control Pressurizer Level. The higher Containment Pressure can be attributed to a SGTR which is considered to be an RCS Leak and loss of the RCS Barrier within the boundaries of Containment. 2nd part is incorrect but plausible if the acceptance criteria for Containment Environment is misinterpreted as the same value as the SIAS/CIS setpoint of 2.8 psig.

B. Incorrect - 1st part is correct as stated below. 2nd part is incorrect as stated above.

C. Incorrect - 1st part is incorrect as stated above. 2nd part is correct as stated below.

D. Correct - 1st part is correct since lowering Pressurizer Pressure and Level as well as high Containment Pressure can all be attributed to an RCS Leak inside Containment. For a LOCA, a transition to EOP-5 is appropriate based on comparing the Safety Function Acceptance Criteria for Containment Pressure in EOP-5 and EOP-6. EOP-5 allows a Containment Pressure less than 50 psig, but EOP-6 requires a Containment pressure less than 0.7 psig. Implementation of EOP-6 would be incorrect for this reason. 2nd part is correct since the PIC acceptance criteria is not met based on PZR pressure and level and the Containment Environment acceptance criteria is not met for Containment pressure.

Question Information

Topic	Q94 - SRO-Small Break LOCA EOP Transition				
User ID	Q2479494			**System ID**	2479494
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Performing the Safety Function Status Checks in EOPs, directing a transition to another EOP, and directing EOP Block Steps are all SRO-Only functions.
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-5-1 Rev 03200 EOP-6-1 Rev 02000		
Training Objective	Given a loss of Reactor Coolant, identify and understand the basis and actions to mitigate the event in accordance with EOP-5.		
Previous NRC Exam Use	2019 NRC SRO Exam		

K/A Reference(s)

P2.4.04	Safety Function	Tier 3	Group	RO Imp: 4.5	SRO Imp: 4.7
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures (CFR: 41.10 / 43.2 / 45.6)					
G.EPE.009	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Small-Break LOCA					

Learning Objective(s)

SRO NRC Test

User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 95	ID: 2479873	Points: 1.00

SRO ONLY

- A Loss of All Feedwater event from full power is in progress
- Once Through Core Cooling initiation criteria is met
- No HPSI Pumps are available

What decision will the Unit Supervisor make regarding the operation of the PORVs?

A. Open the PORVs to prevent the high pressure ejection of a damaged core.

B. Open the PORVs to allow AFW Pump injection when a pump becomes available.

C. Do not open the PORVs to minimize cycling which over time can cause damage.

D. Do not open the PORVs to prevent damaging the Quench Tank rupture disc

Answer	A

Answer Explanation

A. Correct as stated in the EOP-3 Basis document for Block Step J.

B. Incorrect but plausible since EOP-3 contains steps to prepare for the availability of an AFW Pump.

C. Incorrect but plausible since minimizing cycling is a stated reason in the EOP-3 Basis document for Block Step J.

D. Incorrect but plausible since the rupturing of the Quench Tank rupture disc is discussed in the EOP-3 Basis document for Block Step J.

Question Information

Topic	Q95 - SRO-PORV Decision in EOP-3				
User ID	Q2479873			System ID	2479873
Status	Active	Point Value	1.00	Time (min)	3

Open or Closed Reference	CLOSED	Cognitive Level	HIGH
Operator Discipline	LO-I	Operator Type	SRO
10CFR55 Content	CFR: 43.1 Conditions and limitations in the facility license.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	EOP-3 Technical Basis Rev 02900		
Training Objective	Determine required actions and strategies during a Loss of All Feedwater event.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.4.18	Safety Function	Tier 3	Group	RO Imp: 3.3	SRO Imp: 4.0
Knowledge of the specific bases for emergency and abnormal operating procedures (CFR: 41.10 / 43.1 / 45.13)					
G.SYS.007	Safety Function	Tier	Group	RO Imp:	SRO Imp:
PRTS Pressurizer Relief Tank/Quench Tank System					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 96	ID: 2479879	Points: 1.00

SRO ONLY

When it is determined that more information is needed to determine if an Issue Report for a SSC covered by the Technical Specifications affects the SSC, Operations will request engineering to perform a(an) ___(1)___.
The engineering product should be completed within ___(2)___ working days unless approval from the Site Vice President is obtained.

A. (1) Operability Evaluation
(2) 3

B. (1) Operability Evaluation
(2) 7

C. (1) Functionality Evaluation
(2) 3

D. (1) Functionality Evaluation
(2) 7

Answer	A

Answer Explanation

A. Correct. (1) Per OP-AA-108-115, Operability Evaluations are requested for TS SSCs. (2) Per OP-AA-108-115, the OpEval should be completed in 3 working days.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since this is the request for equipment that may impact TS related SSCs. (2) Plausible since 7 days is a common number of days used in the TS to allow for time to access the situation.

Question Information

Topic	Q96 - SRO-OpEval information				
User ID	Q2479879			**System ID**	2479879
Status	Active	**Point Value**	1.00	**Time (min)**	2

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.1 Conditions and limitations in the facility license.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	OP-AA-108-115, Rev 024		
Training Objective	Apply the requirements of OP-AA-108-115, Operability Determinations.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.2.37	Safety Function	Tier 3	Group	RO Imp: N/A	SRO Imp: 4.6
Ability to determine operability or availability of safety related equipment (SRO Only) (CFR: 43.2 / 43.5 / 45.12)					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 97	ID: 2479917	Points: 1.00

SRO ONLY

Unit-2 is at 100% power.

- Large Break LOCA occurs
- EOP-5, Loss of Coolant Accident, is implemented

Then the following is noted:

- All SRW Pumps are found not running
- Containment Pressure is 15 PSIG and steady

(1) What is the impact of restoring SRW flow under the stated conditions?

(2) What direction will the Unit Supervisor give regarding the 21 SRW header?

A. (1) Water hammer damage from SRW header voiding may occur.
(2) Start the desired SRW Pump on 21 SRW header.

B. (1) Water hammer damage from SRW header voiding may occur.
(2) Isolate SRW to 21 and 23 Containment Air Coolers.

C. (1) SRW Pump minimum flow requirements not being met.
(2) Start the desired SRW Pump on 21 SRW header.

D. (1) SRW Pump minimum flow requirements not being met.
(2) Isolate SRW to 21 and 23 Containment Air Coolers.

Answer	A

Answer Explanation

A. Correct. (1) Correct as stated in the EOP-5 Caution prior to Step 6 in Block Step H. (2) Correct as stated in Step 6 of Block Step H. As long as Containment pressure remained less than 25 PSIG, 21 SRW header will be restored without the need to isolate any CACs.

B. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible since this action is directed in Block Step H if Containment pressure exceeded 25 PSIG.

C. Incorrect. (1) Incorrect but plausible since this is the Caution prior to the alternate action Step 6.a.1(1) if 21 SRW header is restored with 21 and 22 CACs isolated. The SRW flow to the 2A EDG does not meet minimum flow requirements. (2) Correct as stated above.

D. Incorrect. (1) Incorrect as stated above. (2) Incorrect as stated above.

Question Information

Topic	Q97 - SRO-CAC Strategy During LOCA with no SRW				
User ID	Q2479917			**System ID**	2479917
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-5-2, Rev 03200		
Training Objective	Given a value of containment pressure, determine if a SRW header may be returned to service.		
Previous NRC Exam Use	None		

K/A Reference(s)

SYS.022.A2.04	Safety Function	Tier 2	Group	RO Imp: 3.7	SRO Imp: 3.4
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: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Cooling water system malfunction					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 98	ID: 2479918	Points: 1.00

SRO ONLY

The plant is operating at 100% power with the following:

- At time 0100 on March 1st, a common cause failure is discovered which makes multiple pieces of Technical Specification required equipment inoperable.
- The affected LCOs are applicable in Modes 1, 2, and 3.
- The Shift Manager immediately enters Technical Specification LCO 3.0.3.
- Corrective measures will take at least 3 days to restore compliance with Technical Specifications.

What is the latest time by which Technical Specifications require the plant to be in Mode 5?

A. 1400 on March 1st

B. 2100 on March 1st

C. 1400 on March 2nd

D. 1000 on March 3rd

Answer	C

Answer Explanation

A. Incorrect. Plausible if the LCO requirements are misinterpreted and the time to mode 5 vice mode 3 is thought to be 7 hours.

B. Incorrect. Plausible if the LCO requirements are misinterpreted and the time to mode 5 vice mode 4 is thought to be 13 hours and if the 7 and 13 hour time requirements were added together and run in series not parallel.

C. Correct. LCO 3.0.3 requires initiating actions within 1 hour to place the place in Mode 5 within 37 hours.

D. Incorrect. Plausible if the 7, 13, and 37 hours time requirements were added together and run in series not parallel.

Question Information

Topic	Q98 - SRO-LCO 3.0.3 time requirements				
User ID	Q2479918			**System ID**	2479918
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	HIGH
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.2 Facility operating limitations in the technical specifications and their bases.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	Tech Spec 3.0.3, Amend 285		
Training Objective	Given a set of plant or system conditions, evaluate the conditions and apply the appropriate actions in accordance with the Tech Specs and/or TRM.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.2.38	Safety Function	Tier 3	Group	RO Imp: 3.6	SRO Imp: 4.5
Knowledge of conditions and limitations in the facility license (CFR: 41.7 / 41.10 / 43.1 / 45.13)					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 99	ID: 2479678	Points: 1.00

SRO ONLY

- A Site Area Emergency has been declared for a LOCA with inadequate RCS inventory.
- The Shift Manager has assumed responsibility as the Emergency Director.
- The Emergency Offsite Facility and Technical Support Center are NOT yet activated.

What task or responsibility may the Emergency Director delegate to another individual per EP procedures?

A. Reclassifying the event in progress.

B. Making Protective Action Recommendations.

C. Notification and activation of the ERO.

D. Determination of offsite emergency notifications to state and local authorities.

Answer	C

Answer Explanation

A. Incorrect. Plausible because once the event is classified it could be misinterpreted that another qualified SRO would be allowed to perform this function, however, The SM/ED is required to make all classifications or changes to classifications until relieved by another ED.

B. Incorrect. Plausible if the operator misinterprets that the Dose Assessor could make the PARs since they help during PAR declaration.

C. Correct. The Emergency Director may request another person to direct notification of Offsite Agencies such as NRC Headquarters. This is not designated as a non-delegable duty per the checklist.

D. Incorrect. Plausible if the operator misinterprets what actions the STA is allowed to perform.

Question Information

Topic	Q99 - SRO-Non-delegable duties of SM/ED				
User ID	Q2479678			**System ID**	2479678
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	Bank	**Difficulty**	2
Technical Reference and Revision #	EP-AA-112-100-F-01, Rev AD EP-AA-112-400-F-02, Rev V		
Training Objective	Given any event the trainee will respond to the event and take appropriate actions to protect the public and plant personnel in accordance with ERO procedures.		
Previous NRC Exam Use	None		

K/A Reference(s)

P2.4.40	Safety Function	Tier 3	Group	RO Imp: N/A	SRO Imp: 4.5
Knowledge of SRO responsibilities in emergency plan implementing procedures (SRO Only) (CFR: 43.5 / 45.11)					
G.EPE.074	Safety Function	Tier	Group	RO Imp:	SRO Imp:
Inadequate Core Cooling					

Learning Objective(s)

SRO NRC Test
User (Sys) ID N/A (1545505)

Cross Reference Links

None

Question 100	ID: 2479885	Points: 1.00

SRO ONLY

Which one of the following describes a responsibility of the Unit Supervisor when signing a Gaseous Release Permit?

A. Enter termination criteria into the Plant Computer for high flow rate and high activity as indicated on the Release Permit.

B. Ensure required plant systems are in operation and the required plant configuration has been established.

C. Ensure radiation monitor alarm setpoints on 1C22 have been adjusted to the appropriate values.

D. Perform the pre-release source check and channel check.

Answer	B

Answer Explanation

A. Incorrect - Entering termination criteria into the computer is plausible since the SM/SRO would perform an IV that the termination criteria have properly been entered into the Plant Computer. However, entry of the data into the Plant Computer is the responsibility of the CRO.

B. Correct - Per CY-CA-170-604 Attachment 4, the SM/SRO signature verifies the release criteria are understood, that required plant systems are in operation and that the required plant configuration has been established.

C. Incorrect - Adjusting RMS setpoints is plausible to the Operator since the RMS Warning/Alert/Critical alarm values from the permit are inputted to the Plant Computer for termination criteria. However, RMS alarm setpoints are not adjusted during a discharge.

D. Incorrect - Performing pre-release source check and channel checks is plausible to the Operator since these are actions required and initialed for on the Waste Gas Permit. However, these checks are the responsibility of the CRO. Per CY-CA-170-604 Attachment 4, the SM/SRO signature verifies the release criteria are understood, that required plant systems are in operation and that the required plant configuration has been established.

Question Information

Topic	Q100 - SRO-Approve a Gaseous Release Permit				
User ID	Q2479885			**System ID**	2479885
Status	Active	**Point Value**	1.00	**Time (min)**	3

Open or Closed Reference	CLOSED	**Cognitive Level**	LOW
Operator Discipline	LO-I	**Operator Type**	SRO
10CFR55 Content	CFR: 43.4 Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.		

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	CY-CA-170-604 Rev 0		
Training Objective	Determine when a Waste Gas Decay Tank can be isolated and/or released.		
Previous NRC Exam Use	2016 NRC		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Reference(s)

P2.3.06	Safety Function	Tier 3	Group	RO Imp: 2.0	SRO Imp: 3.8
Ability to approve liquid or gaseous release permits (CFR: 41.13 / 43.4 / 45.10)					

Learning Objective(s)

SRO NRC Test

User (Sys) ID N/A (1545505)

Cross Reference Links

None