Form 4.1-PWR Pressurized-Water Reactor Examination Outline

Facility: Seabrook **K/A Catalog Rev. 3** **Rev.** DRAFT **Date of Exam:** 7/18/2023

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

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

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

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				(000007EK3.02) Knowledge of the reasons for the following responses and/or actions as they apply to (EPE 7) REACTOR TRIP, STABILIZATION, RECOVERY (CFR: 41.5 / 41.10 / 45.6 / 45.13): Verifying a reactor trip	4.5	8
2	(000009) (EPE 9) Small Break LOCA				X			(000009EA1.17) Ability to operate and/or monitor the following as they apply to (EPE 9) SMALL-Break LOCA (CFR: 41.5 / 41.7 / 45.5 to 45.8): PRT/quench tank	3.2	75
3	(000015) (APE 15) Reactor Coolant Pump Malfunctions						X	(000015) (APE 15) Reactor Coolant Pump Malfunctions (G2.2.44) EQUIPMENT CONTROL: Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions (CFR: 41.5 / 43.5 / 45.12)	4.2	29
4	(000022) (APE 22) Loss of Reactor Coolant Makeup	X						(000022AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 22) LOSS OF REACTOR Coolant Makeup (CFR: 41.5 / 41.7 / 45.7 / 45.8): Relationship between charging flow and PZR level	3.6	58
5	(000025) (APE 25) Loss of Residual Heat Removal System				X			(000025AA1.03) Ability to operate and/or monitor the following as they apply to (APE 25) LOSS OF RESIDUAL Heat Removal System (CFR: 41.5 / 41.7 / 45.5 to 45.8): RHR	4.0	53
6	(000026) (APE 26) Loss of Component Cooling Water		X					(000026AK2.05) Knowledge of the relationship between (APE 26) LOSS OF Component Cooling Water and the following systems or components (CFR: 41.8 / 41.10 / 45.3): RMS	3.0	37
7	(000027) (APE 27) Pressurizer Pressure Control System Malfunction		X					(000027AK2.10) Knowledge of the relationship between (APE 27) PRESSURIZER PRESSURE Control System Malfunction and the following systems or components (CFR: 41.8 / 41.10 / 45.3): PZR pressure transmitters	3.7	46
8	(000029) (EPE 29) Anticipated Transient Without Scram			X				(000029EK3.11) Knowledge of the reasons for the following responses and/or actions as they apply to (EPE 29) ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) (CFR: 41.5 / 41.10 / 45.6 / 45.13): Initiating emergency boration	4.3	65
9	(000038) (EPE 38) Steam Generator Tube Rupture					X		(000038EA2.07) Ability to determine and/or interpret the following as they apply to (EPE 38) STEAM GENERATOR Tube Rupture (CFR: 41.10 / 43.5 / 45.13): Plant conditions from survey of control room indications	4.0	64
10	(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer					X		(WE12EA2.06) Ability to determine and/or interpret the following as they apply to (W E12) UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS (CFR: 41.10 / 43.5 / 45.13): Core exit temperatures and/or subcooling	3.9	6
11	(000054) (APE 54; CE E06) Loss of Main Feedwater						X	(000054) (APE 54; CE E06) Loss of Main Feedwater (G2.1.20) CONDUCT OF OPERATIONS: Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12)	4.6	31
12	(000056) (APE 56) Loss of Offsite Power			X				(000056AK3.02) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 56) Loss of Offsite Power (CFR: 41.5 / 41.10 / 45.6 / 45.13): Actions contained in AOPs	4.1	16
13	(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	17

14	(000058) (APE 58) Loss of DC Power	X						(000058AK1.05) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 58) LOSS OF DC Power (CFR: 41.5 / 41.7 / 45.7 / 45.8): Prevention of inadvertent system(s) actuation upon restoration of DC power	3.6	48
15	(000065) (APE 65) Loss of Instrument Air	X						(000065AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 65) LOSS OF Instrument Air (CFR: 41.5 / 41.7 / 45.7 / 45.8): Failure modes of air-operated equipment	3.7	60
16	(000077) (APE 77) Generator Voltage and Electric Grid Disturbances						X	(000077) (APE 77) Generator Voltage and Electric Grid Disturbances (G2.4.31) EMERGENCY PROCEDURES/PLAN: Knowledge of annunciator alarms, indications, or response procedures (CFR: 41.10 / 45.3)	4.2	34
17	(W E04) LOCA Outside Containment					X		(WE04EA2.03) Ability to determine and/or interpret the following as they apply to (W E04) LOCA Outside Containment (CFR: 41.10 / 43.5 / 45.13): RCS pressure	3.9	1
18	(BW E04; W E05) Inadequate Heat Transfer – Loss of Secondary Heat Sink		X					(WE05EK2.16) Knowledge of the relationship between (W E05) Loss of Secondary Heat Sink and the following systems or components (CFR: 41.8 / 41.10 / 45.3): CNT	3.1	9
	(000008) (APE 8) Pressurizer Vapor Space Accident									
	(000011) (EPE 11) Large Break LOCA									
	(000026) (APE 26) Loss of Component Cooling Water									
	(000055) (EPE 55) Station Blackout									
	(000062) (APE 62) Loss of Nuclear Service Water									
	(W E11) Loss of Emergency Coolant Recirculation									
K/A Category Totals:		3	3	3	3	3	3	Group Point Total:		18

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

Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
19	(000028) (APE 28) Pressurizer (PZR) Level Control Malfunction					X		(000028AA2.16) Ability to determine and/or interpret the following as they apply to (APE 28) PRESSURIZER (PZR) Level Control Malfunction (CFR: 41.10 / 43.5 / 45.13): RCS leaks	3.4	49
20	(000032) (APE 32) Loss of Source Range Nuclear Instrumentation		X					(000032AK2.03) Knowledge of the relationship between (APE 32) LOSS OF SOURCE RANGE Nuclear Instrumentation and the following systems or components (CFR: 41.8 / 41.10 / 45.3): RPS	3.9	30
21	(000037) (APE 37) Steam Generator Tube Leak			X				(000037AK3.04) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 37) STEAM Generator Tube Leak (CFR: 41.5 / 41.10 / 45.6 / 45.13): Use of “feed and bleed” process	3.2	62
22	(000051) (APE 51) Loss of Condenser Vacuum	X						(000051AK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 51) LOSS OF Condenser Vacuum (CFR: 41.5 / 41.7 / 45.7 / 45.8): Relationship of condenser vacuum to circulating water, flow rate, and temperature	3.3	14
23	(000061) (APE 61) Area Radiation Monitoring System Alarms		X					(000061AK2.06) Knowledge of the relationship between (APE 61) AREA RADIATION MONITORING (ARM) System Alarms and the following systems or components (CFR: 41.8 / 41.10 / 45.3): Refueling pool	3.4	13
24	(000074) (EPE 74; W E06 & E07) Inadequate Core Cooling				X			(000074EA1.08) Ability to operate and/or monitor the following as they apply to (EPE 74) Inadequate Core Cooling (CFR: 41.5 / 41.7 / 45.5 to 45.8): ECCS	4.3	50
25	(000076) (APE 76) High Reactor Coolant Activity						X	(000076) (APE 76) High Reactor Coolant Activity (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	22
26	(W E15) Containment Flooding			X				(WE15EK3.06) Knowledge of the reasons for the following responses and/or actions as they apply to (W E15) Containment Flooding (CFR: 41.5 / 41.10 / 45.6 / 45.13): Determining the containment sump activity level	3.1	24
	(000001) (APE 1) Continuous Rod Withdrawal									
	(000059) (APE 59) Accidental Liquid Radwaste Release									
	(W E01 & E02) Rediagnosis & SI Termination									
	(BW E09; CE A13**; W E09 & E10) Natural Circulation									
	000003 (APE 3) Dropped Control Rod / 1									
	000005 (APE 5) Inoperable/Stuck Control Rod / 1									

	000024 (APE 24) Emergency Boration / 1									
	000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7									
	000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8									
	000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
	000067 (APE 67) Plant Fire On Site / 8									
	000068 (APE 68; BW A06) Control Room Evacuation / 8									
	000069 (APE 69; W E14) Loss of Containment Integrity / 5									
	000078 (APE 78*) RCS Leak / 3									
	(W E13) Steam Generator Overpressure / 4									
	(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 E13 & E14) EOP Rules and Enclosures									
	(CE A11**; W E08) RCS Overcooling – Pressurized Thermal Shock / 4									

	(CE A16) Excess RCS Leakage / 2									
	(CE E09) Functional Recovery									
	(CE E13*) Loss of Forced Circulation / LOOP / Blackout / 4									
K/A Category Totals:		1	2	2	1	1	1	Group Point Total:		8

Plant Systems—Tier 2/Group 1 (RO)

Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
27	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM		X										(003K2.04) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF4P RCP) REACTOR COOLANT PUMP SYSTEM Containment isolation valves for RCP cooling water	3.4	56
28	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM					X							(004K5.30) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM (CFR: 41.5 / 45.3): Relationship between temperature and pressure in CVCS components during solid plant operation	4.0	70
29	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM									X			(005A3.02) Ability to monitor automatic features of the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM, including (CFR: 41.7 / 45.7): RHRS actuation	4.2	43
30	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM					X							(005K5.10) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM (CFR: 41.5 / 45.3): RHRS suction vortexing during reduced RCS inventory	4.2	68
31	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM								X				(006A2.13) Ability to (a) predict the impacts of the following on the (SF2; SF3 ECCS) EMERGENCY CORE 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): Inadvertent ECCS actuation	3.6	36
32	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM		X										(006K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM ECCS pumps	4.1	15
33	(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM							X					(007A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM, including (CFR: 41.5 / 45.5): Containment radiation levels	3.2	32

34	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM			X									(008K3.01) Knowledge of the effect that a loss or malfunction of the (SF8 CCW) COMPONENT COOLING WATER SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): Loads cooled by CCWS	4.0	33
35	(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM										X		(010A4.02) Ability to manually operate and/or monitor the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): PZR heaters	3.6	25
36	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM	X											(012K1.04) Knowledge of the physical connections and/or cause and effect relationships between the (SF7 RPS) REACTOR PROTECTION SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): RPIS	3.8	47
37	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM			X									(012K3.05) Knowledge of the effect that a loss or malfunction of the (SF7 RPS) REACTOR PROTECTION SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): RCPS	3.5	10
38	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM				X								(013K4.15) Knowledge of (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Automatic circuit continuity testing	3.0	39
39	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM								X				(022A2.07) 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): ESFAS actuation	4.0	61
40	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM			X									(022K3.04) Knowledge of the effect that a loss or malfunction of the (SF5 CCS) CONTAINMENT COOLING SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): CNT	3.9	3
41	(026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM									X			(026A3.02) Ability to monitor automatic features of the (SF5 CSS) CONTAINMENT SPRAY SYSTEM, including (CFR: 41.7 / 45.7): Verification that cooling water is supplied to the containment spray heat exchanger	3.5	69

42	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM		X										(039K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM Safety-related MRSS valves	3.1	67
43	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM							X					(059A1.02) 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): MFW pump oil temperatures and MFW pump vibrations	2.9	42
44	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM									X			(059A3.08) Ability to monitor automatic features of the (SF4S MFW) MAIN FEEDWATER SYSTEM, including (CFR: 41.7 / 45.7): S/G water LCS	3.8	26
45	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM						X						(061K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM (CFR: 41.7 / 45.7): AFW flow controller	3.9	35
46	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM	X											(062K1.08) Knowledge of the physical connections and/or cause and effect relationships between the (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): Onsite standby power systems	3.8	27
47	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM											X	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM (191002K1.07) SENSORS AND DETECTORS (CFR: 41.7): (LEVEL) Theory and operation of level detectors	2.6	74
48	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM								X				(064A2.27) 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): Loss of DC power	3.8	5
49	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM						X						(073K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM (CFR: 41.7 / 45.7): PRM component failures	3.2	19

50	(076) (SF4S SW) SERVICE WATER SYSTEM							X					(076A1.02) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4S SW) SERVICE WATER SYSTEM, including (CFR: 41.5 / 45.5): Reactor and turbine building CCW temperatures	3.2	41
51	(076) (SF4S SW) SERVICE WATER SYSTEM			X									(076K3.05) Knowledge of the effect that a loss or malfunction of the (SF4S SW) SERVICE WATER SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): RHRS	3.6	54
52	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM											X	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM (G2.1.1) CONDUCT OF OPERATIONS: Knowledge of conduct of operations requirements (CFR: 41.10 / 43.10 / 45.13)	3.8	72
53	(103) (SF5 CNT) CONTAINMENT SYSTEM										X		(103A4.09) Ability to manually operate and/or monitor the (SF5 CNT) CONTAINMENT SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Containment vacuum system	3.0	38
54	(103) (SF5 CNT) CONTAINMENT SYSTEM				X								(103K4.05) Knowledge of (SF5 CNT) CONTAINMENT SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Containment construction	2.8	57
	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM														
	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM														
	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM														

	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM													
	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM													
	025 (SF5 ICE) ICE CONDENSER SYSTEM													
	053 (SF1; SF4P ICS*) INTEGRATED CONTROL SYSTEM													
K/A Category Totals:		2	3	4	2	2	2	3	3	3	2	2	Group Point Total:		28

Plant Systems—Tier 2/Group 2 (RO)

Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
55	(002) (SF2; SF4P RCS) REACTOR COOLANT SYSTEM											X	(002) (SF2; SF4P RCS) REACTOR COOLANT SYSTEM (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	52
56	(014) (SF1 RPI) ROD POSITION INDICATION SYSTEM										X		(014A4.05) Ability to manually operate and/or monitor the (SF1 RPI) ROD POSITION INDICATION SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): RPI accuracy mode selection (W)	3.1	55
57	(035) (SF4P SG) STEAM GENERATOR SYSTEM									X			(035A3.01) Ability to monitor automatic features of the (SF4P SG) STEAM GENERATOR SYSTEM, including (CFR: 41.7 / 45.7): S/G water level control	3.9	2
58	(041) (SF4S SDS) STEAM DUMP/TURBINE BYPASS CONTROL SYSTEM	X											(041K1.07) Knowledge of the physical connections and/or cause and effect relationships between the (SF4S SDS) STEAM DUMP/TURBINE BYPASS CONTROL SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): RPS	3.3	44
59	(050) (SF9 CRV*) CONTROL ROOM VENTILATION							X					(050A1.01) Ability to predict and/or monitor changes in parameters associated with operation of the (SF9 CRV) CONTROL ROOM VENTILATION, including (CFR: 41.5 / 45.5): Filter D/P	2.5	66
60	(055) (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM								X				(055A2.04) Ability to (a) predict the impacts of the following on the (SF4S CARS) CONDENSER AIR 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): Air in-leakage	3.6	12
61	(056) (SF4S CDS) CONDENSATE SYSTEM					X							(056K5.13) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4S CDS) CONDENSATE SYSTEM (CFR: 41.5 / 45.3): Purpose of low-pressure cleanup valve	2.5	40

62	(071) (SF9 WGS) WASTE GAS DISPOSAL SYSTEM						X						(071K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF9 WGS) WASTE GAS DISPOSAL SYSTEM (CFR: 41.7 / 45.7): Waste gas discharge release valve	3.2	73
63	(086) (SF8 FPS) FIRE PROTECTION SYSTEM				X								(086K4.06) Knowledge of (SF8 FPS) FIRE PROTECTION SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Carbon dioxide	3.2	28
	(001) (SF1 CRDS) CONTROL ROD DRIVE SYSTEM														
	(028) (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM														
	(033) (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM														
	011 (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM														
	015 (SF7 NI) NUCLEAR INSTRUMENTATION SYSTEM														
	016 (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM														
	017 (SF7 ITM) IN CORE TEMPERATURE MONITOR SYSTEM														

	027 (SF5 CIRS) CONTAINMENT IODINE REMOVAL SYSTEM													
	029 (SF8 CPS) CONTAINMENT PURGE SYSTEM													
	034 (SF8 FHS) FUEL HANDLING EQUIPMENT SYSTEM													
	045 (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM													
	068 (SF9 LRS) LIQUID RADWASTE SYSTEM													
	072 (SF7 ARM) AREA RADIATION MONITORING SYSTEM													
	075 (SF8 CW) CIRCULATING WATER SYSTEM													
	079 (SF8 SAS**) STATION AIR SYSTEM													
K/A Category Totals:		1	0	0	1	1	1	1	1	1	1	1	Group Point Total:		9

Form 4.1-COMMON Common Examination Outline

Generic Knowledge and Abilities Outline (Tier 3) (RO)							
				RO		SRO-Only	
Category	K/A #	Topic	Item #	IR	Q#	IR	Q#
1. Conduct of Operations	G2.1.7	(G2.1.7) CONDUCT OF OPERATIONS: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 43.5 / 45.12 / 45.13)	64	4.4	45		
	G2.1.38	(G2.1.38) CONDUCT OF OPERATIONS: Knowledge of the station's requirements for verbal communications when implementing procedures (CFR: 41.10 / 45.13)	65	3.7	63		
	Subtotal				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)	66	4.1	51		
	G2.2.22	(G2.2.22) EQUIPMENT CONTROL: Knowledge of limiting conditions for operation and safety limits (CFR: 41.5 / 43.2 / 45.2)	67	4	71		
	Subtotal				2		
3. Radiation Control	G2.3.11	(G2.3.11) RADIATION CONTROL: Ability to control radiation releases (CFR: 41.11 / 43.4 / 45.10)	68	3.8	18		
	Subtotal				1		
4. Emergency Procedures / Plan	G2.4.35	(G2.4.35) EMERGENCY PROCEDURES / PLAN: Knowledge of nonlicensed operator responsibilities during an emergency (CFR: 41.10 / 43.1 / 43.5 / 45.13)	69	3.8	23		
	Subtotal				1		
Tier 3 Point Total					**6**		

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Seabrook)		
Facility:	Seabrook	Date of Exam:	7/11/2023

Theory (Tier 4) (RO)

Category	K/A #	Topic	Item #	RO IR	RO Q#
Reactor Theory	192006	(192006K1.06) FISSION PRODUCT POISONS (CFR: 41.1): Describe the following processes and state their effect on reactor operations: -- transient xenon	70	3.4	59
	192007	(192007K1.04) FUEL DEPLETION AND BURNABLE POISONS (CFR: 41.1): Describe how and why boron concentration changes over core life	71	3.4	21
	192008	(192008K1.10) REACTOR OPERATIONAL PHYSICS (CFR: 41.1): (CRITICALITY) Describe reactor power and startup rate response once criticality is reached	72	3.4	11
	Subtotal				3
Thermodynamics	193003	(193003K1.25) STEAM (CFR: 41.14): Explain and use saturated and superheated steam tables	73	3.4	4
	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	74	2.8	7
	193009	(193009K1.03) CORE THERMAL LIMITS (CFR: 41.14): Explain local peaking factor	75	2.7	20
	Subtotal				3
Tier 4 Point Total					**6**

Examination Outline Cross-reference:	Level	RO	
Q1	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(W E04) LOCA Outside Containment

(WE04EA2.03) Ability to determine and/or interpret the following as they apply to (W E04) LOCA Outside Containment: RCS pressure

Importance Rating	3.9	

Proposed Question:

The crew is performing ECA-1.2, "LOCA Outside Containment".

After attempting to isolate the leak, if RCS pressure continues to decrease the crew will first transition to _____**(1)**_____.

The first action taken in the procedure transitioned to is to check if _____**(2)**_____ criteria are met.

	(1)	(2)
A.	E-1, "Loss of Reactor or Secondary Coolant"	cold leg recirculation
B.	ECA-1.1, "Loss of Emergency Coolant Recirculation"	cold leg recirculation
C.	E-1, "Loss of Reactor or Secondary Coolant"	ECCS termination
D.	ECA-1.1, "Loss of Emergency Coolant Recirculation"	ECCS termination

Proposed Answer:	B.

Explanation:

B. Correct. For the conditions given the leak is not isolated. (1) The crew will transition to ECA-1.1 (2) where the first step is to check for cold leg recirculation conditions. The student must recall what actions are required based on an interpretation of the given trend in RCS pressure.

A. Incorrect but plausible. (1) If the leak were isolated a transition would be made to E-1. However, with RCS pressure decreasing the leak is not isolated and the crew will transition to ECA-1.1. (2) is correct as explained above. Additionally, the combination of parts (1) and (2) is consistent and thus plausible.

C. Incorrect but plausible. (1) If the leak were isolated a transition would be made to E-1. However, with RCS pressure decreasing the leak is not isolated and the crew will transition to ECA-1.1. (2) ECCS termination criteria will be checked in both E-1 and ECA-1.1 however, in ECA-1.1 ECCS termination criteria will be checked well after other actions are taken. ECCS termination criteria is checked at step 25 in ECA-1.1 while cold leg recirculation criteria are checked at step 1.

D. Incorrect but plausible. (1) is correct as explained above. (2) is incorrect as explained above.

Additionally, the combination of parts (1) and (2) is consistent and thus plausible.				
Technical Reference(s):	ECA-1.2, “LOCA Outside Containment”, rev 26.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1209I 05			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.10		
Justification for RO question: The question is an RO question as it relies on the overall sequence of events and mitigative strategy contained in ECA-1.1 and 1.2.				

Examination Outline Cross-reference:	Level	RO	
Q2	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(035) (SF4P SG) STEAM GENERATOR SYSTEM

(035A3.01) Ability to monitor automatic features of the (SF4P SG) STEAM GENERATOR SYSTEM, including: S/G water level control

	Importance Rating	3.9	

Proposed Question:

Plant conditions:

- 100% power.
- The controlling Steam Flow channel for the ‘A’ Steam Generator fails HIGH.

In response the ‘A’ SG Feed Reg Valve will **_____(1)_____** and **_____(2)_____**.

	(1)	**(2)**
A.	close	level error will overcome the steam flow/feed flow mismatch signal and restore SG level
B.	open	level error will overcome the steam flow/feed flow mismatch signal and restore SG level.
C.	open	level error will be unable to overcome the steam flow/feed flow mismatch signal. The plant will trip on HI-HI SG level
D.	close	level error will be unable to overcome the steam flow/feed flow mismatch signal. The plant will trip on LOW SG level

Proposed Answer:	B.	

Explanation:

B. Correct. The Steam Generator Water Level Control system (SGWLC) is level dominant. (1) With the steam flow channel initially failing high, the Feed Reg Valve (FRV) will initially open until the level error begins to close the FRV. (2) Level will eventually return to initial value. At 100% power the Main Feedwater Speed control d/p circuit is clipped at the max value of 135 psid and there is no effect on the MFPs.

A. Incorrect but plausible. (1) As explained above the FRV will fail open in response to this failure although it failing closed is plausible. (2) is correct as explained above.

C. Incorrect but plausible. (1) is correct as explained above. (2) If the student were not aware of the level dominant nature of the SGWLC system this could be a possible answer making it plausible. Additionally, the combined parts (1) and (2) are consistent and thus the combined answer is plausible.

D. Incorrect but plausible. (1) is incorrect as explained above. (2) would be a correct answer for low failure of a controlling Steam Flow channel. Additionally, the combined parts (1) and (2) are consistent and thus the combined answer is plausible.

Technical Reference(s):	Lesson plan L8046I, “Steam generator Water Level Control”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1193I 06			
Question Source:	Bank #	X	TEB 22060	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q3	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM

(022K3.04) Knowledge of the effect that a loss or malfunction of the (SF5 CCS) CONTAINMENT COOLING SYSTEM will have on the following systems or system parameters: CNT

Importance Rating	3.9	

Proposed Question:

Following a DBA LOCA, a(n) _____**(1)**______ signal will prevent CAH-F-8, "Containment Recirc and Cleanup Filter" from being swapped from RECIRC to FILTER.

The inability to place the unit in FILTER mode will impact its ability to _____**(2)**_____.

	(1)	**(2)**
A.	P	lower radiation levels
B.	P	prevent H_2 stratification
C.	SI	lower radiation levels
D.	SI	prevent H_2 stratification

Proposed Answer:	A.	

Explanation:

A. Correct. (1) The 'P' signal (18 psig containment pressure) will prevent CAH-F-8 from being swapped to the FILTER mode. (2) This will impact the filter's ability to lower radiation levels as this is the primary function of the filter mode of operation.

B. Incorrect but plausible. (1) is correct as explained above. (2) The recirc mode of operation is used to prevent H_2 stratification not the filter mode.

C. Incorrect but plausible. The 'P' signal will prevent the filter from being swapped from recirc to filter not an SI. (2) is correct as explained above.

D. Incorrect but plausible. (1) is incorrect as explained above. (2) is incorrect as explained above.

Technical Reference(s):	Lesson plan, L8038I, "Containment Building HVAC". FR-Z.3, " Response to High Containment Radiation Level", rev 19.

Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8038 04			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge		X	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q4	Tier #	4	
	Group #		

Knowledge and Ability (K/A) Statement:

Thermodynamics 193003

(193003K1.25) STEAM Explain and use saturated and superheated steam tables

Importance Rating	3.4	

Proposed Question:

Plant conditions:

- A LOCA has occurred.
- Pressurizer pressure is 1500 psig.
- RCS wide range pressure is 1550 psig.
- Highest wide range Thot indicates 580°F.
- Highest core exit thermocouple temperature is 622°F.
- Highest average quadrant core exit thermocouple temperature is 612°F.

What is the Subcooling Monitor reading?

A. (-) 20°F.

B. (-) 15°F.

C. (-) 10°F.

D. (+) 22°F.

Proposed Answer:	C.	

Explanation:

C. Correct. For the given plant conditions, Tsat for 1550 psig (1565 psia) is approximately 601.6°F and subcooling is 601.6°F - 612°F = (-) 10.4°F superheated.

A. Incorrect but plausible. This incorrect answer would be obtained if the student used the given highest CETC vs the highest average quadrant CETC.

B. Incorrect but plausible. Tsat for 1500 psig (1515 psia) is approximately 597°F. If the student incorrectly used pressurizer pressure vs wide range pressure this answer would be obtained.

D. Incorrect but plausible. This incorrect answer would be obtained if the student used the given highest wide range Thot vs the highest average quadrant CETC.				
Technical Reference(s):	Steam Tables.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8058I 11			
Question Source:	Bank #	X	TEB 34479	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.14		

Examination Outline Cross-reference:	Level	RO	
Q5	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM

(064A2.27) 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: Loss of DC power

	Importance Rating	3.8	

Proposed Question:

Plant conditions:

- 100% power.
- The following sequence of events occurs:
  - t = 0, loss of DC Bus 11A.
  - t = 30 seconds, reactor trip and SI.
  - t = 45 seconds, loss of offsite power.

At t = 120 seconds, Bus 5 is **_____(1)_____.**

When offsite power is restored, the operator will **_____(2)_____** to close the UAT breaker.

	(1)	(2)
A.	deenergized	reset RMO
B.	deenergized	bypass RMO
C.	energized	reset RMO and parallel the ‘A’ EDG
D.	energized	bypass RMO and parallel the ‘A’ EDG

Proposed Answer:	B.	

Explanation:

B. Correct. (1) With DC bus 11A deenergized the ‘A’ EDG will not receive a start signal and following the loss of offsite power, Bus 5 will be deenergized. (2) When offsite power is restored, operators will use SUP-005 to restore power to Bus 5 from offsite by first bypassing RMO (Remote Manual Override) and then closing the UAT breaker. Bypassing RMO is required because the

EPS (Emergency Power Sequencer) did not sequence due to the loss of DC Bus 11A and thus RMO cannot be reset which is required to close the UAT breaker. This can be seen in SUP-005, “Offsite Power Restoration to Bus 5 and 6” section 1.

A. Incorrect but plausible. (1) is correct as explained above. (2) It is plausible that RMO need to be reset as this is normally required to close the UAT breaker when the system functions correctly. However, with DC Bus 11A lost the EPS will not progress to step 9 and thus RMO cannot be reset. The RMO bypass feature must be used to close the UAT breaker in this case.

C. Incorrect but plausible. (1) It is plausible that for the given plant conditions the ‘A’ EDG has started and repowered Bus 5 as the EDGs use compressed air to start. However, the start signal and control power for the ‘A’ EDG comes from vital DC Bus 11A. For the given conditions then the ‘A’ EDG will not receive a start signal and its output breaker cannot close. (2) If the EDG had started it is necessary to reset RMO and parallel the EDG with offsite power to close the UAT breaker and transfer to offsite power. Thus, the combined parts (1) and (2) in ‘C’ are consistent and thus plausible.

D. Incorrect but plausible. (1) Incorrect as explained above. (2) Incorrect as explained above. It is plausible that RMO need to be bypassed here as this is a common misconception. RMO must be reset to parallel the EDG with offsite power, not bypassed. Additionally, the combined parts (1) and (2) in ‘D’ are consistent and thus plausible.

Technical Reference(s):	E-1, “Loss of Reactor or Secondary Coolant”, rev 45. SUP-005, “Offsite Power Restoration to Bus 5 and 6”, rev 01. Lesson plan L8020, “Emergency Diesel Electrical”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8020I 21			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q6	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer

(WE12EA2.06) Ability to determine and/or interpret the following as they apply to (W E12) UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS: Core exit temperatures and/or subcooling

	Importance Rating	3.9	

Proposed Question:

Step 9 of ECA-2.1, "Uncontrolled Depressurization of all SGs" to isolate SI accumulators, checks if at least two RCS hot leg temperatures are less than 450°F.

What is the basis for this temperature check?

A. It determines if RCS temperature is above the FR-P.1 "Limit A" value.

B. It corresponds to RCS saturation pressure to prevent nitrogen injection after the accumulators have injected.

C. It ensures that accumulator nitrogen will remain in solution in the RCS and not interfere with natural circulation.

D. It corresponds to RCS saturation pressure for P-11 and ensures that the accumulator discharge valves can be closed.

Proposed Answer:	B.	

Explanation:

B. Correct. Per the background document for ECA-2.1, "RCS hot leg temperature to prevent accumulator nitrogen injection, plus a margin for nitrogen heat up, controllability, and typical RCS hot leg temperature instrument uncertainty."

A. Incorrect but plausible. A central concern for an uncontrolled depressurization of all steam generators is the associated cooldown. The Integrity CSFST (P) places a lower limit on RCS temperature for given pressures to prevent a challenge to the integrity of the RCS. It is plausible here that this temperature corresponds to that temperature but is incorrect.

C. Incorrect but plausible. The main concern for nitrogen injection to the RCS is that it interrupts natural circulation. Nitrogen is somewhat soluble in water, although its solubility decreases with increasing temperature.

D. Incorrect but plausible. In step 9 of ECA-2.1 immediately after checking RCS hot leg temperatures less than 450°F, RCS pressure will be verified less than 1950 psig (P-11). If pressure exceeds this value, the accumulator outlet isolation valves cannot be closed. It is plausible that hot leg temperatures are temperature are being checked to confirm this but is incorrect.				
Technical Reference(s):	ECA-2.1, “Uncontrolled Depressurization of all SGs”, rev 41. Background document for ECA-2.1, rev 3.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1207I 12			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.10		
K/A Justification: The question meets the intent of the K/A as hot leg temperatures are functionally equivalent to CETC temperatures in procedure ECA-2.1.				

Examination Outline Cross-reference:	Level	RO	
Q7	Tier #	4	
	Group #		

Knowledge and Ability (K/A) Statement:

Thermodynamics 193004

(193004K1.15) THERMODYNAMIC PROCESS: (THROTTLING AND THE THROTTLING PROCESS) Determine the exit conditions for a throttling process based on the use of steam and/or water

Importance Rating	2.8	

Proposed Question:

Plant conditions:

- The plant is at 100% power.
- An NSO reports a steam leak at the inlet to the 'C' low pressure turbine.
- The steam at the outlet of the MSR is at 175 psig and 500°F

What is the temperature of the steam when it exits to the Turbine Building?

A. 212°F

B. 378°F

C. 470°F

D. 500°F

Proposed Answer:	C.	

Explanation:

C. Correct. The steam at 175 psig (190 psia) and 500°F has an enthalpy of about 1270 Btu/lb. Steam with this enthalpy at standard atmospheric pressure has a temperature of approximately 470°F.

A. Incorrect but plausible. If the student incorrectly thought that steam at standard atmospheric pressure always has a temperature of 212°F this could be a plausible answer.

B. Incorrect but plausible. The saturation temperature of steam at 175 psig is approximately 378°F making this a plausible answer.

D. Incorrect but plausible. If the student incorrectly thought that the steam would maintain its temperature this could be a plausible answer.

Technical Reference(s):	Mollier Diagram.			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	193003 Obj 2			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.14		

Examination Outline Cross-reference:	Level	RO	
Q8	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000007) (EPE 7; BW E02 & E10; CE E02) Reactor Trip, Stabilization, Recovery

(000007EK3.06) Knowledge of the reasons for the following responses and/or actions as they apply to (EPE 7) REACTOR TRIP, STABILIZATION, RECOVERY: Stopping an RCP

	Importance Rating	3.5	

Proposed Question:

E-0, "Reactor Trip or Safety Injection", requires Reactor Coolant Pumps to be tripped if RCS subcooling has decreased to <40°F with either a CCP or SI pump running.

What does this action prevent?

A. Damage to the RCP seal package due to the potential for a two-phase mixture existing in the pump casing.

B. Damage to the RCPs and RCS because of dynamic stresses associated with pumping a two-phase mixture.

C. Loss of the Unit Auxiliary Transformers because of potential RCP motor overloads or faults propagating back through the 13.8 kV buses.

D. Excessive loss of RCS water inventory through an RCS rupture which could lead to severe core uncovery if the RCPs were tripped later in the accident.

Proposed Answer:	D.	

Explanation:

D. Correct. Per Westinghouse Owners Group ERG's, Executive volume, RCP Trip/Restart the reason for purposely tripping the RCPs during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some other reason later in the accident.

A. Incorrect but plausible. The Westinghouse background document discusses various situations where tripping the RCPs is prudent. The document discusses tripping one RCP in procedure FR-C.2 to prevent pump damage due to running under the pumps under two-phase/voided conditions. This pump trip is to save that pump for potential future use. This situation is not applicable to the guidance specific to the 40°F subcooling criteria in E-0. It is plausible though that tripping the RCP due to loss of subcooling is related to protection of the seal package.

B. Incorrect but plausible. The Westinghouse background document discusses various situations where tripping the RCPs is prudent. The document discusses tripping one RCP in procedure FR-C.2 to prevent pump damage due to running under the pumps under two-phase/voided conditions. This pump trip is to save that pump for potential future use. This situation is not applicable to the guidance specific to the 40°F subcooling criteria in E-0.

C. Incorrect but plausible. Per Westinghouse Owners Group ERG's, Executive volume, RCP Trip/Restart RCP operation with 2 phase mixture will affect pump performance, predominately pump current changes. It is plausible this could propagate back to the UATs through the 13.8 kv busses.

Technical Reference(s):	Executive Volume for RCP Trip and Restart, page 8, rev 3.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1202I 03			
Question Source:	Bank #	X	TEB 16539	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2013 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.5 / 41.10		

Examination Outline Cross-reference:	Level	RO	
Q9	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(BW E04; W E05) Inadequate Heat Transfer – Loss of Secondary Heat Sink

(WE05EK2.16) Knowledge of the relationship between (W E05) Loss of Secondary Heat Sink and the following systems or components: CNT

Importance Rating	3.1	

Proposed Question:

Plant conditions:

- A Total Loss of Feedwater has occurred.
- The crew is performing FR-H.1, “Response to Loss of Secondary Heat Sink” and are currently at step 4, “Try to Establish SUFP flow to SG(s)”.
- Containment pressure is 5 psig.
- Reactor Coolant Pumps are tripped.
- Pressurizer Pressure is 2350 psig.

The applicable criteria for immediately establishing bleed and feed based on these conditions is if wide range in any __________.

A. three SG is less than 44%

B. three SG is less than 20%

C. one SG is less than 30%

D. one SG is less than 14%

Proposed Answer:	A.	

Explanation:

A. Correct. With the given plant conditions, containment is adverse (>4 psig). From FR-H.1, “If wide range level in any 3 SGs is less than 20% [44% ADVERSE CONTAINMENT], Steps 11 through 15 should be immediately initiated for bleed and feed (Applicable after procedure Step 1)”.

B. Incorrect but plausible. This would be the correct answer if containment were not adverse and is thus plausible.

C. Incorrect but plausible. 30% wide range in any one SG is the criteria for a “dry SG” with

containment adverse and is thus plausible. D. Incorrect but plausible. 14% wide range in any one SG is the criteria for a “dry SG” with containment not adverse and is thus plausible.					
Technical Reference(s):	FR-H.1, “Response to Loss of Secondary Heat Sink”, rev 39.				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L1211I 03				
Question Source:	Bank #				
	Modified Bank#				
	New	X			
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis			X	
10 CFR Part 55 Content:	55.41	41.8 / 41.10			

Examination Outline Cross-reference:	Level	RO	
Q10	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(012) (SF7 RPS) REACTOR PROTECTION SYSTEM

(012K3.05) Knowledge of the effect that a loss or malfunction of the (SF7 RPS) REACTOR PROTECTION SYSTEM will have on the following systems or system parameters: RCPS

Importance Rating	3.5	

Proposed Question:

The crew is performing a power increase per OS1000.02, “Plant Startup from Hot Standby to Minimum Load”.

- The excore NIs are reading:
  - NI-41 12%
  - NI-42 11%
  - NI-43 9%
  - NI-44 9%

When performing step 4.34.1 to verify the status light on UL-6 at H-4 for P-10 (see panel below) it is found that the light is not lit.

	1	2	3	4	5	6	7	8
A	RCP A UV	RCP A UF	RC LOOP 1 TB411C OTDT	RC LOOP 1 TB411G OPDT	PRESSURIZER PB455C PRESS LO	PRESSURIZER PB455A PRESS HI	PRESSURIZER LB459A LEVEL HI	
B	RCP B UV	RCP B UF	RC LOOP 2 TB421C OTDT	RC LOOP 2 TB421G OPDT	PRESSURIZER PB456C PRESS LO	PRESSURIZER PB456A PRESS HI	PRESSURIZER LB460A LEVEL HI	CONTAINMENT PB936B PRESS HI-1
C	RCP C UV	RCP C UF	RC LOOP 3 TB431C OTDT	RC LOOP 3 TB431G OPDT	PRESSURIZER PB457C PRESS LO	PRESSURIZER PB457A PRESS HI	PRESSURIZER LB461A LEVEL HI	CONTAINMENT PB935B PRESS HI-1
D	RCP D UV	RCP D UF	RC LOOP 4 TB441C OTDT	RC LOOP 4 TB441G OPDT	PRESSURIZER PB458C PRESS LO	PRESSURIZER PB458A PRESS HI		CONTAINMENT PB934B PRESS HI-1
E	RC LOOP 1 FB414 FLOW LO	RC LOOP 2 FB424 FLOW LO	RC LOOP 3 FB434 FLOW LO	RC LOOP 4 FB444 FLOW LO				
F	RC LOOP 1 FB415 FLOW LO	RC LOOP 2 FB425 FLOW LO	RC LOOP 3 FB435 FLOW LO	RC LOOP 4 FB445 FLOW LO				
G	RC LOOP 1 FB416 FLOW LO	RC LOOP 2 FB426 FLOW LO	RC LOOP 3 FB436 FLOW LO	RC LOOP 4 FB446 FLOW LO				
H		IR ABOVE P6 10^{-10} AMP BLOCK SR	10^{5} CPS ↓	POWER ABOVE P10 10% BLOCK TRIPS	10% ↓			20% ↓
J	SR-31D TRIP BYPASSED	SOURCE RANGE TRAIN A BLOCKED	SOURCE RANGE 31D TRIPPED	POWER RANGE 41M ABOVE P10	TURBINE PWR PB505A ABOVE P13	INTERM RANGE TRAIN A BLOCKED	LO PWR RANGE TRAIN A BLOCKED	POWER RANGE 41S ABOVE P9
K	SR-32D TRIP BYPASSED	SOURCE RANGE TRAIN B BLOCKED	SOURCE RANGE 32D TRIPPED	POWER RANGE 42M ABOVE P10	TURBINE PWR PB506A ABOVE P13	INTERM RANGE TRAIN B BLOCKED	LO PWR RANGE TRAIN B BLOCKED	POWER RANGE 42S ABOVE P9
L				POWER RANGE 43M ABOVE P10				POWER RANGE 43S ABOVE P9
M				POWER RANGE 44M ABOVE P10				POWER RANGE 44S ABOVE P9
N				POWER BELOW P7 INTLK TRIPS BLOCKED	TURBINE PWR BELOW P13			RX PWR BELOW P9 TURBINE TRIP BLOCKED

If two RCPs trip, the reactor _____**(1)**_____ automatically trip and _____**(2)**_____.

	(1)	(2)
A.	will not	the DNBR limit could be reached
B.	will not	ΔT limits could be exceeded
C.	will	the core is protected from excessive ΔT
D.	will	the core is protected from DNB

Proposed Answer:	A.	

Explanation:

A. Correct. (1) The NIs indicate that power is above P-10 (2/4 >10%) however, due to a malfunction of the RPS, P-10 is not actuated as indicated by the status light on UL-6 “POWER ABOVE P-10 10% BLOCK TRIPS” not being lit. With RPS sensing that power is below P-10 when it is not, a trip of two RCPs will not result in an automatic reactor trip. (2) The basis for the loss of flow reactor trip is to protect the core from DNB following a loss-of-primary-flow accident, where there is not enough coolant flow to remove the heat generated in the fuel, resulting in DNB.

B. Incorrect but plausible. (1) is correct as explained above. (2) is incorrect but plausible as a reduction in loop flow will cause ΔT to increase however, this is not the basis of the loss of flow trip. The basis of the trip is protection against DNBR.

C. Incorrect but plausible. (1) It is plausible that the loss of flow trip is still enabled based on the observed indications of excore NIs. The NIs however, are the inputs to the RPS. The actual status of P-10 as sensed by RPS is indicated by the status light on UL-6. Thus, if the RCPs trip, the reactor will not automatically trip. (2) is incorrect but plausible as a reduction in loop flow will cause ΔT to increase however, this is not the basis of the loss of flow trip. The basis of the trip is protection against DNBR. Additionally, the combined statement of parts (1) and (2) is consistent.

D. Incorrect but plausible. (1) It is plausible that the loss of flow trip is still enabled based on the observed indications of excore NIs. The NIs however, are the inputs to the RPS. The actual status of P-10 as sensed by RPS is indicated by the status light on UL-6. Thus, if the RCPs trip, the reactor will not automatically trip. (2) This is the correct basis for the loss of flow trip as explained above. Additionally, the combined statement of parts (1) and (2) is consistent.

Technical Reference(s):	Lesson plan L8056I, RPS. OS1000.02, “Plant Startup from Hot Standby to Minimum Load”, rev 41.
Proposed references to be provided to applicants during examination:	Embedded reference: Picture of MCB UL-6 showing RPS bistable status lights.

Learning Objective:	SBK LOP L8056I 17			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q11	Tier #	4	
	Group #		

Knowledge and Ability (K/A) Statement:

Reactor Theory 192008

(192008K1.10) REACTOR OPERATIONAL PHYSICS: (CRITICALITY) Describe reactor power and startup rate response once criticality is reached

Importance Rating	3.4	

Proposed Question:

During a reactor startup, criticality is declared when a **_____(1)_____** startup rate is observed, and reactor power is increasing at a(n) **_____(2)_____** rate.

	(1)	**(2)**
A.	constant positive	linear
B.	constant positive	exponential
C.	continuously increasing	linear
D.	continuously increasing	exponential

Proposed Answer:	B.	

Explanation:

B. Correct. To definitively identify critical conditions, the reactor is made slightly supercritical. In this state, the SUR rate is constant positive (not decaying) and reactor power increases exponentially.

A. Incorrect but plausible. (1) is correct. (2) It is a common misconception that power increases linearly as observed trends are typically viewed on a log scale. The linear increase in power as a function of time on the log scale is an exponential increase.

C. Incorrect but plausible. (1) When criticality is declared, SUR is positive however it is not increasing as the constant positive SUR that is indicative of critical conditions must be observed with no rod motion. (2) is incorrect as explained above.

D. Incorrect but plausible. (1) is incorrect as explained above. (2) is correct as explained above.

Technical Reference(s):	INPO Reactor Theory, 192008, “Reactor Operational Physics”.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	192008 Obj. 2			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.1		

Examination Outline Cross-reference:	Level	RO	
Q12	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(055) (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM

(055A2.04) Ability to (a) predict the impacts of the following on the (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Air in-leakage

Importance Rating	3.6	

Proposed Question:

Plant conditions:

- Initially at 100% power.
- Alarm B8198, “CONDSR A VACUUM LOW” actuates.
- Vacuum is 27” Hg and degrading.

The next automatic action that will occur is the **_____(1)_____**.

Per ON1233.01, “Loss of Condenser Vacuum”, if vacuum continues to degrade to less than 25 “Hg following a load reduction to 360 MWe, operators will trip the **_____(2)_____**.

	(1)	**(2)**
A.	steam dumps are blocked	reactor
B.	steam dumps are blocked	main turbine
C.	standby vacuum pump starts	reactor
D.	standby vacuum pump starts	main turbine

Proposed Answer:	C.

Explanation:

C. Correct. (1) Alarm B8198, “CONDSR A VACUUM LOW” actuates at 27 “Hg. The next automatic action to occur if vacuum continues to degrade is the standby mechanical vacuum pump will auto start at 26” Hg. (2) Per AOP ON1233.01, “Loss of Condenser Vacuum” if vacuum continues to degrade to less than 25 “Hg following a load reduction to 360 MWe, operators will be directed to trip the reactor.

A. Incorrect but plausible. (1) The steam dumps are automatically blocked at 25” Hg. It is plausible but incorrect that blocking of the steam dumps is the next automatic action to occur. The standby vacuum pump will start before this at 26” Hg. (2) is correct as explained above.

B. Incorrect but plausible. (1) is incorrect as explained above. (2) It is plausible that AOP ON1233.01, “Loss of Condenser Vacuum” would direct that the turbine be tripped as 360 MWe corresponds to about 28% power, which is below P-9. Various AOPs direct a turbine trip for serious degraded conditions if the plant is below P-9 and a reactor trip if the plant is above P-9.

D. Incorrect but plausible. (1) is correct as explained above. (2) is incorrect as explained above.

Technical Reference(s):	VPRO for B8198, “CONDSR A VACUUM LOW”. ON1233.01, “Loss of Condenser Vacuum”, rev 13. Lesson plan L1188I, “Loss of Vacuum”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1188I 08			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q13	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(000061) (APE 61) Area Radiation Monitoring System Alarms

(000061AK2.06) Knowledge of the relationship between (APE 61) AREA RADIATION MONITORING (ARM) System Alarms and the following systems or components: Refueling pool

	Importance Rating	3.4	

Proposed Question:

Plant conditions:

- Core off-load is in progress.
- The fuel handlers were moving irradiated fuel to a location in the spent fuel pool.
- ‘A’ Train FAH is in the Fuel Handling mode of operation.
- A spent fuel assembly was dropped in the spent fuel pool.
- RM-6518, “Spent Fuel Pool High Range” radiation monitor is in alarm.

What actions are required per procedure?

A. Enter OS1215.06, "Fuel Handling Accident" and place both trains of FAH in the fuel handling mode.

B. Enter OS1215.06, "Fuel Handling Accident" and evacuate non-essential personnel from the Fuel Storage Building and place CBA in the filter recirc mode.

C. Enter OS1252.02, "Airborne High Radiation" and verify containment ventilation isolation.

D. Enter OS1252.02, "Airborne High Radiation" and immediately evacuate non-essential personnel from the Fuel Storage Building.

Proposed Answer:	B.	

Explanation:

B. Correct. The conditions given meet the entry criteria for OS1215.06, “ Fuel Handling Accident”. The AOP will direct the crew to evacuate non-essential personnel from the building and align Control Building Air (CBA) to the filter recirc mode.

A. Incorrect but plausible. The conditions given meet the entry criteria for OS1215.06, “ Fuel Handling Accident”. The AOP will direct the crew to evacuate non-essential personnel from the building and align CBA to the filter recirc mode. It will also verify that one train of the Fuel Storage Building Ventilation system is in the “Fuel Handling” mode of operation. Only one train need be in operation, though it is plausible that the AOP would direct that both trains be placed into service.

C. Incorrect but plausible. It is plausible that for the given plant conditions OS1252.02 "Airborne High Radiation" would be required to be entered. However, RM-6518 is an Area Radiation Monitor not an Airborne Radiation Monitor therefore OS1252.03 “Area High Radiation” may apply not OS1252.02. Additionally, it is plausible that a containment ventilation isolation (CVI) would need to be verified here as radiation monitors associated with refueling will cause an automatic CVI. These rad monitors however are on the refueling machine and not in the fuel storage building.

D. Incorrect but plausible It is plausible that for the given plant conditions OS1252.02 "Airborne High Radiation" would be required to be entered. However, RM-6518 is an Area Radiation Monitor not an Airborne Radiation Monitor therefore OS1252.03 “Area High Radiation” may apply not OS1252.02. Additionally, OS1252.02 will evacuate an area that is experiencing high radiation. The evacuation will not be immediate though, it will be made after notifications are made and consultation with RP is performed.

Technical Reference(s):	OS1215.06, "Fuel Handling Accident", rev 16. OS1252.03, "Area High Radiation", rev 15.
Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L1192I 04			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.8 / 41.10		

Examination Outline Cross-reference:	Level	RO	
Q14	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(000051) (APE 51) Loss of Condenser Vacuum

(000051AK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 51) LOSS OF Condenser Vacuum: Relationship of condenser vacuum to circulating water, flow rate, and temperature

Importance Rating	3.3	

Proposed Question:

Plant conditions:

- 100% power.
- Ocean temperature is 60°F.

Due to ocean debris, one CW pump trips on high screen d/p.

In response to this malfunction, Main Generator output will be __________

A. zero, because a manual reactor trip is required.

B. stable, but below the initial value because of the degraded vacuum.

C. initially lower but will return to the initial value when CW-V-40 is opened.

D. unchanged, because the two remaining pumps are sufficient to maintain vacuum and load.

Proposed Answer:	B.	

Explanation:

B. Correct. With the loss of one running CW pump, CW flow through the condenser will be reduced lowering condenser vacuum. With the degraded vacuum, main generator output will be reduced but stable.

A. Incorrect but plausible. ON1233.01, “Loss of Condenser Vacuum” will direct a reactor trip if at least two CW pumps are not running, thus making this plausible.

C. Incorrect but plausible. CW-V-40 is opened to maintain condenser vacuum with low ocean water temperatures, not for reduced CW system flow.

D. Incorrect but plausible. Two CW pumps can maintain plant power provided CW ΔT can be maintained. Typically with two CW pumps running, load will need to be reduced to maintain CW ΔT within limits. For the conditions given with ocean temperature of 60°F, load would need to be reduced to maintain ΔT within limits.

Technical Reference(s):	ON1233.01, “Loss of Condenser Vacuum”, rev 13. ON1038.01, “CW System Operation”, rev 50.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1188I 03, 06			
Question Source:	Bank #	X	NUSEC 53577	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Millstone 2 2014 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5 / 41.7		

Examination Outline Cross-reference:	Level	RO	
Q15	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM

(006K2.01) Knowledge of electrical power supplies to the following (CFR:): (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM ECCS pumps

Importance Rating	4.1	

Proposed Question:

Plant conditions:

- 100% power.
- DBA LOCA coincident with a loss of off-site power (SI/LOP).
- The following alarm occurs:
    - D6329, “Bus E5 UAT INC LN BKR TRIP & L/O”

What, if anything will be powering the SI pumps?

	‘A’ SI pump	**‘B’ SI pump**
A.	‘A’ EDG	SEPS
B.	‘A’ EDG	‘B’ EDG
C.	No power	SEPS
D.	No power	‘B’ EDG

Proposed Answer:	D.

Explanation:

D. Correct. With the alarms as given, Bus 5 is unavailable The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the ‘A’ EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the ‘A’ SI pump has no available power source. The DG ‘B’ will power Bus 6 and the ‘B’ SI pump. SEPS would be used to power Bus 6 if the ‘B’ EDG were unavailable.

A. Incorrect but plausible. The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the ‘A’ EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the ‘A’ SI

pump has no available power source. The DG ‘B’ will power Bus 6 and the ‘B’ SI pump. B. Incorrect but plausible. The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the ‘A’ EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the ‘A’ SI pump has no available power source. The DG ‘B’ will power Bus 6 and the ‘B’ SI pump. C. Incorrect but plausible. The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the ‘A’ EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the ‘A’ SI pump has no available power source. The DG ‘B’ will power Bus 6 and the ‘B’ SI pump.				
Technical Reference(s):	VPRO D6329, “Bus E5 UAT INC LN BKR Trip & LO”, rev 05.			
Proposed references to be provided to applicants during examination:			None	
Learning Objective:	SBK LOP L8034I 14			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2020 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q16	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000056) (APE 56) Loss of Offsite Power

(000056AK3.02) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 56) Loss of Offsite Power: Actions contained in AOPs

Importance Rating	4.1	

Proposed Question:

Plant conditions:

- Loss of off-site power.
- Buses E5 and E6 are de-energized and cannot be re-powered.
- The crew is performing ECA-0.0, “Loss of All AC Power”.
- Step 18 "Depressurize Intact SGs to Reduce RCS Leakage" is in progress.
- ECA-0.0 cautions the operators <u>not</u> to depressurize SGs below 250 psig.

The cooldown rate specified for dumping steam in ECA-0.0 is _____**(1)**_____ and 250 psig is specified as the minimum pressure to _____**(2)**_____?

	(1)	**(2)**
A.	<100°F/hr	prevent accumulator nitrogen injection
B.	<100°F/hr	prevent challenging the RCS Integrity CSF
C.	max rate	prevent accumulator nitrogen injection
D.	max rate	ensure that the reactor does not return to a critical condition

Proposed Answer:	A.	

Explanation:

A. Correct. (1) Step 18 of ECA-0.0 directs a cooldown rate of <100°F/hr. (2) Per the caution before step 18, “SG pressures should not be decreased to less than 250 psig to prevent injection of accumulator nitrogen into the RCS.” With regard to the K/A/ match, ECA-0.0 is being used as equivalent to a Loss of Offsite Power AOP which Seabrook does not have.

B. Incorrect but plausible. (1) Correct as explained above. (2) It is plausible that the limitation on SG pressure is to prevent challenging the RCS Integrity CSF as lower SG pressures corelate to lower RCS temperatures and the RCS Integrity CFS places limitations on how low RCS temperature may be for a given pressure.

C. Incorrect but plausible. (1) Incorrect as explained above. The max cooldown rate is plausible as previous revisions of ECA-0.0 used this cooldown rate. Also, procedures such as E-3 utilize a maximum cooldown rate. (2) Correct as explained above.

D. Incorrect but plausible. (1) is incorrect as explained above. The max cooldown rate is plausible as previous revisions of ECA-0.0 used this cooldown rate. Also, procedures such as E-3 utilize a maximum cooldown rate. (2) Step 19 of ECA-0.0 will check that the reactor remains subcritical and if not, to allow for RCS temperature to increase until the reactor returns to a subcritical state. Thus, this is plausible and consistent with SG pressures below 250 psig as this would correlate to low RCS temperatures.

Technical Reference(s):	ECA-0.0, “Loss of All AC Power”, rev 57 (current revision). ECA-0.0, “Loss of All AC Power”, rev 31 (older revision with max cooldown rate).

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8067I 04, 10			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		

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

10 CFR Part 55 Content:	55.41	41.5 / 41.10

Examination Outline Cross-reference:	Level	RO	
Q17	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000057) (APE 57) Loss of Vital AC Instrument Bus

(000057AA1.02) Ability to operate and/or monitor the following as they apply to (APE 57) LOSS OF VITAL AC ELECTRICAL INSTRUMENT BUS: Manual control of PZR level

Importance Rating	3.8	

Proposed Question:

Plant conditions:

- 100% power.
- RC-LT-459/460 are selected for control/backup.
- Charging flow is 110 gpm
- Letdown flow is 103 gpm.
- Subsequently, alarm D5800, “VITAL INST PANEL 1A POWER LOST” actuates.

In response, CS-FCV-121, “Charging Flow Control Valve” will **_____(1)_____**.

The PSO will **_____(2)_____** to control pressurizer level.

	(1)	(2)
A.	open	reduce charging flow to the initial value
B.	close	raise charging flow to the initial value
C.	open	reduce charging flow to the RCP seals only
D.	close	raise charging flow to the RCP seals only

Proposed Answer:	C.

Explanation:

C. Correct. (1) With the loss of PP-1A, RC-LT-459 will fail low. The pressurizer level control system will see pressurizer level as off scale low and in response will open LCV-121 to attempt to restore pressurizer level. (2) The loss of PP-1A will also result in a letdown isolation. In response, the PSO will reduce charging to seals only to mitigate the rate of pressurizer level increase.

A. Incorrect but plausible. (1) is correct as explained above. (2) The PSO will reduce charging flow in response to the PP-1A failure however, flow will be reduced to the RCPs only (approximately 32 - 40 gpm) not to the initial value of 110 gpm.

B. Incorrect but plausible. (1) is incorrect as explained above. (2) is incorrect as explained above. Additionally, the combined parts (1) and (2) in ‘B’ are consistent and therefore plausible.

D. Incorrect but plausible. (1) is incorrect as explained above. (2) is incorrect as explained above. Additionally, the combined parts (1) and (2) in ‘D’ are consistent and therefore plausible.

Technical Reference(s):	Lesson plan L1182I, “Primary instrument Failures”.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L1182I 03			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5 / 41.7		

Examination Outline Cross-reference:	Level	RO	
Q18	Tier #	3	
	Group #		

Knowledge and Ability (K/A) Statement:

Radiation Control

(G2.3.11) RADIATION CONTROL: Ability to control radiation releases

Importance Rating	3.8	

Proposed Question:

Which of the following correctly describes immediate equipment response and/or required operator action associated with a high alarm on RM-6505, “Condenser Air Evacuation Radiation Monitor”?

A. All condenser Mechanical Vacuum Pumps will automatically trip.

B. All condenser Mechanical Vacuum Pumps will be shut off per OS1227.02, “Steam Generator Tube Leak”.

C. AR-FV-5004, “Condenser Mechanical Vacuum Pump Discharge to ATM/PAB” automatically realigns to the PAB.

D. AR-FV-5004, “Condenser Mechanical Vacuum Pump Discharge to ATM/PAB” will be verified aligned to the PAB per OS1227.02, “Steam Generator Tube Leak”.

Proposed Answer:	D.	

Explanation:

D. Correct. Per OS1227.02, Steam Generator Tube Leak”, Attachment C, step 2 directs the operator to verify that the mechanical vacuum pump discharge is aligned to the PAB.

A. Incorrect but plausible. In the event of a steam generator tube leak the mechanical vacuum pumps would draw radiation from the condenser, however there is no automatic pump trip signal generated from RM-6505, “Condenser Air Evacuation Radiation Monitor”.

B. Incorrect but plausible. In the event of a steam generator tube leak the mechanical vacuum pumps would draw radiation from the condenser, however the guidance in OS1227.02, Attachment C is to verify that the vacuum pump discharge is aligned to the PAB vice stopping the pumps.

C. Incorrect but plausible. It is true that the desired vacuum pump discharge flowpath is to the PAB, however there is no automatic valve alignment signal generated from RM-6505, “Condenser Air Evacuation Radiation Monitor”.

Technical Reference(s):	OS1227.02, “Steam Generator Tube Leak”, rev 21.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1190I 02			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2015 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.11		

Examination Outline Cross-reference:	Level	RO	
Q19	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM

(073K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM: PRM component failures

	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- 100% power.
- Steam Generator Blowdown (SGBD) is aligned to the ocean.
- RM-6519, "Blowdown Flash Tank Outlet Radiation Monitor", is out of service due to a component failure.
- A tube leak develops in the 'B' Steam Generator.
- RM-6511, "'B' SG Blowdown Radiation Monitor", goes into HIGH ALARM.

How will the SGBD system respond, if at all?

A. No automatic actions will occur.

B. SB-LV-1909, "Flash Tank Level Control Valve", will close on the high radiation signal ultimately resulting in a blowdown isolation.

C. SB-CV-6519, "Flash Tank Discharge Isolation Valve", will close on the high radiation signal ultimately resulting in a blowdown isolation.

D. SB-V-3, "SGBD Loop 'B' IRC Isolation Valve", will close on the high radiation signal. SB will remain in service with the remaining 3 loops.

Proposed Answer:	C.	

Explanation:

C. Correct. High radiation on RM-6511 will cause SB-CV-6519 to auto close. This will occur regardless of the operability of RM-6519. SGBD flash tank level will then increase and the high level in the flash tank will cause SGBD to isolate.

A. Incorrect but plausible. With RM-6519 out of service, it is plausible that no automatic actions were to occur on high radiation if the student were not aware the relationship between the individual SGBD line radiation monitors and SB-CV-6519.

B. SB-LV-1909 is used when SGBD is aligned to the ocean. It functions as the outlet of the SGBD flash tank and is located downstream of SB-CV-6519. It is plausible but incorrect that this valve closes on high radiation.

D. Incorrect but plausible. SB-V-3 is the inside reactor containment isolation valve for 'B' SGBD. It will close on a SGBD isolation after SB-CV-6519 closes on high radiation and flash tank level increases to the point of isolation. However, SB-V-3 will not close on the high radiation signal, it closes on high flash tank level.

Technical Reference(s):	Lesson plan L8063I, "SGBD".			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8063I 11			
Question Source:	Bank #	X	TEB 32179	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q20	Tier #	4	
	Group #		

Knowledge and Ability (K/A) Statement:

Thermodynamics 193009

(193009K1.03) CORE THERMAL LIMITS: Explain local peaking factor

Importance Rating	2.7	

Proposed Question:

Technical Specifications defines AFD (Axial Flux Difference) as _____**(1)**_____.
Maintaining AFD within limits ensures that design limits on _____**(2)**_____ are not exceeded.

	(1)	(2)
A.	top half NI power – bottom half NI power	cladding temperature
B.	top half NI power – bottom half NI power	local power density and min DNBR
C.	bottom half NI power – top half NI power	cladding temperature
D.	bottom half NI power – top half NI power	local power density and min DNBR

Proposed Answer:	B.

Explanation:

B. Correct. (1) AFD is top – bottom NI power. A positive AFD indicates the power in the upper half of the core is greater than that in the bottom while a negative indicates that power in the lower half is greater than in the top. (2) The basis for AFD is that “the limits on AXIAL FLUX DIFFERENCE (AFD) specified in the CORE OPERATING LIMITS REPORT (COLR) assure that the design limits on peak local power density and minimum DNBR are not exceeded during normal operation and the consequences of any Non-LOCA event would be within specified acceptance criteria.”

A. Incorrect but plausible. (1) is correct as explained above. (2) The Technical Specification basis for other peaking factors such as FNΔH and FQ(z) are related to cladding temperatures and the 10CFR50.67 ECCS Acceptance criteria however, the basis for AFD is related to local power density and DNBR.

C. Incorrect but plausible. (1) AFD is top – bottom NI power. It is however plausible that the AFD would be bottom – top power as the convention is arbitrary and is a common misconception. (2) is incorrect as explained above.

D. Incorrect but plausible. (1) is incorrect but plausible as explained above. (2) is correct as explained above.

Technical Reference(s):	Technical Specification 3.2.1 and basis, rev 149.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8030I 12			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.14		

Examination Outline Cross-reference:	Level	RO	
Q21	Tier #	4	
	Group #		

Knowledge and Ability (K/A) Statement:

Reactor Theory 192007

(192007K1.04) FUEL DEPLETION AND BURNABLE POISONS: Describe how and why boron concentration changes over core life

	Importance Rating	3.4	

Proposed Question:

As the reactor core ages over an entire cycle from BOL to EOL, operators will ___________

A. continually decrease boron concentration to offset fuel depletion only.

B. continually decrease boron concentration to offset fuel depletion and buildup of fission product poisons.

C. initially increase boron concentration to offset conversion of U^{238} to Pu^{239}, then decrease boron concentration to offset fuel depletion and buildup of fission product poisons.

D. initially increase boron concentration to offset burnout of burnable poisons, then decrease boron concentration to offset fuel depletion and buildup of fission product poisons.

Proposed Answer:	D.	

Explanation:

D. Correct. Over the life of the reactor core, the critical boron concentration initially increases followed by a decrease toward the end of core life. The initial increase in boron concentration is due to the reduction in negative reactivity from the burnout of the burnable poisons (Integral Fuel Burnable Assemblies at Seabrook). Following this, boron concentration is reduced to offset fuel depletion and the buildup of fission product poisons.

A. Incorrect but plausible. It is plausible that boron concentration continually decreases over core life because the overall concentration decreases from BOL to EOL. However, initially the concentration increases followed by a decrease. It is plausible that if boron concentration were to continually decrease that this is to offset fuel depletion only. However, the overall decrease in boron concentration is due to both fuel depletion and the buildup of fission product poisons.

B. Incorrect but plausible. It is plausible that boron concentration continually decreases over core life because the overall concentration decreases from BOL to EOL. However, initially the concentration increases followed by a decrease. It is plausible that if boron concentration were to continually decrease that this is to offset fuel depletion and buildup of fission product poisons.

C. Incorrect but plausible. Over the life of the reactor core, the critical boron concentration initially increases followed by a decrease toward the end of core life. The initial increase in boron concentration is due to the reduction in negative reactivity from the burnout of the burnable poisons (Integral Fuel Burnable Assemblies at Seabrook). However, it is plausible that this initial increase in boron concentration is to done to offset the positive reactivity added from conversion of U^{238} to Pu^{239}. This conversion does occur as the core ages but it is not responsible for the initial increase in boron concentration early in core life. Following this, boron concentration is reduced to offset fuel depletion and the buildup of fission product poisons.

Technical Reference(s):	Lesson plan for Reactor Theory, 192007, “Fuel Depletion and Burnable Poisons”. RE-1 Critical Boron Concentration, rev 01.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	192007 obj 01			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.1		

Examination Outline Cross-reference:	Level	RO	
Q22	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(000076) (APE 76) High Reactor Coolant Activity

(G2.2.2) EQUIPMENT CONTROL: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels (CFR: / 45.2)

Importance Rating	4.6	

Proposed Question:

Plant conditions:

- The crew has entered OS1202.05, “RCS High Activity”.
- Chemistry has just verified the high activity by re-sampling the RCS.
- Required notifications have been made.
- Letdown filters and mixed bed demineralizers are in their normal alignment.

What action will the crew take next in accordance with OS1202.05, “RCS High Activity”?

A. Maximize letdown flow.

B. Bypass the mixed bed demineralizers.

C. Place the cation demineralizer in service.

D. Isolate letdown and establish minimum RCP seal flow.

Proposed Answer:	A.	

Explanation:

A. Correct. Per OS1202.05, “Reactor Coolant System High Activity”, step 3b. will direct the crew to maximize letdown flow to lower letdown activity.

B. Incorrect but plausible. It is plausible that the next action that the crew would take is to bypass the demineralizers as this would prevent high activity from building up in the filter thus reducing dose to plant personnel. However, OS1202.05, “Reactor Coolant System High Activity”, step 3b. will direct the crew to maximize letdown flow to lower letdown activity.

C. Incorrect but plausible. The cation demineralizer is an additional (normally in standby) demineralizer that is used to remove lithium from the RCS periodically. Because it is a standby demineralizer it is plausible but incorrect that it is placed in service to lower RCS activity.

D. Incorrect but plausible. It is plausible that in response to high RCS activity the AOP would direct the operators to isolate letdown as this will minimize contamination of areas outside of containment. However, OS1202.05, “Reactor Coolant System High Activity”, step 3b. will direct the crew to maximize letdown flow to lower letdown activity.

Technical Reference(s):	OS1202.05, “Reactor Coolant System High Activity”, rev 15.				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L1181I 09				
Question Source:	Bank #	X	TEB 13691		
	Modified Bank#				
	New				
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	41.6 / 41.7			

Examination Outline Cross-reference:	Level	RO	
Q23	Tier #	3	
	Group #		

Knowledge and Ability (K/A) Statement:

Emergency Procedures / Plan

(G2.4.35) EMERGENCY PROCEDURES / PLAN: Knowledge of nonlicensed operator responsibilities during an emergency

	Importance Rating	3.8	

Proposed Question:

In accordance with the Emergency Plan, NSOs will report to the **_____(1)_____** when an **_____(2)_____** is first declared in accordance with ER-1.1.

	(1)	**(2)**
A.	Technical Support Center	Unusual Event
B.	Operational Support Center	Unusual Event
C.	Technical Support Center	Alert
D.	Operational Support Center	Alert

Proposed Answer:	D.	

Explanation:

D. Correct. (1) NSOs will report to the OSC once it is activated. (2) Per ER 3.2, "Operational Support Center Operations" the OSC is activated on an Alert or higher emergency plan classification. The term "first" is used here to eliminate a subset issue between the Alert and Unusual Event.

A. Incorrect but plausible. (1) The Technical Support Center is normally located within the Main Control Room envelope making NSO response to the TSC incorrect but plausible. The TSC however does not dispatch personnel to respond to the event, the OSC does. (2) The OSC is only activated on an Alert or higher. Operators will potentially be involved in response to an Unusual Event thus making (2) plausible here.

B. Incorrect but plausible. (1) is correct as explained above. (2) The OSC is only activated on an Alert or higher. Operators will potentially be involved in response to an Unusual Event thus making (2) plausible.

C. Incorrect but plausible. (1) The Technical Support Center is normally located within the Main Control Room envelope making NSO response to the TSC incorrect but plausible. The TSC however does not dispatch personnel to respond to the event, the OSC does. (2) is correct as explained above.

Technical Reference(s):	ER 3.2, “Operational Support Center Operations”, Rev 56.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1308I 02			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.10		

Examination Outline Cross-reference:	Level	RO	
Q24	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(W E15) Containment Flooding

(WE15EK1.04) Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Containment Flooding: Design-basis flood level in containment

	Importance Rating	3.1	

Proposed Question:

Plant conditions:

- A Loss of Coolant Accident has occurred.
- Refueling Water Storage Tank (RWST) level is at 115,000 gallons and slowly lowering.
- Containment building pressure is 15 psig and slowly lowering.
- Containment Building Level is 5.1 feet and slowly rising.
- The crew has entered ES-1.3, “Transfer to Cold Leg Recirculation” and is performing Step 2, “Verify Containment Recirculation Valves Full Open”.

Which one of the following actions should the crew take and why?

A. Immediately transfer to FR-Z.2, “Response to Containment Flooding” because the high containment building water level indicates that there is an additional flooding volume that could potentially compromise critical systems or components needed for plant recovery.

B. Continue performing the remainder of ES-1.3, “Transfer to Cold Leg Recirculation”. Once ES-1.3 is complete then transfer to FR-Z.2, “Response to Containment Flooding” because the high containment building water level indicates that there is an additional flooding volume that could potentially compromise critical systems or components needed for plant recovery.

C. Continue performing the remainder of ES-1.3, “Transfer to Cold Leg Recirculation”. Once ES-1.3 is complete, then transfer to ECA-1.2, “LOCA Outside Containment” because the low containment building water level indicates that there is a loss of water inventory outside of the containment structure which could potentially compromise the ability for ECCS to cool the reactor core.

D. Continue performing ES-1.3, “Transfer to Cold Leg Recirculation”. Once the first three steps of ES-1.3 are complete then transfer to ECA-1.2, “LOCA Outside Containment” because the low containment building water level indicates that there is a loss of water inventory outside of the containment structure which could potentially compromise the ability for ECCS to cool the reactor core.

Proposed Answer:	B.

Explanation:

B. Correct. The conditions in the stem of the question meet the FR-Z status tree criteria for entry into FR-Z.2, “Response to Containment Flooding”. Per the Westinghouse Background Document for FR-Z.2, a containment building water level greater than the status tree criteria of 4.7 feet is indicative of an additional flooding volume that could potentially compromise critical systems or components needed for plant recovery. Additionally, one of the key Caution statements in ES-1.3, “Transfer to Cold Leg Recirculation” states “Steps 1 through 3 must be performed within 3 minutes after receiving RWST Lo Lo level alarm. The remainder of the procedure should be performed without delay. <u>FRPs should not be implemented prior to completion of this procedure</u>.”

A. Incorrect but plausible. The basis for performing FR-Z.2 is correct. It is plausible that the crew should immediately transition to FR-Z.2, as it is an ORANGE PATH procedure, however one of the key Caution statements in ES-1.3, “Transfer to Cold Leg Recirculation” states “Steps 1 through 3 must be performed within 3 minutes after receiving RWST Lo Lo level alarm. The remainder of the procedure should be performed without delay. FRPs should not be implemented prior to completion of this procedure.”

C. Incorrect but plausible. It is true that the crew should continue performing the remainder of ES-1.3. It is incorrect but plausible that a procedure transition to ECA-1.2, “LOCA Outside Containment” would be made, as the safety function status tree for FR-F, Emergency Recirculation (F) does utilize “Containment Building Level vs RWST Level” as a criteria, however a low building level would be indicative of a LOCA outside containment.

D. Incorrect but plausible. It is true that the crew should continue to process ES-1.3, however it should be processed in its entirety. Additionally, it is incorrect but plausible that a procedure transition to ECA-1.2, “LOCA Outside Containment” would be made, as the safety function status tree for FR-F, Emergency Recirculation (F) does utilize “Containment Building Level vs RWST Level” as a criteria, however a low building level would be indicative of a LOCA outside containment.

Technical Reference(s):	FR-Z.2, “Response to Containment Flooding”, rev 19. ES-1.3, “Transfer to Cold Leg Recirculation”, rev 30.
Proposed references to be provided to applicants during examination:	None
Learning Objective:	SBK LOP L1203I 07, 09

Question Source:	Bank #	X	NUSEC 62568	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2018 NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	41.5 / 41.7	

Examination Outline Cross-reference:	Level	RO	
Q25	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM

(010A4.02) Ability to manually operate and/or monitor the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM in the control room: PZR heaters

Importance Rating	3.6	

Proposed Question:

Plant conditions:

- All PZR Backup Heater control switches are in AUTO.
- A Loss of Offsite Power occurs.
- Busses 5 and 6 are powered from the Emergency Diesel Generators.

What action is required to energize Group ‘A’ and ‘B’ Backup heaters?

A. Place the switches in the ON position, only.
B. Reset RMO. Place the switches in the ON position.
C. Reset RMO. Verify the heaters energize on demand automatically.
D. Place the switches in the OFF position, then go to the ON position.

Proposed Answer:	B.	

Explanation:

B. Correct. PZR Backup Group ‘A’ and ‘B’ are the only heaters available after an LOP with Bus E5 and E6 energized by the EDGs. After the EPS has completed sequencing loads on Bus E5 and E6, RMO must be reset, and the control switches taken to ‘ON’ position to energize the backup heaters. There is no automatic operation of the backup heaters until busses E5 and E6 are powered from offsite power.

A. Incorrect but plausible. RMO and the need to reset RMO is a common student misconception. Normally the backup heaters are energized by placing the control switches to ON. However, with the EDGs powering the emergency busses E5 and E6, RMO must be reset to allow manual operation of the heaters.

C. Incorrect but plausible. The heaters will not operate in auto until offsite power is restored to busses E5 and E6.

D. Incorrect but plausible. RMO and the need to reset RMO is a common student misconception. Also, the control group heaters must be taken to 'OFF' the back to 'ON' to be re-energized after tripping off further adding to the plausibility of this distractor. The student could apply this knowledge incorrectly to the backup heaters.					
Technical Reference(s):	Lesson plan L8027I, "Pressurizer Pressure and Level Control".				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L8027I 5, 6, 8				
Question Source:	Bank #	X	TEB 31634		
	Modified Bank#				
	New				
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	41.7			

Examination Outline Cross-reference:	Level	RO	
Q26	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(059) (SF4S MFW) MAIN FEEDWATER SYSTEM

(059A3.08) Ability to monitor automatic features of the (SF4S MFW) MAIN FEEDWATER SYSTEM, including: S/G water LCS

	Importance Rating	3.8	

Proposed Question:

Plant conditions:

- 100% power.
- SG ‘A’ feed regulating valve is in manual.
- FW-PT-508 “Main Feed Header Pressure” transmitter fails low.

What is the effect on the Main Feed Pumps and what automatic action(s) will take place?

The Main Feed Pumps will _____**(1)**_____ and _____**(2)**_____.

	(1)	(2)
A.	slow down	lowering SG water levels cause reactor to trip on SG level low-low
B.	slow down	the turbine trips on SG low-low level and the reactor trips on main turbine trip
C.	speed up	trip on overspeed causing a reactor trip on SG level low-low
D.	speed up	the reactor will trip on a turbine trip due to SG high-high level

Proposed Answer:	D.

Explanation:

D. Correct. (1) With PT-508 failed low the Main Feed Pumps (MFPs) will speed up due to the lower d/p input to the MFP speed control circuit which is part of the SG level control circuit. (2) The ‘A’ SG feed regulating valve will not reposition as it is in manual and SG level will increase until at 90.8% a turbine trip will occur which will cause a reactor trip.

A. Incorrect but plausible. (1) With PT-508 failed low the Main Feed Pumps (MFP) will speed up due to the lower d/p input to the MFP speed control circuit which is part of the SG level control circuit. However, if the student incorrectly recalled the MFP speed control circuit it is plausible that it slows down. (2) If the MFP were to slow down, lowering SG levels would cause a reactor trip on SG level at 20%.

B. Incorrect but plausible. (1) is incorrect as explained above. (2) If the MFP were to slow down, it is plausible that the turbine would trip on SG level low-low (20%) and this would cause a reactor trip. However, low SG level would directly cause a reactor trip.

C. Incorrect but plausible. (1) is correct as explained above. (2) The MFPs overspeed trip is at 5940 rpm. It is true that the pumps will increase in speed as explained above however, they will not trip on overspeed. If the pumps were to trip on overspeed a reactor trip on SG level low-low would occur.

Technical Reference(s):	Lesson plan L8046I, “Steam Generator Water Level Control”.			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8046I 11			
Question Source:	Bank #	X	TEB 26717	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q27	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM

(062K1.08) Knowledge of the physical connections and/or cause and effect relationships between the (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM and the following systems: Onsite standby power systems

Importance Rating	3.8	

Proposed Question:

Plant conditions:

- The plant was initially at 100% power.
- At 1200, Bus E6 de-energized due to a bus fault.
- At 1202, there was a Loss of Offsite Power.
- ‘A’ Emergency Diesel started and tripped. It cannot be restarted by any means.
- At 1203, the crew entered ECA-0.0, “Loss of All AC Power”.

What actions per ECA-0.0 are required to energize Bus E5 using the Supplemental Emergency Power System (SEPS)?

A. Check SEPS incoming bus voltage and frequency is normal. CLOSE the SEPS Bus E5 breaker at the MCB, only.

B. Manually start the SEPS diesels. Check SEPS incoming bus voltage and frequency is normal. CLOSE the SEPS Bus E5 breaker at the MCB.

C. Transfer SEPS supply from Bus E6 to Bus E5. Check SEPS incoming bus voltage and frequency normal. CLOSE the SEPS Bus E5 breaker at the MCB.

D. Manually start the SEPS diesels. Transfer SEPS supply from Bus E6 to Bus E5. Check SEPS incoming bus voltage and frequency normal. CLOSE the SEPS Bus E5 breaker at the MCB.

Proposed Answer:	C.	

Explanation:

C. Correct. SEPS will automatically start and run unloaded given the initial conditions presented in

the question stem. Per ECA-0.0, to power bus E5 from SEPS requires transferring SEPS supply from Bus E6 to Bus E5 and then closing the SEPS Bus 5 breaker at the MCB A. Incorrect but plausible. The student could mistake that the normal alignment of SEPS is bus E5. B. Incorrect but plausible. The student could mistake that the normal alignment of SEPS is bus E5 and that SEPS needs to be manually started. D. Incorrect but plausible. The student could mistake that SEPS needs to be manually started.					
Technical Reference(s):	Lesson plan L8066I, "SEPS".				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L8066I 02				
Question Source:	Bank #	X	TEB 31622		
	Modified Bank#				
	New				
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis			X	
10 CFR Part 55 Content:	55.41	41.2 to 41.9			

Examination Outline Cross-reference:	Level	RO	
Q28	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(086) (SF8 FPS) FIRE PROTECTION SYSTEM

(086K4.02) Knowledge of (SF8 FPS) FIRE PROTECTION SYSTEM design features and/or interlocks that provide for the following: Maintaining fire header pressure

Importance Rating	3.3	

Proposed Question:

The following plant conditions occurred:

- Fire Brigade Leader was in the process of flushing a protected area fire hydrant.
- All Fire Protection system components are aligned to normal auto/standby condition.
- During the flush, fire main header pressure dropped to 125 psig.
- Total time of the flush evolution was 12 seconds.
- After flush was complete, system pressure increased to 145 psig and stabilized.
- Fire Brigade Leader returned to the Fire Pump House 15 minutes later.

Which component(s) were running, if any?

A. None.

B. Both jockey pumps <u>and</u> electric fire pump.

C. Electric fire pump <u>and</u> one diesel fire pump.

D. Both jockey pumps, the electric fire pump <u>and</u> one diesel fire pump.

Proposed Answer:	C.	

Explanation:

C. Correct. For the conditions given, upon return of the Fire Brigade Leader the electric pump and one diesel driven pump will be running. The jockey pumps start at 131 psig however they will auto shut down when pressure is above 135 psig and hence will not be found running upon return of the Fire Brigade Leader. The electric pump starts at 127 psig and has no auto shut down hence it will be found running. The 'A' diesel driven fire pump will auto start at 127 psig after a 10 second delay hence it will start during the 12 second flush. It does not auto shutdown and hence will be found running. The 'B' diesel driven fire pump will auto start at 127 psig after a 20 second delay and will not start during the flush and will not be found running.

A. Incorrect but plausible. With system pressure restored to 145 psig it is plausible that all the Fire Protection pumps will be found secured however, as explained above upon return of the Fire Brigade Leader the electric pump and one diesel driven pump will be running.

B. Incorrect but plausible. It is plausible that upon return of the Fire Brigade Leader that both jockey pumps and one diesel pump are found running however, as explained above upon return of the Fire Brigade Leader the electric pump and one diesel driven pump will be running.

D. Incorrect but plausible. It is plausible that upon return of the Fire Brigade Leader that both jockey pumps, the electric pump and one diesel pump are found running however, as explained above upon return of the Fire Brigade Leader the electric pump and one diesel driven pump will be running.

Technical Reference(s):	Lesson plan L8089I, “Fire Protection”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8089I 05			
Question Source:	Bank #	X	TEB 29920	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q29	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000015) (APE 15) Reactor Coolant Pump Malfunctions

(G2.2.44) EQUIPMENT CONTROL: Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions

	Importance Rating	4.2	

Proposed Question:

Plant conditions:

- A plant down power from 100% is in progress.
  - NI-41 is reading 50%
  - NI-42 is reading 51%
  - NI-43 is reading 47%
  - NI-44 is reading 48%
- D5775, “RCP A Shaft Vibration High” is in alarm.
- The BOP is performing Attachment ‘A’ of OS1201.01, “RCP Malfunction” to determine the RCP vibration rack module status at the back of the MCB.
- Channel for ‘A’ RCP is determined to be in alarm.
- ‘A’ RCP vibrations are above the danger levels.

After the BOP depresses the RESET pushbutton on the vibration rack module, the alarm will _____**(1)**_____.

Based on the vibration levels and plant power the crew will _____**(2)**_____.

	(1)	(2)
A.	reset and remain reset	trip the reactor and when immediate actions are complete, stop the ‘A’ RCP
B.	reset momentarily then re-alarm	trip the reactor and when immediate actions are complete, stop the ‘A’ RCP
C.	reset and remain reset	stop the ‘A’ RCP per Attachment ‘D’, “RCP Trip

		Actions Below P-8" of OS1201.01
D.	reset momentarily then re-alarm	stop the 'A' RCP per Attachment 'D', "RCP Trip Actions Below P-8" of OS1201.01

Proposed Answer:	B.	

Explanation:

B. Correct. (1) When the BOP depresses the RESET pushbutton the RCP vibration rack module, the alarm will momentarily reset and immediately re-alarm. This is done to verify the status of the alarm. (2) For the given plant conditions, P-8 is not reset. Thus, before stopping the RCP, the reactor must be tripped, and immediate actions performed.

A. Incorrect but plausible. (1) The alarm will momentarily reset and immediately re-alarm however, it is plausible that resetting the alarm would cause it to remain reset. (2) is correct as explained above.

C. Incorrect but plausible. (1) The alarm will momentarily reset and immediately re-alarm however, it is plausible that resetting the alarm would cause it to remain reset. (2) If the student were to incorrectly determine that P-8 is reset this could be chosen. With two NIs indicating below 50% power it is plausible that P-8 is reset however, P-8 reset requires 3/4 NIs to indicate below 48%.

D. Incorrect but plausible. (1) is correct as explained above. (2) If the student were to incorrectly determine that P-8 is reset could be chosen. With two NIs indicating below 50% power it is plausible that P-8 is reset however, P-8 reset requires 3/4 NIs to indicate below 48%.

Technical Reference(s):	OS1201.01, RCP Malfunction", rev 20.
Proposed references to be provided to applicants during examination:	None
Learning Objective:	SBK LOP L8021I 35

Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q30	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(000032) (APE 32) Loss of Source Range Nuclear Instrumentation

(000032AK2.03) Knowledge of the relationship between (APE 32) LOSS OF SOURCE RANGE Nuclear Instrumentation and the following systems or components: RPS

Importance Rating	3.9	

Proposed Question:

Plant conditions:

- A plant shutdown is in progress.
- Reactor power is 6% and decreasing.
- Nuclear Instrumentation Intermediate Range Channel N-36 fails HIGH.

How will this failure affect the plant shutdown and subsequent operation of the Nuclear Instrumentation Source Range Channels?

A. The reactor will trip. The Source Range NI's will have to be manually energized.

B. The reactor will NOT trip. The Source Range NI's will have to be manually energized.

C. The reactor will NOT trip. The Source Range NI's will automatically energize when Intermediate Range Channel N-35 decreases to the proper setpoint.

D. The reactor will trip. The Source Range NI's will automatically energize when Intermediate Range Channel N-35 decreases to the proper setpoint.

Proposed Answer:	A.	

Explanation:

A. Correct. The reactor will trip when Intermediate Range Channel N-36 fails HIGH. An Intermediate Range reactor trip signal occurs when 1 of 2 Intermediate Range Channels is >25% equivalent current. A failed high IR channel would exceed the 25% equivalent current. This trip is active when below the P-10 setpoint (10% reactor power). The Source Range NI's will not automatically energize thus constituting a loss of Source Range instrumentation. During a reactor shutdown the Intermediate Range Channels will decrease down to the P-6 reset value of $5x10^{-11}$ A at which time the Source Range Channels will energize within the normal 1 decade of overlap indication with the intermediate range. Automatic energization of the Source Range NI's requires <u>both</u> Intermediate Range NI's to reduce to the P-6 reset value.

B. Incorrect but plausible. As explained above the reactor will trip when the failure occurs however, considering common misconceptions associated with P-6 and P-10 it is plausible that a reactor trip would not occur given the low power conditions. There are common operator misconceptions about NI related permissive signals, particularly P-6 and P-10. The stem of the question states that reactor power is at 6% and decreasing. The candidate may incorrectly interpret this as being above P-10. If the candidate makes this error, then they may interpret that the Intermediate Range channels are still above the interlock setpoint where they would initiate a reactor trip. This interlock is P-10 (setpoint 10%) as described above for answer A. The second half of the answer is correct.

C. Incorrect but plausible. As stated in the plausibility statement for answer B, there are common operator misconceptions about NI related permissive signals, particularly P-6, P-8, and P-10. The stem of the question states that reactor power is at 6% and decreasing. The candidate may incorrectly interpret this as being above P-8 (which is actually a permissive for a reactor trip based on a single loop loss of coolant flow with a setpoint of 50% on the Power Range NI instrumentation). If the candidate makes this error, then they may interpret that the Intermediate Range channels are still above the interlock setpoint where they would initiate a reactor trip. This interlock is P-10 (setpoint 10%) as described above for answer A. Additionally, the candidate may have a misconception that the P-6 reset only requires 1 of 2 Intermediate Range channels to decrease to the P-6 reset value of $5x10^{-11}$ A, however this requires both channels to do so.

D. Incorrect but plausible. The reactor will trip when Intermediate Range Channel N-36 fails HIGH. An Intermediate Range reactor trip signal occurs when 1 of 2 Intermediate Range Channels is >25% equivalent current. A failed high IR channel would exceed the 25% equivalent current. This trip is active when below the P-10 setpoint (10% reactor power). The candidate may have a misconception that the P-6 reset only requires 1 of 2 Intermediate Range channels to decrease to the P-6 reset value of $5x10^{-11}$ A, however this requires both channels to do so.

Technical Reference(s):	Precautions, Limitations and Setpoints for Nuclear Steam Supply Systems, pgs. 6 and 12, rev 27.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8030I 02, 08, 11			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2010 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.8 / 41.10		

Examination Outline Cross-reference:	Level	RO	
Q31	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000054) (APE 54; CE E06) Loss of Main Feedwater

(G2.1.20) CONDUCT OF OPERATIONS: Ability to interpret and execute procedure steps

Importance Rating	4.6	

Proposed Question:

Step 1 of FR-H-1, “Response to Loss of Secondary Heat Sink” checks if RCS pressure is greater than any non-faulted SG pressure because if RCS pressure is less than intact SG pressures __________.

A. strategies in FR-H.1 will not be successful and the crew will transition to E-0.

B. a loss of coolant accident is occurring and the crew will immediately transition to ES-1.3.

C. the steam generators no longer function as a heat sink and the crew will return to the procedure and step in effect, only.

D. the steam generators no longer function as a heat sink and the crew will first verify that bleed and feed is not required and then return to the procedure and step in effect.

Proposed Answer:	C.	

Explanation:

C. Correct. From the background document of FR-H.1, “before implementing actions to restore flow to the steam generators, the operator should check if secondary heat sink is required. For larger LOCA break sizes, the RCS will depressurize below the intact steam generator pressures. The steam generators no longer function as a heat sink and the core decay heat is removed by the RCS break flow.” The crew will be directed per step 1 RNO to return to procedure and step in effect.

A. Incorrect but plausible. It is plausible that if RCS pressure is less than SG pressure that the strategies in FR-H.1 will not be successful and that this is the reason that the crew will transition out of FR-H.1 and back to E-0.

B. Incorrect but plausible. In accordance with the background document for FR-H.1, the reason that RCS pressure might be less than the intact SG pressures is due to a LOCA and if this is the case the steam generators no longer function as a heat sink. However, the crew will not immediately transition to ES-1.3 in response to this determination.

D. Incorrect but plausible. From the background document of FR-H.1, “before implementing actions to restore flow to the steam generators, the operator should check if secondary heat sink is required. For larger LOCA break sizes, the RCS will depressurize below the intact steam generator pressures. The steam generators no longer function as a heat sink and the core decay heat is removed by the RCS break flow.” It is plausible but incorrect that the crew will first check if bleed and feed is required before transitioning out of FR-H.1 and return to procedure and step in effect.

Technical Reference(s):	FR-H.1, “Response to Loss of Secondary Heat Sink”, rev 38. Background document for FR-H.1, rev 3.				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L1211I 03				
Question Source:	Bank #				
	Modified Bank#				
	New	X			
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	41.10			

Examination Outline Cross-reference:	Level	RO	
Q32	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM

(007A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM, including: Containment radiation levels

Importance Rating	3.2	

Proposed Question:

Plant conditions:

- 100% power.
- The Pressurizer pressure control system is operating in automatic with pressure channels 455/456 selected for control/backup.

Without operator action, which of the following will result in RM-6526, “Containment Particulate Radiation Monitor” going into alarm?

A. channel 458 failing high
B. channel 457 failing high
C. channel 456 failing low
D. channel 455 failing low

Proposed Answer:	D.	

Explanation:

D. Correct. Instrument 455 is selected as the controlling channel. The controlling channel provides input into the pressurizer spray valves, control group heaters, and PCV-456A, “PORV A”. If instrument 455 fails low the control group heaters would energize and the spray valves and “A” PORV would never open. The “B” PORV receives its open signal from the backup instrument (in this case instrument 456) provided it also receives an “open interlock” signal from instrument 457 (hardwired). Given the conditions in the question stem, the “B” PORV would open. Discharge from the “B” PORV is routed to the Pressurizer Relief Tank (PRT). The PRT is protected from overpressure by a rupture disc which relieves to the containment atmosphere. When the rupture disks eventually relieve pressure, RM6526, “Containment Particulate Radiation Monitor” will go into alarm.

A. Incorrect but plausible. If the student understood the effects of a failed instrument on heaters and sprays but had a misconception of the instrument requirements for opening the PORVs then may choose this distractor.

B. Incorrect but plausible. If the student understood the effects of a failed instrument on heaters and sprays but had a misconception of the instrument requirements for opening the PORVs then may choose this distractor.

C. Incorrect but plausible. If the student understood the effects of a failed instrument on heaters and sprays but had a misconception of the instrument requirements for opening the PORVs then may choose this distractor.

Technical Reference(s):	1-NHY-509026, Pressurizer Pressure Control, rev 9.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8027I 14			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2018 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q33	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM

(008K3.01) Knowledge of the effect that a loss or malfunction of the (SF8 CCW) COMPONENT COOLING WATER SYSTEM will have on the following systems or system parameters: Loads cooled by CCWS

Importance Rating	4.0	

Proposed Question:

While operating at 100% power the following trends are observed:

- C0768, “Containment Average Temperature” is 111°F and slowly increasing.
- A0285, “RCP Thermal Barrier Inlet Temperature” is 94°F and slowly increasing.
- CS-TI-130, “Letdown HX Outlet Temperature” is 118°F and increasing.
- CS-TK-130, “Letdown HX Temperature Controller” output is 100% and stable.

Which of the following could be the cause of these indications?

A. 1-CC-TK-2171, “PCCW Loop ‘A’ Supply Header Temperature Controller” output failing high.

B. 1-CC-TK-2171, “PCCW Loop ‘A’ Supply Header Temperature Controller” output failing low.

C. 1-CC-TK-2271, “PCCW Loop ‘B’ Supply Header Temperature Controller” output failing high.

D. 1-CC-TK-2271, “PCCW Loop ‘B’ Supply Header Temperature Controller” output failing low.

Proposed Answer:	B.	

Explanation:

B. Correct. ‘A’ train PCCW temperature controller (CC-TK-2171) output failing low would cause TV-2171-1 (HX outlet) to close and CC-TV-2171-2 (HX bypass) to open. ‘A’ train PCCW temperature would increase. Containment and RCP thermal barrier systems are cooled by both trains of PCCW. ‘A’ train cooling water temperature increase would cause these temperatures to increase. Letdown HX is cooled by ‘A’ train of PCCW only. Question stem has letdown temperature increasing and the controller has increased to maximum trying to maintain it at setpoint.

A. Incorrect but plausible. ‘A’ train PCCW temperature controller (CC-TK-2171) output failing high would cause TV-2171-1 (HX outlet) to open and CC-TV-2171-2 (HX bypass) to close. ‘A’ train

PCCW temperature would decrease. This would result in temperature decrease of the supplied components. Question stem has temperatures increasing not decreasing.

C is incorrect but plausible. ‘B’ train PCCW temperature controller (CC-TK-2271) output failing high would cause TV-2271-1 (HX outlet) to open and CC-TV-2271-2 (HX bypass) to close. ‘B’ train PCCW temperature would decrease. This would result in Temperature decrease of the supplied components. Question stem has temperatures increasing not decreasing. Letdown NRHX is cooled by ‘A’ train of PCCW only, changes to ‘B’ train of PCCW would have no effect on letdown.

D. Incorrect but plausible. ‘B’ train PCCW temperature controller (CC-TK-2271) output failing low would cause TV-2271-1 (HX outlet) to close and CC-TV-2271-2 (HX bypass) to open. ‘B’ train PCCW temperature would increase. Containment and RCP thermal barrier systems are cooled by both trains of PCCW. ‘B’ train cooling water temperature increase would cause these temperatures to increase. Letdown is cooled by ‘A’ train of PCCW and would be unaffected by this condition.

Technical Reference(s):	Lesson plan L8036I, “PCCW”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8036I 10			
Question Source:	Bank #	X	32897	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q34	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000077) (APE 77) Generator Voltage and Electric Grid Disturbances

(G2.4.31) EMERGENCY PROCEDURES/PLAN: Knowledge of annunciator alarms, indications, or response procedures

Importance Rating	4.2	

Proposed Question:

Given the following plant conditions:

- An electrical grid disturbance occurs resulting in a loss of the 345kV Tewksbury (394) line.
- Generator reactive load is at 75 MVAR "Leading".
- Alarm point D6442 GEN UEL LIMIT-LIMITER ON is in alarm.

According to the alarm response procedure, Main Generator excitation will be **_____(1)_____** in order to increase VAR loading in the **_____(2)______** direction.

	(1)	(2)
A.	raised	leading
B.	raised	lagging
C.	lowered	leading
D.	lowered	lagging

Proposed Answer:	B.

Explanation:

B. Correct. The conditions stated in the question stem are indicative of an under-excited condition in the main generator. As generator excitation is raised VAR loading will move in the "lagging" direction.

A. Incorrect but plausible. Raising excitation is done such that VAR loading trends in the "lagging" direction.

C. Incorrect but plausible. It is true that VAR loading must be adjusted however VARS should be adjusted in the "lagging" direction vice increasing "leading" VAR's.

D. Incorrect but plausible. It is true that VAR loading must be in the "lagging" direction however this would be accomplished by increasing generator excitation.

Technical Reference(s):	VPRO for D6442, rev 9.			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8061I 03			
Question Source:	Bank #	X	TEB 34899	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	2010 Seabrook NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.10		

Examination Outline Cross-reference:	Level	RO	
Q35	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM

(061K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM: AFW flow controller

Importance Rating	3.9	

Proposed Question:

Plant conditions:

- Reactor trip due to loss of offsite power.
- ES-0.1, “Reactor Trip response” is in progress.
- EFW flow to all SGs has been isolated to control SG level and RCS temperature by closing the upstream flow control valves on the MCB apron.
- ‘B’ EDG trips due to a fault on Bus 6.

What is the current status of the ability to control SG levels?

A. All SGs can be controlled from the MCB.

B. All SGs can only be controlled by an operator in the EFW pump house.

C. ‘A’ and ‘C’ SGs can be controlled from the MCB. ‘B’ and ‘D’ SGs can only be controlled by an operator in the EFW pump house.

D. ‘B’ and ‘D’ SGs can be controlled from the MCB. ‘A’ and ‘C’ SGs can only be controlled by an operator in the EFW pump house.

Proposed Answer:	C.	

Explanation:

C. Correct. The question stem states that there was a loss of offsite power and that EFW flow to all four SGs was isolated. This means that the four EFW control valves on the main control board apron were taken to CLOSE and that the emergency busses were being powered by the emergency diesel generators. When the ‘B’ EDG subsequently trips only the ‘A’ and ‘C’ EFW control valves are capable of being controlled from their control board switches. The ‘B’ and ‘D’ EFW control valves would have to be controlled locally utilizing the clutch level and manual valve wheel.

A. Incorrect but plausible. All SG’s do have EFW flow control valves that are powered from Bus 6, however, given the conditions in the question stem, the ‘B’ and ‘D’ SGs could not be controlled from the MCB because their Train ‘B’ powered control valves (“apron valves”) were closed prior to the ‘B’ EDG tripping.

B. Incorrect but plausible. All SG’s do have EFW flow control valves that are powered from Bus 6, however, given the conditions in the question stem, the ‘A’ and ‘C’ SGs could be controlled from the MCB because their Train ‘A’ powered control valves (“apron valves”) are still powered from the “A” EDG.

D. Incorrect but plausible. The question requires the student to know the train related power supplies to the two in-series EFW control valves for each SG. If the student had the power supplies reversed, then this would be the correct answer.

Technical Reference(s):	UFSAR section 6.8, page 6, rev 20.			
Proposed references to be provided to applicants during examination:			None	
Learning Objective:	SBK LOP L8045I 01			
Question Source:	Bank #	X	TEB 34730	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2015 NRC Exam, same K/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q36	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM

(006A2.13) Ability to (a) predict the impacts of the following on the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Inadvertent ECCS actuation

Importance Rating	3.6	

Proposed Question:

Plant conditions:

- Reactor trip due to an inadvertent SI.
- The crew has stopped one charging pump per E-0 step 12.
- Pressurizer level is 96% and rising.

What actions will the crew take in accordance with E-0, “Reactor Trip or Safety Injection” to mitigate the increase in pressurizer level?

A. Check at least one thermal barrier pump running, then stop the second charging pump and establish normal letdown.

B. Check at least one thermal barrier pump running, then stop the second charging pump and establish excess letdown.

C. Stop the second charging pump only and establish normal letdown.

D. Stop the second charging pump only and establish excess letdown.

Proposed Answer:	B.	

Explanation:

B. Correct. Per step 13 of E-0 if pressurizer level exceeds 95% following an inadvertent SI, operators will check one thermal barrier pump running, stop the second charging pump and establish excess letdown. Step 13 of E-0 is typically only exercised following an inadvertent SI and this is a specific action unique to this event. The student must predict that when stopping both charging pumps, normal letdown is no longer available and excess letdown will be used to lower pressurizer level.

A. Incorrect but plausible. Per step 13 of E-0 if pressurizer level exceeds 95% following an inadvertent SI, operators will check one thermal barrier pump running, stop the second charging pump and establish excess letdown. Normal letdown is a plausible method of lowering pressurizer level as this is the normal means to do so however, with both charging pumps secured normal letdown is not available and excess letdown must be used.

C. Incorrect but plausible. Per step 13 of E-0 if pressurizer level exceeds 95% following an inadvertent SI, operators will check one thermal barrier pump running, stop the second charging pump and establish excess letdown. The student must recall that to secure the second charging pump a thermal barrier pump must first be verified to be running as seal injection flow will be lost when both charging pumps are secured. Normal letdown is a plausible method of lowering pressurizer level as this is the normal means to do so however, with both charging pumps secured normal letdown is not available and excess letdown must be used.

D. Incorrect but plausible. Per step 13 of E-0 if pressurizer level exceeds 95% following an inadvertent SI, operators will check one thermal barrier pump running, stop the second charging pump and establish excess letdown. The student must recall that to secure the second charging pump a thermal barrier pump must first be verified to be running as seal injection flow will be lost when both charging pumps are secured. It is correct that excess letdown will be established.

Technical Reference(s):	E-0, “Reactor Trip or Safety Injection” Attachment A, Rev 62.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L1202I 11			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	41.5	

Examination Outline Cross-reference:	Level	RO	
Q37	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000026) (APE 26) Loss of Component Cooling Water

(000026AK2.05) Knowledge of the relationship between (APE 26) LOSS OF Component Cooling Water and the following systems or components: RMS

Importance Rating	3.0	

Proposed Question:

‘A’ Train PCCW head tank level is decreasing due to a leak. What occurs when level reaches less than 42%?

A. Isolates CC to the letdown Hx (CC-341) and
Isolates CC to the SF Hx’s (CC-V-32) and
WPB Train ‘A’ supply valves isolate (CC-V-426 and CC-V-427)

B. WPB Train ‘A’ supply valves isolate (CC-V-426 and CC-V-427) and
Train ‘A’ Radiation Monitor isolates (CC-V-975 and CC-V-1298)

C. Train ‘A’ Thermal Barrier Supply Valves isolate (CC-V-1101 and CC-V-1109)

D. Train ‘A’ PCCW supply and return valves to containment isolate (CC-V-168, 57, 121, 122)

Proposed Answer:	B.	

Explanation:

B. Correct. At 42% decreasing, WPB Train ‘A’ supply valves isolate and Train ‘A’ Radiation Monitor isolates.

A. Incorrect but plausible. The valves in this distractor are related to the PCCW system but isolate on a ‘T’ signal, not low tank level.

C. Incorrect but plausible. The valves in this distractor are related to the PCCW system but do not automatically isolate.

D. Incorrect but plausible. The valves in this distractor are related to the PCCW system but isolate on a ‘P’ signal, not low tank level.

Technical Reference(s):	Lesson plan L8036I, PCCW.			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8036I 11			
Question Source:	Bank #	X	TEB 32174	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge		X	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.8 / 41.10		

Examination Outline Cross-reference:	Level	RO	
Q38	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(103) (SF5 CNT) CONTAINMENT SYSTEM

(103A4.09) Ability to manually operate and/or monitor the (SF5 CNT) CONTAINMENT SYSTEM in the control room: Containment vacuum system

Importance Rating	3.0	

Proposed Question:

Plant conditions:

- Containment building pressure is being reduced in accordance with OS1023.69, "Containment On-Line Purge (COP) System Operation."
- COP Exhaust Containment Isolation Valves COP-V-3 and COP-V-4 have been opened.
- The crew is establishing COP flow through COP-V-8, "COP Exhaust Throttle Valve (Coarse Control)".
- RM-6527A-1 and RM-6527A-2, "Train 'A' COP Rad Monitor" go into HIGH alarm.

Which of the following describes how the control room crew will control the radiological release?

A. Control room operators must ensure COP-V-4 automatically closes to stop the release.

B. Control room operators must ensure COP-V-3 and COP-V-8 automatically close to stop the release.

C. Control room operators must ensure COP-V-4 and COP-V-8 automatically close to stop the release.

D. Control room operators must manually close COP-V-3 and COP-V-4 since no automatic actions will occur.

Proposed Answer:	A.	

Explanation:

A. Correct. COP-V-3 and 4 receive an automatic Containment Ventilation Isolation (CVI) signal to close when high radiation is sensed. COP Valves 1 & 4 receive a CVI signal from Train 'A'. COP Valves 2 & 3 receive a CVI signal from Train 'B'.

B. Incorrect but plausible. COP V-3 does receive a CVI signal, however it is from Train 'B' radiation monitors. It is a common operator misconception that the COP exhaust throttle valves also receive a CVI signal, but this is incorrect (COP-V-8 will not close automatically). C. Incorrect but plausible. COP-V-4 is a Train 'A' valve and will receive a CVI signal to close. It is a common operator misconception that the COP exhaust throttle valves also receive a CVI signal, but this is incorrect (COP-V-8 will not close automatically). D. Incorrect but plausible. Both COP-V-3 and 4 receive a CVI signal to close, however COP-V-3 receives it's signal from Train 'B'.				
Technical Reference(s):	Lesson plan, L8038I, "Containment Building HVAC".			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8038I 25			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2013 and 2020 NRC Exams		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q39	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

(013K4.13) Knowledge of (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM design features and/or interlocks that provide for the following: MFW isolation/reset

Importance Rating	3.7	

Proposed Question:

The plant was at 100% power when the following events occur:

- The ‘C’ Main Feedwater Regulating Valve fails open.
- The crew manually trips the reactor.
- Reactor Trip Breaker ‘B’ fails to open.
- ‘C’ Steam Generator narrow range level instruments are indicating:
  - 92% and increasing
  - 89% and increasing
  - 91% and increasing
  - 88% and increasing
- Reactor Coolant System Tavg instruments are indicating:
  - 559°F and decreasing
  - 559°F and decreasing
  - 553°F and decreasing
  - 558°F and decreasing

What is the status of the Feedwater Isolation Signal (FWI) and its basis?

A. Neither train FWI has actuated because no setpoints have been reached.
B. Both trains of FWI have actuated based on “P-14”.
C. Only the Train ‘A’ FWI signal has actuated based on “P-4 combined with P-14”.
D. Only the Train ‘A’ FWI signal has actuated based on “P-4 combined with a low Tavg”.

Proposed Answer:	B.	

Explanation:

B. Correct. The setpoint for P-14 is 2 of 4 Steam Generator narrow range levels >90.8% on any Steam Generator. There is no P-4 input associated with the P-14 logic.

A. Incorrect but plausible. It is true that FWI would not have actuated based on “P-4 combined with Lo Tavg” as the setpoint is < 557°F (2 of 4 logic). A FWI signal is generated from any Safety Injection signal, so it is true that there was no SI signal. The signal would, however, be generated based on P-14. The student could choose answer “A” if they had a misconception regarding the P-14 setpoint. NOTE: There is a common operator misconception regarding FWI signals, particularly the P-14 and Low Tavg setpoints and the association with P-4.

C. Incorrect but plausible It is true that the P-14 setpoint has been reached, however the P-14 signal logic does not include input from P-4. NOTE: There is a common operator misconception regarding FWI signals, particularly the P-14 and Low Tavg setpoints and the association with P-4.

D. Incorrect but plausible. It is true that based on “P-4 combined with a low Tavg” that only the Train ‘A’ signal would actuate as the ‘B’ Reactor Trip Breaker failed to open and generate a P-4 signal for that train, however the “P-4 combined with a low Tavg” setpoint was not reached. NOTE: There is a common operator misconception regarding FWI signals, particularly the P-14 and Low Tavg setpoints and the association with P-4.

Technical Reference(s):	Lesson Plan, L8056I, “RPS”

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8056I 18			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2018 NRC Exam		

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

10 CFR Part 55 Content:	55.41	41.7

Examination Outline Cross-reference:	Level	RO	
Q40	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(056) (SF4S CDS) CONDENSATE SYSTEM

(056K5.08) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4S CDS) CONDENSATE SYSTEM: Chemistry specifications for secondary system

	Importance Rating	2.7	

Proposed Question:

Plant conditions:

- 100% power.
- VAS alarm B5957, CONDENSATE PUMP DISCHARGE CONDUCTIVITY HIGH is in alarm.
- Chemistry has sampled the secondary system in accordance with procedure CD0905.07, Seawater In-Leakage.
- Chemistry has confirmed that there is a valid saltwater intrusion, and that Condensate Pump discharge conductivity is 1.5 µmhos (micro mhos).
- The crew has entered procedure OS1234.02, “Condenser Tube or Tube Sheet Leak.”

What action will the crew take in accordance with OS1234.02?

A. Commence a power decrease to isolate the affected water box.

B. Trip the Reactor and go to procedure E-0, “Reactor Trip or Safety Injection”.

C. Remain at 100% power and continue plant operation while monitoring the leak rate trend.

D. Commence a plant shutdown to Hot Standby per procedure OS1231.04, “Rapid Down Power”.

Proposed Answer:	B.	

Explanation:

B. Correct. Procedure OS1234.02 contains continuous action step #7 which evaluates the need to trip the reactor. The threshold value for tripping the reactor is >1.0 micromho and that Chemistry has determined that there has been a valid saltwater intrusion.

A. Incorrect but plausible. If the Condensate Pump discharge conductivity is less than 1.0 micromho then the procedure directs performing a plant down power per management recommendation to isolate the affected water box.

C. Incorrect but plausible. If the Condensate Pump discharge conductivity is less than 1.0 micromho then the procedure contains the option of continuing plant operation per management recommendation and continuing to monitor leak rate trends.

D. Incorrect but plausible. If the Condensate Pump discharge conductivity is less than 1.0 micromho then the procedure contains additional guidance for shutting the plant down to Hot Standby per management recommendation.

Technical Reference(s):	OS1234.02, “Condenser Tube or Tube Sheet Leak”, rev			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1188I 03			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2010 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q41	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(076) (SF4S SW) SERVICE WATER SYSTEM

(076A1.02) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4S SW) SERVICE WATER SYSTEM, including: Reactor and turbine building CCW temperatures

	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- 100% power.
- TA has occurred in both trains.
- SW-V-20, “SW Train ‘A’ to DTS Isolation” failed to close and cannot be closed locally.
- SW-P-110A, Cooling Tower pump has just been stopped.

Which of the following describes the effect of these conditions on SCCW and ‘A’ Train PCCW temperatures?

	SCCW	**‘A’ Train PCCW**
A.	Increase	Increase
B.	Increase	Stable
C.	Stable	Stable
D.	Stable	Increase

Proposed Answer:	A.

Explanation:

A. Correct. When both trains of SW receive a Tower Actuation (TA) signal, SW-V-4 and 5 will close, isolating all SW to the Turbine Building. With no cooling, SCCW (Secondary Component Cooling Water) temperatures will increase. Response to SW-V-20 failing to close is to secure SW-P-110A to prevent pumping the cooling tower inventory to the ocean which would render both trains of SW inoperable. With no SW pumps running in the ‘A’ train, ‘A’ PCCW temperatures will increase.

B. Incorrect but plausible. SCCW temperature will increase as explained above. ‘A’ Train PCCW temperatures could be believed to be stable if the student did not understand that securing SW-P-110A will stop all SW flow to the ‘A’ PCCW heat exchanger.

C. Incorrect but plausible. If the student did not realize that the dual train TA will close both SW-V-4 and 5 isolating all flow to the SCCW heat exchangers it is plausible that SCCW temperatures remain stable. ‘A’ Train PCCW temperatures could be believed to be stable if the student did not understand that securing SW-P-110A will stop all SW flow to the ‘A’ PCCW heat exchanger.

D. If the student did not realize that the dual train TA will close both SW-V-4 and 5 isolating all flow to the SCCW heat exchangers it is plausible that SCCW temperatures remain stable. Response to SW-V-20 failing to close is to secure SW-P-110A to prevent pumping the cooling tower inventory to the ocean which would render both trains of SW inoperable. With no SW pumps running in the ‘A’ train, ‘A’ PCCW temperatures will increase.

Technical Reference(s):	OS1216.01, “Degraded Ultimate Heat Sink”, rev 24.			
Proposed references to be provided to applicants during examination:			None	
Learning Objective:	SBK LOP L8037I 13			
Question Source:	Bank #	X	NUSEC 46676	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2013 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q42	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(059) (SF4S MFW) MAIN FEEDWATER SYSTEM

(059A1.02) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4S MFW) MAIN FEEDWATER SYSTEM, including (CFR: / 45.5): MFW pump oil temperatures and MFW pump vibrations

	Importance Rating	2.9	

Proposed Question:

Plant conditions:

- 100% power.
- Alarm D4723, “MFP B TURB HP BRG VIBRATION HI-HI” actuates.

Per the VPRO for D4723, Operators will monitor vibrations on DCS and if vibrations are sustained above 5 mils they will **___________.**

A. trip the reactor and transition to E-0, “Reactor Trip or safety Injection”, then manually trip the pump

B. manually trip the pump, then transition to OS1231.03, “Turbine Runback/Setback”

C. trip the reactor and transition to E-0, “Reactor Trip or safety Injection”, verify that the pump trips automatically due to high vibrations

D. verify that the pump trips automatically due to high vibrations, then transition to OS1231.03, “Turbine Runback/Setback”

Proposed Answer:	A.	

Explanation:

A. Correct. The MFPs will not automatically trip on high vibrations. With the plant at 100% power a reactor trip is required followed by a manual trip of the pump. The plant cannot sustain a single main feedwater pump trip from 100% power.

B. Incorrect but plausible. The MFPs will not automatically trip on high vibrations. With the plant at 100% power a reactor trip is required followed by a manual trip of the pump. The plant cannot sustain a single main feedwater pump trip from 100% power. It is plausible that the crew would transition to OS1231.03, “Turbine Runback/Setback” as the trip of the pump will cause a setback signal to actuate which is an entry condition to OS1231.03. However, the plant cannot sustain a trip of a single feedwater pump from full power and the crew will be required to trip the reactor and then manually trip the pump.

C. Incorrect but plausible. The main feedwater pumps will not trip on high vibrations. Other components will trip on high vibrations thus making this plausible e.g., the main turbine. With the plant at 100% power a reactor trip is required followed by a manual trip of the pump. The plant cannot sustain a single main feedwater pump trip from 100% power.

D. Incorrect but plausible. The main feedwater pumps will not trip on high vibrations. Other components will trip on high vibrations thus making this plausible e.g., the main turbine. It is plausible that the crew would transition to OS1231.03, “Turbine Runback/Setback” as the trip of the pump will cause a setback signal to actuate which is an entry condition to OS1231.03. However, the plant cannot sustain a trip of a single feedwater pump from full power and the crew will be required to trip the reactor and then manually trip the pump.

Technical Reference(s):	VPRO D4723, “MFP B TURB HP BRG VIBRATION HI-HI”, rev 1. Lesson plan L8062I, “Feedwater”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8062I 11			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q43	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM

(005A3.02) Ability to monitor automatic features of the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM, including: RHRS actuation

Importance Rating	4.2	

Proposed Question:

Plant conditions:

- Refueling outage cooldown is in progress per the appropriate MPE.
- One RCP is running.
- RCS temperature is 320°F
- RHR Train 'B' is in service for shutdown cooling.
- Safety Injection signal is actuated coincident with a Loss of Offsite Power (SI/LOP).

How will RH-P-8B and RH-P-8A automatically respond to this event?

	RH-P-8B	**RH-P-8A**
A.	Breaker stays closed. Pump restarted when Bus 6 is repowered.	Pump started by the EPS.
B.	Breaker tripped by Bus 6 UV stripping scheme. Pump not restarted.	Pump started by the EPS.
C.	Breaker tripped by Bus 6 UV stripping scheme. Pump restarted by the EPS.	Pump started by the EPS.
D.	Breaker tripped by Bus 6 UV stripping scheme. Pump restarted by the EPS.	Pump will not start.

Proposed Answer:	C.

Explanation:

C. Correct. Following the LOP/SI, the 'B' RHR pump will be stripped from Bus 6 and restarted by the Emergency Power Sequencer (EPS). For the given plant conditions, the 'A' RHR pump will be in standby and will be automatically started by the EPS.

A. Incorrect but plausible. If the student were not aware of the function of the EPS and the automatic stripping of the bus following an SI/LOP this could be a plausible answer. For the given plant conditions, the ‘A’ RHR pump will be in standby and will be automatically started by the EPS.

B. Incorrect but plausible. Following the LOP/SI, the ‘B’ RHR pump will be stripped from Bus 6 and restarted by the Emergency Power Sequencer (EPS). With the ‘B’ RHR pump in shutdown cooling mode, it is plausible that it is not restarted on the SI. For the given plant conditions, the ‘A’ RHR pump will be in standby and will be automatically started by the EPS.

D. Incorrect but plausible. Following the LOP/SI, the ‘B’ RHR pump will be stripped from Bus 6 and restarted by the Emergency Power Sequencer (EPS). If the student were not aware of the required alignment of the ‘A’ RHR pump for the given plant conditions, they may incorrectly believe that it is not in standby and will not start following the SI/LOP.

Technical Reference(s):	Lesson plan L8033I, “RHR” .

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8033I 05			
Question Source:	Bank #	X	TEB 28933	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q44	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(041) (SF4S SDS) STEAM DUMP/TURBINE BYPASS CONTROL SYSTEM

(041K1.07) Knowledge of the physical connections and/or cause and effect relationships between the (SF4S SDS) STEAM DUMP/TURBINE BYPASS CONTROL SYSTEM and the following systems: RPS

Importance Rating	3.3	

Proposed Question:

Plant conditions:

- A plant cooldown is in progress.
- RSC Loop Tavg:
  - Loop 1: 550°F decreasing.
  - Loop 2: 548°F decreasing.
  - Loop 3: 551°F decreasing.
  - Loop 4: 548°F decreasing.
- Steam header pressure is 1030 psig and decreasing.
- Steam Dump Mode Selector Switch is in Steam Pressure Mode.
- Steam Dump Controller is in Manual set at 30% demand.
- The BOP momentarily places the Train ‘A’ and ‘B’ Steam Dump Bypass Interlock Switches to BYPASS.

What is the status of the Steam Dump valves?

A. All valves full closed.

B. Bank 1 valves are partly open.

C. Bank 1 valves are fully open and valves in Bank 2 are fully closed.

D. Bank 1 valves are fully open and valves in Bank 2 valves are partly open.

Proposed Answer:	C.	

Explanation:

C. Correct. The RPS P-12 Low-Low setpoint is 550°F (2/4). For the conditions given the P-12 setpoint has been reached and RPS will activate the protective feature to close the steam dumps. When the BOP places the Steam Dump Bypass Interlock Switches to BYPASS, bank 1 will reopen but bank 2 is blocked from opening by P-12. With the Steam Dump Controller output at 30% bank 1 will be full open as the full open position of bank 1 corresponds to 25% output.

A. Incorrect but plausible. If the student had a misconception of the Steam Dump Bypass Interlock Switches this could be a possible answer as they may believe that once closed by P-12 the valves remain closed.

B. Incorrect but plausible. It is correct that the bank 1 valves will be open however, based on controller output of 30% they will be fully open not partly open.

D. Incorrect but plausible. It is correct that the bank 1 valves will be fully open however, bank 2 is blocked from opening by the P-12 protective feature.

Technical Reference(s):	1-NHY-509050, Steam Dump Functional, rev 12. Lesson plan L8047I, Steam Dumps.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8047I 12			
Question Source:	Bank #	X	NUSEC 62604	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	2018 Seabrook NRC Exam		

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

10 CFR Part 55 Content:	55.41	41.2 to 41.9

Examination Outline Cross-reference:	Level	RO	
Q45	Tier #	3	
	Group #		

Knowledge and Ability (K/A) Statement:

Conduct of Operations

(G2.1.7) CONDUCT OF OPERATIONS: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Importance Rating	4.4	

Proposed Question:

Plant conditions:

- 75% power.
- CO-P-30A and B are in service with 30C in standby.
- The BOP observes the following:
  - Condensate discharge header pressure 450 psig.
  - CO-P-30A current = 500 Amps.
  - CO-P-30B current = 400 Amps.
  - All hotwell levels are (-) 9 inches.
- CO-P-30A trips and CO-P-30C starts automatically.

(1) What was the cause of the 'A' Condensate pump trip and **(2)** what is the required procedure?

	(1)	(2)
A.	overcurrent	OS1231.03, "Turbine Runback/Setback"
B.	low hotwell level	OS1231.03, "Turbine Runback/Setback"
C.	overcurrent	OS1290.02, Response to Secondary System Transient"
D.	low hotwell level	OS1290.02, Response to Secondary System Transient"

Proposed Answer:	C.	

Explanation:

C. Correct. (1) The auto start of a condensate pump is only caused by an electrical trip of the running pump. (2) OS1290.02 is the required procedure transition.

A. Incorrect but plausible. (1) is correct as explained above. (2) Provided the standby condensate pump starts, suction pressure to the main feedwater pumps will not be lost and a setback will not occur.

B. Incorrect but plausible. (1) The condensate pumps do not trip on low hotwell level, though this is plausible. (2) Provided the standby condensate pump starts, suction pressure to the main feedwater pumps will not be lost and a setback will not occur.

D. Incorrect but plausible. (1) incorrect as explained above. (2) correct as explained above.

Technical Reference(s):	OS1290.02, "Response to Secondary System Transient", rev 11. CO Pump Logic, 1-NHY-503320, rev 5.				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L1191I 07				
Question Source:	Bank #	X	NUSEC 43181		
	Modified Bank#				
	New				
Question History:	Last NRC Exam	2007 Seabrook NRC Exam, same K/A			
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis			X	
10 CFR Part 55 Content:	55.41	41.5			

Examination Outline Cross-reference:	Level	RO	
Q46	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000027) (APE 27) Pressurizer Pressure Control System Malfunction

(000027AK2.10) Knowledge of the relationship between (APE 27) PRESSURIZER PRESSURE Control System Malfunction and the following systems or components: PZR pressure transmitters

Importance Rating	3.7	

Proposed Question:

Plant conditions:

- A plant startup is in progress.
- 6% power.
- Pressurizer pressure channel PT-456 fails low to 1600 psig.
- The crew trips the required bistables in accordance with OS1201.06, “Pressurizer Pressure Instrument/Component Failure.”
- Power is then lost to EDE-PP-1D.

Which of the following describes the expected status of the plant?

A. Plant trip occurs due to OPΔT.

B. The plant will remain at power.

C. Plant trip occurs due to SI actuation.

D. Plant trip occurs due to Low Pressurizer Pressure.

Proposed Answer:	C.	

Explanation:

C. Correct. With the failure of PT-465 all protective features that this channel 2 instrument inputs into will have had the associated bistables tripped per OS1201.06. These include high pressure reactor trip, low pressure reactor trip (blocked below P-10 (10% power)), pressurizer pressure low SI (blocked below P-11 (1950 psig)), OTΔT, and P-11. When PP-1D is lost all channel 4 bistables will deenergize to the tripped condition. Coincidence will then be made for the pressurizer pressure low SI (1800 psig) and the low pressurizer pressure reactor trip (1945 psig). However, with plant power below P-10, the low pressure trip is blocked and the reactor/plant will trip on the low pressurizer pressure SI.

A. Incorrect but plausible. PT-456 inputs to the OTΔT trip but not the OPΔT trip. This is a common misconception.

B. Incorrect but plausible. It is plausible that no automatic trip occurs here with the given plant power level and nature of the malfunction. However, a reactor/plant trip will occur on low pressurizer pressure SI.

D. Incorrect but plausible. The coincidence for the low pressurizer pressure trip is met here as described above however, below P-10 (10% power) this trip is blocked.

Technical Reference(s):	OS1201.06, "Pressurizer Pressure Instrument/Component Failure", rev 16.				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L8027I 04				
Question Source:	Bank #	X	TEB 22067		
	Modified Bank#				
	New				
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis			X	
10 CFR Part 55 Content:	55.41	41.8 / 41.10			

Examination Outline Cross-reference:	Level	RO	
Q47	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(012) (SF7 RPS) REACTOR PROTECTION SYSTEM

(012K1.05) Knowledge of the physical connections and/or cause and effect relationships between the (SF7 RPS) REACTOR PROTECTION SYSTEM and the following systems: ESFAS

Importance Rating	4.2	

Proposed Question:

Plant conditions:

- The plant is initially at 100% power.
- An inadvertent Safety Injection occurs.
- The Train ‘B’ Reactor Trip Breaker will not open.
- All other systems and equipment respond as expected.
- Per E-0, “Reactor Trip or Safety Injection” the crew has reset the Safety Injection signal on both trains.

What is the status of the Safety Injection circuitry?

A. Both trains of SI are reset. SI automatic actuation is blocked on both trains.
B. Only Train ‘A’ SI is reset. Only Train ‘A’ SI automatic actuation is blocked.
C. Both trains of SI are reset. Only Train ‘A’ SI automatic actuation is blocked.
D. Neither trains SI signal is reset. SI automatic actuation is not blocked on either train.

Proposed Answer:	C.	

Explanation:

C. Correct. A train specific P-4 signal allows for blocking of further automatic SI actuations once that train’s SI signal is reset. Failure of Reactor Trip Breaker ‘B’ to open will inhibit the ability to block a further automatic SI on Train ‘B’ once SI is reset. The SI reset is still as it does not have a P-4 interface.

A. Incorrect but plausible. Both trains of SI will reset as the SI switches are dual train and have no P-4 interface. It is plausible that the auto SI signals would be blocked on both trains if the blocking signal were dual train however that signal is train specific with a P-4 signal being generated from the specific train related reactor trip breaker actuation.

B. Incorrect but plausible. It is plausible that both the SI reset and auto SI block would both require a P-4 signal input however both trains of SI will reset as the SI switches are dual train and have no P-4 interface.

D. Incorrect but plausible. It is plausible that the P-4 signal would be dual train with both train related P-4 signals required to be reset.

Technical Reference(s):	Lesson plan, L8056I, “RPS”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8056I 29			
Question Source:	Bank #	X	TEB 34841	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2010 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.2 to 41.9		

Examination Outline Cross-reference:	Level	RO	
Q48	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000058) (APE 58) Loss of DC Power

(000058AK1.06) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 58) LOSS OF DC Power: Loss of remote or automatic operation

	Importance Rating	3.7	

Proposed Question:

Following a loss of 125 VDC control power, a running RCP's breaker will **_____(1)_____** and the breaker **_____(2)_____**.

	(1)	**(2)**
A.	trip open	cannot be closed from the MCB
B.	remain closed	cannot be tripped from the MCB or locally from switchgear
C.	remain closed	can be tripped from the MCB and locally from switchgear
D.	remain closed	can be tripped locally from switchgear but not from the MCB

Proposed Answer:	D.	

Explanation:

D. Correct. (1) Loss of 125 VDC control power will not cause the RCP breaker to open, it will remain closed. (2) The breaker can be opened local from switchgear but not from the MCB (Main Control Board).

A. Incorrect but plausible. (1) is incorrect as explained above. (2) It is true that the breaker can not be closed from the MCB, but the combined statement of parts (1) and (2) is consistent but incorrect.

B. Incorrect but plausible. (1) is correct as explained above. (2) The breaker can be tripped locally at switchgear but not from the MCB.

C. Incorrect but plausible. (1) is correct as explained above. The breaker can be tripped locally from switchgear but not from the MCB.

Technical Reference(s):	Lesson plan, L8012I, “13.8 KV Distribution System”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8012I 18			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.5 / 41.7		

Examination Outline Cross-reference:	Level	RO	
Q49	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(000028) (APE 28) Pressurizer (PZR) Level Control Malfunction

(000028AA2.16) Ability to determine and/or interpret the following as they apply to (APE 28) PRESSURIZER (PZR) Level Control Malfunction: RCS leaks

Importance Rating	3.4	

Proposed Question:

Plant conditions:

- RC-LT-459 has failed low.
- The crew has entered OS1201.07, “Pressurizer Level Instrument Failure”.
- PSO has reduced charging to seals only.
- Seal Injection flow is 8 gpm to each RCP.
- Seal return flow is 1.25 from each RCP.
- Pressurizer level is 65% and rising.
- Letdown has not yet been restored.

After 5 minutes the PSO notes that pressurizer level has increased 1.5%.

Based on this, the PSO should determine that there **_____(1)_____** an RCS leak and that **_____(2)_____**.

A. is — leak is approximately 9 gpm

B. is not — pressurizer level is increasing at the expected rate after manually isolating letdown

C. is — leak is approximately 27 gpm

D. is not — pressurizer level is increasing at the expected rate after letdown automatically isolated

Proposed Answer:	A.	

Explanation:

A. Correct. With LT-459 failed low, letdown is isolated. The difference between seal injection and seal return is, (8 gpm * 4) - (1.25 gpm * 4) = 27 gpm. This is the net inventory addition to the pressurizer if there were no leakage. Pressurizer level is observed to increase by 18 gpm; (1.5% / 5 min)*(60 gal / %) = 18 gal/min. The difference between the inventory addition rate and the observed level increase is 27 - 18 = 9 gpm, which is the leak rate.

B. Incorrect but plausible. (1) This is a possible answer if the student were to incorrectly recall the conversion between pressurizer % and gallons to be 90 gallons/% versus the correct 60 gallons/%. This is plausible as the temperature correction factor for the pressurizer is 90 gallons/°F. If 90 gallons/% is used, (1.5% / 5 min)*(90 gal / %) = 27 gal/min which balances the difference in seal injection and seal return resulting in no leakage with the pressurizer increasing at the expected rate. (2) With RC-LT-459 failed low letdown will isolate automatically by RC-LCV-459 automatically closing. It is plausible that letdown would have been manually isolated here as this is a common action in other AOPs. For example, if a charging pump trips, operators will manually isolate letdown by closing CS-V-145. This is typically done via skill of the operator before entering the AOP therefore based on the conditions given in the stem of the question, the student may incorrectly determine that the operators manually isolated letdown by closing CS-V-145 before entering OS1201.07.

C. Incorrect but plausible. This answer would be obtained if the student incorrectly considered the difference in seal injection and seal return only, (8 gpm * 4) - (1.25 gpm * 4) = 27 gpm. Additionally, it is plausible that pressurizer level is increasing with this leak rate if the student incorrectly believed that the net mass addition were due to the 32 gpm from seal injection and did not consider seal return.

D. Incorrect but plausible. (1) This is a possible answer if the student were to incorrectly recall the conversion between pressurizer % and gallons to be 90 gallons/% versus the correct 60 gallons/%. This is plausible as the temperature correction factor for the pressurizer is 90 gallons/°F. If 90 gallons/% is used, (1.5% / 5 min)*(90 gal / %) = 27 gal/min which balances the difference in seal injection and seal return resulting in no leakage with the pressurizer increasing at the expected rate. (2) With RC-LT-459 failed low letdown will isolate automatically by RC-LCV-459 automatically closing. This is by itself correct however with part (1) incorrect distractor ‘D’ is incorrect.

Technical Reference(s):	Lesson plan L8024, “CVCS”.			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8024I 08			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.10		

Examination Outline Cross-reference:	Level	RO	
Q50	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(000074) (EPE 74; W E06 & E07) Inadequate Core Cooling

(000074EA1.08) Ability to operate and/or monitor the following as they apply to (EPE 74) Inadequate Core Cooling: ECCS

	Importance Rating	4.3	

Proposed Question:

Plant conditions:

- CS-P-2A is danger tagged out of service.
- CS-P-2B cannot be started due to a loss of power to bus E6.
- RCS pressure is 1900 psig and slowly increasing.
- CETCs are 750°F and slowly increasing.
- All other ECCS equipment is functioning as designed.
- ECCS flow cannot be verified in either train.
- The crew has transitioned to FR-C.1, “Response to Inadequate Core Cooling”.

What is the <u>first</u> action that will be taken to establish core cooling?

A. Start one RCP.

B. Depressurize all intact SGs to 250 psig.

C. Open one PORV to depressurize the RCS.

D. Start the PDP. Establish flow through SI-V-138 or SI-V-139.

Proposed Answer:	D.	

Explanation:

D. Correct. The first strategy implemented in FR-C.1 is to restore ECCS injection flow. With the Charging pumps unavailable, the crew will start the positive displacement pump (PDP) per Attachment ‘A’. In the attachment, SI-V-138 or 139 will be opened and the PDP will be started.

A. Incorrect but plausible. Starting RCPs to establish core cooling is a strategy in FR-C.1 however, it is only implemented if CETCs are greater than 1100°F.

B. Incorrect but plausible. Depressurize all intact SGs to 250 psig is a strategy in FR-C.1. It is performed to cause SI Accumulator injection into the RCS as a means of core cooling. It is only performed if the first strategy of establishing ECCS injection flow is not successful. C. Incorrect but plausible. The PORVs will be opened in FR-C.1 as a core cooling strategy only if all RCPs have been started and CETS remain greater than 1100°F.					
Technical Reference(s):	FR-C.1, “Response to Inadequate Core Cooling”, rev 28.				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	SBK LOP L1227I 02, 03				
Question Source:	Bank #	X	TEB 23171		
	Modified Bank#				
	New				
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis			X	
10 CFR Part 55 Content:	55.41	41.5 / 41.7			
RO Justification:	The question relies on the overall strategy and sequence of events in FR-C.1 and is thus an RO question.				

Examination Outline Cross-reference:	Level	RO	
Q51	Tier #	3	
	Group #		

Knowledge and Ability (K/A) Statement:

Equipment Control

(G2.2.13) EQUIPMENT CONTROL: Knowledge of tagging and clearance procedures

Importance Rating	4.1	

Proposed Question:

Per ODI.89, "Tagging Group Instructions", the boundary between PSNH and Seabrook in the 345 kV Switchyard is defined as the RAT disconnects T2003A and B, and __________.

A. the Main Generator Breaker

B. Motor Operated Disconnect G106

C. 345 kV Switchyard Breakers 11 and 12

D. 345 kV Switchyard Breakers 294, 941, 163, 632, 695, and 169

Proposed Answer:	B.	

Explanation:

B. Correct. Per ODI.89, "Tagging Group Instructions", page 23, "the BOUNDARY between jurisdictional responsibility areas for FPLE Seabrook and PSNH shall be defined as FPLE Seabrook's and PSNH's respective side of the 345 kV Boundary Device Switches with designations of G-106, T-2003A, T-2003B…"

A. Incorrect but plausible. It is plausible that the Main Generator Breaker functions as the delineation point between PSNH and Seabrook as this is the last circuit breaker Seabrook's electrical system before the 345 kV switchyard.

C. Incorrect but plausible. It is plausible that 345 kV Switchyard Breakers 11 and 12 function as the delineation point between PSNH and Seabrook as this is the connection point between the output of the main generator and the 345 kV switchyard.

D. Incorrect but plausible. It is plausible that 345 kV Switchyard Breakers 294, 941, 163, 632, 695, and 169 function as the delineation point between PSNH and Seabrook as these are the breakers that connect the three offsite distribution lines to the 345 kV switchyard.

Technical Reference(s):	ODI.89 “Tagging Group Instructions”, rev 34, page 23.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1306I 05			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.10		
K/A Justification: The question directly examines the student’s knowledge of tagging and clearance procedures. This definition of the boundary between PSNH and Seabrook station is related to tagging as it delineates the jurisdictional responsibility for the 345 kV switchyard.				

Examination Outline Cross-reference:	Level	RO	
Q52	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(002) (SF2; SF4P RCS) REACTOR COOLANT SYSTEM

(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

	Importance Rating	3.1	

Proposed Question:

Plant conditions:

- Mode 3.
- Plant startup is in progress.
- RCS temperature is 557°F
- Control Banks are fully inserted
- Shutdown Banks are fully withdrawn
- Two RCPs have just been stopped due to a problem with 13.8 kV Bus 1.
- The remaining two RCPs are running.

What Technical Specification action applies, if any?

A. No action is required, mode change to Mode 2 is not allowed.

B. Within 1 hour place the control rod drive system in a condition incapable of rod withdrawal.

C. Restore the two loops to operable status within 1 hour or be in Hot Shutdown within the next 12 hours.

D. Suspend all operations which could cause reduction in RCS boron concentration. Immediately initiate corrective action to restore required RCS loops.

Proposed Answer:	A.	

Explanation:

A. Correct. Technical Specifications LCO 3.4.1.2 is met with the two remaining RCPs running. However, entry into mode 2 is prohibited as TS 3.4.1.1 requires all RCPs to be in operation in

Mode 1 and 2.

B. Incorrect but plausible. This would be correct per TS 3.4.1.2 Action ‘b’ if only one RCP were running and is thus plausible.

C. Incorrect but plausible. This is similar to TS 3.4.1.2 and is thus plausible. However, with two RCPs in service TS 3.4.1.2 LCO is met.

D. Incorrect but plausible. This is similar to TS 3.4.1.2 allowance for all RCPs being deenergized for 1 hour per 8 hour period provided no operations are permitted that would cause dilution of the RCS. This requirement is denoted by the “*” in TS 3.4.1.2.

Technical Reference(s):	Technical Specifications 3.4.1.1 and 3.4.1.2, rev 149.			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8021I 09			
Question Source:	Bank #	X	TEB 6313	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.10		

Examination Outline Cross-reference:	Level	RO	
Q53	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000025) (APE 25) Loss of Residual Heat Removal System

(000025AA1.03) Ability to operate and/or monitor the following as they apply to (APE 25) LOSS OF RESIDUAL Heat Removal System: RHR

Importance Rating	4.0	

Proposed Question:

Plant conditions:

- Mode 5.
- Reactor vessel level is minus 18 inches.
- The ‘A’ RHR pump is in standby.
- The ‘B’ RHR pump motor current, discharge pressure and flow are fluctuating.
- The crew is implementing OS1213.01, “Loss of RHR During Shutdown Cooling”.

What is the first action the crew will take in OS1213.01?

A. Transition to OS1213.02, “Loss of RHR While at Reduced Inventory or Midloop Conditions”.

B. Place the control switch for ONLY the ‘B’ RHR pump in Pull to lock.

C. Place the control switches for BOTH RHR pumps in Pull to Lock.

D. Start the ‘A’ RHR pump.

Proposed Answer:	C.	

Explanation:

C. Correct. OS1213.01, “Loss of RHR During Shutdown Cooling”, requires that both RHR pumps be placed in PTL when one show signs of cavitation. The standby pump will be started later in the procedure.

A. Incorrect but plausible. OS1213.01, “Loss of RHR During Shutdown Cooling” will direct transition to OS1213.02, “Loss of RHR While at Reduced Inventory or Midloop Conditions” only if reactor vessel level is not above -36 inches.

B. Incorrect but plausible. Initially both RHR pumps are placed in PTL. Later in the procedure, the standby will be started. It is a common misconception that the standby pump will not be placed in

PTL. This is required for equipment protection.

D. Incorrect but plausible. The standby RHR pump will be started in this procedure, but not before its control switch is placed in PTL and conditions are established to start it.

Technical Reference(s):	OS1213.01, “Loss of RHR During Shutdown Cooling”, rev 20.		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1705I 06		
Question Source:	Bank #	X	
	Modified Bank#		
	New		
Question History:	Last NRC Exam	Seabrook 2020 NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	41.5 / 41.7	

Examination Outline Cross-reference:	Level	RO	
Q54	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(076) (SF4S SW) SERVICE WATER SYSTEM

(076K3.11) Knowledge of the effect that a loss or malfunction of the (SF4S SW) SERVICE WATER SYSTEM will have on the following systems or system parameters: EDGS

Importance Rating	4.1	

Proposed Question:

Following a loss of SW cooling to an Emergency Diesel Generator, the high jacket coolant temperature trip will be bypassed following a(n) __________.

A. SI signal only

B. SI or LOP signals only

C. SI, LOP or manual emergency start signal

D. LOP or manual emergency start signal only

Proposed Answer:	C.	

Explanation:

C. Correct. The hi jacket cooling temperature trip of 195°F is bypassed on and SI via the RLA relay and any emergency start via the emergency start relay (ESX) which is actuated by an SI, LOP or emergency start pushbutton.

A. Incorrect but plausible. The generator back up protection 86DB is bypassed on an SI only, thus making this distractor plausible.

B. Incorrect but plausible. An SI and LOP will cause the high jacket coolant temperature trip to be bypassed however, the manual emergency start will also.

D. An LOP and manual emergency start will cause the high jacket coolant temperature trip to be bypassed however, and SI will also.

Technical Reference(s):	Lesson plan, L8019I, “Emergency Diesel Mechanical”.

Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8019I 13, 15, 16, 19			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge		X	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q55	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(014) (SF1 RPI) ROD POSITION INDICATION SYSTEM

(014A4.05) Ability to manually operate and/or monitor the (SF1 RPI) ROD POSITION INDICATION SYSTEM in the control room: RPI accuracy mode selection (W)

Importance Rating	3.1	

Proposed Question:

A DRPI invalid data condition occurs on one control rod.

Per OS1210.07, “RPI Malfunction” the crew will place_____**(1)**_____ and this will result in _____**(2)**_____.

	(1)	(2)
A.	the accuracy mode selector switch in “A ONLY” or “B ONLY” position	half accuracy for the one rod
B.	the accuracy mode selector switch in “A ONLY” or “B ONLY” position	half accuracy for all rods
C.	the accuracy mode selector switch in the “A+B” position	the invalid data clearing
D.	rod control in manual	prevention of unplanned rod motion

Proposed Answer:	B.

Explanation:

B. Correct. (1) Per OS1210.07, “RPI Malfunction” step 2b and c if an invalid data condition occurs, the accuracy mode selector switch will be manipulated per Attachment C. The switch is normally in the A+B position so that DRPI obtains data from both the A and B data cabinets. Placing the switch in the A ONLY or B ONLY position will be done to bypass the failed cabinet. (2) When this is done only data from that respective cabinet will be available to DRPI for ALL control rods resulting in half accuracy for all the rods.

A. Incorrect but plausible. (1) is correct as explained above. (2) it is plausible but incorrect that placing the accuracy mode selector switch in the A ONLY or B ONLY position impacts the rod with bad data only however, as explained above only data from that respective cabinet will be available to DRPI for ALL control rods resulting in half accuracy for all the rods. This is a common misconception.

C. (1) If the student were not able to manually operate the DRPI Accuracy Mode selector switch it may be believed that the A+B position be used to bypass failed data. This is incorrect as explained above. (2) The A+B is the normal position for this switch so that DRPI reads data from both data cabinets. This will not clear the invalid data, although this is plausible.

D. (1) Placing rod control in manual is a possible answer here as it is a required action in OS1210.07 for multiple DRIP indications lost in one rod group. See step 3a RNO. (2) Per the Tech Spec basis this action is carried out here to “prevent unplanned rod motion”. Thus the combination of parts (1) and (2) here is consistent and plausible.

Technical Reference(s):	Lesson plan 8032I, “Digital Rod Position Indication System”. OS1210.07, “RPI Malfunction”, rev 17.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8032I 07			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q56	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM

(003K2.01) Knowledge of electrical power supplies to the following: (SF4P RCP) REACTOR COOLANT PUMP SYSTEM: RCPS

	Importance Rating	3.7	

Proposed Question:

Bus voltage on 13.8 kV Bus 1 begins to steadily decrease.

What component(s) will automatically trip <u>first</u>?

A. ‘C’ CW pump.

B. ‘A’ and ‘B’ RCPs.

C. ‘A’ and ‘C’ RCPs.

D. ‘A’ and ‘B’ CW pumps.

Proposed Answer:	B.	

Explanation:

B. Correct. ‘A’ and ‘B’ RCPs are powered from 13.8KV Bus 1. As bus voltage steadily decreases. The RCPs trip at 70% nominal bus voltage after 1/3 second. There is a common misconception as to the power supply to RCPs and CW pumps. These pumps are power by 13.8kv bus 1 and 2. The ‘A’ and ‘B’ RCPs and the ‘A’ and ‘C’ CW pumps are powered from Bus 1. The ‘C’ and ‘D’ RCPs and ‘B’ CW pump is powered from bus 2. The RCPs UV trip has a 1/3 second time delay. The CW pumps are stripped from the bus after a 1.5 second time delay.

A. Incorrect but plausible as the ‘C’ CW pump is powered from 13.8KV bus 1 and does get stripped from the bus on under voltage.

C. Incorrect but plausible due the common misconception of power source.

D. Incorrect but plausible as CW pumps are stripped from the bus on under voltage and the common misconception as to power source.

Technical Reference(s):	Lesson plan, L8012I, “13.8 kV”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8012I 27			
Question Source:	Bank #	X	TEB 32811	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2013 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q57	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(103) (SF5 CNT) CONTAINMENT SYSTEM

(103K4.04) Knowledge of (SF5 CNT) CONTAINMENT SYSTEM design features and/or interlocks that provide for the following: Personnel access hatch and emergency access hatch

Importance Rating	3.3	

Proposed Question:

Operability of the Containment Personnel Hatch Air Lock is determined by a surveillance performed at the local test panel. The hatch test panel is designed to test operability based on___________.

A. air leakage rate across the door seals

B. differential pressure across the door seals

C. rate of change of differential pressure across the door seals

D. proper limit switch indication for the mechanical door latches

Proposed Answer:	A.	

Explanation:

A. Correct. Per OX1460.01, “Weekly Air Lock Door Seal Air Flow Rate” the inner and outer door seals are tested and verified operable utilizing the air flow rate value across the inner and outer door seals.

B. Incorrect but plausible. Differential pressure across the seals could be an indicator of seal integrity, however that parameter is not used.

C. Incorrect but plausible. Rate of change of differential pressure across the seals could be an indicator of seal integrity, however that parameter is not used.

D. Incorrect but plausible. It is true that the inner and outer doors have a rotating mechanical latch feature however there is no latch position testing feature.

Technical Reference(s):	OX1460.01, “Weekly Air Lock Door Seal Air Flow Rate” rev 12.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8010I 04			
Question Source:	Bank #	X	NUSEC 62598	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2018 NRC Exam, same K/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q58	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000022) (APE 22) Loss of Reactor Coolant Makeup

(000022AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 22) LOSS OF REACTOR Coolant Makeup: Relationship between charging flow and PZR level

Importance Rating	3.6	

Proposed Question:

Plant conditions:

- 100% power.
- Pressurizer level is 62%, stable.
- VCT level is 50%, stable.
- Charging flow is 97 gpm.
- Letdown flow is 80 gpm.

VCT level transmitter, CS-LT-112 fails low.

What are the impacts of this failure?

A. Automatic makeup is disabled. Charging pump suction will swap to the RWST. Charging flow will decrease to maintain Pressurizer level stable.

B. Automatic makeup is disabled. Charging pump swap over to the RWST is disabled. As VCT level decreases, charging pumps will lose suction.

C. Automatic makeup initiates. VCT divert is disabled. Charging system flow increases when VCT fills causing Pressurizer level to increase. The plant will trip on high pressurizer level.

D. Automatic makeup initiates. Makeup will not automatically terminate. CS-LCV-112A and LV-112A letdown divert valves will open to divert letdown to the PDT. Pressurizer level is unaffected.

Proposed Answer:	D.	

Explanation:

D. Correct. CS-LT-112 provides the input for auto makeup and when low level is sensed an auto makeup is initiated. With 112 failed low, the makeup will not terminate. CS-LT-185 will cause LCV-112A and LV-112A letdown divert valves to open based on actual level which will divert to the PDT (Primary Drain Tank).

A. Incorrect but plausible. Auto makeup is only actuated from LT-112, thus it is not disabled, it initiates the makeup. Charging pump swap to the RWST requires both LT-112 and 185 <5% VCT level. Charging flow is unaffected.

B. Incorrect but plausible. Auto makeup is only actuated from LT-112, thus it is not disabled, it initiates the makeup. Charging pump swap to the RWST requires both LT-112 and 185 <5% VCT level. Charging flow is unaffected and the charging pumps will not lose suction.

C. Incorrect but plausible. Auto makeup is initiated but VCT divert is not disabled. If VCT level were to increase from the failure it is plausible that charging flow would increase to compensate and lead to a pressurizer level high trip. C

Technical Reference(s):	Lesson plan L8024I, “CVCS”.
Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8024I 03, 05, 07			
Question Source:	Bank #	X	NUSEC 44760	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2009 NRC Exam, same K/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5 / 41.7		

Examination Outline Cross-reference:	Level	RO	
Q59	Tier #	4	
	Group #		

Knowledge and Ability (K/A) Statement:

Reactor Theory 192006

(192006K1.06) FISSION PRODUCT POISONS: Describe the following processes and state their effect on reactor operations: -- transient xenon

	Importance Rating	3.4	

Proposed Question:

The plant was shut down to repair a Main Feedwater pump at middle of life (MOL).

7 days following the shutdown a reactor startup using control rods is performed.

100% power is reached after 16 hours.

CBD is at 228 steps.

Without rod motion, over the next 24 hours operators will ___________ to account for changing transient xenon concentration.

A. borate only

B. borate, then dilute

C. dilute only

D. dilute, then borate

Proposed Answer:	C.	

Explanation:

C. Correct. As power reaches 100% xenon concentration will be increasing. Xenon concentration will reach an equilibrium value over approximately 40-50 hours. Thus for 24 hours, operators will dilute to offset this increase in xenon concentration.

A. Incorrect but plausible. If the student were not aware of the dynamics of xenon concentration following a power increase, this would be a plausible answer as operators will at times borate to account for changing xenon concertation.

B. Incorrect but plausible. If the student were not aware of the dynamics of xenon concentration following a power increase, this would be a plausible answer as operators will at times borate followed by dilution to account for changing xenon concertation.

D. Incorrect but plausible. If the student were not aware of the dynamics of xenon concentration following a power increase, this would be a plausible answer as operators will at times dilute

followed by boration to account for changing xenon concertation.				
Technical Reference(s):	192006 - Fission Product Poisons, INPO Training Material.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	192006 Obj. 1			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.1		

Examination Outline Cross-reference:	Level	RO	
Q60	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000065) (APE 65) Loss of Instrument Air

(000065AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 65) LOSS OF Instrument Air: Failure modes of air-operated equipment

Importance Rating	3.7	

Proposed Question:

On a loss of Instrument Air, CS-FCV-121, "Charging Flow Control Valve" will fail **_____(1)_____**, and CS-HCV-182, "RCP Seal Flow Control Valve" will fail **_____(2)_____**.

	(1)	(2)
A.	open	open
B.	open	closed
C.	closed	open
D.	closed	closed

Proposed Answer:	A.	

Explanation:

A. Correct. Per ON1242.01, "Loss of Instrument Air", CS-FCV-121 and HCV 182 both fail open.

B, C and D. Incorrect but plausible. Each of these distractors are plausible but incorrect combinations of the possible failure positions of each of these valves.

Technical Reference(s):	ON1242.01, "Loss of Instrument Air", rev 17.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L8024I 03			
Question Source:	Bank #			
	Modified Bank#			

	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.5 / 41.7		

Examination Outline Cross-reference:	Level	RO	
Q61	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM

(022A2.07) 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: ESFAS actuation

Importance Rating	4.0	

Proposed Question:

Plant conditions:

- Small break LOCA coincident with an LOP
- Containment pressure reached 16 psig before lowering.

The Containment Structure Cooling Units CAH-FN-1A-F are stopped because of the **_____(1)_____** signal and the operator must reset **_____(2)_____** to restart them.

	(1)	(2)
A.	SI	RMO
B.	SI	SI
C.	P	RMO
D.	P	P

Proposed Answer:	A.

Explanation:

A. Correct. (1) On the LOP and coincident SI, the CAH fans will trip and be blocked from auto starting by the SI signal. (2) The operator must reset RMO to manually restart the fans.

B. Incorrect but plausible. (1) is correct as explained above. (2) With the SI signal responsible for preventing the EPS from automatically restarting the fans it is plausible that the SI need to be reset before manually starting the fans. However, RMO needs to be reset not the SI.

C. Incorrect but plausible. (1) The CAH fans will trip on a ‘P’ signal due to the loss of PCCW to containment making this plausible. However, for the given plant conditions a P signal has not actuated as containment pressure has not exceeded 18 psig. (2) is correct as explained above.

D. Incorrect but plausible. (1) The CAH fans will trip on a ‘P’ signal due to the loss of PCCW to containment making this plausible. However, for the given plant conditions a P signal has not

actuated as containment pressure has not exceeded 18 psig. (2) It is plausible that if a P signal had actuated causing the fans to trip that it would need to be reset to reestablish PCCW flow to containment.				
Technical Reference(s):	Lesson plan L8038I, “Containment Building HVAC”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8038I 04			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q62	Tier #	1	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(000037) (APE 37) Steam Generator Tube Leak

(000037AK3.04) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 37) STEAM Generator Tube Leak: Use of “feed and bleed” process

Importance Rating	3.2	

Proposed Question:

Plant conditions:

- SGTR has occurred.
- The crew has transitioned from E-3, "Steam Generator Tube Rupture" to ES-3.1 "Post SGTR-Cooldown Using Backfill”.
- Step 7 of ES-3.1 has the operator check SG NR level >25%, and refill to 85% if necessary.

Why is the ruptured SG refilled in batches?

A. Allows for a rapid depressurization of the ruptured SG.

B. Provides adequate secondary heat sink for RCS decay heat removal.

C. Provides better control over ruptured SG pressure when steam dump is initiated.

D. Allows for more effective heat removal from SG upper internal components and maintains the ruptured tubes covered.

Proposed Answer:	D.	

Explanation:

D. Correct. Per ES-3.1 Background document the ruptured SG is refilled in batches to maintain the ruptured SG tubes covered and to aid in cooling the internals of the ruptured SG.

A. Incorrect but plausible. The intent of maintaining water level above the ruptured SG tubes is to prevent rapid depressurization of the ruptured SG, not to allow for a rapid depressurization.

B. Incorrect but plausible. SG level control criteria is normally intended to ensure a secondary heat sink is available. However, in the case of “feed and bleed” in response to a steam generator tube leak or rupture, the ruptured SG is being refilled to remove heat from the SG and to maintain the U tubes covered.

C. Incorrect but plausible. It is plausible that adequate level in the ruptured SG will allow for better control when initiating steam dumps as mass will be removed from the SG during this process.

Technical Reference(s):	ES-3.1, "Post-SGTR Cooldown Using Backfill", rev 30. ES-3.1 Background, rev 3.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1206I 03			
Question Source:	Bank #	X	TEB 30516	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.5 / 41.10		
K/A Justification:	The SGTR EOP is being used here as equivalent to the SGTL AOP. Both procedures utilize the "feed and bleed" process however; documentation for the basis of the process only exists for the SGTR EOP.			

Examination Outline Cross-reference:	Level	RO	
Q63	Tier #	3	
	Group #		

Knowledge and Ability (K/A) Statement:

Conduct of Operations

(G2.1.38) CONDUCT OF OPERATIONS: Knowledge of the station's requirements for verbal communications when implementing procedures

Importance Rating	3.7	

Proposed Question:

In accordance with OP 9.2, "Transient Response Procedures User's Guide", when reading notes and cautions during transients the protocol for three way communication is:

A. Three way communication is always required for notes and cautions.

B. Three way communication is required for cautions, "understood" is sufficient for notes.

C. Three way communication should be used if the note or caution gives direction, otherwise "understood" is sufficient.

D. Three way communication is required for notes and cautions when information about potential hazards to personnel or equipment is given.

Proposed Answer:	C.	

Explanation:

C. Correct. From OP 9.2, "Transient Response Procedures User's Guide", 4.11 "three way communication is not required when reading Notes and Cautions during transients. A simple response, such as "understood" is sufficient. If direction is given as part of a Note or Caution, three way communication should be used for the direction."

A. Incorrect but plausible. Notes and cautions contain administrative or advisory information, and information about potential hazards to personnel or equipment. For this reason, it is plausible that three way communication is required.

B. Incorrect but plausible. Notes typically contain administrative information thus, "understood" is plausibly sufficient for communications. Cautions contain information about potential hazards to personnel or equipment thus, three way communication is plausibly required.

D. Incorrect but plausible. Cautions contain information about potential hazards to personnel or equipment thus, three way communication is plausibly required. Notes typically contain administrative or advisory information.

Technical Reference(s):	OP 9.2, “Transient Response Procedures User’s Guide”, rev 20.			
Proposed references to be provided to applicants during examination:				none
Learning Objective:	SBK LOP L1305I 12			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge		X	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.10		

Examination Outline Cross-reference:	Level	RO	
Q64	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000038) (EPE 38) Steam Generator Tube Rupture

(000038EA2.07) Ability to determine and/or interpret the following as they apply to (EPE 38) STEAM GENERATOR Tube Rupture: Plant conditions from survey of control room indications

	Importance Rating	4.0	

Proposed Question:

Plant conditions:

- SG tube rupture occurred at 100% power.
- E-3, “Steam Generator Tube Rupture” is being implemented.
- ‘A’ SG has been identified as the ruptured SG.
- The crew is performing step 17, “Depressurize RCS to Minimize Break Flow and Refill PZR”.
- Both PZR spray valves are fully open, per step 17b.
- The PSO notes:
  - ‘A’ SG pressure is 1125 psig and stable.
  - RCS pressure is 1400 psig and decreasing.
  - RCS subcooling is 110°F and decreasing.
  - PZR level is 38% and increasing.
  - Containment pressure is 0 psig and stable.

What action is required per E-3? **(reference provided)**

A. Stop RCS depressurization. Fully close both PZR spray valves.

B. Stop RCS depressurization. Maintain RCS pressure stable at present value using PZR spray.

C. Continue RCS depressurization until RCS pressure is less than “A” SG pressure. Then fully close both PZR spray valves.

D. Continue RCS depressurization until RCS pressure is less than “A” SG pressure. Then maintain RCS pressure stable using PZR spray.

Proposed Answer:	A.	

Explanation:					
A. Correct. Conditions given meet the second bullet in E-3 step 17b. RCS pressure of 1400 psig is within 300 psig of the ruptured SG pressure of 1125 psig, and PZR level is greater than 37%. The PSO should proceed to step 17c. to fully close the spray valves. B. Incorrect but plausible. As explained above, conditions are met to stop the depressurization per step 17b. When the depressurization is stopped the PSO will fully close the spray valves per step 17c., not maintain pressure stable. C. Incorrect but plausible. If the student were to incorrectly apply the criteria for stopping the depressurization in step 17b., this could be a possible answer. D. Incorrect but plausible. If the student were to incorrectly apply the criteria for stopping the depressurization in step 17b., this could be a possible answer. When the depressurization is stopped the PSO will fully close the spray valves per step 17c., not maintain pressure stable.					
Technical Reference(s):	E-3, “Steam Generator Tube Rupture”, rev 48.				
Proposed references to be provided to applicants during examination:				E-3 step 17	
Learning Objective:	SBK LOP L1205I 02				
Question Source:	Bank #				
	Modified Bank#				
	New	X			
Question History:	Last NRC Exam	N/A			
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis			X	
10 CFR Part 55 Content:	55.41	41.10			

Examination Outline Cross-reference:	Level	RO	
Q65	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000029) (EPE 29) Anticipated Transient Without Scram

(000029EK3.06) Knowledge of the reasons for the following responses and/or actions as they apply to (EPE 29) ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS): Verifying a main turbine trip; methods

Importance Rating	4.0	

Proposed Question:

In FR-S.1, "Response to Nuclear Power Generation/ATWS" the basis for verifying a turbine trip is to _____**(1)**_____ and a successful automatic turbine trip is indicated by all _____**(2)**_____ closed.

	(1)	**(2)**
A.	cause auto control rod insertion at max speed	MSIVs
B.	cause auto control rod insertion at max speed	stop or control valves
C.	prevent an uncontrolled cooldown and preserve SG inventory	MSIVs
D.	prevent an uncontrolled cooldown and preserve SG inventory	stop or control valves

Proposed Answer:	D.

Explanation:

D. Correct. (1) Per the Background Document for FR-S.1, "the turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require. For an ATWS event where a loss of normal feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain SG inventory." (2) FR-S.1 verifies that the Turbine is tripped by checking that either all stop or control valves are closed.

A. Incorrect but plausible. (1) It is plausible that the basis of tripping the turbine is to cause auto control rod insertion at max speed by maximizing the Tavg/Tref differential as this does occur.

However, the basis per the background document is “the turbine is tripped to prevent an uncontrolled cooldown of the RCS due to steam flow that the turbine would require. For an ATWS event where a loss of normal feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain SG inventory.”(2) It is plausible that the MSIVs are required to be closed for a turbine trip as the MSIVs will be closed if the turbine cannot be tripped.

B. Incorrect but plausible. (1) is incorrect as explained above. (2) is correct as explained above.

C. Incorrect but plausible. (1) is correct as explained above. (2) is incorrect as explained above.

Technical Reference(s):	FR-S.1, “Response to Nuclear Power Generation/ATWS”, rev 32. Background Document for FR-S.1, rev 3.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1200I 01, 02			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.5 / 41.10		

Examination Outline Cross-reference:	Level	RO	
Q66	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(050) (SF9 CRV*) CONTROL ROOM VENTILATION

(050A1.04) Ability to predict and/or monitor changes in parameters associated with operation of the (SF9 CRV) CONTROL ROOM VENTILATION, including: Control room pressure

Importance Rating	3.0	

Proposed Question:

Plant conditions:

- CBA-DP-28, "Control Room Exhaust Modulate Damper/Controller" is in auto with a setpoint of 0.3 Inches Water Column (INWC).
- Outside air pressure and Cable Spreading Room pressure are both stable at 0.1 INWC.
- Control Room pressure is currently stable at 0.4 INWC.
- Cable Spreading Room pressure suddenly rises to 0.3 INWC.

What is the response of the CBA-DP-28?

A. Controller selects outside pressure for its input signal and modulates CBA-DP-28, Control Room Modulate Exhaust Damper in the close direction to raise Control Room pressure.

B. Controller selects Cable Spreading Room for its input signal and modulates CBA-DP-28, Control Room Modulate Exhaust Damper in the open direction to lower Control Room pressure.

C. Controller selects Cable Spreading Room for its input signal and modulates CBA-DP-28, Control Room Modulate Exhaust Damper in the closed direction to raise Control Room pressure.

D. Controller selects outside pressure for its input signal; Control Room pressure is currently at the desired value above outside pressure, therefore the Controller will maintain CBA-DP-28, Control Room Modulate Exhaust Damper in its current position.

Proposed Answer:	C.	

Explanation:

C. Correct. DP-28 controller compares outside pressure and Cable Spreading Room pressure with the Control Room's pressure. The controller automatically adjusts DP-28 to maintain 0.3 INWC (normal setpoint) above outside air pressure or cable spreading room pressure, whichever is higher. For the conditions given when cable spreading room pressure increases this will be selected as input to the controller which will throttle DP-28 closed to raise control room pressure to 0.3 INWC above cable spreading room pressure.

A. Incorrect but plausible. With the control room pressure control scheme maintaining control room pressure positive to prevent infiltration of radiological contamination, it is plausible that the pressure controller would only use outside pressure as the comparison to control room pressure.

B. Incorrect but plausible. DP-28 controller compares outside pressure and Cable Spreading Room pressure with the Control Room's pressure. The controller automatically adjusts DP-28 to maintain 0.3 INWC (normal setpoint) above outside air pressure or cable spreading room pressure, whichever is higher. For the conditions given when cable spreading room pressure increases this will be selected as input to the controller which will throttle DP-28 closed to raise control room pressure to 0.3 INWC above cable spreading room pressure however, it is plausible that the controller open DP-28 as cable spreading room pressure has increased.

D. Incorrect but plausible. With control room pressure stable at 0.4 INWC it is plausible that DP-28 maintain its current position.

Technical Reference(s):	Lesson plan, L8039I, "Control Building Air Handling".			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8039I 03			
Question Source:	Bank #	X	TEB 26077	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q67	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM

(039K1.03) Knowledge of the physical connections and/or cause and effect relationships between the Main and Reheat Steam System and the following systems: IAS

Importance Rating	3.1	

Proposed Question:

The BOP operator places the MCB jog switch for MS-PV-3003,"'C' ASDV" in the OPEN position.

What occurs because of this action?

A. Control air is directed to the underside of the operating piston by the positioner.

B. Nitrogen is directed to the underside of the operating piston by the positioner.

C. Backup air is directed to the underside of the operating piston by solenoids that are energized when the switch is placed in OPEN.

D. Nitrogen is directed to the underside of the operating piston by solenoids that are energized when the switch is placed in OPEN.

Proposed Answer:	D.	

Explanation:

D. Correct. When placing the jog switch to open for the 'C' ASDV, control air is prevented from reaching the pneumatic actuator due to solenoid valves #1 & #2 being closed. Nitrogen bypasses the valve positioner by flowing thru jog switch solenoid valves #3 & #5 to the ASDV pneumatic actuator thereby fully opening the ASDV.

A. Incorrect but plausible. This distractor would be correct if the positioner was in use via the M/A station. The jog switch does not use the positioner.

B. Incorrect but plausible. This distractor would be correct if the positioner was in use via the M/A station and control air were unavailable. The jog switch does not use the positioner.

C. Incorrect but plausible. Backup air is only available on the 'A' and 'B' ASDVs. It is not available to the 'C' and 'D'. This would be the correct answer if the ASDV in question were the 'A' or 'B'.

Technical Reference(s):	Lesson plan L8041I, “Main Steam”.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8041I 06			
Question Source:	Bank #	X	TEB 20201	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		
K/A Justification: The question examines the student’s knowledge of power supplies to the ASDVs which are safety related valves in the Main Steam system. The ASDVs are air operated valves using either instrument air or nitrogen for motive force. The flow path for the air or nitrogen is controlled via electrical solenoid valves that are operated by either a jog switch or M/A control station. While the question is asking about AOVs, it is the electrical power to the solenoids that is the required piece of knowledge and thus meets the intent of the K/A statement.				

Examination Outline Cross-reference:	Level	RO	
Q68	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM

(005K5.10) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM: RHRS suction vortexing during reduced RCS inventory

	Importance Rating	4.2	

Proposed Question:

Plant conditions:

- OS1000.12, “Operation with RCS at Reduced Inventory/Midloop Conditions” is in effect.
- Reactor vessel level is at minus 74 inches and stable.
- ‘A’ RHR pump is in operation with flow at 3500 gpm.
- ‘B’ RHR pump is in standby.
- ‘A’ RHR pump subsequently trips and cannot be restarted.
- The crew enters OS1213.02, “Loss of RHR While Operating at Reduced Inventory/Midloop Conditions”.

Which action should be taken for ‘B’ RHR pump?

A. Should not be started until vessel level is raised above minus 71 inches using charging and letdown.

B. Should not be started until an RWST gravity fill has restored vessel level above minus 36 inches.

C. Should not be started. Air entrainment could result in a loss of both RHR pumps.

D. Should be started. Vessel level is above the start criteria of minus 83.5 inches.

Proposed Answer:	D.	

Explanation:

D. Correct. Per OS OS1213.02, “Loss of RHR While Operating at Reduced Inventory/Midloop Conditions” step 2, with no RHR pumps running a standby pump will be started if vessel level is above 83.5 inches. This level criteria is to ensure that the pump once started will not suffer from cavitation.

A. Incorrect but plausible. Charging and letdown flow adjustments are used to adjust vessel level when level is above –83.5 inches in accordance with OS1213.02. Vessel level does not have to be raised above the top of the hot leg nozzle before the RHR pump can be started.

C. Incorrect but plausible. It is generally prudent not to start a standby pump at mid loop conditions, however the vessel level of the question stem is above mid loop conditions.

B. Incorrect but plausible. Increasing vessel level to greater than –36 inches brings level above reduced inventory conditions but this is not required for starting the standby RHR pump. RWST gravity fill is only utilized if no emergency bus has power.

Technical Reference(s):	OS1213.02, "Loss of RHR While Operating at Reduced Inventory/Midloop Conditions", rev 18.			
Proposed references to be provided to applicants during examination:			None	
Learning Objective:	SBK LOP L1705I 06			
Question Source:	Bank #	X	TEB 29949	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2007 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q69	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM

(026A3.02) Ability to monitor automatic features of the (SF5 CSS) CONTAINMENT SPRAY SYSTEM, including: Verification that cooling water is supplied to the containment spray heat exchanger

Importance Rating	3.5	

Proposed Question:

Following a Design Basis Accident, a(n) __________ signal will cause the PCCW isolation valves to the CBS heat exchangers, CC-V-137 and 226 to open.

A. ‘CSAS’ (Containment Spray Actuation Signal)

B. ‘P’

C. ‘S’

D. ‘T’

Proposed Answer:	B.	

Explanation:

B. Correct. A ‘P’ signal is actuated at 18 psig (2/4) containment pressure. The ‘P’ signal will directly cause the CBS heat exchanger PCCW isolation valves to open.

A. Incorrect but plausible. A ‘CSAS’ (Containment Spray Actuation Signal) is also actuated at 18 psig (2/4) containment pressure. The CSAS causes the CBS pumps to start. It does not directly cause the CBS heat exchanger PCCW isolation valves to open.

C. Incorrect but plausible. An ‘S’ signal is expected to be actuated following a Design Basis Accident. It will cause numerous systems and components to realign. It will not cause the CBS heat exchanger PCCW isolation valves to open.

D. Incorrect but plausible. A ‘T’ signal is expected to be actuated following a Design Basis Accident. It will cause numerous systems and components to realign e.g., the RHR heat exchanger PCCW isolation valves. It will not cause the CBS heat exchanger PCCW isolation valves to open.

Technical Reference(s):	Logic Print for PCCW Isolation Valves, 1-NHY-503271, rev 9

Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8035I 07			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge		X	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q70	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM

(004K5.30) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM (CFR: / 45.3): Relationship between temperature and pressure in CVCS components during solid plant operation

	Importance Rating	4.0	

Proposed Question:

Plant conditions:

- Mode 5.
- 'A' train RHR is in shut down cooling mode.
- RCS temperature is 166°F.
- The pressurizer is water solid.
- Letdown Pressure Control Valve CS-PCV-131 is in automatic.
- RCS pressure is being maintained at 275 psig.
- Air supply line to RH-HCV-606, 'A' RHR temperature control valve, is broken off.
- Assume no operator action.

In response to the failure, PCV-131 will _____**(1)**_____ to maintain RCS pressure constant and letdown flow through RH-HCV-128 will _____**(2)**_____.

	(1)	(2)
A.	open	increase
B.	close	decrease
C.	open	remain the same
D.	close	remain the same

Proposed Answer:	B.	

Explanation:

B. Correct. Per OS1000.04, Plant Cooldown from Hot Standby to Cold Shutdown, with the plant in a solid condition RHR letdown flow is through CS-HCV-128 to CS-PCV-131. CS-PCV-131 is in automatic and the setpoint is adjusted to the desired pressure for the shutdown condition. When the air supply line is broken off, RH-HCV-606 will fail open. This will cause a cool down of the RCS. When solid, the RCS cool down causes the contraction of water decreasing RCS pressure. To maintain setpoint CS-PCV-131 will close to offset the loss of volume in the RCS and flow through CS-HCV-128 will decrease.

A. Incorrect but is plausible. If the student were not aware of the failure mode of RH-HCV-606 it would be plausible for them to determine that the valve fails closed causing an increase in RCS temperature and subsequently pressure. An increase in pressure will cause PCV-131 to open and letdown flow to increase through HCV-128.

C. Incorrect but plausible. If the student were not aware of the failure mode of RH-HCV-606 it would be plausible for them to determine that the valve fails closed causing an increase in RCS temperature and subsequently pressure. An increase in pressure will cause PCV-131 to open and with RH-HCV-128 being a manual valve it is plausible that letdown flow does not change.

D. Incorrect but plausible. RH-HCV-606 fails open. This will cause a cool down of the RCS. While solid if the RCS cools down the contraction of water will decrease RCS pressure. To maintain setpoint CS-PCV-131 will close to offset the loss of volume in the RCS and flow through CS-HCV-128 will decrease. Although it is plausible that with RH-HCV-128 being a manual valve it is plausible that letdown flow does not change.

Technical Reference(s):	Lesson plan L8024I, “CVCS”.			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8024I 02			
Question Source:	Bank #	X	TEB 23140	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2013 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5		

Examination Outline Cross-reference:	Level	RO	
Q71	Tier #	3	
	Group #		

Knowledge and Ability (K/A) Statement:

Equipment Control

(G2.2.22) EQUIPMENT CONTROL: Knowledge of limiting conditions for operation and safety limits

Importance Rating	4.0	

Proposed Question:

In Mode 3 which of the following RCS overpressure systems are required to be operable by Technical Specifications?

A. Both Pressurizer PORVs.

B. Both RHR suction relief valves.

C. One Pressurizer PORV and the RCS head vent.

D. One Pressurizer PORV and one RHR suction relief valve.

Proposed Answer:	A.	

Explanation:

A. Correct. Per Technical Specification 3.4.4, both PORVs are required to be operable in Mode 3.

B. Incorrect but plausible. Both RHR suction relief valves is a condition to satisfy Technical Specifications however, this is applicable in Mode 4.

C. Incorrect but plausible. The RCS head vent is used for overpressure protection however, it is only applicable in Modes 5 and 6.

D. Incorrect but plausible. The combination of 1 PORV and 1 RHR suction relief valve is a condition to satisfy Technical Specifications however, this is only applicable in Mode 4.

Technical Reference(s):	Technical Specifications 3.4.4, rev 146.

Proposed references to be provided to applicants during examination:	None

Learning Objective:	SBK LOP L1306I 14			
Question Source:	Bank #	X		

	Modified Bank#				
	New				
Question History:	Last NRC Exam	Seabrook 2007 NRC Exam			
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	41.5			

Examination Outline Cross-reference:	Level	RO	
Q72	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(078) (SF8 IAS) INSTRUMENT AIR SYSTEM

(G2.1.1) CONDUCT OF OPERATIONS: Knowledge of conduct of operations requirements

	Importance Rating	3.8	

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- You are the BOP control room operator.
- At 2335 a loss of main plant computer occurs immediately followed by a decreasing instrument air pressure trend.
- Your crew has entered ON1242.01, “Loss of Instrument Air” AND ON1251.01, “Loss of Plant Computer.
- Per ON1251.01, “Loss of Plant Computer” your crew has been recording local area temperature readings on Attachment C, the local temperature reading log sheet.
- Per ON1242.01, “Loss of Instrument Air”, Step 7, “Initiate Area Temperature Monitoring”, at 2345 the Unit Supervisor directed you to implement Attachment G, “Area Temperature Monitoring Actions”**.**
- The time is now 0200.

Utilizing ON1242.01, Attachment G, “Area Temperature Monitoring Actions”**,** which of the following correctly describes the required action or actions that must be taken?

(references provided)

A. Go to ON1223.01, “Loss of Control Room Ventilation or Air Conditioning” ONLY.

B. Go to the VAS alarm hard copy for B6543, “Battery Room A Temp High” ONLY.

C. Go to ON1223.01, “Loss of Control Room Ventilation or Air Conditioning” AND go to the VAS alarm hard copy for B6543, “Battery Room A Temp High”.

D. Continue to monitor area temperature VAS alarms for the Control Room , Essential Switchgear (Train A and B), and Vital Battery Rooms A, B, C, and D. If any of the area temperature alarms actuates then perform the actions of the associated VAS Alarm Response Procedure.

Proposed Answer:	C.	

Explanation:
C. Correct. Per ON1242.01, Attachment G, “Area Temperature Monitoring Actions , if the MPCS is not available then the action is to perform ON1251.01, Attachment D, “Area Temperature Monitoring Actions”, which in turn directs implementing attachment C, the area temperature log sheet. At the 0200 the data on the log sheet indicates that the Control Room area and Battery Room “A” temperatures exceed the prescribed limits. In this case Attachment D directs the operator to go to ON1223.01, “Loss of Control Room Ventilation or Air Conditioning” AND go to the VAS alarm hard copy for B6543, “Battery Room A Temp High”. A. Incorrect but plausible. It is true that ON1251.01, “Loss of Plant Computer, Attachment D would direct the operator to go to ON1223.01, “Loss of Control Room Ventilation or Air Conditioning”, however the operator should also go to the VAS alarm hard copy for B6543, “Battery Room A Temp High”. B. Incorrect but plausible. It is true that ON1251.01, “Loss of Plant Computer, Attachment D would direct the operator to go to the VAS alarm hard copy for B6543, “Battery Room A Temp High”, however the operator should also go to ON1223.01, “Loss of Control Room Ventilation or Air Conditioning”. D. Incorrect but plausible. This distractor describes the action that would be taken per ON1242.01, Attachment G, “Area Temperature Monitoring Actions if the MPCS were available.

Technical Reference(s):	ON1242.01, Attachment G, “Area Temperature Monitoring Actions”, rev 17. ON1251.01, “Loss of Plant Computer, Attachment D, “Area Temperature Monitoring Actions”, rev 25.

Proposed references to be provided to applicants during examination:	ON1242.01, Attachment G, “Area Temperature Monitoring Actions” ON1251.01, “Loss of Plant Computer, Attachment C, Local area temperature reading log sheet. (With data filled in) ON1251.01, “Loss of Plant Computer, Attachment D, “Area Temperature Monitoring Actions”

Learning Objective:	SBK LOP L1194I 02			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2018 NRC Exam		
Question Cognitive	Memory or Fundamental			

Level:	Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	41.10	

Examination Outline Cross-reference:	Level	RO	
Q73	Tier #	2	
	Group #	2	

Knowledge and Ability (K/A) Statement:

(071) (SF9 WGS) WASTE GAS DISPOSAL SYSTEM

(071K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF9 WGS) WASTE GAS DISPOSAL SYSTEM: Waste gas discharge release valve

Importance Rating	3.2	

Proposed Question:

Plant conditions:

- 100% power.
- The Waste Gas/Vent Gas (WG/VG) system is operating normally.
- Both PAH-FN-8A and 8B, “PAB Cleanup Filter Exhaust Fans 8 A/B” are inadvertently shut down.

How does the WG/VG system automatically respond, if at all?

A. VG-V-57, “PAB Hydrogenated Vent Header Isolation” closes.

B. The WG/VG system operates in recirculation. No automatic actions occur.

C. VG-V-50, “Hydrogenated Vent Header Pressure Relief” auto opens to discharge directly to the Plant Vent.

D. Discharge header pressure increases causing the WG Compressor to trip. The WG/VG system automatically shuts down.

Proposed Answer:	A.	

Explanation:

A. Correct. VG-V-57 requires either PAH-FN-8A or B to be running to be open. If both fans are secured, VG-V-57 will automatically close.

B. Incorrect but plausible. The Waste Gas/Vent Gas system originally was designed and operated as a recirculation system thus, this distractor is plausible.

C. Incorrect but plausible. VG-V-50 will auto open at 12 psig waste gas system pressure. It is not interlocked with the PAH-FN-8A and B directly though.

D. Incorrect but plausible. The waste gas compressors are not currently used with the system operating as a 'once through' flow path as opposed to the original design as a recirculation system. The compressors are bypassed normally. When the system was operated as a recirculation system the WG compressor would auto trip on high system pressure making this distractor plausible				
Technical Reference(s):	Lesson plan L8064I, "Waste Gas and Waste Liquid".			
Proposed references to be provided to applicants during examination:			None	
Learning Objective:	SBK LOP L8064I 02			
Question Source:	Bank #	X	TEB 20388	
	Modified Bank#			
	New			
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		X	
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q74	Tier #	2	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM

(191008K1.08) BREAKERS RELAYS AND DISCONNECTS: Effects of closing breakers with the current out of phase, different frequencies, high-voltage differential, low current, or too much load

	Importance Rating	3.5	

Proposed Question:

The operating procedure for transferring a Vital 125VDC bus to its Alternate Battery Supply utilizes a Kirk Key interlock device.

This interlock feature is intended to prevent **_____(1)_____** in the same train from being placed in parallel which could cause **_____(2)_____**.

	(1)	**(2)**
A.	battery chargers	low voltage
B.	battery chargers	excessive current
C.	battery banks	low voltage
D.	battery banks	excessive current

Proposed Answer:	D.	

Explanation:

D. Correct. The normal battery supply breaker is opened before the alternate battery supply breaker is closed to prevent excessive current conditions associated with parallel configuration of both battery banks on the same train.

B. Incorrect but plausible. (1) It is plausible that the intent of the Kirk Key interlock is to prevent the battery chargers from being placed in parallel however, it is the batteries that is the concern. (2) It is also plausible that paralleling battery chargers at different voltages would cause excessive current.

A. Incorrect but plausible. (1) It is plausible that the intent of the Kirk Key interlock is to prevent the battery chargers from being placed in parallel however, it is the batteries that is the concern. It is also plausible that paralleling battery chargers at different voltages would cause system voltage to degrade.

C. Incorrect but plausible. (1) is correct as explained above. (2) It is also plausible that paralleling batteries at different voltages would cause system voltage to degrade.

Technical Reference(s):	OS1048.13, "Vital Bus 11A Operation", Section 4.1, "Transferring 125 VDC Bus 11A To Its Alternate Battery Supply", rev 21. Lesson plan, L8017I, "125 VDC".			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L8017 04			
Question Source:	Bank #			
	Modified Bank#			
	New	X		
Question History:	Last NRC Exam	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	41.7		

Examination Outline Cross-reference:	Level	RO	
Q75	Tier #	1	
	Group #	1	

Knowledge and Ability (K/A) Statement:

(000009) (EPE 9) Small Break LOCA

(000009EA1.17) Ability to operate and/or monitor the following as they apply to (EPE 9) SMALL-Break LOCA: PRT/quench tank

	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- A loss of offsite power has occurred
- Due to a small break LOCA the crew is performing ES-1.2, “Post LOCA Cooldown and Depressurization”.
- Normal charging has been established.
- Letdown is not in service.
- The crew is preparing to depressurize the RCS to minimize subcooling using a PORV.
- The PRT rupture disk is intact.
- As the crew opens the PORV it is found to be ineffective at decreasing RCS pressure.

What is the reason that the PORV is ineffective and what actions should be taken per ES-1.2?

A. With the PRT rupture disk intact, the PORV loses effectiveness as PZR pressure approaches PRT pressure. Use Aux Spray to perform the depressurization.

B. With the PRT rupture disk intact, the PORV loses effectiveness as PZR pressure approaches PRT pressure. Use normal spray to perform the depressurization.

C. The PORV loses effectiveness when the RCPs are shut down because of the reduced vessel d/p. Use Aux Spray to perform the depressurization.

D. The PORV loses effectiveness when the RCPs are shut down because of the reduced vessel d/p. Use normal spray to perform the depressurization.

Proposed Answer:	A.	

Explanation:

A. Correct. From the background document for ES-1.2 “a PRZR PORV may not be effective for

RCS depressurization at low temperature and pressure conditions that could exist in the pressurizer when the PRT rupture disk is still intact, and RCS pressure approaches PRT pressure." This is the reason that the PORV is ineffective under the given conditions. With the PORV ineffective, ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization.

B. Incorrect but plausible. From the background document for ES-1.2 "a PRZR PORV may not be effective for RCS depressurization at low temperature and pressure conditions that could exist in the pressurizer when the PRT rupture disk is still intact, and RCS pressure approaches PRT pressure." This is the reason that the PORV is ineffective under the given conditions. With the PORV ineffective, ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization. Normal spray is not available without RCPs running but is a plausible distractor as it is typically the preferred method of depressurization.

C. Incorrect but plausible. The PORV is has lost effectiveness here because the PRT rupture disk is intact, not because the RCPs are shut down. This is plausible because a reduced d/p renders the pressurizer spray valves ineffective. ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization.

D. Incorrect but plausible. The PORV is has lost effectiveness here because the PRT rupture disk is intact, not because the RCPs are shut down. This is plausible because a reduced d/p renders the pressurizer spray valves ineffective. ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization.

Technical Reference(s):	ES-1.2, "POST LOCA Cooldown and Depressurization", rev 42. Background document for ES-1.2, rev 3.			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1204I 03			
Question Source:	Bank #	X		
	Modified Bank#			
	New			
Question History:	Last NRC Exam	Seabrook 2020 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	41.5 / 41.7		