Form 4.1-BWR Boiling-Water Reactor Examination Outline (K/A Catalog Rev. 3)

Facility: Browns Ferry **Date of Exam:** 06/12/2023

Tier	Group	RO K/A Category Points											SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1	3	3	3				4	3			4	20	3	4	7
	2	1	1	1				1	1			1	6	1	2	3
	Tier Totals	4	4	4				5	4			5	26	4	6	10
2. Plant Systems	1	2	2	2	3	3	3	2	2	2	2	3	26	3	2	5
	2	1	1	1	0	1	1	1	2	1	1	1	11	0 / 1	2	3
	Tier Totals	3	3	3	3	4	4	3	4	3	3	4	37	4	4	8

3. Generic Knowledge and Abilities Categories	CO	EC	RC	EP	
	2	2	1	1	6

CO	EC	RC	EP	
2	2	1	2	7

4. Theory	Reactor Theory	Thermodynamics	
	3	3	6

Notes: CO=Conduct of Operations; EC = Equipment Control; RC = Radiation Control; EP =Emergency Procedures/Plan

ES-4.1-BWR	BWR Examination Outline (Browns Ferry) Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/**SRO**)									
Item #	E/APE # / Name / Safety Function	K 1	K 2	K 3	A1	A2	G*	K/A Topic(s)	IR	Q#
1	(295001) (APE1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION						X	(295001) (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION (G2.4.4) EMERGENCY PROCEDURES/PLAN Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures	4.5	1
2	(295003) (APE3) PARTIAL OR COMPLETE LOSS OF AC POWER		X					(295003AK2.05) Knowledge of the relationship between the (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER and the following systems or components: Decay heat removal systems	4.2	2
3	(295004) (APE4) PARTIAL OR COMPLETE LOSS OF DC POWER					X		(295004AA2.03) Ability to determine or interpret the following as they apply to (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER: Battery parameters	3.8	3
4	**(295004) (APE4) PARTIAL OR COMPLETE LOSS OF DC POWER**					**X**		**(295004AA2.01) Ability to determine or interpret the following as they apply to (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER: Partial or complete loss of DC power**	**4.1**	**76**
5	(295005) (APE5) MAIN TURBINE GENERATOR TRIP						X	(295005) (APE 5) MAIN TURBINE GENERATOR TRIP (G2.1.19) CONDUCT OF OPERATIONS Ability to use available indications to evaluate system or component status	3.9	4
6	(295006) (APE6) SCRAM	X						(295006AK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 6) SCRAM: Decay heat generation and removal	4	5
7	**(295006) (APE6) SCRAM**						**X**	**(295006) (APE 6) SCRAM (G2.2.38) EQUIPMENT CONTROL Knowledge of conditions and limitations in the facility license**	**4.5**	**77**
8	(295016) (APE16) CONTROL ROOM ABANDONMENT						X	(295016) (APE 16) CONTROL ROOM ABANDONMENT (G2.1.31) CONDUCT OF OPERATIONS Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup	4.6	6
9	**(295016) (APE16) CONTROL ROOM ABANDONMENT**						**X**	**(295016) (APE 16) CONTROL ROOM ABANDONMENT (G2.4.45) EMERGENCY PROCEDURES/PLAN Ability to prioritize and interpret the significance of each annunciator or alarm**	**4.3**	**78**
10	(295018) (APE18) PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)				X			(295018AA1.05) Ability to operate or monitor the following as they apply to (APE 18) PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): Reactor recirculation system	3.6	7
11	(295019) (APE19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR		X					(295019AK2.03) Knowledge of the relationship between the (APE 19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following systems or components: Feedwater system	3.7	8
12	**(295019) (APE19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR**					**X**		**(295019AA2.01) Ability to determine or interpret the following as they apply to (APE 19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air pressure**	**3.9**	**79**

13	(295021) (APE21) LOSS OF SHUTDOWN COOLING			X				(295021AK3.04) Knowledge of the reasons for the following responses or actions as they apply to (APE 21) LOSS OF SHUTDOWN COOLING: Maximizing reactor water cleanup flow	3.5	9
14	(295023) (APE23) REFUELING ACCIDENTS				X			(295023AA1.03) Ability to operate or monitor the following as they apply to (APE 23) REFUELING ACCIDENTS: Fuel handling equipment	3.1	10
15	(295024) (EPE1) HIGH DRYWELL PRESSURE	X						(295024EK1.01) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (EPE 1) HIGH DRYWELL PRESSURE: Drywell Integrity	4.3	11
16	**(295024) (EPE1) HIGH DRYWELL PRESSURE**						**X**	**(295024) (EPE 1) HIGH DRYWELL PRESSURE (G2.4.23) EMERGENCY PROCEDURES/PLAN Knowledge of the bases for prioritizing emergency operating procedures implementation**	**4.4**	**80**
17	(295025) (EPE2) HIGH REACTOR PRESSURE				X			(295025EA1.09) Ability to operate or monitor the following as they apply to (EPE 2) HIGH REACTOR PRESSURE: RPS	4.2	12
18	**(295025) (EPE2) HIGH REACTOR PRESSURE**					**X**		**(295025EA2.01) Ability to determine or interpret the following as they apply to (EPE 2) HIGH REACTOR PRESSURE: Reactor pressure**	**4.3**	**81**
19	(295026) (EPE3) SUPPRESSION POOL HIGH WATER TEMPERATURE		X					(295026EK2.01) Knowledge of the relationship between the (EPE 3) SUPPRESSION POOL HIGH WATER TEMPERATURE and the following systems or components: RHR/LPCI	3.8	13
	(295027) (EPE4) HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) / 5							NOT APPLICABLE		
20	(295028) (EPE5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY)			X				(295028EK3.01) Knowledge of the reasons for the following responses or actions as they apply to (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY): Emergency depressurization	4.1	14
21	(295030) (EPE7) LOW SUPPRESSION POOL WATER LEVEL					X		(295030EA2.03) Ability to determine or interpret the following as they apply to (EPE 7) LOW SUPPRESSION POOL WATER LEVEL: Reactor pressure	4.1	15
22	(295031) (EPE8) REACTOR LOW WATER LEVEL					X		(295031EA2.03) Ability to determine or interpret the following as they apply to (EPE 8) REACTOR LOW WATER LEVEL: Reactor pressure	4.2	16
23	(295037) (EPE14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN						X	(295037) (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN (G2.1.30) CONDUCT OF OPERATIONS Ability to locate and operate components, including local controls	4.4	17
24	(295038) (EPE15) HIGH OFFSITE RADIOACTIVITY RELEASE RATE				X			(295038EA1.08) Ability to operate or monitor the following as they apply to (EPE 15) HIGH OFFSITE RADIOACTIVITY RELEASE RATE: MSIV leakage control	3.1	18

25	(600000) (APE24) PLANT FIRE ON SITE			X				(600000AK3.04) Knowledge of the reasons for the following responses or actions as they apply to (APE 24) PLANT FIRE ON SITE: Actions contained in the fire response procedures for a plant fire on site	3.6	19
26	**(600000) (APE24) PLANT FIRE ON SITE**						**X**	**(600000) (APE 24) PLANT FIRE ON SITE (G2.2.14) EQUIPMENT CONTROL Knowledge of the process for controlling equipment configuration or status**	**4.3**	**82**
27	(700000) (APE25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES	X						(700000AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Under-excitation	3.2	20
K/A Category Totals:		3	3	3	4	3	4	Group Point Total:		20/**7**
						3	**4**			

ES-4.1-BWR	BWR Examination Outline (Browns Ferry) Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/**SRO**)									
Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
28	(295002) (APE2) LOSS OF MAIN CONDENSER VACUUM			X				(295002AK3.11) Knowledge of the reasons for the following responses or actions as they apply to (APE 2) LOSS OF MAIN CONDENSER VACUUM: Bypass augmented offgas system	3.1	21
29	(295007) (APE7) HIGH REACTOR PRESSURE					X		(295007AA2.02) Ability to determine or interpret the following as they apply to (APE 7) HIGH REACTOR PRESSURE: Reactor power	4.5	22
30	**(295008) (APE8) HIGH REACTOR WATER LEVEL**					**X**		**(295008AA2.04) Ability to determine or interpret the following as they apply to (APE 8) HIGH REACTOR WATER LEVEL: Heatup rate**	**3.3**	**83**
	(295009) (APE9) LOW REACTOR WATER LEVEL / 2									
31	**(295010) (APE10) HIGH DRYWELL PRESSURE**						**X**	**(295010) (APE 10) HIGH DRYWELL PRESSURE (G2.1.20) CONDUCT OF OPERATIONS Ability to interpret and execute procedure steps**	**4.6**	**84**
	(295011) (APE11) HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) / 5									
	(295012) (APE12) HIGH DRYWELL TEMPERATURE / 5									
	(295013) (APE13) HIGH SUPPRESSION POOL TEMPERATURE. / 5									
32	**(295014) (APE14) INADVERTENT REACTIVITY ADDITION**						**X**	**(295014) (APE 14) INADVERTENT REACTIVITY ADDITION (G2.1.37) CONDUCT OF OPERATIONS Knowledge of procedures, guidelines, or limitations associated with reactivity management**	**4.6**	**85**
33	(295017) (APE17) HIGH OFFSITE RELEASE RATE	X						(295017AK1.06) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 17) HIGH OFFSITE RELEASE RATE: Filtered vs. nonfiltered release	3.6	23
	(295020) (APE20) INADVERTENT CONTAINMENT ISOLATION / 5 & 7									
	(295022) (APE22) LOSS OF CONTROL ROD DRIVE PUMPS / 1									

	(295029) (EPE6) HIGH SUPPRESSION POOL WATER LEVEL / 5									
34	(295032) (EPE9) HIGH SECONDARY CONTAINMENT AREA TEMPERATURE						X	(295032) (EPE 9) HIGH SECONDARY CONTAINMENT AREA TEMPERATURE (G2.4.50) EMERGENCY PROCEDURES/PLAN Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure	4.2	24
35	(295033) (EPE10) HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS		X					(295033EK2.02) Knowledge of the relationship between the (EPE 10) HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following systems or components: Process radiation monitoring system	3.5	25
	(295034) (EPE11) SECONDARY CONTAINMENT VENTILATION HIGH RADIATION / 9									
	(295035) (EPE12) SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE / 5									
36	(295036) (EPE13) SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL				X			(295036EA1.03) Ability to operate or monitor the following as they apply to (EPE 13) SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Radwaste system	2.7	26
	(500000) (EPE16) HIGH CONTAINMENT HYDROGEN CONCENTRATION / 5									
K/A Category Totals:		1	1	1	1	1	1	Group Point Total:		6/**3**
						1	**2**			

ES-4.1-BWR	BWR Examination Outline (Browns Ferry) Plant Systems—Tier 2/Group 1 (RO/**SRO**)														
Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
37	(203000) (SF2, SF4 RHR/LPCI) RHR/LPCI: INJECTION MODE				X								(203000K4.13) Knowledge of (SF2, SF4 RHR/LPCI) RHR/LPCI: INJECTION MODE design features and/or interlocks that provide for the following: Prevention of leakage to the environment through LPCI/RHR heat exchanger	3	27
38	(205000) (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE)											X	(205000) (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) (291005K1.06) MOTORS AND GENERATORS Reason for limiting the number of motor starts in a given time period	3.1	28
39	(206000) (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM				X								(206000K4.01) Knowledge of (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM design features and/or interlocks that provide for the following: Turbine trips	3.8	29
40	**(206000) (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM**											**X**	**(206000) (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM (G2.2.42) EQUIPMENT CONTROL Ability to recognize system parameters that are entry-level conditions for technical specifications**	**4.6**	**86**
	(207000) (SF4 IC) ISOLATION (EMERGENCY) CONDENSER												NOT APPLICABLE		
41	(209001) (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM									X			(209001A3.02) Ability to monitor automatic operation of the (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM including: Pump start	4.3	30
	(209002) (SF2, SF4 HPCS) HIGH PRESSURE CORE SPRAY SYSTEM														
42	(211000) (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM											X	(211000) (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM (G2.2.3) EQUIPMENT CONTROL (Multi-unit license) Knowledge of the design, procedural, or operational differences between units	3.8	31

43	(211000) (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM					X							(211000K5.03) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM: Shutdown margin	3.7	32
44	(212000) (SF7 RPS) REACTOR PROTECTION SYSTEM							X					(212000A1.04) Ability to predict and/or monitor changes in parameters associated with operation of the (SF7 RPS) REACTOR PROTECTION SYSTEM including: RPS bus status	3.7	33
45	(212000) (SF7 RPS) REACTOR PROTECTION SYSTEM						X						(212000K6.06) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF7 RPS) REACTOR PROTECTION SYSTEM : Process radiation monitoring system	3	34
46	(215003) (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM		X										(215003K2.01) (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM Knowledge of electrical power supplies to the following: IRM channels/detectors	3.4	35
47	**(215003) (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM**								**X**				**(215003A2.03) Ability to (a) predict the impacts of the following on the (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Stuck detector**	**3**	**87**
48	(215004) (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM										X		(215004A4.02) Ability to manually operate and/or monitor the (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM in the control room: SRMS recorder	3.4	36
49	(215005) (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR										X		(215005A4.04) Ability to manually operate and/or monitor the (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR in the control room: LPRM back panel switches, meters, and indicating lights	3.4	37
50	(215005) (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR						X						(215005K6.02) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR: TIP	2.9	38

51	(217000) (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM		X									(217000K2.02) (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM Knowledge of electrical power supplies to the following: Initiation/isolation logic	3.7	39
52	(217000) (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM				X							(217000K4.17) Knowledge of (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM design features and/or interlocks that provide for the following: Bypass trips and isolation logic	3.7	40
53	(218000) (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM										X	218000 (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM (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	4.2	41
54	(223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF								X			(223002A3.01) Ability to monitor automatic operation of the (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF including: System indicating lights and alarms	4	42
55	(239002) (SF3 SRV) SAFETY RELIEF VALVES							X				(239002A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the (SF3 SRV) SAFETY RELIEF VALVES including: Reactor water level	4.2	43
56	(259002) (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM	X										(259002K1.16) Knowledge of the physical connections and/or cause and effect relationships between the (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM and the following systems: ECCS	3.5	44
57	**(259002) (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM**								**X**			**(259002A2.05) Ability to (a) predict the impacts of the following on the (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of applicable plant air systems**	**3.2**	**88**

58	(261000) (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM					X							(261000K5.01) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM : Heat removal mechanisms	3	45
59	(262001) (SF6 AC) AC ELECTRICAL DISTRIBUTION			X									(262001K3.03) Knowledge of the effect that a loss or malfunction of the (SF6 AC) AC ELECTRICAL DISTRIBUTION will have on the following systems or system parameters: DC electrical distribution	3.9	46
60	**(262001) (SF6 AC) AC ELECTRICAL DISTRIBUTION**								**X**				**(262001A2.10) Ability to (a) predict the impacts of the following on the (SF6 AC) AC ELECTRICAL DISTRIBUTION and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Exceeding current limitations**	**3.4**	**89**
61	(262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC)								X				(262002A2.01) Ability to (a) predict the impacts of the following on the (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal voltage	3.2	47
62	**(262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC)**											**X**	**(262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC) (G2.4.16) EMERGENCY PROCEDURES/PLAN Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as operating procedures, abnormal operating procedures, or severe accident management guidelines**	**4.4**	**90**
63	(263000) (SF6 DC) DC ELECTRICAL DISTRIBUTION	X											(263000K1.01) Knowledge of the physical connections and/or cause and effect relationships between the (SF6 DC) DC ELECTRICAL DISTRIBUTION and the following systems: AC electrical distribution	4	48

64	(264000) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET)								X				(264000A2.11) Ability to (a) predict the impacts of the following on the (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Failure of emergency generator to start/load	4.6	49
65	(300000) (SF8 IA) INSTRUMENT AIR SYSTEM			X									(300000K3.12) Knowledge of the effect that a loss or malfunction of the (SF8 IA) INSTRUMENT AIR SYSTEM will have on the following systems or system parameters: Feedwater system	3.5	50
66	(400000) (SF8 CCS) COMPONENT COOLING WATER SYSTEM						X						(400000K6.11) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF8 CCS) COMPONENT COOLING WATER SYSTEM : Recirculation system	2.9	51
67	(510000) (SF4 SWS*) SERVICE WATER SYSTEM					X							(510000K5.02) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4 SWS*) SERVICE WATER SYSTEM: Radiation alarm response	3	52
K/A Category Totals:		2	2	2	3	3	3	2	2	2	2	3	Group Point Total:		26/**5**
									3			**2**			

ES-4.1-BWR	BWR Examination Outline (Browns Ferry) Plant Systems—Tier 2/Group 2 (RO/**SRO**)														
Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
68	(201001) (SF1 CRDH) CRD HYDRAULIC SYSTEM											X	(201001) (SF1 CRDH) CRD HYDRAULIC SYSTEM (291003K1.04) CONTROLLERS AND POSITIONERS Function and operation of pressure and temperature controllers, including pressure and temperature control valves	3.3	53
	(201002) (SF1 RMCS) REACTOR MANUAL CONTROL SYSTEM														
	(201003) (SF1 CRDM) CONTROL ROD AND DRIVE MECHANISM														
	(201004) (SF7 RSCS) ROD SEQUENCE CONTROL SYSTEM														
	(201005) (SF1, SF7 RCIS) ROD CONTROL AND INFORMATION SYSTEM														
69	(201006) (SF7 RWMS) ROD WORTH MINIMIZER SYSTEM							X					(201006A1.04) Ability to predict and/or monitor changes in parameters associated with operation of the (SF7 RWMS) ROD WORTH MINIMIZER SYSTEM including: Rod withdrawal blocks	3.6	54
	(202001) (SF1, SF4 RS) RECIRCULATION SYSTEM														
70	(202002) (SF1 RSCTL) RECIRCULATION FLOW CONTROL SYSTEM								X				(202002A2.12) Ability to (a) predict the impacts of the following on the (SF1 RSCTL) RECIRCULATION FLOW CONTROL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: APRM/LPRM	3.8	55
71	(204000) (SF2 RWCU) REACTOR WATER CLEANUP SYSTEM										X		(204000A4.04) Ability to manually operate and/or monitor the (SF2 RWCU) REACTOR WATER CLEANUP SYSTEM in the control room: Heat exchanger temperature	3.1	56

72	**(214000) (SF7 RPIS) ROD POSITION INFORMATION SYSTEM**											**X**	**(214000) (SF7 RPIS) ROD POSITION INFORMATION SYSTEM (G2.4.21) EMERGENCY PROCEDURES/PLAN Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions**	**4.6**	**91**
73	(215001) (SF7 TIP) TRAVERSING IN CORE PROBE					X							(215001K5.02) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF7 TIP) TRAVERSING IN CORE PROBE: Area radiation monitor indications	2.8	57
74	(215002) (SF7 RBMS) ROD BLOCK MONITOR SYSTEM								X				(215002A2.03) Ability to (a) predict the impacts of the following on the (SF7 RBMS) ROD BLOCK MONITOR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of associated reference APRM channel	3.3	58
	(216000) (SF7 NBI) NUCLEAR BOILER INSTRUMENTATION														
	(219000) (SF5 RHR SPC) RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE														
	(223001) (SF5 PCS) PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES														
	(226001) (SF5 RHR CSS) RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE														
	(230000) (SF5 RHR SPS) RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE														

	(233000) (SF9 FPCCU) FUEL POOL COOLING/CLEANUP														
	(234000) (SF8 FH) FUEL HANDLING														
75	(239001) (SF3, SF4 MRSS) MAIN AND REHEAT STEAM SYSTEM		X										(239001K2.01) (SF3, SF4 MRSS) MAIN AND REHEAT STEAM SYSTEM Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids	3.6	59
	(239003) (SF9 MSVLCS) MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM														
	(241000) (SF3 RTPRS) REACTOR/TURBINE PRESSURE REGULATING SYSTEM														
	(245000) (SF4 MTGEN) MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS														
76	(256000) (SF2 CDS) CONDENSATE SYSTEM	X											(256000K1.30) Knowledge of the physical connections and/or cause and effect relationships between the (SF2 CDS) CONDENSATE SYSTEM and the following systems: Oxygen injection system	2.7	60
	(259001) (SF2 FWS) FEEDWATER SYSTEM														
	(268000) (SF9 RW) RADWASTE SYSTEM														
77	(271000) (SF9 OG) OFFGAS SYSTEM			X									(271000K3.02) Knowledge of the effect that a loss or malfunction of the (SF9 OG) OFFGAS SYSTEM will have on the following systems or system parameters: Offsite release rate	3.7	61

78	**(272000) (SF7, SF9 RMS) RADIATION MONITORING SYSTEM**								**X**				**(272000A2.06) Ability to (a) predict the impacts of the following on the (SF7, SF9 RMS) RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Downscale trips**	**3.1**	**92**
79	(286000) (SF8 FPS) FIRE PROTECTION SYSTEM									X			(286000A3.05) Ability to monitor automatic operation of the (SF8 FPS) FIRE PROTECTION SYSTEM including: Fire doors	2.8	62
	(288000) (SF9 PVS) PLANT VENTILATION SYSTEMS														
80	**(290001) (SF5 SC) SECONDARY CONTAINMENT**											**X**	**(290001) (SF5 SC) SECONDARY CONTAINMENT (G2.1.9) CONDUCT OF OPERATIONS Ability to direct licensed personnel activities inside the control room (SRO Only)**	**4.5**	**93**
	(290002) (SF4 RVI) REACTOR VESSEL INTERNALS														
	(290003) (SF9 CRV) CONTROL ROOM VENTILATION														
81	(510001) (SF8 CWS*) CIRCULATING WATER SYSTEM						X						(510001K6.04) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF8 CWS*) CIRCULATING WATER SYSTEM: Valve malfunctions	3.0	63
K/A Category Totals:		1	1	1	0	1	1	1	2	1	1	1	Group Point Total:		11/**3**
									1			**2**			

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Browns Ferry)					
Facility: Browns Ferry					Date of Exam: 06/12/2023	

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

Category	K/A #	Topic	Item #	RO IR	RO Q#	SRO-Only IR	SRO-Only Q#
1. Conduct of Operations	G2.1.7	(G2.1.7) CONDUCT OF OPERATIONS Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	82	4.4	64		
	G2.1.45	(G2.1.45) CONDUCT OF OPERATIONS Ability to identify and interpret diverse indications to validate the response of another indication	83	4.3	65		
	G2.1.6	**(G2.1.6) CONDUCT OF OPERATIONS Ability to manage the control room crew during plant transients (SRO Only)**	**84**			**4.8**	**94**
	G2.1.40	**(G2.1.40) CONDUCT OF OPERATIONS Knowledge of refueling administrative requirements**	**85**			**3.9**	**95**
	Subtotal			N/A	2	N/A	**2**
2. Equipment Control	G2.2.18	(G2.2.18) EQUIPMENT CONTROL Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments and work prioritization	86	2.6	66		
	G2.2.39	(G2.2.39) EQUIPMENT CONTROL Knowledge of less than or equal to 1 hour technical specification action statements (This K/A does not include action statements of 1 hour or less that follow the expiration of a completion time for a technical specification condition for which an action statement has already been entered.)	87	3.9	67		
	G2.2.19	**(G2.2.19) EQUIPMENT CONTROL Knowledge of maintenance work order requirements**	**88**			**3.4**	**96**
	G2.2.38	**(G2.2.38) EQUIPMENT CONTROL Knowledge of conditions and limitations in the facility license**	**89**			**4.5**	**97**
	Subtotal			N/A	2	N/A	**2**
3. Radiation Control	G2.3.5	(G2.3.5) RADIATION CONTROL Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms or personnel monitoring equipment	90	2.9	68		
	G2.3.12	**(G2.3.12) RADIATION CONTROL Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters**	**91**			**3.7**	**98**
	Subtotal			N/A	1	N/A	**1**
4. Emergency Procedures / Plan	G2.4.14	(G2.4.14) EMERGENCY PROCEDURES/PLAN Knowledge of general guidelines for emergency and abnormal operating procedures usage	92	3.8	69		
	G2.4.3	**(G2.4.3) EMERGENCY PROCEDURES/PLAN Ability to identify post-accident instrumentation**	**93**			**3.9**	**99**

	G2.4.43	(G2.4.43) EMERGENCY PROCEDURES/PLAN Knowledge of emergency communications systems and techniques	94			3.8	100
	Subtotal			N/A	1	N/A	2
Tier 3 Point Total				N/A	**6**	N/A	**7**

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Browns Ferry)	
Facility: Browns Ferry		Date of Exam: 06/12/2023

Theory (Tier 4) (RO)

Category	K/A #	Topic	Item #	RO IR	RO Q#
Reactor Theory	292003	(292003K1.07) REACTOR KINETICS AND NEUTRON SOURCES Explain prompt critical, prompt jump, and prompt drop	95	3.3	70
	292007	(292007K1.03) FUEL DEPLETION AND BURNABLE POISONS Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points	96	2.7	71
	292008	(292008K1.29) REACTOR OPERATIONAL PHYSICS (NORMAL REACTOR SHUTDOWN) Define decay heat	97	3.6	72
	Subtotal				3
Thermodynamics	293007	(293007K1.09) HEAT TRANSFER (CONDENSER APPLICATIONS OF HEAT TRANSFER) Discuss operational implications of condensate	98	2.7	73
	293008	(293008K1.26) THERMAL HYDRAULICS (RECIRCULATION SYSTEM) Explain the jet pump operating principle	99	3.1	74
	293010	(293010K1.05) BRITTLE FRACTURE AND VESSEL THERMAL STRESS State the effect of fast neutron irradiation on reactor vessel metals	100	2.8	75
	Subtotal				3
Tier 4 Point Total				N/A	6

Form 3.2-1 **Administrative Topics Outline**

Facility Browns Ferry NPP			Date of Examination: 6/12/23
Examination Level: RO ☒ SRO ☐			Operating Test Number: 23-04
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)		Type Code* (Step 3)
Conduct of Operations	JPM 760	Main Generator Voltage Regulator Operation	R, N
	K/A 2.1.20 (RO 4.6)	Ability to interpret and execute procedure steps.	
Conduct of Operations	JPM 551	Work Hour Limitations	R, D
	K/A 2.1.5 (RO 2.9)	Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations.	
Equipment Control	JPM 510	Evaluate Recombiner Performance	R, D, P
	K/A 2.2.44 (RO 4.2)	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.	
Radiation Control	JPM 688	Determine ALARA Path for Emergency Radiation Exposure	R, D
	K/A 2.3.12 (RO 3.2)	Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters.	
Emergency Plan	N/A		N/A

Instructions for completing Form 3.2-1, "Administrative Topics Outline"

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	Number of JPMs	
	RO*	SRO and RO Retakes
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., "Equipment Control," "Radiation Control," or "Emergency Plan" can be omitted by doubling up on "Conduct of Operations"), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:

Location:

(C)ontrol room, (S)imulator, or Class(R)oom

Source and Source Criteria:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)

(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)

(N)ew or Significantly (M)odified from bank (no fewer than one)

Reactor Operator

1. **Conduct of Ops** – Main Generator Voltage Regulator Operation

   Using OI-47, Turbine-Generator System, determines the notification requirements and grace period (if any) when operating the Main Generator Voltage Regulator in MANUAL.

   K/A 2.1.20: Ability to interpret and execute procedure steps. (RO 4.6)

2. **Conduct of Ops** – Work Hour Limitations

   Using NPG-SPP-03.21, Fatigue Rule and Work Hour Limits, and given an operator's work schedule, determines that the work hour limit requiring a continuous break of at least 34 hours in any 9-day period is not met and that the schedule needs to be revised or a waiver granted.

   K/A 2.1.5: Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations. (RO 2.9)

3. **Equipment Control** – Evaluate Recombiner Performance

   Using OI-66, Off-Gas System, and given Recombiner operating parameters, evaluates Off-Gas Recombiner Performance to determine if Acceptance Criteria is met.

   K/A 2.2.44. 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. (RO 4.2)

4. **Radiation Control** – Determine ALARA Path for Emergency Radiation Exposure

   Given three paths with dose rates to complete a job in the Drywell, calculates total dose for each path and decides which path would be the best with respect to maintaining the dose as low as possible.

   K/A 2.3.12: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters. (RO 3.2)

5. **Emergency Plan** – N/A

Form 3.2-1 **Administrative Topics Outline**

Facility Browns Ferry NPP			Date of Examination: 6/12/23
Examination Level: RO ☐ SRO ☒			Operating Test Number: 23-04
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)		Type Code* (Step 3)
Conduct of Operations	JPM 507	Fitness for Duty Determination	R, D
	K/A 2.1.23 (SRO 4.4)	Ability to perform general or normal operating procedures during any plant condition	
Conduct of Operations	JPM 665	NRC Event Notification	R, M
	K/A 2.1.18 (SRO 3.8)	Ability to make accurate, clear, and concise logs, records, status boards, and reports.	
Equipment Control	JPM 761	Review Drywell Average Air Temperature Surveillance	R, N
	K/A 2.2.37 (SRO 4.6)	Ability to determine operability or availability of safety-related equipment (SRO Only)	
Radiation Control	JPM 688	Determine ALARA Path for Emergency Radiation Exposure	R, D
	K/A 2.3.12 (SRO 3.7)	Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters	
Emergency Plan	JPM 752	Emergency Action Level Classification	R, N
	K/A 2.4.41 (SRO 4.6)	Knowledge of the Emergency Action Level thresholds and Classifications (SRO Only)	

Instructions for completing Form 3.2-1, "Administrative Topics Outline"

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	Number of JPMs	
	RO*	SRO and RO Retakes
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., "Equipment Control," "Radiation Control," or "Emergency Plan" can be omitted by doubling up on "Conduct of Operations"), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:

Location:

(C)ontrol room, (S)imulator, or Class(R)oom

Source and Source Criteria:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)

(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)

(N)ew or Significantly (M)odified from bank (no fewer than one)

Senior Reactor Operator

1. **Conduct of Ops** – Fitness for Duty Determination

   Using NPG-SPP-14.1, Fitness for Duty and Fatigue Management, and given Fitness for Duty information on three employees, determines which individuals can be called in to work and any action that must be taken upon arrival at the plant.

   K/A 2.1.23: Ability to perform general or normal operating procedures during any plant condition (SRO 4.4)

2. **Conduct of Ops** – NRC Event Notification

   Given a failure of 2-PCV-1-22, MAIN STEAM LINE 'B' RELIEF VALVE, with HPCI already tagged out for scheduled maintenance, determine Technical Specification 3.5.1 Action(s) and Nuclear Regulatory Reporting requirements in accordance with NPG-SPP-3.5, Regulatory Reporting Requirements.

   K/A 2.1.18: Ability to make accurate, clear, and concise logs, records, status boards, and reports. (SRO 3.8)

3. **Equipment Control** – Review Drywell Average Air Temperature Surveillance

   Using SR-2, Instrument Checks and Observations, reviews Bulk Volumetric Average Drywell Air Temperature readings and determines if Acceptance Criteria (AC) is met. Using the information obtained in the Surveillance Requirement, determines the Technical Specification Required Action(s) that must be taken (if any).

   K/A 2.2.37 Ability to determine operability or availability of safety-related equipment (SRO Only) (SRO 4.6)

4. **Radiation Control** – Determine ALARA Path for Emergency Radiation Exposure

   Given three paths with dose rates to complete a job in the Drywell, calculates total dose for each path and decides which path would be the best with respect to maintaining the dose as low as possible. Using EPIP-15, Emergency Exposure, determines if an individual could be allowed to perform the job and fills out required paperwork.

   K/A 2.3.12: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters. (SRO 3.7)

5. Emergency Plan – Emergency Action Level Classification

Given EPIP-1, Emergency Classification Procedure, and plant conditions, classifies an event and fills out the required notification form.

K/A 2.4.41: Knowledge of the Emergency Action Level thresholds and Classifications (SRO Only) (SRO 4.6)

Form 3.2-2 **Control Room/In-Plant Systems Outline**

Facility: Browns Ferry NPP Date of Examination: 6/12/23

Exam Level: ☒ RO ☐ SRO-I ☐ SRO-U Operating Test No.: 23-04

Control Room Systems			
System / JPM Title		**Type Code**	**Safety Function**
a. JPM 15	Maximize Control Rod Drive Injection to the Reactor Vessel per EOI-Appendix-5B, Injection System Lineup CRD	D, L, S	1
b. JPM 274A	Remove a Reactor Feedwater Pump from Service per OI-3, Reactor Feedwater System	A, D, S	2
c. JPM 756A	Inject to the Reactor per EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System	A, EN, N, S	4
d. JPM 669	Purge the Primary Containment per EOI-Appendix-14A, N2 Makeup to Primary Containment	D, EN, S	5
e. JPM 759A	Parallel an Emergency Diesel Generator to a Shutdown Board per OI-82, Standby Diesel Generator System	A, D, EN, S	6
f. JPM 272	Prepare Source Range Monitors for Startup per OI-92, Source Range Monitors	D, L, S	7
g. JPM 757A	Start a Condenser Circulating Water Pump per OI-27, Condenser Circulating Water System	A, N, S	8
h. JPM 55A	Emergency Vent Primary Containment per EOI Appendix-13, Emergency Venting Primary Containment	A, D, EN, L, P, S	9
In-Plant Systems			
i. JPM 758	Bypass Reactor Core Isolation Cooling High RPV Water Level Trips per EOI-Appendix-16H, Bypassing RCIC High RPV Water Level Shutdown Interlocks	E, L, N, R	2
j. JPM 734	Bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock per EOI-Appendix-8A, Bypassing Group 1 RPV Low Low Low Level Isolation Interlocks	E, L, D	5
k. JPM 673	Manually Start ‘A’ and ‘C’ Fire Pumps per 0-OI-26, High Pressure Fire System	D	8

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	Control Room	In-Plant	Total
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

   Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, "Heat Removal from Reactor Core," in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant's license level (see the table in step 1).

   For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.

   For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee's site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5.
   The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

   Apply the following specific task selection criteria:

   - At least one of the tasks shall be related to a shutdown or low-power condition.
   - Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee's operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee's operating procedures. At least one alternate path JPM must be new or modified from the bank.
   - At least one of the tasks conducted in the plant shall evaluate the applicant's ability to implement actions required during an emergency or abnormal condition.
   - At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

   If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

Code	License Level Criteria		
	RO	SRO-I	SRO-U
(A)lternate path	4–6	4–6	2–3
(C)ontrol room			
(D)irect from bank	≤ 9	≤ 8	≤ 4
(E)mergency or abnormal in-plant	≥ 1	≥ 1	≥ 1
(EN)gineered safety feature (for control room system)	≥ 1	≥ 1	≥ 1
(L)ow power/shutdown	≥ 1	≥ 1	≥ 1
(N)ew or (M)odified from bank (must apply to at least one alternate path JPM)	≥ 2	≥ 2	≥ 1
(P)revious two exams (randomly selected)	≤ 3	≤ 3	≤ 2
(R)adiologically controlled area	≥ 1	≥ 1	≥ 1
(S)imulator			

Reactor Operator

Job Performance Measures

a. JPM 15

Title: Maximize Control Rod Drive Injection to the Reactor Vessel per EOI-Appendix-5B, Injection System Lineup CRD

Description: The candidate will perform operations necessary to align the Control Rod Drive (CRD) Hydraulic System to maximize injection to the Reactor Vessel in accordance with EOI-Appendix 5B.

K/A: 201001 SF1 CRDH Control Rod Drive Hydraulic System A4.03; Ability to manually operate and/or monitor in the control room: CRD system flow control valve (3.6)

b. JPM 274A

Title: Remove a Reactor Feedwater Pump from Service per OI-3, Reactor Feedwater System

Description: The candidate will perform operations necessary to remove 'A' Reactor Feedwater Pump (RFPT) from service in accordance with OI-3, and respond to a failure of the RFPT Discharge Check Valve to automatically close.

K/A 259001 SF2 Reactor Feedwater System A3.03; System flow: Ability to monitor automatic operation of the Feedwater System, including: System flow (3.8)

c. JPM 756A

Title: Inject to the Reactor per EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System

Description: The candidate will perform the actions necessary in accordance with EOI-Appendix-7L to start an Emergency High Pressure Makeup Pump (EHPM) to raise Reactor Water Level to (+)2 to (+)51 inches. The candidate will be required to take action to provide a power source to the EHPM in accordance with Attachment 3, Supplemental Diesel Generator Operation from MCR.

K/A 295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to Low Reactor Water Level: Reactor Water Level Control (4.2)

d. JPM 669

Title: Purge the Primary Containment per EOI-Appendix-14A, N2 Makeup to Primary Containment

Description: The candidate will perform operations necessary to purge the Primary Containment with Nitrogen in accordance with EOI-Appendix-14A.

K/A

223001 SF5 PCS Primary Containment System and Auxiliaries A2.04; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High containment/drywell hydrogen concentration (3.8)

e. JPM 759A

Title: Parallel an Emergency Diesel Generator to a Shutdown Board per OI-82, Standby Diesel Generator System

Description: The candidate will perform operations necessary to parallel an Emergency Diesel Generator (EDG) to its associated 4KV Shutdown Board in accordance with OI-82, and respond to a low oil pressure condition.

K/A

264000 SF6 EGE Emergency Generators A1.10; Ability to predict and/or monitor changes in parameters associated with operation of the Emergency Generators, including: Lights and alarms (3.5)

f. JPM 272

Title: Prepare Source Range Monitors for Startup per OI-92, Source Range Monitors

Description: The candidate will prepare the Source Range Monitors (SRMs) for Reactor Startup in accordance with OI-92, Source Range Monitors.

K/A

215004 SF7 SRMS Source Range Monitor System A4.04; Ability to manually operate and/or monitor in the control room: SRMS drive control switches (3.5)

g. JPM 757A

Title: Start a Condenser Circulating Water Pump per OI-27, Condenser Circulating Water System

Description: The candidate will start a Condenser Circulating Water (CCW) Pump in accordance with OI-27, Condenser Circulating Water System, and respond to a failure of the pump discharge valve to automatically open.

K/A

400000 SF8 CCW Component Cooling Water System A3.07; Ability to monitor automatic operations of the Component Cooling Water System, including: System realignments (3.3)

h. JPM 55A

Title: Emergency Vent Primary Containment per EOI-Appendix-13, Emergency Venting Primary Containment

Description: The candidate will emergency vent Primary Containment in accordance with EOI-Appendix-13. The Suppression Chamber vent lineup will fail, and an alternate vent method must be used.

K/A

288000 SF9 PVS Plant Ventilation Systems A2.01; Ability to (a) predict the impacts of the following on the Plant Ventilation Systems and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High Drywell Pressure (3.5)

i. JPM 758

Title: Bypass Reactor Core Isolation Cooling High RPV Water Level Trips per EOI-Appendix-16H, Bypassing RCIC High RPV Water Level Shutdown Interlocks

Description: The candidate will perform operations necessary to bypass the Reactor Core Isolation (RCIC) High RPV Water Level trips in accordance with EOI-Appendix-16H.

K/A

217000 SF2 RCIC Reactor Core Isolation Cooling System A2.02; Ability to (a) predict the impacts of the following on the Reactor Core Isolation Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Turbine trips (4.2)

j. JPM 734

Title: Bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock per EOI-Appendix-8A, Bypassing Group 1 RPV Low Low Low Level Isolation Interlocks

Description: The candidate will perform operations necessary to bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock in accordance with EOI-Appendix-8A.

K/A

223001 SF5 PCS Primary Containment System and Auxiliaries A2.01; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of coolant accident (4.4)

k. JPM 673

Title: Manually Start ‘A’ And ‘C’ Fire Pumps per 0-OI-26, High Pressure Fire Protection System

Description: The candidate will perform operations necessary to manually start ‘A’ and ‘C’ Fire Pumps in accordance with 0-OI-26, High Pressure Fire Protection System.

K/A

286000 SF8 Fire Protection System A2.08; Ability to (a) predict the impacts of the following on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to actuate when required (3.7)

Form 3.2-2 **Control Room/In-Plant Systems Outline**

Facility: Browns Ferry NPP Date of Examination: 6/12/23

Exam Level: ☐ RO ☒ SRO-I ☐ SRO-U Operating Test No.: 23-04

Control Room Systems			
System / JPM Title		**Type Code***	**Safety Function**
a. JPM 15	Maximize Control Rod Drive Injection to the Reactor Vessel per EOI-Appendix-5B, Injection System Lineup CRD	D, L, S	1
b. JPM 274A	Remove a Reactor Feedwater Pump from Service per OI-3, Reactor Feedwater System	A, D, S	2
c. JPM 756A	Inject to the Reactor per EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System	A, EN, N, S	4
d. JPM 669	Purge the Primary Containment per EOI-Appendix-14A, N2 Makeup to Primary Containment	D, EN, S	5
e. JPM 759A	Parallel an Emergency Diesel Generator to a Shutdown Board per OI-82, Standby Diesel Generator System	A, D, EN, S	6
f. N/A			
g. JPM 757A	Start a Condenser Circulating Water Pump per OI-27, Condenser Circulating Water System	A, N, S	8
h. JPM 55A	Emergency Vent Primary Containment per EOI Appendix-13, Emergency Venting Primary Containment	A, D, EN, L, P, S	9
In-Plant Systems			
i. JPM 758	Bypass Reactor Core Isolation Cooling High RPV Water Level Trips per EOI-Appendix-16H, Bypassing RCIC High RPV Water Level Shutdown Interlocks	E, L, N, R	2
j. JPM 734	Bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock per EOI-Appendix-8A, Bypassing Group 1 RPV Low Low Low Level Isolation Interlocks	E, L, D	5
k. JPM 673	Manually Start ‘A’ and ‘C’ Fire Pumps per 0-OI-26, High Pressure Fire System	D	8

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	Control Room	In-Plant	Total
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, "Heat Removal from Reactor Core," in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant's license level (see the table in step 1).
For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.
For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee's site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5.

The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

Apply the following specific task selection criteria:

- At least one of the tasks shall be related to a shutdown or low-power condition.
- Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee's operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee's operating procedures. At least one alternate path JPM must be new or modified from the bank.
- At least one of the tasks conducted in the plant shall evaluate the applicant's ability to implement actions required during an emergency or abnormal condition.
- At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

Code	License Level Criteria		
	RO	**SRO-I**	**SRO-U**
(A)lternate path	4–6	4–6	2–3
(C)ontrol room			
(D)irect from bank	≤ 9	≤ 8	≤ 4
(E)mergency or abnormal in-plant	≥ 1	≥ 1	≥ 1
(EN)gineered safety feature (for control room system)	≥ 1	≥ 1	≥ 1
(L)ow power/shutdown	≥ 1	≥ 1	≥ 1
(N)ew or (M)odified from bank (must apply to at least one alternate path JPM)	≥ 2	≥ 2	≥ 1
(P)revious two exams (randomly selected)	≤ 3	≤ 3	≤ 2
(R)adiologically controlled area	≥ 1	≥ 1	≥ 1
(S)imulator			

Senior Reactor Operator (Instant)

Job Performance Measures

a. JPM 15

Title: Maximize Control Rod Drive Injection to the Reactor Vessel per EOI-Appendix-5B, Injection System Lineup CRD

Description: The candidate will perform operations necessary to align the Control Rod Drive (CRD) Hydraulic System to maximize injection to the Reactor Vessel in accordance with EOI-Appendix 5B.

K/A: 201001 SF1 CRDH Control Rod Drive Hydraulic System A4.03; Ability to manually operate and/or monitor in the control room: CRD system flow control valve (3.6)

b. JPM 274A

Title: Remove a Reactor Feedwater Pump from Service per OI-3, Reactor Feedwater System

Description: The candidate will perform operations necessary to remove 'A' Reactor Feedwater Pump (RFPT) from service in accordance with OI-3, and respond to a failure of the RFPT Discharge Check Valve to automatically close.

K/A 259001 SF2 Reactor Feedwater System A3.03; System flow: Ability to monitor automatic operation of the Feedwater System, including: System flow (3.8)

c. JPM 756A

Title: Inject to the Reactor per EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System

Description: The candidate will perform the actions necessary in accordance with EOI-Appendix-7L to start an Emergency High Pressure Makeup Pump (EHPM) to raise Reactor Water Level to (+)2 to (+)51 inches. The candidate will be required to take action to provide a power source to the EHPM in accordance with Attachment 3, Supplemental Diesel Generator Operation from MCR.

K/A 295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to Low Reactor Water Level: Reactor Water Level Control (4.2)

d. JPM 669 **Title:** Purge the Primary Containment per EOI-Appendix-14A, N2 Makeup to Primary Containment

Description: The candidate will perform operations necessary to purge the Primary Containment with Nitrogen in accordance with EOI-Appendix-14A.

K/A 223001 SF5 PCS Primary Containment System and Auxiliaries A2.04; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High containment/drywell hydrogen concentration (3.8)

e. JPM 759A **Title:** Parallel an Emergency Diesel Generator to a Shutdown Board per OI-82, Standby Diesel Generator System

Description: The candidate will perform operations necessary to parallel an Emergency Diesel Generator (EDG) to its associated 4KV Shutdown Board in accordance with OI-82, and respond to a low oil pressure condition.

K/A 264000 SF6 EGE Emergency Generators A1.10; Ability to predict and/or monitor changes in parameters associated with operation of the Emergency Generators, including: Lights and alarms (3.5)

f. N/A

g. JPM 757A **Title:** Start a Condenser Circulating Water Pump per OI-27, Condenser Circulating Water System

Description: The candidate will start a Condenser Circulating Water (CCW) Pump in accordance with OI-27, Condenser Circulating Water System, and respond to a failure of the pump discharge valve to automatically open.

K/A 400000 SF8 CCW Component Cooling Water System A3.07; Ability to monitor automatic operations of the Component Cooling Water System, including: System realignments (3.3)

h. JPM 55A **Title:** Emergency Vent Primary Containment per EOI-Appendix-13, Emergency Venting Primary Containment

Description: The candidate will emergency vent Primary Containment in accordance with EOI-Appendix-13. The Suppression Chamber vent lineup will fail, and an alternate vent method must be used.

K/A 288000 SF9 PVS Plant Ventilation Systems A2.01; Ability to (a) predict the impacts of the following on the Plant Ventilation Systems and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High Drywell Pressure (3.6)

i. JPM 758 **Title:** Bypass Reactor Core Isolation Cooling High RPV Water Level Trips per EOI-Appendix-16H, Bypassing RCIC High RPV Water Level Shutdown Interlocks

Description: The candidate will perform operations necessary to bypass the Reactor Core Isolation (RCIC) High RPV Water Level trips in accordance with EOI-Appendix-16H.

K/A 217000 SF2 RCIC Reactor Core Isolation Cooling System A2.02; Ability to (a) predict the impacts of the following on the Reactor Core Isolation Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Turbine trips (4.0)

j. JPM 734 **Title:** Bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock per EOI-Appendix-8A, Bypassing Group 1 RPV Low Low Low Level Isolation Interlocks

Description: The candidate will perform operations necessary to bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock in accordance with EOI-Appendix-8A.

K/A 223001 SF5 PCS Primary Containment System and Auxiliaries A2.01; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of coolant accident (4.5)

k. JPM 673

Title: Manually Start ‘A’ And ‘C’ Fire Pumps per 0-OI-26, High Pressure Fire Protection System

Description: The candidate will perform operations necessary to manually start ‘A’ and ‘C’ Fire Pumps in accordance with 0-OI-26, High Pressure Fire Protection System.

K/A

286000 Fire Protection System A2.08; Ability to (a) predict the impacts of the following on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to actuate when required (3.3)

Facility: Browns Ferry NPP Date of Examination: 6/12/23

Exam Level: ☐ RO ☐ SRO-I ☒ SRO-U Operating Test No.: 23-04

Control Room Systems: @ 8 for RO, 7 for SRO-I, 2 or 3 for SRO-U, including 1 ESF			
System / JPM Title		**Type Code***	**Safety Function**
a. JPM 15	Maximize Control Rod Drive Injection to the Reactor Vessel per EOI-Appendix-5B, Injection System Lineup CRD	D, L, S	1
b. N/A			
c. JPM 756A	Inject to the Reactor per EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System	A, EN, N, S	4
d. N/A			
e. N/A			
f. N/A			
g. N/A			
h. JPM 55A	Emergency Vent Primary Containment per EOI Appendix-13, Emergency Venting Primary Containment	A, D, EN, L, P, S	9
In-Plant Systems: @ 3 for RO, 3 for SRO-I, 3 or 2 for SRO-U			
i. JPM 758	Bypass Reactor Core Isolation Cooling High RPV Water Level Trips per EOI-Appendix-16H, Bypassing RCIC High RPV Water Level Shutdown Interlocks	E, L, N, R	2
j. JPM 734	Bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock per EOI-Appendix-8A, Bypassing Group 1 RPV Low Low Low Level Isolation Interlocks	E, L, D	5
k. N/A			

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	Control Room	In-Plant	Total
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

   Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, "Heat Removal from Reactor Core," in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant's license level (see the table in step 1).

   For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.

   For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee's site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5.
   The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

   Apply the following specific task selection criteria:

   - At least one of the tasks shall be related to a shutdown or low-power condition.
   - Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee's operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee's operating procedures. At least one alternate path JPM must be new or modified from the bank.
   - At least one of the tasks conducted in the plant shall evaluate the applicant's ability to implement actions required during an emergency or abnormal condition.
   - At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

   If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

Code	License Level Criteria		
	RO	SRO-I	SRO-U
(A)lternate path	4–6	4–6	2–3
(C)ontrol room			
(D)irect from bank	≤ 9	≤ 8	≤ 4
(E)mergency or abnormal in-plant	≥ 1	≥ 1	≥ 1
(EN)gineered safety feature (for control room system)	≥ 1	≥ 1	≥ 1
(L)ow power/shutdown	≥ 1	≥ 1	≥ 1
(N)ew or (M)odified from bank (must apply to at least one alternate path JPM)	≥ 2	≥ 2	≥ 1
(P)revious two exams (randomly selected)	≤ 3	≤ 3	≤ 2
(R)adiologically controlled area	≥ 1	≥ 1	≥ 1
(S)imulator			

Senior Reactor Operator (Upgrade)

Job Performance Measures

a. JPM 15 **Title:** Maximize Control Rod Drive Injection to the Reactor Vessel per EOI-Appendix-5B, Injection System Lineup CRD

Description: The candidate will perform operations necessary to align the Control Rod Drive (CRD) Hydraulic System to maximize injection to the Reactor Vessel in accordance with EOI-Appendix 5B.

K/A: 201001 SF1 CRDH Control Rod Drive Hydraulic System A4.03; Ability to manually operate and/or monitor in the control room: CRD system flow control valve (3.6)

b. N/A

c. JPM 756A **Title:** Inject to the Reactor per EOI-Appendix-7L, Alternate RPV Injection System Lineup EHPM System

Description: The candidate will perform the actions necessary in accordance with EOI-Appendix-7L to start an Emergency High Pressure Makeup Pump (EHPM) to raise Reactor Water Level to (+)2 to (+)51 inches. The candidate will be required to take action to provide a power source to the EHPM in accordance with Attachment 3, Supplemental Diesel Generator Operation from MCR.

K/A 295009 Low Reactor Water Level AA1.02; Ability to operate and/or monitor the following as they apply to Low Reactor Water Level: Reactor Water Level Control (4.2)

d. N/A

e. N/A

f. N/A

g. N/A

h. JPM 55A

Title: Emergency Vent Primary Containment per EOI-Appendix-13, Emergency Venting Primary Containment

Description: The candidate will emergency vent Primary Containment in accordance with EOI-Appendix-13. The Suppression Chamber vent lineup will fail, and an alternate vent method must be used.

K/A

288000 SF9 PVS Plant Ventilation Systems A2.01; Ability to (a) predict the impacts of the following on the Plant Ventilation Systems and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High Drywell Pressure (3.6)

i. JPM 758

Title: Bypass Reactor Core Isolation Cooling High RPV Water Level Trips per EOI-Appendix-16H, Bypassing RCIC High RPV Water Level Shutdown Interlocks

Description: The candidate will perform operations necessary to bypass the Reactor Core Isolation (RCIC) High RPV Water Level trips in accordance with EOI-Appendix-16H.

K/A

217000 SF2 RCIC Reactor Core Isolation Cooling System A2.02; Ability to (a) predict the impacts of the following on the Reactor Core Isolation Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Turbine trips (4.0)

j. JPM 734

Title: Bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock per EOI-Appendix-8A, Bypassing Group 1 RPV Low Low Low Level Isolation Interlocks

Description: The candidate will perform operations necessary to bypass Primary Containment Isolation System Group 1 Low RPV Level Interlock in accordance with EOI-Appendix-8A.

K/A

223001 SF5 PCS Primary Containment System and Auxiliaries A2.01; Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of coolant accident (4.5)

k. N/A

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1&4

G2.4.4 (10CFR 55.41.10)

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295001G2.4.4	
Importance Rating	4.5	------

Proposed Question: **# 1**

Unit 1 is operating at 100% Rated Thermal Power (RTP), when 1A Recirc Pump trips. In accordance with 1-AOI-68-1, Recirc Pump Trip / Core Flow Decrease, the required Operator action is to _____.

A. insert a Manual Reactor SCRAM

B. insert Control Rods until Load Line is AT LEAST< 67%

C. raise Core Flow until Region II of the Power to Flow Map is exited

D. insert Control Rods until Load Line is < 74%

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that if both Recirc Pumps are tripped in MODES 1 or 2, then the Operator is required to SCRAM the Reactor.

B **CORRECT**: *(See attached)* In accordance with 1-AOI-68-1, Recirc Pump Trip/Core Flow Decrease, if a single Recirc Pump has tripped then if Load Line is greater than 67.0%, then IMMEDIATELY take actions to insert Control Rods to less than 67.0% Load Line.

C INCORRECT: Incorrect but plausible in that Core Flow is raised to greater than 45% in accordance with OI-68 for a single Recirc Pump. Additionally, in accordance with the Reactivity Control Plan Urgent Load Reduction actions for a Recirc Pump Trip, the Operator will change Core Flow with the Operating Recirc Pump to exit POWER/FLOW Region 2 and until Operating Recirc Pump drive flow is below 46,600 gpm.

D INCORRECT: Incorrect but plausible in that previous revisions of AOI-68-1A (now consolidated into AOI-68-1, current revision) required inserting Control Rods to less than 74% loadline.

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's ability to recognize abnormal indications that are entry-level conditions for abnormal operating procedures as it relates to partial or complete loss of forced Core Flow circulation due to a trip of a Recirc Pump. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to required actions for Recirc Pump Trip(s).

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-AOI-68-1, Rev. 3 (Attach if not previously provided)
1-AOI-68-1A, Rev. 5 (old revision)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.007 Obj. 28 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.007-19.k 001 #422
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

422. OPL171.007-19.K 001

Unit 1 is at 100% Reactor Power **AND** Core Flow is 92%. A trip of 1A Recirc Pump results in Operation in Region II of the Core Power to Flow Map.

Which ONE of the following completes the statement below?

The required action(s) in accordance with 1-AOI-68-1A, "Recirc Pump Trip / Core Flow Decrease," is (are) to **IMMEDIATELY** _____.

A. insert a Manual Reactor Scram

B. raise Core Flow until Region II of the Power to Flow Map is exited

C. insert Control Rods until Region II of the Power to Flow Map is exited

D. insert Control Rods until Load Line is < 74%; then, raise Core Flow to > 45%

Excerpts from 1-AOI-68-1: (Supports Distractor A)

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease	1-AOI-68-1 Rev. 0003 Page 7 of 14

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** both Recirc Pumps are tripped in modes 1 or 2, **THEN** (Otherwise **MARK** NA)

SCRAM the Reactor.

4.2 Subsequent Actions

CAUTION

[NER/C] Failure to restart Reactor Recirculation pumps in a timely manner may result in exceeding the differential temperature limit for pump start and subsequently require plant depressurization to avoid exceeding pressure-temperature limits for the reactor vessel. [SER 93-005]

[1] **IF** both Recirc Pumps are tripped in modes 1 or 2, **THEN** (Otherwise **MARK** NA)

PERFORM the following:

[1.1] **RESTART** affected Reactor Recirculation pumps. **REFERENCE** 1-OI-68.

[1.2] **IF** the ΔT between the Rx vessel bottom head temperature and the moderator temperature precludes restart of a Recirc pump,

OR

Forced Recirculation flow CANNOT be established for any reason, **THEN** (Otherwise **MARK** NA)

[1.2.1] **INITIATE** a plant cooldown to prevent exceeding the pressure limit for the Rx vessel bottom head temperature indicated on 1-PNL-9-47, REACTOR VESSEL METAL TEMPERATURE, 1-TR-56-4 pt. 10 and based on Tech Specs Figure 3.4.9-1.

[1.2.2] **INFORM** the Unit SRO, Tech Spec 3.4.1 requires the Reactor be placed in Mode 3 in 12 hours. **REFERENCE** 1-GOI-100-12A and Tech Specs 3.4.1.B.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** both Recirc Pumps are tripped in modes 1 or 2, **THEN** (Otherwise **MARK** NA)

SCRAM the Reactor.

4.2 Subsequent Actions

CAUTION

[NER/C] Failure to restart Reactor Recirculation pumps in a timely manner may result in exceeding the differential temperature limit for pump start and subsequently require plant depressurization to avoid exceeding pressure-temperature limits for the reactor vessel. [SER 93-005]

[1] **IF** both Recirc Pumps are tripped in modes 1 or 2, **THEN** (Otherwise **MARK** NA)

PERFORM the following:

[1.1] **RESTART** affected Reactor Recirculation pumps. **REFERENCE** 1-OI-68.

[1.2] **IF** the ΔT between the Rx vessel bottom head temperature and the moderator temperature precludes restart of a Recirc pump,

OR

Forced Recirculation flow CANNOT be established for any reason, **THEN** (Otherwise **MARK** NA)

[1.2.1] **INITIATE** a plant cooldown to prevent exceeding the pressure limit for the Rx vessel bottom head temperature indicated on 1-PNL-9-47, REACTOR VESSEL METAL TEMPERATURE, 1-TR-56-4 pt. 10 and based on Tech Specs Figure 3.4.9-1.

[1.2.2] **INFORM** the Unit SRO, Tech Spec 3.4.1 requires the Reactor be placed in Mode 3 in 12 hours. **REFERENCE** 1-GOI-100-12A and Tech Specs 3.4.1.B.

Supports Correct Answer B and Distractor C:

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease	1-AOI-68-1 Rev. 0003 Page 9 of 14

4.2 Subsequent Actions (continued)

[2.2] **IF** loadline is greater than 67.0%, **THEN**
(Otherwise **MARK** N/A)

IMMEDIATELY take actions to insert control rods to less than 67.0% loadline. **REFERENCE** 0-TI-464,Reactivity Control Plan and 1-OI-85.

[2.3] **IF** OPRM Upscale Trip Function is inoperable, **THEN**
(Otherwise **MARK** N/A)

PERFORM 1-SR-3.3.1.1.I, Core Thermal Hydraulic Stability.

[2.4] [NER/C] **WHEN** the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], **THEN**
(**MARK** N/A if Recirc Pump will be isolated for maintenance).

OPEN Recirc Pump 1A(1B) discharge valve, 1-FCV-068-0003(0079) as necessary to maintain Recirc Loop in thermal equilibrium.

[2.5] **RAISE** core flow to greater than 45%. **REFERENCE** 1-OI-68.

[2.6] **MAINTAIN** operating Recirc pump flow less than 46,600 gpm. **REFERENCE** 1-OI-68.

[2.7] [NER/C] **WHEN** plant conditions allow, **THEN**,
(Otherwise **MARK** N/A)

MAINTAIN operating jet pump loop flow greater than 41 x 10^6 lbm/hr (1-FI-68-46 or 1-FI-68-48). [GE SIL 517]

[2.8] **NOTIFY** Reactor Engineer to **PERFORM** the following:

[2.8.1] 1-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation

[2.8.2] 0-TI-248, Core Flow Determination in Single Loop Operation

Excerpt from Reactivity Control Plan: (Supports Distractor C)

BFN Unit 2	Reactivity Control Plan	

Attachment 7
(Page 1 of 5)
Reactivity Control Plan Form

BFN Unit: 2 **Valid Date(s):** Current **Reactivity Maneuver Plan #:** U2 Simulator

Are Multiple Activations Allowed: Yes (If yes, US may make additional copies)

Prepared by: *R. Engineer* / *current* (Reactor Engineer / Date)
Reviewed by: *2. R. Engineer* / *current* (Qualified Reactor Engineer / Date)

Approved by: *R. E Manager* / *current* (RE Manager / Date)
Concurrence: *W. C Risk* / *current* (WCC/Risk/Us SRO / Date)

Approved by: *O. P. Manager* / *current* (Ops Manager **OR** Supt. / Date)
Authorized by: *S. Manager* / *current* (Shift Manager / Date)

RCP Activated: _____ / _____ (Unit Supervisor / Date)
RCP Terminated: _____ / _____ (Unit Supervisor / Date)

Title of Evolution: **Urgent Load Reductions**

Purpose/Overview of Evolution: ***This plan is a contingency for the following maneuvers:***

Plan A: Recirculation Pump Trip Contingency Plans

Plan B: Urgent Load Reduction

Plan C: Planned Load Reduction

Maneuver Steps

Recirculation Pump Trip

A1. IF both pumps trip, SCRAM the reactor per 2-AOI-68-1A

For a single recirculation pump trip, IMMEDIATELY insert rod groups to lower rod line below 67%.

A2. Change core flow with the Operating recirculation pump to exit POWER/FLOW region 2 and until Operating Recirculation pump drive flow is below 46,600 gpm.

Urgent Load Reduction

B1. Lower power by reducing core flow until either the desired operating power level is reached or core flow is 55 - 60% of rated core flow. (If a recirculation pump trip is imminent, only lower using the pump to be tripped).

B2. Insert control rods per Shove Sheets until load line is between 48% and 55%.

B3. Lower recirculation pump speed to ~28% (~480 rpm).

B4. Insert control rods per Shove Sheets until desired power level is reached or SCRAM the reactor.

Planned Load Reduction

C1. Insert first 4 control rods total, per the shove sheets to lower reactor power.

C2. Lower power by reducing core flow until either the desired operating power level is reached or core flow is 55 - 60% of rated core flow.

C3. Insert control rods per Shove Sheets until Loadline is between 48% and 55%.

Excerpt from 1-AOI-68-1**A** (previous revision consolidated into current revision of 1-AOI-68-1): supports Distractor D

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease OPRMs Operable	1-AOI-68-1A Rev. 0005 Page 8 of 13

4.2 Subsequent Actions (continued)

NOTES

1) Step 4.2[2] through Step 4.2[17.3] apply to any core flow lowering event.
2) Power To Flow Map is maintained in 0-TI-248, Station Reactor Engineer and on ICS.
3) If a cell bypasses while a recirc pump is running, a drop of ≈200 rpm will occur. The drive will remain at the speed at which it stopped. Operator action is required to raise speed following a cell bypass.

[2] **IF** a single Recirc Pump has tripped, **THEN**

CLOSE tripped Recirc Pump 1A(1B) discharge valve 1-FCV-068-0003(0079). ☐

[3] [NER/C] **WHEN** the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], **THEN** (N/A if Recirc Pump will be isolated for maintenance).

OPEN Recirc Pump 1A(1B) discharge valve, 1-FCV-068-0003(0079) as necessary to maintain Recirc Loop in thermal equilibrium. ☐

[4] **IF** loadline is greater than 74%, **THEN** (Otherwise N/A)

IMMEDIATELY take actions to insert control rods to less than 74% loadline AND **REFER TO** 0-TI-464, Reactivity Control Plan and 1-OI-85. ☐

[5] **RAISE** core flow to greater than 45% in accordance with 1-OI-68. ☐

[6] **INSERT** control rods to exit regions if **NOT** already exited AND **REFER TO** 0-TI-464, Reactivity Control Plan and 1-OI-85. ☐

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

215004 (SF7 SRMS) Source-Range Monitor System

A4.02 (10CFR 55.41.7)

Ability to manually operate and/or monitor in the control room:

- SRMS recorder

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	215004A4.02	
Importance Rating	3.4	------

Proposed Question: **# 2**

A Reactor Startup is in progress on Unit 1.

In accordance with 1-GOI-100-1A, Unit Startup, Control Rods are limited to single notch withdrawal at the **(1)** count rate doubling.

Source Range Monitors are procedurally required to be fully withdrawn during the startup when **(2)**.

A. (1) fourth
(2) IRMs are on Range 3

B. (1) fourth
(2) the MODE SWITCH is in RUN

C. (1) fifth
(2) IRMs are on Range 3

D. (1) fifth
(2) the MODE SWITCH is in RUN

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) in accordance with 1-GOI-100-1A, Unit Startup, the fourth count rate doubling has been chosen as a starting point to limit Control Rod withdrawal to single notch movement. For second part, in accordance 1-GOI-100-1A, SRMs are fully withdrawn when IRMs are on Range 3 or above and indicating above their downscale trip point.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that in accordance with 1-OI-92, SRMs, numerous SRM trip signals can be bypassed when the REACTOR MODE SWITCH is in RUN.

C INCORRECT: First part is incorrect but plausible in that in accordance with 1-GOI-100-1A, states that when count rate doubles five times, criticality is imminent. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability to manually operate and/or monitor SRM recorders in the control room. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to the operation of Source Range Monitors.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): 1-GOI-100-1A, Rev. 72 (Attach if not previously provided)
1-OI-92, Rev.11

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.019 Obj. 9b (As available)

Question Source:
Bank # ILT EXAM BANK OPL171.059-04 005 #2143
Modified Bank # (Note changes or attach parent)
New

Question History:
Last NRC Exam

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

2143. OPL171.059-04 005

Unit 1 Startup is in progress in accordance with 1-GOI-100-1A, Unit Startup.

When is control rod withdrawal limited to single notch withdrawal and when may continuous rod withdrawal resume?

A. At the fourth SRM count rate doubling; once IRMs are middle of scale on Range 7.

B. At the fifth SRM count rate doubling; once IRMs are middle of scale on Range 7.

C. At the fourth SRM count rate doubling; once IRM/APRM overlap is verified.

D. At the fifth SRM count rate doubling; once IRM/APRM overlap is verified.

Excerpts from 1-GOI-100-1A:

5.4 Withdrawal of Control Rods while in Mode 2 (continued)

[2] **RECORD** SOURCE RANGE MONITORS readings in Step 5.4[1] on PIP 95-119 on Panel 1-9-5.

(R) _____ _____ _____
Initials Date Time
1st

(R) _____ _____ _____
Initials Date Time
Reactor Engineer

NOTE

[NER/C] A review of startup data has revealed that when count rate doubles five times, criticality is imminent. As an added precaution, the fourth count rate doubling has been chosen as a starting point to limit rod withdrawal to single notch movement. This requirement, along with close monitoring of neutron monitoring instrumentation should assure a slow controlled approach to criticality. Criticality should be expected at all times. [SOER 88-002]

[3] **CALCULATE** SRM count rate at which notch withdrawal limitations are to be imposed by multiplying pre-startup count rate, recorded in Step 5.4[1], by a factor of 16.

(R) _____ _____ _____
Initials Date Time
1st

(R) _____ _____ _____
Initials Date Time
Reactor Engineer

[4] **RECORD** results below and at Step 5.4[14] (N/A readings for inoperable SRM):

CHANNEL A LEVEL _____ cps
CHANNEL C LEVEL _____ cps
CHANNEL B LEVEL _____ cps
CHANNEL D LEVEL _____ cps

(R) _____ _____ _____
Initials Date Time

5.4 Withdrawal of Control Rods while in Mode 2 (continued)

NOTE

Once required, Control rod withdrawal is limited to single-notch withdrawal until Reactor power is in the heating range.

CAUTIONS

1) Near end of core life, criticality may occur before five doublings due to a stronger top peak flux and buildup of plutonium.

2) [NER/C] When rod movement is restricted to notch withdrawal, failure to stop at each notch position may result in high notch worth. [GE SIL 316]

[14] **WHEN** SRMs indicate the calculated values recorded below (N/A for inoperable SRM):

CHANNEL A LEVEL _____ cps

CHANNEL C LEVEL _____ cps

CHANNEL B LEVEL _____ cps

CHANNEL D LEVEL _____ cps

THEN

START single-notch withdrawal of control rods.

(R)	_____	_____	_____
	Initials	Date 1st	Time
(R)	_____	_____	_____
	Initials	Date Reactor Engineer	Time

Excerpt from 1-OI-92:

BFN Unit 1	Source Range Monitors	1-OI-92 Rev. 0011 Page 20 of 20

Attachment 1
(Page 1 of 1)

SRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
SRM High	6.8 X 10^4 counts per second	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Downscale	5 counts per second	Rod block unless IRMs on range 3 (or higher) or REACTOR MODE SWITCH in RUN
SRM Detector Wrong Position	145 counts per second	Rod block unless detector full-in, IRMs on range 3 (or higher), or REACTOR MODE SWITCH in RUN
SRM High-High	2 x 10^5 counts per second	Scram if shorting links removed

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215005 (SF7 PRMS) Average Power Range Monitor / Local Power Range Monitor	Tier #	2	------
K6.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Average Power Range Monitor / Local Power Range Monitor System:	K/A #	215005K6.02	
• TIP	Importance Rating	2.9	------

Proposed Question: **# 3**

Unit 1 is operating at 100% RTP with a Traversing In-Core Probe (TIP) trace in progress when a PCIS Group 8 Isolation occurs.

TIP trace is used to maintain the calibrated accuracy of **(1)** and as a result of the Isolation signal, in accordance with 1-AOI-64-2E, TIP Isolation, TIPs retract to the in-shield position and their respective **BALL** Valves **(2)** automatically close.

A. (1) IRMs
(2) will

B. (1) IRMs
(2) will NOT

C. (1) LPRMs
(2) will

D. (1) LPRMs
(2) will NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that IRMs monitor and record core neutron flux levels between the startup range and the power range during Reactor startup and shutdown. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that numerous systems/components are associated with PCIS Group Isolations. Each Group contains various isolation signals which impact numerous automatically component repositioning/isolating which are easily and often confused.

C **CORRECT:** *(See attached)* In accordance with 1-OI-94, TIP System, TIP calibration (trace) system is used to maintain the calibrated accuracy of LPRMs. LPRMs are hard wired to each APRM channel for APRM power detection and for OPRM detection and suppression of thermal hydraulic instabilities. For second part, in accordance with 1-AOI-64-2E, TIP Isolation, if a Group 8 Isolation occurs, then the TIP Ball Valves receive an automatic closure signal or close after TIP probes are withdrawn to their IN-SHIELD position.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the effect of the Traversing In-Core Probe System malfunctions, or component malfunctions on the Average Power Range Monitor / Local Power Range Monitor System. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to PCIS Group Isolations.

Technical Reference(s): 1-OI-94, Rev. 20 (Attach if not previously provided)
1-AOI-64-2e, Rev. 1
OPL171.023, Rev. 7
OPL171.020, Rev. 13U1
OPL171.148, Rev. 18U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.023 Obj. 1 (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK OPL171.023-07 002 #926 (Note changes or attach parent)
New	

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

926. OPL171.023-07 002

Which ONE of the following completes the statement below?

The Traversing In-Core Probe System (TIPS) probes are ___(1)___ and are used to calibrate the __(2)__.

A. (1) gamma sensitive ion chambers
(2) source range and intermediate range monitors (SRMs/IRMs)

B. (1) Geiger-Mueller Tubes
(2) local power range monitors (LPRMs)

C. (1) Geiger-Mueller Tubes
(2) source range and intermediate range monitors (SRMs/IRMs)

D✓ (1) gamma sensitive ion chambers
(2) local power range monitors (LPRMs)

Excerpts from 1-OI-94:

3.0 PRECAUTIONS AND LIMITATIONS

A. To prevent accidental exposure to personnel, immediately evacuate the area if the TIP drive area radiation monitor alarms.

B. Only one TIP at a time should be operated when maintenance is being performed in TIP drive area.

C. Continuous voice communication should be maintained between TIP operator or maintenance personnel in control room and drive mechanism area while maintenance is being performed and TIP detector driving is necessary.

D. [NER/C] Length of time detector is left in core should be minimized to limit activation of detector and cable. [GE SIL-166]

E. [NER/C] When TIP System operation is NOT desired, detectors should be retracted and stored in chamber shield with ball valves closed. [GE SIL-166]

F. Detector may be stored at Indexer for approximately 24 hours or as directed by RP personnel following removal of TIP detector from core region PROVIDED the automatic withdrawal and isolation capability of TIPs is maintained. Temporary storage of detector at the Indexer is allowed only for ALARA concerns and to prevent unnecessary masking of multiple inputs to annunciator RX BLDG AREA RADIATION HIGH 1-RA-90-1D (1-XA-55-3A, Window 22).

G. The following conditions are to be ensured when TIP detector is stored at the Indexer (Parked):

1. Drive remains energized.

2. Ball Valve operation normal and remains energized.

3. Movement in reverse direction is NOT prohibited (problem may be indicated by one or all of the following):

   "In-shield" logic sealed in ("In-shield" message illuminated on display or "R" contactor de-energized due to K2 "in-shield" relay dropped out).

   Drive mechanically bound.

   Self-Test Fault

H. [NER/C] Upon receipt of a PCIS signal (low reactor water level or high drywell pressure), verify any detector inserted beyond its shield chamber automatically shifts to reverse mode and begins withdrawal. Once in shield, ball and purge valves close. [GE SIL-166] Ball valve can NOT reopen until PCIS is reset on Panel 1-9-4 and manual reset of TIP ISOLATION RESET pushbutton, 1-HS-94-7D/S2, located on Panel 1-9-13.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

X. Local AUTO-TIP operations, Section 6.2, are performed for NUMAC testing only. Data retrieved from a local AUTO-TIP scan is not used by the Core Monitoring System program for LPRM gain adjustments. The ATCU may be used locally to move the probe to specific core locations, to obtain flux data, to obtain torque data, or to perform a checkout of the TIP calibration System. Data for LPRM gain adjustments is obtained by performing Section 6.1.

Y. The ATCU initiates an automatic TIP probe withdrawal to the PARKED position if the probe has been in the reactor core for more than 10 minutes. (OPERATE mode only)

Z. The DC drive motor provides a measured torque, as identified on the upper display of the ATCU. If torque exceeds 200 lbf-in, the motor is stopped preventing possible damage to the probe or helical cable. Torque readings within 50 lbf-in of the setpoint are displayed in inverse video.

AA. In the unlikely event that a failure of the NUMAC ATCU (Automated TIP Control Unit) prevents the ball valve from closing, coincident with the need for primary containment isolation, and the operator is unable to correct the situation the ball valve can be closed by manually removing power to the NUMAC ATCU chassis ie (breaker/fuse), which will de-energize the ball valve relay. If this action fails to restore isolation, manual actuation of the TIP isolation shear valve will restore isolation, but should be considered a last resort. [GE SIL-662]

Attachment 1
(Page 1 of 1)

LPRM to TIP Machine Assignment

Indexer Position	LPRM Location	Indexer Position	LPRM Location	Indexer Position	LPRM Location
A-1	16-41	C-1	16-09	E-1	40-41
A-2	16-49	C-2	24-09	E-2	32-41
A-3	16-57	C-3	24-17	E-3	32-49
A-4	08-49	C-4	24-25	E-4	24-41
A-5	24-57	C-5	32-25	E-5	48-49
A-6	24-49	C-6	32-17	E-6	48-41
A-7	32-57	C-7	32-09	E-7	56-41
A-8	40-57	C-8	40-09	E-8	56-33
A-9	40-49	C-9	SPARE	E-9	Spare
A-10(1)	32-33	C-10(1)	32-33	E-10(1)	32-33
B-1	08-17	D-1	40-17		
B-2	08-25	D-2	40-25		
B-3	08-33	D-3	40-33		
B-4	08-41	D-4	48-33		
B-5	16-33	D-5	48-25		
B-6	16-25	D-6	48-17		
B-7	16-17	D-7	48-09		
B-8	24-33	D-8	56-25		
B-9	Spare	D-9	56-17		
B-10(1)	32-33	D-10(1)	32-33		

(1) Common guide tube

Excerpt from 1-AOI-64-2e:

BFN Unit 1	Traversing Incore Probe Isolation	1-AOI-64-2e Rev. 0001 Page 3 of 7

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 8, Traversing Incore Probe (TIP) Isolation and detection of a reactor coolant leak in a TIP guide tube.

2.0 SYMPTOMS

NOTE

A PCIS Group 8 isolation is initiated by either of the following:

- Reactor Vessel Water Level Low
- Drywell High Pressure

[1] Any one or more of the following annunciators in alarm:

- RX VESSEL WTR LEVEL LOW HALF SCRAM (1-XA-55-4A, Window 2) in alarm (Group 8 Isolation).
- DRYWELL PRESSURE HIGH HALF SCRAM (1-XA-55-4A, Window 8) in alarm (Group 8 Isolation).
- AIR PARTICULATE MONITOR RADIATION HIGH 1-RA-90-50A (1-XA-55-3A, Window 2) in alarm (indicative of TIP guide tube leak).
- RX BLDG AREA RADIATION HIGH 1-RA-90-1D (1-XA-55-3A, Window 22) in alarm (indicative of TIP guide tube leak).

3.0 AUTOMATIC ACTIONS

[1] **IF** a Group 8 isolation occurred, **THEN** the following are automatic actions:

- **IF** TIP probes are outside their shields, **THEN** TIP withdrawal initiated to IN-SHIELD position.
- TIP Ball Valves receive a close signal, or close after TIP probes are withdrawn to their IN-SHIELD position.
- TIP Purge Valves closes (no indications provided).

Excerpt from OPL171.023 Lesson Plan:

OPL171.023 , Traversing In-Core Probe System, Rev.# 07

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
I. General Introduction 1. This lesson plan will provide information dealing with the function and operation of the Traversing In-Core Probe (TIP) System at Browns Ferry. The TIP system is used periodically to monitor and record the axial thermal neutron flux for the purpose of calibrating the local power range monitors. 2. The TIP systems for Unit 2/3 are identical. Unit 1 TIP system is different in the type of control system utilized (NUMAC). 3. Upon completion of this lesson, the operator will demonstrate satisfactory knowledge of the Traversing In-Core Probe system (TIP) by scoring at least 80% on a written exam. Initial License Candidates and Licensed Operators will additionally be evaluated on the practical application of the learning objectives by simulator scenarios. Non licensed operators may be evaluated on the practical aspects covered in this lesson by performing in-plant evaluations as applicable.	Information in this column is optional and for use at the instructor's discretion.
II. Presentation B. Purpose 1. Five identical traversing in-core probes (TIP) are used to: a. Measure and record the axial thermal neutron flux profile at 43 radial locations. b. Calibrate the fixed in-core detectors (local power range monitors (LPRMs). 2. Loss of periodic calibration using the TIPs would result in LPRM inaccuracy and PRNM operability issues.	Objective numbers are the same for ILT/LOR//NLO/NLOR except for objective 8 which only applies to NLO/NLOR. Objective 1

Excerpt from OPL171.148 Lesson Plan:

II. Presentation

A. Power Range Neutron Monitor System (PRNM)

1. Overview
   a. The PRNM equipment consists of four APRM channels and two RBM channels.
   b. Each APRM channel consists of one APRM and one LPRM instrument with their support equipment.
   c. Each APRM channel has the capacity to perform the OPRM functions, as well.
      1) Therefore, there are four OPRM channels, one contained within each APRM channel.
   d. Each RBM channel consists of one RBM instrument with their support equipment.
2. FSAR Safety and Performance Objectives
   a. Safety Objective
      1) Detect conditions in the core that threaten the overall integrity of the fuel barrier due to excessive power generation.
      2) Provides signals to the Reactor Protection System, so that the release of radioactive materials from the fuel barrier is limited.
   b. Power Generation Objective
      1) Provide information for the efficient, expedient operation and control of the reactor.
      2) Detect conditions that could lead to local fuel damage and provide signals that can be used to prevent such damage, so that plant availability is not reduced.
3. Local Power Range Monitoring System (LPRMs)
   a. The LPRMs are hard wired to each APRM channel for APRM power detection and for OPRM detection and suppression of thermal hydraulic instabilities.
4. Average Power Range Monitoring System (APRMs)
   a. Provides average power level indication, alarms, and automatic reactor scram functions.
   b. One LPRM in each LPRM string is assigned to each APRM.

Excerpt from OPL171.020 Lesson Plan: Supports Distractors A1, B1

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
d) Consequences - a manual scram occurred after a human performance error and equipment failure.	
III. Presentation	
A. General Description	
1. Purpose	
a) Monitor and record core neutron flux levels between the startup range and the power range during reactor startup and shutdown.	NLO/NLOR Objective 1, 4
b) Provide protective trips (control rod blocks and reactor scrams) to protect against rapid changes in reactor power which might otherwise result in fuel damage.	
2. Safety Purpose (from FSAR Section 7.5)	
a) The safety objective of the Neutron Monitoring System is to detect conditions in the core that threaten the overall integrity of the fuel barrier due to excessive power generation and to provide signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier is limited.	
3. Power Generation Objective	
a) The power generation objective of the Neutron Monitoring System is to provide information for the efficient, expedient operation and control of the reactor.	
b) Specific power generation objectives of the Neutron Monitoring System are to detect conditions that could lead to local fuel damage and to provide signals that can be used to prevent such damage, so that plant availability is not reduced.	
4. Basic Components	
a) IRM Detector (8 total per Unit)	
b) IRM Detector drive (8 total per Unit)	

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff

A3.01 (10CFR 55.41.7)

Ability to monitor automatic operation of the Primary Containment Isolation System/Nuclear Steam Shutoff, including:

- System indicating lights and alarms

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	223002A3.01	
Importance Rating	4.0	------

Proposed Question: **# 4**

The following conditions exist on Unit 1:

- A Loss of Coolant Accident (LOCA) is in progress
- Drywell Pressure is 2.2 psig and rising
- Reactor Water Level is (-) 40 inches and lowering
- Main Steam Tunnel Temperature is 175 °F

Primary Containment Isolation System (PCIS) Groups _____ have received an automatic isolation signal.

A. 1, 3, 6, and 8

B. 3, 4, 5, and 6

C. 1, 2, 5, and 8

D. 2, 3, 6, and 8

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that a PCIS Group 1 Isolation automatically occurs when Main Steam Tunnel Temperature rises to 189 °F resulting in the MSIVs to automatically close. This would be indicated by MAIN STEAM LINE LEAK DETECTION TEMPERATURE HIGH (1-9-3D, Window 24), alarming.

B INCORRECT: Incorrect but plausible in that PCIS Groups 4(5) Isolations occur when HPCI(RCIC) reach their respective automatic isolation setpoints. For HPCI(RCIC), both isolate at Steamline Space Torus Area Temperature of 165 °F or HPCI(RCIC) Pump Room Temperature of 185 °F (165 °F). This would be indicated by MAIN STEAM LINE LEAK DETECTION TEMPERATURE HIGH (1-9-3D, Window 24), alarming.

C INCORRECT: Incorrect but plausible in that PCIS Groups 1(5) Isolation setpoints have not been reached (*See A(B)).*

D **CORRECT**: (*See attached)* In accordance with REACTOR VESSEL WATER LEVEL LOW HALF SCRAM (1-9-4A, Window 2), when Reactor Water Level lowers to less than or equal to 2 inches, an automatic Reactor SCRAM occurs along with PCIS Groups 2, 3, 6 and 8 Isolations if both RPS A and B related sensors actuate.

RO Level Justification: Tests the candidate's ability to monitor automatic operation of the Primary Containment Isolation System as it applies to system indicating lights and alarms. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution as it relates to Primary Containment Isolation System setpoints and indications.

Technical Reference(s): 1-ARP-9-4A, Rev. 33 (Attach if not previously provided)
1-ARP-9-3D, Rev. 34
1-ARP-9-3F, Rev. 29

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.017 Obj. 2a,d,e,n (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.017-01 002 #726
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of bank question:

726. OPL171.017-01 002

The following conditions exist on Unit 1:

- A Loss of Coolant Accident is in progress
- Drywell pressure is 2.2 psig
- Reactor water level is (-) 40 inches
- Main Steam Tunnel temperature is 175° F

Which ONE of the following sets of PCIS groups should have received automatic isolation signals?

A. Groups 1, 3, 6, 8

B. Groups 3, 4, 5, 6

C. Groups 1, 2, 5, 8

D. ✔ Groups 2, 3, 6, 8

Excerpt from 1-ARP-9-4A:

BFN Unit 1	Panel 9-4 1-XA-55-4A	1-ARP-9-4A Rev. 0033 Page 5 of 48

RX VESSEL
WTR LEVEL LOW
HALF SCRAM

2

(Page 1 of 1)

Sensor/Trip Point:

1-LIS-003-0203 A, B, C & D — 2" above inst. zero (530" above vessel zero.)

Sensor Location:
1-LIS-003-0203A, 1-PNLA-009-0083, Auxiliary Instrument Room
1-LIS-003-0203B, 1-PNLA-009-0084, Auxiliary Instrument Room
1-LIS-003-0203C, 1-PNLA-009-0085, Auxiliary Instrument Room
1-LIS-003-0203D, 1-PNLA-009-0086, Auxiliary Instrument Room

Probable Cause:
A. Low level in the reactor.
B. SI/SR in progress.

Automatic Action:
A. Half scram if one sensor actuates.
B. Reactor scram and group 2, 3, 6 and 8 isolations, if one sensor per channel actuates.

Operator Action:
A. **CONFIRM** alarm by multiple indications.
B. **IF** level is less than or equal to 2.0" **AND** reactor has NOT scrammed, **THEN**
PERFORM the following:
1. **SCRAM** the reactor.
2. **ENTER** 1-EOI-1 FLOWCHART.

C. **DISPATCH** personnel to 1-LIS-003-0203 A, B, C, and D to check for abnormal condition.
D. **IF** alarm is NOT valid or initiating condition is corrected, **THEN**
PERFORM the following:
1. **RESET** Half Scram with SRO permission.
2. **REFER TO** 1-OI-99.

E. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function.
REFER TO NPG-SPP-18.3.5.

References:
1-730E915-9, 10
FSAR Sections 7.2.3.1, 7.2.3.5, 13.6.2

Excerpts from 1-ARP-9-3D: Supports Distractors A, C

BFN Unit 1	Panel 9-3 XA-55-3D	1-ARP-9-3D Rev. 0034 Page 29 of 42

MAIN STEAM LINE
LEAK DETECTION
TEMP HIGH
1-TA-1-60

24

(Page 1 of 2)

Sensor/Trip Point:

1-TE-1-60A (1-TIS-1-60A)	170°F
1-TE-1-60B through D (1-TR-69-29)	160°F

Sensor Location:

1-TE-1-60A	Main Steam Tunnel
1-TE-1-60B	Main Steam Line Steam Vault Area Elev 565
1-TE-1-60C	Main Steam Line Bypass Vlv Area Elev 586
1-TE-1-60D	Main Steam Line Control Vlv Area Elev 617

Probable Cause:

A. Main Steam, RWCU, Feedwater, RCIC, or HPCI discharge (only with HPCI in service and elevated Suppression Pool water Temp.) line break.
B. TB or RB Coolers out of service
C. Sensor malfunction
D. Steam Vault Exhaust Booster Fan out of service.

Automatic Action:

Impending MSIV Isolation at 189°F area temp.

Operator Action:

NOTE

The following steps may be performed in any order or concurrently, as necessary.

A. **CHECK** the following temperature indicators:
- Main Steam temperature elements on LEAK DETECTION SYSTEM TEMPERATURE Recorder, 1-TR-69-29 (Points 13-15) on Panel 1-9-21,
- MN STEAM TUNNEL TEMP, 1-TIS-1-60A on 1-9-3,
- RWCU Piping in the Main Steam Tunnel temperature indicators, 1-TIS-69-834A-D, Aux Inst Room 1-PNLA-009-0083(84)(85)(86) or "HPTURB" mimic on ICS.

B. **CHECK** the following flow indications:
- MAIN STEAM LINE FLOW A(B, C, D), 1-FI-46-1(2, 3, 4) on Panel 1-9-5.
- RFW FLOW LINE A(B) 1-FI-3-78A(B) on Panel 1-9-5.
- RFP 1A(B, C) flow indicators, 1-FI-3-20(13, 6) on Panel 1-9-6.

Continued on Next Page

Supports Distractor B

BFN Unit 1	Panel 9-3 XA-55-3D	1-ARP-9-3D Rev. 0034 Page 13 of 42

RCIC STEAM LINE
LEAK DETECTION
TEMP HIGH
1-TA-71-41

10

(Page 1 of 1)

Sensor/Trip Point:

1-TE-071-0041A through -0041D	138°F

Sensor Location:	1-TE-071-0041A	RCIC Turbine Area	Elev 519
	1-TE-071-0041B	RCIC Steam Supply Area	Elev 519
	1-TE-071-0041C	RCIC Steam Supply Area	Elev 519
	1-TE-071-0041D	RCIC Steam Supply Area	Elev 519

Probable Cause:

A. Steam line leak
B. Area cooler out of service

Automatic Action:

If temperature continues to rise it will cause a RCIC Isolation @ 165°F in Torus area or RCIC Pump Room temp.

Operator Action:

A. **CHECK** following instrumentation:
   1. RCIC temperature elements on LEAK DETECTION SYSTEM TEMPERATURE Recorder, 1-TR-69-29 (Points 9-12) on 1-9-21.
   2. CS / RCIC ROOM EL. 519 RX BLDG, 1-RI-90-26A, on Panel 1-9-11.
   3. RX REFUEL ZONE EXHAUST RADIATION, 1-RR-90-144, on Panel 1-9-2

B. **IF** RCIC is **NOT** in service **AND** 1-FI-71-1A(B), RCIC STEAM FLOW indicates flow, **THEN**
   ISOLATE RCIC AND check temperatures lowering.
C. **IF** high temperature is confirmed, **THEN**
   ENTER 1-EOI-3 Flowchart.
D. **IF** a rise in radiation level is indicated, **THEN**
   NOTIFY RADCON and **REFER TO** OI-90.
E. **DISPATCH** personnel to investigate consistent.
F. **REFER TO** Tech Spec Sections 3.3.6.1, 3.5.3.
G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References:

1-45E620-2-2 1-47E610-71-1 GE 920D351
Tech Spec 3.3.6.1, 3.5.3

Excerpt from 1-ARP-9-3F: Supports Distractor B

BFN Unit 1	Panel 9-3 XA-55-3F	1-ARP-9-3F Rev. 0029 Page 14 of 41

HPCI
LEAK DETECTION
TEMP HIGH
1-TA-73-55
10

(Page 1 of 1)

Sensor/Trip Point:
1-TR-69-29

Point 16 (TE-73-55A)	150°F
Point 17 (TE-73-55B)	150°F
Point 18 (TE-73-55C)	150°F
Point 19 (TE-73-55D)	150°F

Sensor Location: Panel 1-9-21 (1-TR-69-29)
Main Control Room El. 617'

Probable Cause: A. HPCI Steam line leak

Automatic Action: Continued rise will cause the following valves to isolate (at Steamline Space Temperature of 165°F Torus Area or 185°F HPCI Pump Room):

- HPCI STEAM LINE INBD ISOL VALVE, 1-FCV-73-2
- HPCI STEAM LINE OUTBD ISOL VALVE, 1-FCV-73-3
- HPCI STEAM LINE WARM-UP VALVE, 1-FCV-73-81

NOTE

SUPP CHBR WTR LVL, 1-LT-64-159B, may give erroneous indications due to high temperatures experienced by the instrument during a HPCI high-energy line break in Secondary Containment.

Operator Action:

A. **CHECK** HPCI temperature indications on LEAK DETECTION SYSTEM TEMPERATURE recorder, 1-TR-69-29, points 16 through 19, on Panel 1-9-21.
B. **IF** high temperature is confirmed, **THEN** **ENTER** 1-EOI-3 Flowchart.
C. **CHECK** HPCI RM RAD MONITOR, 1-RI-90-24, and WEST RHR PUMP RM RAD MONITOR, 1-RI-90-25A, on Panel 1-9-11 and **NOTIFY** RADCON if necessary.
D. **DISPATCH** personnel to investigate for leaks consistent with ALARA considerations in HPCI turbine area (El. 519) and HPCI steam supply area (El. 550).
E. **EVALUATE** equipment associated with this alarm that is out of service to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References: 1-45E620-1-2 1-47E610-73-1

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

292007 Reactor Theory - Fuel Depletion and Burnable Poisons

K1.03 (10CFR 55.41.1)

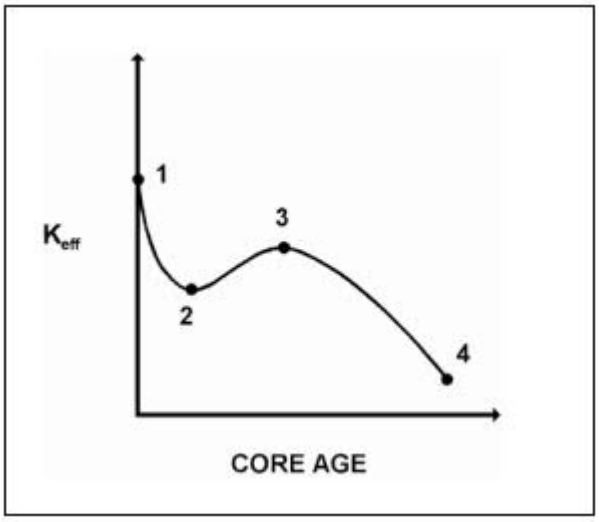
Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points.

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292007K1.03	
Importance Rating	2.7	------

Proposed Question: **# 5**

Referring to the drawing of effective neutron multiplication factor (k_{eff}) versus Core Age, which **ONE** of the following completes the statement below?

The major cause for the change in k_{eff} from point 3 to point 4 is the _____.



A. depletion of fuel

B. depletion of Control Rods

C. burnout of burnable poisons

D. burnout of fission product poisons

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* From point 3 to 4, fuel depletion and fission product poison buildup (other than xenon and samarium) continuously insert negative reactivity. Excess reactivity (k_{excess}) lowers at a rising rate to the end of the fuel cycle.

B INCORRECT: Incorrect but plausible in that Control Rods may deplete over their service life but do not deplete at a rate that coincides with the curve between points 3 and 4.

C INCORRECT: Incorrect but plausible in that burnable poisons are loaded into the core; however, the burnout of the burnable poisons causes k_{excess} to increase from 25 days to about one-third to one-half of the fuel cycle (points 2 to 3). This occurs because the burnable poison depletes faster than the fuel.

D INCORRECT: Incorrect but plausible in that following a Reactor startup, during the first 25 days of the cycle (points 1 to 2), fission product poisons (samarium and xenon) build to equilibrium levels. Excess reactivity (k_{excess}) decreases until the equilibrium concentrations at point 2 are established.

RO Level Justification: Tests the candidate's knowledge of Generic Fundamentals, specifically why k-effective changes over core life. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): BWR Reactor Theory, Ch. 7, Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **k_{eff} vs. Core Age Graph**

Learning Objective: GFES Reactor Theory, Ch. 7, Obj. 3 (As available)

Question Source:

Bank #	BFN 2204 NRC #72
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2022

Question Cognitive Level:

Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Copy of Bank Question:

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

292007 Reactor Theory - Fuel Depletion and Burnable Poisons

K1.03 (10CFR 55.41.1)

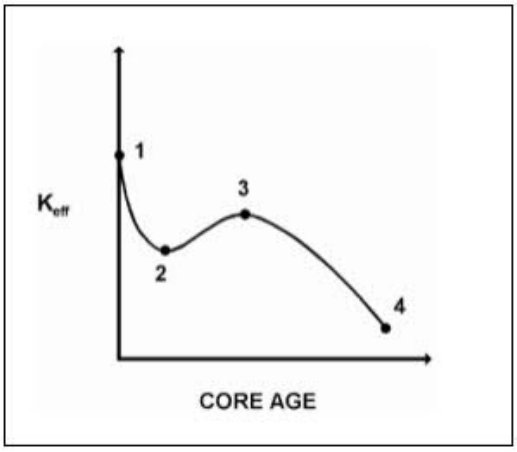
Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points.

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292007K1.03	
Importance Rating	2.7	------

Proposed Question: **# 72**

Referring to the drawing of effective neutron multiplication factor (k_{eff}) versus Core Age, which **ONE** of the following completes the statement below?

The major cause for the change in k_{eff} from point 3 to point 4 is the _____.



A. depletion of fuel

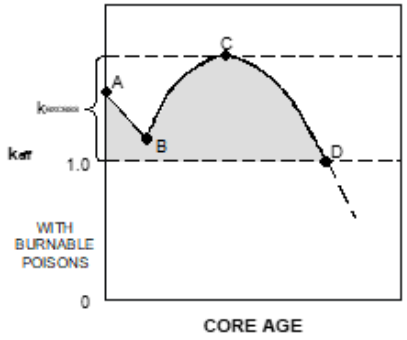
B. depletion of Control Rods

C. burnout of burnable poisons

D. burnout of fission product poisons

Proposed Answer: **A**

Excerpt from BWR GFES Reactor Theory, Ch. 7:

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Objective 3 **Figure 7-1 / TP 7-8**  	B. Figure 7-1 illustrates total core k_{eff} over core life. The combination of control rods and burnable poisons controls positive reactivity in core due to excess fuel (k_{excess}). C. For this discussion, assume that reactor is taken to 100% power at beginning of core life and maintained at that power level until EOC 1. To maintain constant power, operator must change control rod density in response to negative and positive reactivity changes that occur in core during cycle. D. During first 25 days of cycle (points A to B), fission product poisons (samarium and xenon) build to equilibrium levels 1. Excess reactivity (k_{excess}) decreases until equilibrium concentrations at point B are established E. The burnout of burnable poisons causes k_{excess} to increase from 25 days to about one-third to one-half of fuel cycle (points B to C) 1. This occurs because burnable poison depletes faster than fuel. As core operation approaches point C, rate of fuel depletion and rate of poison depletion become same 2. At this point, core has smallest value of shutdown margin. F. From point C to D, fuel depletion and fission product poison buildup (other than xenon and samarium) continuously insert negative reactivity

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
239001 (SF3, SF4 MRSS) Main and Reheat Steam	Tier #	2	------
K2.01 (10CFR 55.41.7)	Group #	2	------
Knowledge of electrical power supplies to the following:	K/A #	239001K2.01	
• Main steam isolation valve solenoids	Importance Rating	3.6	------

Proposed Question: **# 6**

The Main Steam Isolation Valve (MSIV) AC solenoids are **NORMALLY** powered by _____.

A. I & C 'A'

B. Unit Preferred

C. Plant Preferred

D. Reactor Protection System

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that I & C 'A' is a 120V AC system that provides power for the operation and indication of numerous systems, including valve positions for Safety Related Systems. The Browns Ferry 120V AC System is complex and easily and often confused, especially the 120V AC power supplies.

B INCORRECT: Incorrect but plausible in that Unit Preferred is the power supply to the Main Steam Isolation Valve (MSIV) AC solenoids when the transfer switch at the Backup Control Panel (1-9-25-32) is placed in EMERGENCY. The Browns Ferry 120V AC System is complex and easily and often confused, especially the 120V AC power supplies.

C INCORRECT: Incorrect but plausible in that Plant Preferred is a 120V AC system that provides power for the operation and indication of numerous systems. The Browns Ferry 120V AC System is complex and easily and often confused, especially the 120V AC power supplies.

D **CORRECT:** *(See attached)* **In** accordance with 1-OI-1, Main Steam System, and 1-OI-99, Reactor Protection System, the power supply to the Main Steam Isolation Valve (MSIV) AC solenoids is the Reactor Protection System (RPS).

RO Level Justification: Tests the candidate's knowledge of the electrical power supplies to the Main Steam Isolation Valve (MSIV) AC solenoids. This question is rated as Memory due to the requirement to strictly recall facts in relation to the power supplies to the MSIV AC solenoids.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): 1-OI-1, Rev. 17 (Attach if not previously provided)
1-OI-99, Rev. 64
1-AOI-57-4, Rev. 39
1-AOI-57-5A, Rev. 55
0-AOI-57-3, Rev. 60
OPL171.009, Rev. 17U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.009, Obj. 16 (As available)

Question Source:

Bank #	
Modified Bank #	BFN 1306 AUDIT #55
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Copy of Bank Question:

Proposed Question: **# 55**

Control for the Main Steam Isolation Valves (MSIVs) and Main Steam Relief Valves (MSRVs) has been transferred to panel 25-32.

Based on the above condition, which ONE of the following completes the statement below?

The Main Steam Isolation Valve (MSIV) solenoids are powered from ___(1)___, and the Main Steam Relief Valve (MSRV) solenoids are powered from ___(2)___.

A. (1) 250V DC and 120VAC RPS
(2) 250V DC

B. (1) 250V DC
(2) 120VAC Unit Preferred and 250V DC

C. (1) 120VAC Unit Preferred
(2) 120VAC Unit Preferred

D. (1) 250V DC and 120VAC Unit Preferred
(2) 250V DC

Proposed Answer: **D**

Excerpt from 1-OI-1:

BFN Unit 1	Main Steam System	1-OI-1 Rev. 0017 Page 19 of 67

4.0 PRESTARTUP/STANDBY READINESS REQUIREMENTS (continued)

[1.2] The following system lineup checklists are current:

A. External Attachment 1, Main Steam System Valve Lineup Checklist, Unit 1

B. External Attachment 2, Main Steam System Panel Lineup Checklist, Unit 1

C. External Attachment 3, Main Steam System Electrical Lineup Checklist, Unit 1

D. External Attachment 4, Main Steam System Instrument Inspection Checklist, Unit 1

[2] **ENSURE** the following support system requirements are satisfied:

A. Control Air System is in service to provide control air pressure for air-operated valves. REFER TO 0-OI-32.

B. Drywell Control Air System is in service to provide pneumatic pressure for the SRVs and MSIVs in the drywell, REFER TO 1-OI-32A.

C. Primary Containment Isolation System (PCIS) group one is reset. REFER TO Section 8.1.

D. Reactor Protection System energized to provide power to MSIV solenoids. REFER TO 1-OI-99.

E. Condenser Cooling Water (CCW) System is in service to provide cooling water to the condenser. REFER TO 1-OI-27.

Excerpts from 1-OI-99:

Attachment 1
(Page 3 of 4)
RPS Bus A or B Power Transfer

B. Loss of power to RPS Bus A only will result in the following events in addition to those listed for RPS Bus A or B power loss

VALVE	FUNCTION/SYSTEM	ACTION
1-FCV-074-0048	RHR SHUTDN CLG SUCT INBD VLV	CLOSES
1-FCV-074-0053	RHR SYSTEM I LPCI INBD INJECT VALVE	CLOSES
1-FCV-075-0057	PSC PUMP SUCTION INBD ISOL	CLOSES
1-FCV-077-0015A	DRYWELL EQ DR SUMP INBD FCV	CLOSES
1-FCV-077-0002A	DRYWELL FD SUMP INBD ISOLATION VLV	CLOSES
1-FCV-069-0001	RWCU INBD SUCT ISOLATION VALVE	CLOSES
1-FCV-069-0002	RWCU OUTBD SUCT ISOL VLV	CLOSES
1-FCV-069-0012	RWCU SYS RETURN ISOL VLV	CLOSES
1-FCV-001-0014	MSIV LINE A INBOARD AC control power	DE-ENERGIZES
1-FCV-001-0026	MSIV LINE B INBOARD AC control power	DE-ENERGIZES
1-FCV-001-0037	MSIV LINE C INBOARD AC control power	DE-ENERGIZES
1-FCV-001-0051	MSIV LINE D INBOARD AC control power	DE-ENERGIZES
1-FCV-001-0055	MN STM LINE DRAIN INBD ISOL VLV	CLOSES
1-FCV-043-0013	RX RECIRC SAMPLE INBD ISOL VLV	CLOSES

Attachment 1
(Page 4 of 4)
RPS Bus A or B Power Transfer

C. Loss of power to RPS Bus B only will result in the following events in addition to those listed for RPS Bus A or B power loss

VALVE	FUNCTION/SYSTEM	ACTION
1-FCV-074-0047	RHR SD CLG SUCT OUTBD ISOL VLV	CLOSES
1-FCV-074-0067	RHR SYS II LPCI INBD INJECT VLV	CLOSES
1-FCV-075-0058	PSC PUMP SUCTION OUTBD ISOL	CLOSES
1-FCV-077-0015B	DRYWELL EQ DR SUMP OUTBD FCV	CLOSES
1-FCV-077-0002B	DRWELL FD SUMP OUTBD ISOLATION VLV	CLOSES
1-FCV-069-0002	RWCU OUTBD SUCT ISOL VLV	CLOSES
1-FCV-069-0012	RWCU SYS RETURN ISOL VLV	CLOSES
1-FCV-001-0015	MAIN STEAM LINE A OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0027	MAIN STEAM LINE B OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0038	MAIN STEAM LINE C OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0052	MAIN STEAM LINE D OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0056	MAIN STEAM LINE DRAIN OUTBD ISOL VLV	CLOSES
1-FCV-043-0014	REACTOR RECIRC OUTBD ISOLATION VLV	CLOSES

Excerpt from OPL171.009 Lesson Plan:

(3) Close valve operation
- (a) Air supplied to top of air operator piston plus spring force causes valve to close.
- (b) Valve closure speed set by throttle valve on hydraulic dash pot.
  - i. Valve closes in 3-5 seconds.
  - ii. Fast enough (5 seconds) to prevent gross release of fission products to the environs.
  - iii. Slow enough (3 seconds) to minimize the pressure transient resulting from isolation.
- (c) Also steam flow assists closing.

Figure-8

Which AOI addresses MSIV closure (AOI-1-3)
Obj. ILT 8

What plant parameters would we expect to change and why
Obj. NLO 4

(4) Characteristics
- (a) Air to open.
- (b) Air and/or spring to close
- (c) Valve fails closed on loss of air.
- (d) Inboard valves supplied normally by the drywell control air system.
- (e) Outboard valves supplied normally by the control air system.
- (f) One AC and one DC solenoid operated valve set controls the air supply.
  - i The DC solenoids are powered from 250V DC RMOV Boards, when operating from Panel 9-3 or from Panel 25-32.
  - ii The AC solenoids are powered from RPS when operating at Panel 9-3, and from unit preferred when operating at Panel 25-32.
- (g) If both solenoids deenergize, the valves will close.

Obj. ILT 9.a
Obj. NLO 5.a
Obj. ILT 9.b
Obj. NLO 5.b
Obj. ILT 9.c
Obj. NLO 5.c

TMOD BFN-2-2021-001-002 temporarily modified the power supply to inboard MSIV 2C due to a faulty ground on the DC solenoid. Will be removed during next refuel outage. Reference Training Dispatch OPS-TD-2021-012 for details.

Excerpt from 1-AOI-57-5A:

BFN Unit 1	Loss of I&C Bus A	1-AOI-57-5A Rev. 0055 Page 4 of 44

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a loss of I&C Bus A (Battery Board 1 Panel 8, Control Room Panel 1-9-9 Cabinet 2).

NOTES

1) A loss of I&C Transformer A results in the following boards being de-energized:
   - Battery Board 1 Panel 8
   - Panel 1-9-9, Cabinet 2, I&C Bus A
2) Power to I&C Bus A, Panel 1-9-9 Cabinet 2, is supplied via 480V S/D Board 1A, Compt 3D, disconnect switch 1A1, disconnect switch 1A4, and Breaker 1A5. Breaker 1A5 has instantaneous and timed overcurrent fault trip targets, which should be reset by depressing either the overload fault trip target or the short circuit fault trip target. Breakers and disconnects are located in A 4kV S/D Board Room. This series of breakers also supplies Battery Board 1 Panel 8 with the exception that disconnect switch 1A2 is required to also be closed.
3) There is no alternate supply to Battery Board 1, Panel 8, I&C Bus A although some loads are backed up by D.C. power. The fire protection supplied by the supervised and the unsupervised distribution Breakers 808 and 807 will **NOT** automatically initiate while Panel 8 is de-energized but can be manually initiated.

2.0 SYMPTOMS

A. I&C BUS A VOLTAGE ABNORMAL (1-XA-55-8C, Window 21).

B. SJAE A isolation resulting in lowering condenser vacuum.

C. RFP 1C Minimum Flow Valve 1-FCV-3-6 FAILS OPEN

D. Reactor Building/Refuel Zone ventilation static switch isolation.

E. RWCU filter demineralizer A isolation.

F. Loss of instrument power and remote position indication switches for Core Spray Division I, (Panel 1-9-3).

G. Loss of instrument power and remote position indication switches for RHR Division I, (Panel 1-9-3).

2.0 SYMPTOMS (continued)

H. Loss of power to 1-LPNL-925-0023, Recirc Drive Motor Monitoring Relay 1-RLY-068-MMR1(2)(3)/A, and Digital Frequency Relay 1-RLY-068-DFR1(2)(3)/A. These relays will fail to initiate a trip in the 2 out of 3 logic trip system.

I. Loss of Drywell Leak Detection System Division I, (Panel 1-9-2).

J. Loss of Main Steam Relief Valve position indication.

K. Loss of power to RCIC and HPCI Turbine Vibration circuitry and position indication for HPCI testable check valve, (Panel 1-9-3).

L. Loss of position indication for RCIC Pump Discharge Check Valve.

M. Loss of RHRSW and EECW Division I instrumentation, (Panel 1-9-3, 1-9-20).

N. Loss of SBGT A flow and differential pressure indication, (Panel 1-9-25).

O. Loss of SLC A and B amber ready lights and valve position indication, (Panel 1-9-5).

P. Loss of LPRM meter lights and APRM alarm lights, (Panel 1-9-5).

Q. Loss of Condensate - Feedwater and Heater Drains instrumentation, (Panel 1-9-6).

R. Loss of SRM/IRM detector drive power and position indication, (Panel 1-9-5).

S. Loss of one-half the blue scram lights and accumulator low pressure-high level light indications (Panel 1-9-5, 1-LPNL-0925-0004).

T. Loss of Control Bay Emergency Ventilation System Division I.

U. Loss of AC Supply to ± 24V NEUTRON MONITORING BATT CHGR A1-1 NEG SIDE, 1-CHGD-283-0000A1-1 and ± 24V NEUTRON MONITORING BATT CHGR A2-1 POS SIDE, 1-CHGD-283-0000A2-1. (The Neutron Monitoring Battery System is rated to carry loads for 3 hours. STACK GAS CH1 RAD MON RTMR, 0-RM-090-0147B will be lost after this time period)

2.0 SYMPTOMS (continued)

V. "B" Fuel Pool Demin valves:

1. 1-FCV-078-0063, FPC F/D OUTBD ISOL VLV Closes,
2. 1-FCV-078-0068, RX WELL INFL INBD VLV Closes,
3. 1-FCV-078-0066, FPC F/D 1A BYP VLV Opens.

These actions result from loss of power to A and C skimmer surge tank low-low level switches.

W. Short Cycle valves 1-FCV-002-0029A and1- FCV-002-0029B fail closed.

X. Loss of Relief Valve Acoustic Monitor Indication.

Y. Drywell to Suppr. Chamber ΔP Compressor starts due to de-energization of 1-PS-64-137B.

Z. Loss of ENG SAFEGUARDS DIV I supply to Panel 1-25-31.

Excerpts from 1-AOI-57-4:

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0039 Page 4 of 32

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for Loss of Unit Preferred (Battery Board 1 Panel 11, Control Room Panel 1-9-9 Cabinet 6 Unit Preferred and Cabinet 5 Non-preferred 120 VAC).

NOTE

A loss of the Unit Preferred power source results in the following boards being de-energized:

- Battery Board 1 Panel 11
- Panel 1-9-9 Cabinet 6 Unit Preferred (if panel does **NOT** auto transfer to alternate).
- Panel 1-9-9 Cabinet 5 Non-preferred

2.0 SYMPTOMS

A. UNIT PREF SUPPLY ABNORMAL (1-XA-55-8B, Window 35) on Panel 1-9-8 is in alarm.

B. Loss of indication to CRD rod control indicator lights Panel 1-9-5 (i.e.: rod out permissive, timer switch malfunction - select block, rod withdraw, insert and settle, notch override and stabilizing valve indicator lights.

C. The following recorder fails downscale Panel 1-9-6:

1. RFPT/RFP VIB AND ECC Recorder (1-XR-3-177).

D. The following recorders fail downscale Panel 1-9-7:

1. TURBINE VIBRATION (1-XR-47-15),

2. TURBINE GEN ECC / SPEED / VALVE POSN (1-XR-47-16).

E. The following recorder fails downscale Panel 1-9-8:

1. TURB EXP and TEMP MN XFMR WINDING TEMP (1-TR-47-20).

F. PANEL 9-9 CABINET 1, 2, 3 OR 6 CONTROL POWER TRANSFER (1-XA-55-7A, Window 15) on Panel 1-9-7, is in alarm on transfer of Unit Preferred Cabinet 6.

G. PNL 9-21 STEAM LEAK DETECTION POWER FAILURE (1-XA-55-3D, Window 31) on Panel 1-9-3 is in alarm on loss of power to Steam Leak Detection Panel (Panel 1-9-21).

2.0 SYMPTOMS (continued)

H. The following EHC Control System Annunciators alarm on Panel 1-9-7 due to a loss of power to either Unit Preferred Panel 1-9-9 Cabinet 6 OR Non-preferred Panel 1-9-9 Cabinet 5. A loss of power to both cabinets will NOT result in an alarm.

1. EHC POWER ABNORMAL 1-EA-57-163 (1-XA-55-7B, Window 5)
2. EHC/TSI SYSTEM TROUBLE 1-EA-57-165 (1-XA-55-7B Window 6)

I. The following rod control annunciators are in alarm simultaneously due to a loss of power to the associated circuits.

1. CONTROL ROD WITHDRAWAL BLOCK (1-XA-55-5A, Window 7) on Panel 1-9-5 is in alarm.
2. CONTROL ROD OVERTRAVEL (1-XA-55-5A, Window 14) on Panel 1-9-5 is in alarm.
3. RPIS INOPERATIVE (1-XA-55-5A, Window 35) on Panel 1-9-5 is in alarm.
4. FUSE FAILURE (1-XA-55-5B, Window 34) on Panel 1-9-5 is in alarm.
5. LA-85-45M EAST SDV LEVEL HIGH ROD BLOCK (1-XA-55-6A, Window 28) on Panel 1-9-6 is in alarm.
6. LA-85-45L WEST SDV LEVEL HIGH ROD BLOCK (1-XA-55-6A, Window 34) on Panel 1-9-6 is in alarm.

J. Reactor Feed Water Control System Panel Display Stations on Panel 1-9-5 disabled. PDS Controls are inoperative and displays become blank. The Reactor Feed Water Control System continues to control system parameters according to water level setpoint.

K. The following Reactor Feed Water Control System annunciator in alarm on Panel 1-9-6:

1. 1-XA-46-8, 9, 10 RFPT GOVERNOR POWER FAILURE OR GOVERNOR ABNORMAL (1-XA-55-6C, Window 12).

L. Loss of power to Cabinet 6 will cause a loss of flow signal to CNDS FLOW CONTROL SHORT CYCLE, 1-FIC-2-29 when in auto. This will cause 1-FCV-2-29A/B to open resulting in rising condensate flow. This can adversely affect Condensate & Feedwater system NPSH and Reactor water level.

M. EHC Control System HMI on 1-PNLA-009-0031 may become blank if power is lost to Panel 1-9-9 Cabinet 6. An uninterruptible power supply will keep this component energized for approximately 15 minutes after normal power is lost.

2.0 SYMPTOMS (continued)

N. 1-XA-96-10 RECIRC FLOW SYSTEM TROUBLE ALARM (1-XA-55-4A, Window 23).

O. Loss of CISS DATA ACQUISITION PANEL 1-9-96A&B.

P. PNL 9-9 PFD OR NON-PFD BKR TRIPOUT (1-XA-55-8C, Window 6) on Panel 1-9-8 is in alarm on distribution breaker trip on Panel 1-9-9 (partial loss of Unit Preferred or Unit Non-Preferred).

Q. Loss of power (after approximately 5 minutes) to HWC Off-Gas Panel 1-LPNL-925-0588 and HWC Control Panel 1-LPNL-925-0589. Both panels are fitted with a UPS. The UPS battery should keep the panel energized for approximately 5 minutes.

Excerpts from 0-AOI-57-3:

BFN Unit 0	Loss of Plant Preferred	0-AOI-57-3 Rev. 0060 Page 4 of 30

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, and operator actions for Loss of Plant Preferred (Battery Board 2 Panel 13; Panel 1, 2, and 3-9-9 Cabinet 4 and Electrical Distribution Panel 2-9-24).

NOTES

1) A loss of Plant Preferred can also result in a loss of Plant Nonpreferred. If both Plant Nonpreferred and Plant Preferred are lost, 0-AOI-57-6 should be performed concurrently with this instruction.
2) During transfer of Plant Preferred power expect numerous spurious alarms due to the momentary loss of power to recorders. These alarms should clear when power is restored.
3) This procedure is written for all units. Unless the Unit or Unit Panel number is specified, the actions described are applicable to each unit.

2.0 SYMPTOMS

NOTE

The following symptoms occur simultaneously on a Loss of Plant Preferred due to failure of the Plant Preferred Auto Transfer switch to transfer or a Plant Preferred MG set starting or loading failure (Units 1, 2 and 3 loss of power).

A. Panel 1(2)(3)-9-9 Cabinet 4 PLANT PFD 120VAC NORMAL and ALTERNATE SUPPLY lights XI-57-400A and XI-57-400B extinguish.

B. DIESEL GEN A CONTROL CKT GROUND (0-XA-55-23A, Window 43) on Panel 0-9-23-7 is in alarm on loss of power to ground indicator.

C. DIESEL GEN B CONTROL CKT GROUND (0-XA-55-23B, Window 43) on Panel 0-9-23-7 is in alarm on loss of power to ground indicator.

D. DIESEL GEN C CONTROL CKT GROUND (0-XA-55-23C, Window 43) on Panel 0-9-23-8 is in alarm on loss of power to ground indicator.

E. DIESEL GEN D CONTROL CKT GROUND (0-XA-55-23D, Window 43) on Panel 0-9-23-8 is in alarm on loss of power to ground indicator.

F. DIESEL GEN 3A CONTROL CIRCUIT GROUND (3-XA-55-23A, Window 35) on Panel 3-9-23 is in alarm on loss of power to ground indicator.

2.0 SYMPTOMS (continued)

G. DIESEL GEN 3B CONTROL CIRCUIT GROUND (3-XA-55-23B, Window 35) on Panel 3-9-23 is in alarm on loss of power to ground indicator.

H. DIESEL GEN 3C CONTROL CIRCUIT GROUND (3-XA-55-23C, Window 35) on Panel 3-9-23 is in alarm on loss of power to ground indicator.

I. DIESEL GEN 3D, CONTROL CIRCUIT GROUND (3-XA-55-23D, Window 35) on Panel 3-9-23 is in alarm on loss of power to ground indicator.

J. STACK GAS RADIATION MONITOR DNSC/INOP (XA-55-3A, Window 20) on Panel 3-9-3 is in alarm on loss of Stack Radiation Monitoring System.

NOTE

The following annunciators occur on a loss of a single unit's Plant Preferred System.

K. PNL 9-9 PFD OR NON-PFD BKR TRIPOUT (XA-55-8C, Window 6) on Panel 2-9-8 is in alarm on feeder breaker trip on Panel 2-9-9.

L. BATT BD 2 BKR TRIPOUT OR GROUND (2-XA-55-8C, Window 7) on Panel 2-9-8 is in alarm on feeder breaker trip on Battery Board 2 Panel 13.

M. Heat balance data turns blue from loss of Recirc System inputs to ICS.

Examination Outline Cross-reference:

G2.2.18 (10CFR 55.41.10)
Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments and work prioritization.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.18	
Importance Rating	2.6	------

Proposed Question: **# 7**

Which **ONE** of the following types of tags would be used to identify components that are designated as Protected Equipment on Unit 1?

A. Blue Tag

B. Green Tag

C. Orange Tag

D. Hot Pink Tag

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that Blue is the color of the tag if Unit 3 is aligned for Shutdown Cooling.

B INCORRECT: Incorrect but plausible in that Green is the color of the tag if Unit 1 is aligned for Shutdown Cooling.

C INCORRECT: Incorrect but plausible in that Orange is the color of the tag if Unit 2 is aligned for Shutdown Cooling.

D **CORRECT**: (*See attached) In* accordance with BFN-ODM-4.5, Operator Aids and Operator Information Systems, in order to enhance the Operator's awareness of off-normal equipment conditions or system alignments, Operations Information Tags provide highly visible indication on affected components or systems. Hot Pink is the designated color of the tags used to identify Protected Equipment on **all** of the Units. Protected Equipment is governed by BFN-ODM-4.18 and NPG-SPP-07.3.4.

RO Level Justification: Tests the candidate's knowledge of the process for managing maintenance activities in accordance with Browns Ferry Operations Directive Manual. This question is rated as Memory due to the requirement to strictly recall facts.

Technical Reference(s): BFN-ODM-4.5, Rev. 6 (Attach if not previously provided)
BFN-ODM-4.18, Rev. 31
NPG-SPP-07.3.4, Rev. 13

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.113 Obj. 8 (As available)

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.065-09 003 #2183	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

2183. OPL171.065-09 003

Unit 1 is in Mode 4 with the following:

- Loop 1 of RHR is aligned for Shutdown Cooling.
- All other equipment is aligned as required for the plant conditions.

Which one of the following types of tags would be used to identify component that are aligned for Shutdown Cooling on Unit 1?

A. Orange Tag

B. Hot Pink tag

C. Green Tag

D. Blue Tag

Excerpt from BFN-ODM-4.5:

1.0 ODM-4.5, OPERATOR AIDS AND OPERATOR INFORMATION SYSTEMS

INTENT: This directive describes the process set forth for the implementation and control of Operator Aids, Operations Information Tags, Identification of "Out of Service" Instrumentation and Indications, and the Light Lens Cover Convention.

1.1 ODM-4.5 DIRECTIVE

A. Operator Aids

Refer to 0-TI-414 Component Labeling, Signs, Operator Aids, and Permanent Information Postings.

B. Operations' Information Tags

1. In order to enhance Operator's awareness of off-normal equipment conditions or system alignments, Operations Information Tags have been provided to provide a highly visible indication on affected components or systems. The types of tags are as follows:

   - Aligned for Shutdown Cooling (Green - U1, Orange - U2, Blue - U3)
   - Temporary Lift (White)
   - Aligned Per S.I. (SR) (Manila)
   - Information (General Use) (Manila)
   - Protected Equipment (Hot Pink)
   - Blank (Fluorescent Green)

2. These tags are to be used judiciously at the discretion of the Unit Supervisor or at direction of plant procedures in those situations where a system is aligned differently from normal standby alignment. Information such as the applicable procedure section or WO number is to be entered on the tag as appropriate.

3. These tags are not to be used to provide directives associated with equipment/component operation or to track official system status. The tags may be used to display information to the operator of unusual conditions which exist on the system or component. They are to be used to convey information ONLY. These type tags are designed for a short duration, but may be used to convey information which will remain throughout the operating cycle.

Excerpt from BFN-ODM-4.18:

BFN Operations Directive Manual	Protected Equipment	BFN-ODM-4.18 Rev. 0031 Page 4 of 27

1.0 ODM-4.18, PROTECTED EQUIPMENT

INTENT: The intent of protecting systems is to provide additional administrative barriers to guard against inadvertently rendering a component or system inoperable or unavailable that is important to safety. It is also applicable to those systems and activities that pose a potential risk to generation.

EXPECTATIONS: Protected Equipment barriers for boards shall utilize physical barriers such as stanchions and ropes around the board unless Protective Equipment ropes are unavailable. Signs on doors and equipment should be avoided if at all possible. When posting protected equipment, the normal point of control (handswitch), power supply (bkr) and the physical equipment (pump, motor, generator, etc) should be protected.

2.0 ODM-4.18 DIRECTIVE

A. NPG-SPP-07.3.4 "Protected Equipment" is the governing document to be used for protected equipment.

B. BFN-ODM-4.18 Attachment 1, contains a list of the recommended equipment which needs to be protected for a given out of service component.

3.0 ATTACHMENTS

Attachment 1, General Guidance for Protecting Power Supplies

Attachment 2, Protected Equipment Matrix

Excerpts from NPG-SPP-07.3.4:

3.1.4 Work Week Senior Reactor Operator (SRO) Staff review

A. Review the Work Order or Activity and identify if any protected systems are required. Normally this will be complete by the end of T-4 week and presented at the T-3 Challenge Meeting in support of site aggregate Risk..

B. When a system is designated as "protected," notify the Work Week Manager (WWM) to add a schedule activity to place and remove equipment protection.

C. Ensure that planned work that could impact a protected system is not scheduled within the window of protection.

3.1.5 Risk Mitigating Implementer On Shift (SRO)

A. Maintains status of Protected Equipment, normally using the ESOMS Clearance Module. Protected Equipment placements may be documented utilizing Attachment 2, if desired or if ESOMS is unavailable.

B. Responsible as the reviewer for accuracy and consistency for each of the Unit MCR Turnover sheets and the AUO watchstation sheets.

C. Ensures the communication between the site Plan of the Day and the operating crews are consistent and accurate.

D. Ensures Protected Equipment is discussed accurately at Operations Turnover meeting.

3.1.6 Work Week Manager (WWM)

A. When notified by Risk SRO, add schedule activities to place equipment protection on the preceding shift, and to remove the protection following restoration of the redundant component.

B. Ensure that planned work that could impact a protected system is not scheduled within the window of protection.

3.1.7 Unit Operator (UO) / Assistant Unit Operator (AUO) / Outage Specialist

A. Post / Un-Post Protected Equipment and Protected Equipment Areas as directed by the shift manager or designee.

B. Place / remove signs, tags, or barriers as directed by the shift manager or designee.

C. Monitor plant equipment and ensure no inappropriate or unauthorized work is in progress on a Protected Train/Division or Protected Equipment.

D. Be knowledgeable of Protected Equipment on assigned watchstation.

E. For their assigned watchstation, responsible to ensure signs, tags, and barriers are still in place and no work is being performed on or near equipment listed in as Protected Equipment on the Daily Operational Focus Report.

4.0 RECORDS

4.1 QA Records

None

4.2 Non-QA Records

None

5.0 DEFINITIONS

High-Risk Condition - In the context of this document, a high-risk condition refers to an elevated calculated risk, often computed by software programs, used to define trigger points for risk management actions. Calculated risk generally follows the criteria established in NUMARC 93-01. Some stations use site-specific terms or a color code (such as orange or red) to refer to the conditions that correspond to high-risk conditions.

Noteworthy Transient - A noteworthy transient is one in which station expectations require the shift manager to promptly notify operations or station management.

Predetermined Protection Schemes - Protection schemes for Technical Specifications and station generation are determined manually by knowledge of Technical Specification requirements and plant operations. The use of standardized postings ensures a consistent application of the protected equipment program and reduces work activity conflicts associated with the protected train.

Protected Equipment - Structures, Systems, or Components (equipment) redundant to equipment that has failed, is taken out of service, or is otherwise unavailable when the redundant equipment is required for ensuring that either defense-in-depth of a Key Safety Function is maintained, station generation is maintained or overall safety function risk levels are maintained for current plant conditions. Physical barriers or signage is used to alert personnel to maintain a safe distance from the Protected Equipment in order to prevent unintended consequences from operation, maintenance, or nearby activity.

Protected Equipment Area - A protected area is a physical location that contains protected equipment for which access is specifically controlled. This physical area could be a room, pump/motor skid, the switchyard, a control cabinet, an area cordoned off with barrier tape, or any other area that is uniquely posted as containing protected equipment.

Protected Equipment Area Sign - A sign placed at the entrance or boundary to an area or room where greater control of work activities is required for managing risk for the safe operation of the plant. With the exception of normal Operations, Security, and Fire Watch rounds. Access to this equipment or work in this area is allowed only with Shift Manager Permission. See Attachment 1 or similar, “Protected Equipment Area” signs.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
Conduct of Operations	Tier #	3	------
G2.1.45 (10CFR 55.41.7)	Group #	------	------
Ability to identify and interpret diverse indication to validate the response of another indication.	K/A #	G2.1.45	
	Importance Rating	4.3	------

Proposed Question: **# 8**

Unit 2 is operating at 100% RTP with the following plant conditions:

- MAIN STEAM RELIEF VALVE OPEN (2-9-3C, Window 25) alarms
- SRV 1-5 indicates full OPEN on the SRV TAILPIPE FLOW MONITOR
- 2-AOI-1-1, Relief Valve Stuck Open, actions are in progress



The given annunciator is **DIRECTLY** actuated by the tailpipe **(1)**.

In accordance with 2-AOI-1-1, a failed open SRV can be confirmed by observing that 2-XR-57-57, GENERATOR LOAD, recorder on Panel 2-9-8 **(2)**.

A. (1) acoustic monitor
(2) lowers

B. (1) acoustic monitor
(2) rises

C. (1) discharge temperature
(2) lowers

D. (1) discharge temperature
(2) rises

Proposed Answer: A

Explanation (Optional):

A **CORRECT:** (*See Attached*) First part is correct in that in accordance with 2-AOI-1-1, Relief Valve Stuck Open, annunciator 2-9-3C, Window 25, MAIN STEAM RELIEF VALVE OPEN, is actuated by the Steam Relief Valve (SRV) tailpipe acoustic monitor. Second part is correct in that in accordance with 2-AOI-1-1, main generator load lowers as a result of an SRV being open.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that when the SRV closes, Reactor Power will rise, which is why Reactor Power is lowered prior taking immediate actions.

C INCORRECT: First part is incorrect but plausible in that in accordance with the Immediate Actions of 2-AOI-1-1, operators will identify which SRV is open by monitoring SRV Discharge Tailpipe Temperatures. Second part is correct (*See A*).

D INCORRECT**:** First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests candidate's ability to use indications to verify that an SRV has opened. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-AOI-1-1, Rev. 30 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **MAIN STEAM RELIEF VALVE OPEN, (2-9-3C, Window 25)**

Learning Objective: OPL171.009 Obj. 6 (As available)

Question Source:

Bank #	
Modified Bank #	ILT NRC 1909 #51
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Copy of Bank Question:

Proposed Question: **# 51**

Unit 2 was operating at 100% RTP when the following occurs:



- MAIN STEAM RELIEF VALVE OPEN, (2-9-3C, Window 25) alarms
- SRV 1-31 indicates full OPEN on the SRV TAILPIPE FLOW MONITOR
- 2-AOI-1-1, Relief Valve Stuck Open actions are in progress

Given the conditions above, which **ONE** of the following completes the statements below?

The given annunciator is **DIRECTLY** actuated by the tailpipe ___**(1)**___.

In accordance with 2-AOI-1-1, the Operator is allowed to cycle the control switch for SRV 1-31 ___**(2)**___ from CLOSE to OPEN to CLOSE positions.

A. **(1)** acoustic monitor
(2) UP TO three (3) times

B. **(1)** acoustic monitor
(2) ONLY one (1) time

C. **(1)** discharge temperature
(2) UP TO three (3) times

D. **(1)** discharge temperature
(2) ONLY one (1) time

Proposed Answer: **A**

Excerpts from 2-AOI-1-1:

BFN Unit 2	Relief Valve Stuck Open	2-AOI-1-1 Rev. 0030 Page 3 of 28

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a stuck open relief valve.

2.0 SYMPTOMS

A. Annunciator MAIN STEAM RELIEF VALVE OPEN 2-FA-1-1 (2-XA-55-3C, Window 25) is in alarm due to SRV Tailpipe Flow Monitor sensing flow.

B. GENERATOR LOAD recorder, 2-XR-57-57, Panel 2-9-8, indication is lowering.

C. MAIN STEAM/TURBINE STEAM FLOW, flow recorder 2-FR-46-5, Panel 2-9-5, indication is lowering.

D. SUPPRESSION POOL WATER TEMPERATURE recorder, 2-TR-64-161 and SUPPRESSION POOL WATER TEMPERATURE recorder, 2-TR-64-162, indication is rising.

3.0 AUTOMATIC ACTION

None

4.0 OPERATOR ACTION

NOTE

Once a MSRV is operated, a time delay of 15 to 30 seconds can be expected before a response can be detected on 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE. ICS can be used to monitor the discharge tailpipe temperature, but the appropriate indications on 2-TR-1-1 must be confirmed.

4.1 Immediate Action

[1] **IDENTIFY** stuck open relief valve by

OBSERVING the following:

- SRV TAILPIPE FLOW MONITOR, 2-FMT-1-4, on Panel 2-9-3,

  OR

- MSRV DISCHARGE TAILPIPE TEMPERATURE recorder, 2-TR-1-1 on Panel 2-9-47. ☐

[2] **IF** relief valve transient occurred while operating above 90% power, **THEN**

REDUCE reactor power to ≤90% RTP with recirc flow. (Otherwise N/A) ☐

[3] **WHILE OBSERVING** the indications for the affected Relief valve on the Acoustic Monitor;

CYCLE the affected relief valve control switch as required up to three times:

- CLOSE to OPEN to CLOSE positions ☐

[4] **IF** all SRVs are CLOSED, **THEN**

CONTINUE at Step 4.2.4. (Otherwise N/A) ☐

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
209001 (SF 2 / SF 4 LPCS) Low-Pressure Core Spray	Tier #	2	------
A3.02 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the Low-Pressure Core Spray System, including: • Pump start	K/A #	209001A3.02	
	Importance Rating	4.3	------

Proposed Question: **# 9**

Unit 2 is operating at 100% RTP with the following plant conditions:

- At 1000:
  - Drywell Pressure reaches 3.0 psig due to a LOCA
  - 4KV Shutdown Board C trips due to a fault
  - 'C' EDG will **NOT** start
- At 1015:
  - Reactor Pressure is 400 psig

Given the conditions above, which **ONE** of the following completes the statement below?

At time 1015, Core Spray Pumps **(1)** are running and **(2)** injecting to the Reactor Vessel.

A. (1) 2A, 2B, and 2D
(2) are

B. (1) 2A, 2B, and 2D
(2) are **NOT**

C. (1) 2A and 2C
(2) are

D. (1) 2A and 2C
(2) are **NOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that since 'C' 4KV Shutdown Board C is de-energized, the candidate could conclude that 2C Core Spray Pump is not running. Additionally, if the candidate fails to recognize the loop companion pump interlock requirement, they would conclude that 2B and 2D Core Pumps would be running. Second part is incorrect but plausible in that the Core Spray inboard and outboard injection valves have in-line interlocks which will send automatic opening signals when there is an initiation signal and Reactor Pressure is below 450 psig. Therefore, both injection valves are in fact open.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible (See A).

C INCORRECT: First part is correct *(See D).* Second part is correct *(See D).*

D **CORRECT:** *(See attached)* In accordance with 2-OI-75, Core Spray System, the Core Spray System will automatically initiate when Drywell Pressure reaches 2.45 psig and rising with Reactor Pressure reaching 450 psig and lowering. At 1000, no Core Spray Pumps are running just based on 2.45 psig Drywell Pressure, but once Reactor Pressure reached 400 psig as given at 1015, all Core Spray Pumps were given an automatic start signal. However, since 4KV Shutdown Board C trips due to a fault and 'C' EDG will not start, 'B' Core Spray Pump did not start. This in turn prevented 2D Core Spray from automatically starting since each loop's pumps automatically start based on the other companion pump starting. For second part, Core Spray Pumps are not injecting to the Reactor Vessel since their Shutoff Head Pressure is 330 psig. The Core Spray inboard and outboard injection valves have in-line interlocks which will send automatic opening signals when there is an initiation signal and Reactor Pressure is below 450 psig.

RO Level Justification: Tests the candidate's ability to monitor automatic operations of the Low-Pressure Core Spray System as it relates to pump starting requirements. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution as it relates to the Core Spray System.

Technical Reference(s): 0-OI-82, Rev. 180 (Attach if not previously provided)
2-OI-75, Rev. 75
2-EOI-1, Rev. 20

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.045 Obj. 2f (As available)

Question Source:

Bank #	
Modified Bank #	OPL171.045-04 011 #1860
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1860. OPL171.045-04 011

Given the following conditions:

- Unit 2 was at 100% power.
- At 0900, a small break LOCA has occurred.
- At 0915, Drywell pressure reached 2.45 psig.
- At 0930, the 'A' 4 KV SD Board lost power.
- A' DG started but failed to automatically tie to the bus.
- At 1000, RPV pressure reached 450 psig.
- At 1015, RPV pressure dropped to 300 psig.
- **NO** manual operator actions have been performed.

Which ONE of the following completes the statements below?

At time 1005, ___(1)___Core Spray Pumps are running.

At 1015, operating Core Spray Pumps are___(2)___.

A. (1) 2B, 2C, and 2D
(2) **NOT** injecting

B. (1) 2B, 2C, and 2D
(2) injecting

C✓ (1) **ONLY** 2B and 2D
(2) injecting

D. (1) **ONLY** 2B and 2D
(2) **NOT** injecting

Examination Outline Cross-Reference: Level **RO**
K/A # 209001 A3.02

Excerpts from 0-OI-82:

3.0 PRECAUTIONS AND LIMITATIONS (continued)

P. Personnel working in the D/G rooms should remain aware that the possibility exists of CO_2 discharge into the room. Upon CO_2 initiation, an alarm will sound. Personnel then have 20 seconds to evacuate the area before CO_2 is dispensed. For detection purposes, a wintergreen odor is injected into CO_2 discharge.

Q. Environmental calculations assume DG battery ambient temperatures are within 40°F to 110°F.

R. When the D/G is the only feed to the shutdown board and in single unit operations, starting an RHR Pump with other 4kV motor loads running on the associated board may result in D/G overload.

S. After operation of 4160V breakers, the charging spring is required to be verified to have recharged by verifying locally the breaker closing spring target indicates charged and the amber breaker spring charged light is on to ensure future breaker operation.

T. Diesel Generators will automatically start, as follows:

1. Degraded voltage, undervoltage, OR sustained voltage unbalance on 4-kV Shutdown Board A, B, C, or D will start its associated Diesel Generator.

2. A Pre-Accident Signal (Reactor Vessel Low Low Low water level OR High Drywell pressure) on Unit 1, Unit 2 or Unit 3 will start all eight Diesel Generators.

U. Under normal conditions, any of the following will auto trip the Diesel Generator output breaker:

1. Differential overcurrent
2. Timed overcurrent
3. Reverse power
4. Loss of field
5. Overspeed
6. Common Accident Signal (Low Low Low Reactor water level OR Low Reactor pressure in conjunction with High Drywell pressure on Unit 1, 2 or Unit 3.)

Excerpts from 2-OI-75:

3.3 Equipment (continued)

F. Leakage of Suppression Pool quality water into the RPV may occur when Core Spray System pressure is above RPV pressure due to a 1/4 inch hole drilled into the outlet side disc face of CORE SPRAY SYS I(II) INBD INJECT VALVE, 2-FCV-75-25(53) to eliminate pressure locking concerns associated with these valves.

G. The preferred suction source for Reactor Cavity floodup is Section 8.24 which utilizes the CST Standpipe. (PER 245792).

3.4 Initiations

A. The CS System will auto initiate from the following signals:

1. RPV water level at or below -122 inches.

2. DW pressure at or above 2.45 psig and RPV pressure at or below 450 psig.

B. Manually stopping a Core Spray Pump after auto initiation will disable automatic restart of that pump until the initiation signal is clear and has been reset. The affected Core Spray Pump may still be started manually.

3.5 Isolations

A. PSC PUMP SUCTION INBD and OUTBD ISOL VALVES, 2-FCV-75-57 and 2-FCV-75-58, will close on Group II Isolation, tripping PSC Head Tank Pumps 2A and 2B.

B. The Core Spray test valve receives an auto closure signal on any CS auto initiation.

C. The Core Spray minimum flow valves receive a closure signal when flow is approximately 2600 gpm rising and receives a open signal when flow lowers to approximately 2200 gpm.

3.6 Trips

A. Electrical

3.7 Interlocks

A. The CS inboard and outboard injection valves have in-line valve interlocks to prevent both valves from being opened with RPV pressure at or above 450 psig. Both receive auto open signals when there is a CS initiation signal and RPV pressure is below 450 psig. The inboard valve may be throttled immediately after initiation.

3.8 Power Supplies

A. Core Spray Breaker closure with the breaker racked to the test position will result in an auto start of only the specific EECW pumps below if the NVA or DGVA relay is allowed to time-out prior to opening of the breaker. This is due to the breaker compartment stationary (STA) switch not actuating upon breaker closure with the breaker in the test position.

Core Spray Pump	EECW Pump (if aligned for service)
2A	C1, B3
2B	A1, D3
2C	A1, D3
2D	C1, B3

B. If one pump in a Core Spray System has its 4kV Shutdown Board de-energized, neither pump in that loop will auto start and may NOT be considered operable. REFER TO TS 3.5.1/3.5.2.

C. After operation of 4160V breakers, the charging spring shall be verified to have recharged by checking locally, the breaker closing spring target indicates charged and the amber breaker spring charged light is on to insure future breaker operation.

D. Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161kV source and large pumps are started. Unit SROs should evaluate placing the Capacitor Banks in Manual prior to starting RHR, CS, CBP, CCW, or COND pumps as referenced in 0-OI-57A.

Excerpt from 2-EOI-1:

Table L-1
Preferred Injection Systems

SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available ◇2 ◇3 ◇5	5C, 20M	1200 psig
HPCI with CST suction if available ◇2 ◇5 ◇7	5D, 20N	1200 psig
CNDS	6A	470 psig
CS ◇2	6D, 6E	330 psig
LPCI ◇2	6B, 6C	320 psig

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215001 (SF7 TIP) Traversing In-Core Probe	Tier #	2	------
K5.02 (10CFR 55.41.5)	Group #	2	------
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Traversing In-Core Probe systems:	K/A #	215001K5.02	
• Area radiation monitor indications	Importance Rating	2.8	------

Proposed Question: **# 10**

Unit 2 Emergency Operating Instructions (EOIs) must **FIRST** be entered when 2-RI-90-22A, TIP ROOM EL 565 RX BLDG, reaches its Maximum **(1)** value.

A Group 8 Isolation **(2)** directly caused by the TIP Room Maximum **SAFE** Value.

A. (1) SAFE
(2) is

B. (1) SAFE
(2) is NOT

C. (1) NORMAL
(2) is

D. (1) NORMAL
(2) is NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that when TIP Room Radiation reaches the maximum safe value, Emergency Depressurization of the Reactor is required. Second part is incorrect but plausible in that a Primary Containment Isolation System (PCIS) Group 6 isolation is caused by high radiation in the Reactor or Refuel Zones, so a Group 8 Isolation could be caused by high radiation in the TIP Room.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible (*See A*).

D **CORRECT**: *(See attached)* In accordance with 2-EOI-3, Secondary Containment Control, 2-EOI-3 must be entered when TIP Room Radiation reaches the maximum normal value, which is listed in Table SC-2 of 2-EOI-3. Second part is correct in that High Radiation is not one of the isolation signals for PCIS Group 8. TIP will isolate on High Drywell Pressure and Low Reactor Water Level (+2").

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's knowledge of TIP Room Area Radiation Monitors and the effect on TIP System isolation. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to containment isolations.

Technical Reference(s): 2-EOI-3, Rev. 19 (Attach if not previously provided)
2-AOI-64-2E, Rev. 17
2-AOI-64-2D, Rev.37

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.204, Obj. 2 (As available)
OPL171.017, Obj.4

Question Source: Bank # ____
Modified Bank # 2104 NRC #45 (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2104

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

With respect to the MAXIMUM SAFE radiation value for Traversing Incore Probe (TIP), which **ONE** of the following completes the statements below?

If the MAXIMUM SAFE value for 2-RI-90-22A, TIP ROOM EL 565 RX BLDG is reached during TIP operation with the probes outside their shields, then automatic withdrawal to the in-shield position **(1)** occur.

A Group 8 Isolation **(2)** directly caused by the TIP Room MAXIMUM SAFE Value.

A. (1) will
(2) is

B. (1) will
(2) is NOT

C. (1) will NOT
(2) is

D. (1) will NOT
(2) is NOT

Proposed Answer: **D**

Excerpt from 2-EOI-3:

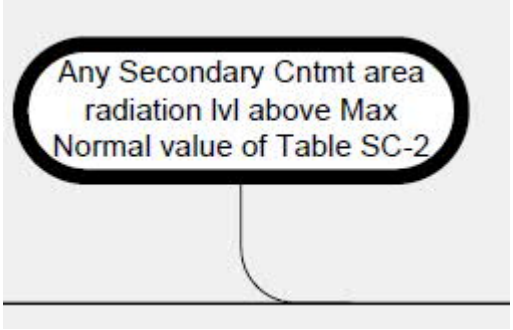


Table SC-2
Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None

Excerpt from2-AOI-64-2E, Traversing Incore Probe Isolation:

BFN Unit 2	Traversing Incore Probe Isolation	2-AOI-64-2E Rev. 0017 Page 4 of 7

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 8, Traversing Incore Probe (TIP) Isolation and detection of a reactor coolant leak in a TIP guide tube.

2.0 SYMPTOMS

NOTES

1) PCIS Group 8 Isolation is initiated by any of the following signals:
   - Reactor Vessel Water Level Low.
   - Drywell High Pressure.

A. Any one or more of the following annunciators in alarm:

1. RX VESSEL WTR LEVEL LOW HALF SCRAM (2-XA-55-4A, Window 2). Group 8 Isolation only.
2. DRYWELL PRESSURE HIGH HALF SCRAM (2-XA-55-4A, Window 8). Group 8 Isolation only.
3. AIR PARTICULATE MONITOR RADIATION HIGH 2-RA-90-50A (2-XA-55-3A, Window 2). TIP guide tube leak only.
4. RX BLDG AREA RADIATION HIGH 2-RA-90-1D (2-XA-55-3A, Window 22). TIP guide tube leak only.

3.0 AUTOMATIC ACTIONS

[1] **IF** a Group 8 isolation occurred, **THEN** the following are automatic actions:

- **IF** TIP probes are outside their shields, **THEN** TIP withdrawal initiated to IN-SHIELD position.
- TIP Ball Valves receive a close signal, or close after TIP probes are withdrawn to their IN-SHIELD position.
- TIP Purge Valves closes (no indications provided).

Excerpt from 2-AOI-64-2D, Group 6 Ventilation System Isolation:

BFN Unit 2	Group 6 Ventilation System Isolation	2-AOI-64-2D Rev. 0037 Page 4 of 17

1.0 PURPOSE

This procedure provides symptoms, automatic actions, and operator actions for a Group 6 Ventilation System Isolation.

2.0 SYMPTOMS

NOTES

1) PCIS Group 6 Isolation is initiated by any one of the following signals:
   - Reactor vessel water level at +2.0"
   - Drywell pressure at 2.45 psig
   - Reactor zone exhaust radiation at 72 mr/hr
   - Refuel zone exhaust radiation at 72 mr/hr
2) High Refuel Zone exhaust radiation causes only the automatic actions listed in Section 3.1.
3) Refuel Zone isolation due to Group 6 isolation initiated on Unit 1 or Unit 3.

A. Any one or more of the following annunciators in ALARM:

1. REACTOR ZONE EXHAUST RADIATION HIGH (2-XA-55-3A, Window 21)
2. REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE (2-XA-55-3A, Window 28)
3. REFUELING ZONE EXHAUST RADIATION HIGH (2-XA-55-3A, Window 34)
4. RX ZONE EXH RADIATION MONITOR DNSC (2-XA-55-3A, Window 35)
5. RX BLDG VENTILATION ABNORMAL (2-XA-55-3D, Window 3)
6. RX VESSEL WTR LEVEL LOW HALF SCRAM (2-XA-55-4A, Window 2)
7. DRYWELL PRESSURE HIGH HALF SCRAM (2-XA-55-4A, Window 8)
8. REACTOR ZONE DIFFERENTIAL PRESSURE LOW (2-XA-55-3D, Window 32)

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
205000 (SF4 SCS) Shutdown Cooling (RHR Shutdown Cooling Mode)	Tier #	2	------
291005 K1.06 (10CFR 55.41.7)	Group #	1	------
Motors and Generators:	K/A #	291005K1.06	
• Reason for limiting the number of motor starts in a given time period	Importance Rating	3.1	------

Proposed Question: **# 11**

2-OI-74, Residual Heat Removal System, limits the number of consecutive starts for 2-MTR-074-0048, RHR SHUTDOWN COOLING SUCTION INBOARD ISOLATION VALVE MOTOR, in order to prevent _____.

A. overheating the motor windings

B. excessive shaft torsional stress

C. overheating the motor supply bus

D. excessive cycling of the motor breaker

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-OI-74, RHR System, for non-emergency operation, 2-MTR-074-0048, RHR SHUTDOWN COOLING SUCTION INBOARD ISOLATION VALVE MOTOR, is limited to 10 consecutive starts and cooldown time limit of 2.5 hours. The starting current in an AC motor is several times normal full load current demands. The insulation on the windings in the motor is designed to handle normal full load conditions, not starting conditions. If the motor is started several times in quick succession, the large current drawn in the starting process contributes to the heat generated from I^2R heat losses. Limitations are placed on large AC motors to prevent overheating and resulting damage to the winding insulations.

B INCORRECT: Incorrect but plausible in that the motor shaft is designed to withstand multiple consecutive starts.

C INCORRECT: Incorrect but plausible in that the supply is rated for the starting current and does not heat up during the start stop cycles.

D INCORRECT: Incorrect but plausible in that the breaker is a current interruption device.

RO Level Justification: Tests the candidate's knowledge of the reason for limiting the number of motor starts in a given time period as it relates to RHR Shutdown Cooling Mode valves.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): 2-OI-74, Rev. 195 (Attach if not previously provided)

GFES Components Motor and Generators Ch. 5

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.044 Obj. 12g (As available)

Question Source:

Bank #	NRC GFES #B1128
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:

TOPIC: 291005
KNOWLEDGE: K1.06 [2.9/3.1]
QID: B1128 (P1131)

The frequency of start/stop cycles for an electrical motor is limited to prevent...

A. overheating the motor windings.

B. excessive shaft torsional stresses.

C. overheating the motor supply bus.

D. excessive cycling of the motor breaker.

ANSWER: A.

Excerpts from GFES Components Motors and Generators Ch.5:

1 / 1 QID: B1128 (P1131) Motor Starts Limits	
BWR TOPIC: 291005	PWR TOPIC: 191005
KNOWLEDGE: BK1.06 [2.9/3.1]	KNOWLEDGE: PK1.06 [3.0/3.1]
Similar questions	
B26, B228 (P2631), B328 (P231), B928, B1826 (P30), B1928 (P1031), B3327 (P3331)	
Recent BWR exams	Recent PWR exams
Sep2007	Jun2007

The frequency of start/stop cycles for an electrical motor is limited to prevent...

A. overheating the motor windings.

B. excessive shaft torsional stresses.

C. overheating the motor supply bus.

D. excessive cycling of the motor breaker.

BPC05Gr4_Motors 8/6/2008 11:01 AM

1 / 1 QID: B1128 (P1131) Motor Starts Limits
Explanation
The frequency of start/stop cycles for an electrical motor is limited to prevent... A. overheating the motor windings. CORRECT - starting current in an AC motor is several time normal full load current demands. The insulation on the windings in the motor is designed to handle normal full load conditions, NOT starting conditions. If the motor is started several time in quick succession the large current drawn in the starting process contributes to the heat generated from I^2R heat losses. Limitations are placed on large AC motor to prevent overheating and the resulting damage to the winding insulations. B. excessive shaft torsional stresses. Incorrect – the motor shaft is designed to withstand multiple successive starts C. overheating the motor supply bus. Incorrect – the supply bus is rated for the starting current and does not heat up during the start stop cycles D. excessive cycling of the motor breaker. Incorrect – the breaker is a current interruption device The correct answer is ANSWER A. overheating the motor windings.

Excerpt from 2-OI-74:

3.7 RHR Service Water (continued)

NOTE

The number of required RHRSW pumps may be reduced by one for each fueled unit that has been in Mode 4 or 5 for ≥ 24 hours.

1. One Unit fueled - four OPERABLE RHRSW pumps.
2. Two Units fueled - six OPERABLE RHRSW pumps.
3. Three Units fueled - eight OPERABLE RHRSW pumps.

3.8 Electrical

A. RHR Pump 2A Motor full load current is 247 amps, 2C Motor full load current is 255 amps, .RHR Pumps 2B & 2D Motor full load current is 247 amps.

B. To ensure operability of 4160V Breakers after closing, the charging spring should be verified to have recharged by verifying locally, breaker closing spring charged, amber light is on and closing spring target indicating charged.

C. Power should not be removed from any RHR MOV unless the valve is in the required position for LPCI initiation (unless it is on a loop that is inoperable).

D. [II/C] For non-emergency operation, RHR SHUTDOWN COOLING SUCT INBD ISOL VALVE MOTOR, 2-MTR-074-0048 is limited to 10 consecutive starts and a cooldown time limit of 2.5 hours. [BFPER940777]

E. [II/C] For actuation of the breaker thermal overload for RHR SHUTDOWN COOLING SUCT INBD ISOL VALVE, 2-FCV-074-0048, a C-Level PER should be generated. [BFPER940777]

F. Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161kV source and large pumps are started. Unit SROs should evaluate placing the Capacitor Banks in Manual prior to starting RHR, CS, CBP, CCW, or COND pumps as referenced in 0-OI-57A.

G. RHR motor heaters are for humidity control in the RHR Pump motor winding insulation systems and minimize the potential of corrosion on the internal motor components. The motor heaters serve no safety-related function. Therefore, the loss of the motor heater will not prevent the RHR pumps from performing their intended design related function. However, due to the potential to shorten the life of the motor, and its qualified insulation system, the time period these heaters are removed from service should be minimal.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295024 (EPE 1) High Drywell Pressure / 5	Tier #	1	------
EK1.01 (10CFR 55.41.8)	Group #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Drywell Pressure:	K/A #	295024EK1.01	
• Drywell integrity	Importance Rating	4.3	------

Proposed Question: **# 12**

Unit 2 is operating at 100% RTP with the following plant conditions:

- An earthquake occurs, resulting in a double-ended shear of 2A Recirculation Pump Suction piping
- Drywell Pressure is 20 psig and rising rapidly

Given these conditions, if Suppression Chamber - to - Drywell vacuum breakers fail **(1)**, the Drywell **DESIGN** Pressure Limit of **(2)** is in danger of being exceeded.

A. **(1)** OPEN
(2) 35 psig

B. **(1)** OPEN
(2) 56 psig

C. **(1)** CLOSED
(2) 35 psig

D. **(1)** CLOSED
(2) 56 psig

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that when Primary Containment Pressure reaches 35 psig, the setpoint for DRYWELL TEMPERATURE/PRESSURE HIGH activates a Main Control Room alarm.

B **CORRECT:** *(See attached)* After a LOCA occurs where Suppression Chamber – to – Drywell vacuum breakers fail OPEN, steam is not forced through the Suppression Pool water resulting in a surface condenser like operation. As a result, Drywell Pressure is likely to exceed the Design Pressure. For second part, in accordance with UFSAR Section 5.2, Primary Containment, the AMSE Boiler and Pressure Vessel Code states that the Drywell Design Pressure is 56 psig.

C INCORRECT: First part is incorrect but plausible in that after a LOCA occurs where Suppression – to – Drywell vacuum breakers failed CLOSED, Drywell Pressure is likely to lower such that the external Design Pressure of 2 psig will be reached. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the operational implications and/or cause and effect relationships of Drywell integrity as it relates to High Drywell Pressure. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): UFSAR Sect. 5.2, Rev. 28, 29 (Attach if not previously provided)
2-ARP-9-3B, Rev. 40
OPL171.016, Rev. 25U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.016 Obj. 2 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.016-04 005 #695
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

MCS Time: 4 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B B B B B B B B B Items Not Scrambled

695. OPL171.016-04 005

Unit 1 was operating at rated power when a DBA LOCA occurred.

Under these conditions, ___(1)___ could cause the drywell to exceed its ___(2)___ design pressure limit.

A✔ (1) one Torus to Drywell Vacuum Breaker failing OPEN
(2) internal

B. (1) one Torus to Drywell Vacuum Breaker failing CLOSED
(2) external

C. (1) one Reactor Building to Torus Vacuum Breaker failing OPEN
(2) external

D. (1) one Reactor Building to Torus Vacuum Breaker failing CLOSED
(2) internal

Excerpts from UFSAR Section 5.2, Primary Containment:

from the drywell and from the water in the pressure suppression chamber, thus cooling the primary containment, when required, under accident conditions. Appropriate isolation valves are actuated during this period to ensure containment of radioactive materials within the primary containment which might be released from the reactor during the course of the accident. If long-term cooling capability is lost, resulting in a pressure increase that would jeopardize the structural integrity of the primary containment, the hardened containment venting system (HCVS) would relieve the corresponding pressure increase. The Primary Containment System is designed as a Class I system (except the drywell pressure suppression chamber pressure differential subsystem -Reference Section 5.2.3.9) in accordance with Appendix C, "Structural Qualification of Subsystems and Components," to the BFNP FSAR.

5.2.3.2 Drywell

The drywell is a steel pressure vessel with a spherical lower portion 67 feet in diameter, and a cylindrical upper portion 38 feet 6 inches in diameter. The overall height is approximately 115 feet. The design, fabrication, inspection and testing of the drywell vessel comply with requirements of the ASME Boiler and Pressure Vessel Code, 1965 edition, Section III, Class B, which pertain to containment vessels for nuclear power plants. The steel head and shell of the drywell are fabricated of SA-516 plate.

The drywell is designed for a maximum internal pressure of 62 psig coincident with a temperature of 281°F, plus the dead, live, and seismic loads imposed on the shell. Thus, in accordance with the ASME Boiler and Pressure Vessel Code, Section III, the drywell design pressure is 56 psig. Thermal stresses in the steel shell due to temperature gradients were taken into account in the design.

Special precautions not required by codes were taken in the fabrication of the steel drywell shell. The plate was preheated to a minimum temperature of 200°F prior to welding of all seams whose thickness exceeds 1-1/4 inches, regardless of surrounding air temperature. Preheat at a minimum of 100°F was applied prior to welding of all seams 1-1/4 inches or less in thickness if the ambient temperature fell below 50°F. Charpy V-notch specimens were used for impact testing of plate and forging material to give assurance of proper material properties. Plates, forgings and pipe associated with the drywell have an initial NDT temperature of 0°F or lower when tested in accordance with the appropriate code for the materials. It is contended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F.

The drywell is enclosed in reinforced concrete for shielding purposes to provide additional resistance to deformation and buckling of the drywell over areas where the concrete backs up the steel shell. Above the transition zone, the drywell is

excessive resistance and distributes this flow effectively and uniformly in the pool following a postulated pipe rupture in the drywell.

The pressure suppression chamber receives this flow, condenses the steam portion of this flow, and contains non-condensable gases and fission products driven into the pressure suppression chamber. The pressure suppression chamber-to-drywell vacuum breakers limit the pressure differential between the drywell and pressure suppression chamber. The pressure suppression chamber is designed for the same leakage rate as the drywell.

Large vent pipes form a connection between the drywell and the pressure suppression chamber. A total of eight circular vent pipes are provided, each having a diameter of 6.75 feet.

The vent pipes are designed for an internal pressure of 56 psig (the ASME Boiler and Pressure Vessel Code, Section III, allows a maximum internal pressure of 62 psig) coincident with a temperature of 281°F and are designed to withstand an external pressure of 2 psi above internal pressure. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces which might accompany a pipe break in the drywell. The vent pipes are fabricated of SA-516 steel, and comply with requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection B. Expansion joints are provided on each vent pipe to accommodate differential motion between the drywell and pressure suppression chamber.

The pressure suppression chamber is a steel pressure vessel in the shape of a torus below and encircling the drywell, with a centerline diameter of approximately 111 feet and a cross-sectional diameter of 31 feet. As a result of the torus modifications on all three units, the maximum water volume in the torus is approximately 131,400 ft^3 (see note on Table 5.2-1). The pressure suppression chamber is held by supports which transmit vertical and seismic loading to the reinforced concrete foundation slab of the Reactor Building. Space is provided outside the pressure suppression chamber for inspection and maintenance. The eight drywell vents are connected to a 4-foot, 9-inch diameter vent header in the form of a torus which is contained within the airspace of the pressure suppression chamber. Projecting downward from the header are 96 downcomer pipes, 24 inches in diameter, and terminating approximately 3 feet 10 inches (maximum, see note on Table 5.2-1) below the water surface of the pool. T-quenchers have been added to replace the ramshead discharge devices at the end of the main steam relief valve discharge pipes to assure the controlled release and condensation of steam and reduce stresses on the torus and tailpipe assemblies. The vent header and vent pipes have the same temperature and pressure design requirements as the pressure suppression chamber. Vacuum breakers discharge from the pressure suppression chamber into the drywell to limit the pressure differential

Excerpts from OPL171.016 Lesson Plan:

(b) Both FCV-64-221 and 222 will remain closed by separate key-locked permissive control switches in the Main Control Room panel 9-3. Automatic isolation signals are not provided to these valves. Figure-4

(i) The function of the key lock switches is to prevent inadvertent operator actuation during normal plant operation and design basis accidents.

(ii) Indication of valve position for valves FCV-64-221, 222, is provided in the Main Control Room via status lights on panel 9- 3

(iii) Annunciator (XA-55-4C window 15) is provided to the MCR should either valve is mis-positioned, i.e. FCV-64-221 or 222 not fully closed

(c) Controls are located on panel 9-3 containment mimic

(d) EOI Appendix 13 covers operation for this vent path.

5. Primary Containment Pressure Response for LOCA

a. The overall initial assumptions are as described in the FSAR.

(1) The reactor is operating at the most severe condition at the time of the LOCA, which maximizes the parameter of interest: primary containment response, fission product release, or core standby cooling system requirements.

(2) A complete loss of normal AC power occurs simultaneously with the LOCA. This additional condition results in the longest delay time for the core standby cooling systems to become operational.

(3) The LOCA assumes that a recirculation loop suction line is instantly severed resulting in the most rapid coolant loss and depressurization with coolant discharge from both ends of the break.

b. Response analysis: A primary containment peak pressure during a LOCA is dependent on amount of energy available to be added to the containment and the amount of energy that can be absorbed by the containment. Figure-4 & 5

Note: The most limiting conditions are used for calculating the peak containment pressure, so the calculated value is larger than that actually expected but less than the design of 56 psig.

(1) Factors affecting containment peak pressure are:

(a) Total energy in the coolant and core - Takes into account power, feed water temperature, and core age.

(b) Drywell wall and atmosphere temperature - Limiting factor in amount of energy the containment can control.

(c) Suppression pool temperature - Will determine amount of steam that can be condensed.

OPL171.016, Primary and Secondary Containment, Rev. 25U1

g. Containment characteristics after LOCA with torus-drywell vacuum breakers failed open	
(1) Steam flows from drywell to torus through vacuum breakers, equalizing pressure between the two immediately.	ILT/LOR-5,
(2) Steam is not forced through the water of the suppression pool, so now it will operate only as a surface condenser.	OF-5 NLO/NLOR-8
(3) As a result, drywell pressure will probably exceed the design pressure.	
(4) To prevent this occurrence, light indication is associated with each vacuum breaker. It indicates if the valve is off its seat and is displayed in the control room. Additionally, Drywell-Torus DP ensures vacuum breakers are closed.	
h. Containment characteristics after LOCA with torus-drywell vacuum breakers failed closed.	ILT/LOR-5,
(1) Steam in drywell will condense and drywell pressure will lower.	OF-5 NLO/NLOR-8
(2) With vacuum breakers failed shut, pressure cannot equalize between the suppression pool and the drywell.	Figure-6
(3) With this condition drywell pressure may lower such that external design pressure is reached.	
(4) To prevent this, the vacuum breakers are operationally checked periodically and 133% capacity is provided with 12 vacuum breakers	
i. Hydrogen Generation due to a Primary Rupture	
(1) Hydrogen is generated from five sources:	
(a) ZIRCALLOY - steam reaction	ILT/LOR-6, NLO-9
(b) Steel - steam reaction	Figure-7
(c) Zinc-steam reaction	
(d) Corium - concrete	
(e) Radiolytic decomposition of water.	
(2) Zirconium Steam Reaction during Accident Conditions	
(a) The zirconium-steam reaction is a major source of hydrogen during degraded core conditions and is most directly related to fuel integrity (or the lack of it).	
(1600°F) (Heat)	ILT/LOR-6, NLO-9
$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + 2800$ Btu/lb-Zr	Figure-6
(b) This equation shows the overall reaction describing the oxidation of zirconium, the production of hydrogen and the accompanying release of heat.	

Excerpt from 2-ARP-9-3B: Supports Distractors A2, C2

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0040 Page 26 of 39

DRYWELL TEMP/PRESS HIGH
2-XA-64-67
22

(Page 1 of 1)

Sensor/Trip Point:

(A)	PT-64-67	35 psig
(B)	TS-64-52A	275°F exists with
	PT-64-58A	DW pressure having exceeded 2.45 psig for 30 minutes.

Sensor Location:	PT-64-67 Panel 25-6A Reactor Building Elevation 593' Column R-12 Q-LINE	TE-64-52A Drywell Elevation 584' 225°F AZ	PT-64-58A Panel 25-6B Reactor Bldg El 593' Column R-12 Q-LINE

Probable Cause:

A. Primary containment pressure reaches 35 psig.
B. A Drywell atmospheric temperature of 275°F exists with Drywell pressure having exceeded 2.45 psig for 30 minutes after an incident begins.
C. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** containment pressure and temperature by multiple indications.

B. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References:	45E620-3 GE 730E937-12	2-47E610-64-1 and 2 GE 730E943-1	47W600-57A, 90

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295025 (EPE 2) High Reactor Pressure / 3

EA1.09 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to High Reactor Pressure:

- RPS

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295025EA1.09	
Importance Rating	4.2	------

Proposed Question: **# 13**

Unit 2 is at 60% RTP when the following conditions occur:

- Main Steam Isolation Valves (MSIVs) A and D inadvertently close
- Reactor Pressure peaks at 1065 psig

NOTE: GIVEN THE CONDITIONS ABOVE, CONSIDER EACH STATEMENT BELOW SEPARATELY.

Based **ONLY** on MSIV Position, a SCRAM **(1)** occur.

Based **ONLY** on the peak Reactor Pressure, a full automatic Reactor SCRAM **(2)** occur.

A. (1) will
(2) will

B. (1) will
(2) will NOT

C. (1) will NOT
(2) will

D. (1) will NOT
(2) will NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that if any 3 Main Steam Lines close (90% open), a Reactor SCRAM occurs. Second part is incorrect but plausible in that a Reactor Pressure of 1058 psig results in a High Reactor Pressure alarm in the Main Control Room.

B INCORRECT**:** First part is incorrect but plausible *(See A).* Second part is incorrect but plausible *(See D).*

C INCORRECT**:** First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See attached)* In accordance with 2-OI-99, Reactor Protection System, any 3 Main Steam Lines close (90% open) results in a Reactor SCRAM Initiation Signal. Isolating MSIVs A and D or B and C does not result in a half or full SCRAM. For second part, in accordance with 2-OI-99, a High Reactor Pressure of 1073 psig results in a Reactor SCRAM Initiation Signal therefore with the given Reactor Pressure peaking at 1065 psig, a Reactor SCRAM will not occur.

RO Level Justification: Tests the candidate's ability to operate and/or monitor the Reactor Protection System as it relates to High Reactor Pressure. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-ARP-9-5A, Rev. 63 (Attach if not previously provided)
2-OI-99, Rev. 100
OPL171.028, Rev. 26U2

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.028 Obj. 4b (As available)

Question Source:

Bank #	
Modified Bank #	(Note changes or attach parent)
New	**X**

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpt from 2-OI-99:

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0100 Page 105 of 116

Attachment 2
(Page 2 of 2)

Unit 2 Reactor Scram Initiation Signals

	Scram	Setpoint	Bypass
10.	OPRM TRIP	Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.	Reactor is NOT operating in the AUTO ENABLE Region of the Power/Flow Map or ABSP Enabled
11.	Low RPV Water Level (Level 3)	+2.0"	NA
12.	Hi RPV Pressure	1073 psig	NA
13.	Hi DW Pressure	2.45 psig	NA
14.	MSIV closure	90% open (3 Main Steam Lines)	Mode Switch NOT in RUN
15.	Scram Discharge Instrument Volume Hi Hi	Thermal level switches 49 gallons (LS-85-45A,B,G,H) Float level switches 45 gallons (LS-85-45C,D,E,F)	Mode Switch in SHUTDOWN or REFUEL with keylock switch in BYPASS
16.	TSV Closure	90% open (3 TSVs)	< 26% Rx Power (≤ 116.7 psig 1st stage pressure)(TR 3.3.1)
17.	TCV Fast Closure (load reject)	40% mismatch (amps to cross-under pressure); 850 psig EHC RETS at TCV (1 or 3) & (2 or 4)	< 26% Rx Power (≤116.7 psig 1st stage pressure)(TR 3.3.1)
18.	Loss of RPS Power	NA	NA
19.	Scram Channel Test Switches	NA (Key-locked in NORMAL Panels 2-9-15 & 2-9-17)	NA

Excerpt from 2-ARP-9-5A:

BFN Unit 2	Panel 9-5 2-XA-55-5A	2-ARP-9-5A Rev. 0063 Page 4 of 48

REACTOR PRESS HIGH
2-PA-3-53
1

(Page 1 of 2)

Sensor/Trip Point:
RFW Control System
2-RLY-003-0053PH 1058 psig
Average of valid Wide Range Pressure signal from RFW Control System. Signals originate from 2-PT-3-54, 3-61 and 3-207.

Sensor Location:
2-RLY-003-0053PH in Panel 2-9-97 (Unit 2 Control Room)
2-PT-3-54 on Panel 2-25-5B (Rx Bldg El 593', R-10-S-LINE)
2-PT-3-61 on Panel 2-25-6B (Rx Bldg El 593', R-12-P-LINE)
2-PT-3-207 on 2-PNLA-925-0426 (Rx Bldg El 593',R-10-T-LINE)

Probable Cause:
A. Group 1 isolation.
B. Maximum combined flow limiting.
C. Failed sensor.
D. Failure of Turbine EHC Bypass Control system.

Automatic Action:
None (Reactor Scram at 1073 psig)

Operator Action:
A. **CHECK** pressure by multiple indications including Reactor Controlling Pressure on 2-PR-3-53 recorder, 2-PI-3-54, 2-PI-3-61 and 2-PI-3-207 on Panel 2-9-5.
B. **IF** 2-PI-3-54, 2-PI-3-61, or 2-PI-3-207 on Panel 2-9-5 has failed or is invalid, **THEN**
with Unit SRO permission, **BYPASS** the affected pressure instrument. **REFER TO** 2-OI-3 Section 8.3.
C. **IF** alarm is valid and bypass valves have **NOT** opened, **THEN** **CHECK** for the following alarms:
- EHC SYSTEM TROUBLE annunciation, 2-XA-55-7B window 6, in ALARM
- BYPASS VALVE CONTROLLER FAILURE annunciation, 2-XA-55-7B Window 12, in ALARM

Continued on Next Page

Excerpt from OPL171.028 Lesson Plan:

d) As with Turbine Stop Valves, a maximum of two steam lines may be isolated (< 90 full open on MSIVs) without SCRAMMING the reactor.

(1) Isolating one steam line does not cause a half-SCRAM.
(2) Isolating steam lines A and D, or B and C, does not cause a half-SCRAM.
(3) Isolating any other combination of two main steam lines will cause a half-SCRAM.
(4) Isolating any combination of three main steam lines will cause a reactor SCRAM.
(5) At high power levels it is possible to cause a SCRAM by fully closing an MSIV. The SCRAM is caused by high reactor pressure, high neutron flux or high steam flow.

2-730E915RF-11
2-730E915RF-12

6. High drywell pressure 2.45 psig

LOR/ILT 5
PIS-64-56A thru 56D
2-730E915 sheets 9, 10

a) a. 2.45 psig indicates a leak in the primary system within the drywell. The SCRAM limits the amount of energy transmitted to the drywell during a LOCA. And if the core is uncovered due to the LOCA, the scram prevents an uncontrolled return to criticality when the core is reflooded.

Drawing
2-730E915RF-11
2-730E915RF-12

(1) Signal comes from four independent pressure switches.
(a) PIS-64-56A
(b) PIS-64-56B
(c) PIS-64-56C
(d) PIS-64-56D

2-730E915RF-11
2-730E915RF-12

7. Reactor high pressure 1073 psig

a) High pressure poses a direct threat of rupture to the nuclear system process barrier.
b) A SCRAM counteracts a pressure rise by quickly reducing the core fission heat generation.

2-730E915 sheets 9, 10
2-730E915RF-11

(1) Signal comes from four independent pressure switches.
(a) PIS-3-22-AA
(b) PIS-3-22-BB
(c) PIS-3-22C
(d) PIS-3-22D

8. Reactor low water level +2 inches, (Low Level 3)

LIS-3-203A thru 203D

a) Low reactor water level indicates that the reactor core is in danger of experience inadequate core cooling.
b) The effect of a lowering reactor water level while the reactor is operating at power is to reduce inlet subcooling.
c) Signal is from 4 level indicating switches.

2-730E915RF-12

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215003 (SF7 IRM) Intermediate-Range Monitor System / 7 **K2.01** (10CFR 55.41.7) Knowledge of the electrical power supplies to the following: • IRM channels/detectors	Tier #	2	------
	Group #	1	------
	K/A #	215003K2.01	
	Importance Rating	3.4	------

Proposed Question: **# 14**

The power supply to the Intermediate Range Monitors is the **(1)**.

When required, IRM 'C' provides a SCRAM signal to RPS bus **(2)**.

A. (1) 24VDC Power Distribution Panel
(2) A

B. (1) 24VDC Power Distribution Panel
(2) B

C. (1) 120VAC RPS Power Supply Distribution Panel
(2) A

D. (1) 120VAC RPS Power Supply Distribution Panel
(2) B

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 2-OI-92A, Intermediate Range Monitors, Illustration 1-IRM Trip Outputs list IRM Inoperable trip signals and setpoints. A loss of 24VDC is listed as an IRM trip signal. For second part, IRMs are separated into two sections for generating SCRAM signals for RPS 'A' and 'B': RPS 'A' – IRM A, C, E, G and RPS 'B' – IRM B, D, F, H.

B INCORRECT: First part is correct (*See A).* Second part is incorrect but plausible in that IRM B, D, F, H input to generate the RPS 'B' SCRAM signal.

C INCORRECT: First part is incorrect but plausible in that opening either Auxiliary Instrument Room's Panel 9-15(17), RPS TRIP SYSTEM 'A'(B) POWER, 120V breaker(s) result in a half SCRAM, opening both results in a full SCRAM. Additionally, all neutron monitoring detectors receive various voltage supplies from 0-350 Volts. Any combination of those voltages could be thought to be correct by the candidate. Although supplied voltages can be varied, the actual 125V distribution panel does not provide power to the neutron monitoring system. Second part is correct (*See A).*

D INCORRECT: First part is incorrect but plausible (*See C).* Second part is incorrect but plausible (*See B).*

RO Level Justification: Tests the candidate's knowledge of electrical power supplies as it relates to Intermediate Range Monitors. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to IRM power supplies.

Technical Reference(s): 2-OI-92A, Rev. 29 (Attach if not previously provided)
2-OI-99, Rev. 100

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.020 Obj. 3__ (As available)

Question Source:

	Bank #	
	Modified Bank #	BFN 2204 #14
	New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2204

Question Cognitive Level:

Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Copy of Bank Question:

Proposed Question: **# 14**

A Unit 3 Reactor Startup is in progress, when the Operator observes the following:



- SRM DOWNSCALE, (3-9-5A, Window 6), alarms
- SRM HIGH/INOP, (3-9-5A, Window 13) alarms
- IRM CH A, C, E, G HI-HI/INOP, (3-9-5A, Window 33), alarms

Given the conditions above, which **ONE** of the following power sources, if lost, would cause these failures?

A. 24VDC Power Distribution Panel

B. 125VDC Power Distribution Panel

C. 120VAC RPS Power Supply Distribution Panel

D. 120VAC Instrument and Control Power Distribution Panel

Proposed Answer: **A**

Excerpts from 2-OI-92A:

BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 20 of 20

Illustration 1
(Page 1 of 1)
IRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
IRM High	> 90 ON 125 SCALE	Rod block unless REACTOR MODE SWITCH in RUN
IRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless REACTOR MODE SWITCH in RUN Reactor Scram unless REACTOR MODE SWITCH in RUN
IRM Downscale	< 7.5 on 125 SCALE	Rod block unless IRMs on range 1 unless REACTOR MODE SWITCH in RUN
IRM Detector Wrong Position	detector not full in	Rod block unless detector full-in, or REACTOR MODE SWITCH in RUN
IRM High-High	> 116.4 ON 125 SCALE	Reactor Scram unless REACTOR MODE SWITCH in RUN

BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 14 of 20

NOTE

1) All IRM Rod Block Trips and IRM Scram Trips are automatically bypassed when the reactor mode switch is in the RUN position.
2) More than one IRM detector may be withdrawn at a time if needed. During a Reactor Startup one IRM Power Indication per channel should used to monitor power during IRM withdraw.
3) The DRIVE IN Circuitry may need to be reset by pressing the DRIVE IN 2-HS-92-7C/S2 if the IRM Detectors fail to withdraw.
4) All operations are performed on Panel 2-9-5 unless specifically stated otherwise.
5) IRMs are separated into two sections for RPS "A" and RPS "B". This section refers to withdrawing IRM detector "A" but each IRM can be substituted using the appropriate detector designator in place of the "A" reference.
   - RPS A- IRM A, C, E, G
   - RPS B- IRM B, D, F, H

6.3 Withdrawing IRMs

[1] **VERIFY** Source Range Monitor System in standby readiness. **REFER TO** Section 4.0. ☐

[2] **REVIEW** all precautions and limitations. **REFER TO** Section 3.0. ☐

[3] **SELECT** the IRM channel to be withdrawn by:

PERFORMING the following: (Refer to notes above.)

[3.1] **DEPRESS** SRM/IRM DETECTOR POSITION 2-HS-92-7C/S1 pushbutton.

AND

OBSERVE the associated pushbutton background light ILLUMINATES. ☐

[3.2] **DEPRESS** the appropriate IRM channel SELECT pushbutton for the IRM(s) to be withdrawn.

AND

OBSERVE the appropriate background SELECT light ILLUMINATES. ☐

Excerpt from 2-OI-99:

3.0 PRECAUTIONS AND LIMITATIONS

A. Failure to reset a half-scram or PCIS trip as soon as possible raises the chances of a full scram or isolation.

B. Loss of control air pressure results in the insertion of control rods.

C. When either RPS Bus A or B is de-energized, ability to bypass the SDV High Level Scram Trip Signal is lost.

D. Opening one of the following 120 VAC breakers results in a half-scram; opening both results in a full scram.

1. At Panel 2-9-15, REACTOR PROTECTION SYSTEM TRIP SYSTEM A POWER, Breaker 5A-CB1A.

2. At Panel 2-9-17, REACTOR PROTECTION SYSTEM TRIP SYSTEM B POWER, Breaker 5A-CB1B.

E. REACTOR MODE SWITCH should not be moved from SHUTDOWN or REFUEL unless directed by an approved operating, surveillance, or test instruction with Shift Manager's approval.

F. Radiation Protection (RP) shall be notified to perform a follow-up survey of the SDV cage areas following the reset of any SCRAM.

G. If a scram setpoint is exceeded and automatic scram does not occur, manual scram should be initiated and Shift Manager notified immediately.

H. The Unit Preferred Transformer, 2-XFA-252-2, is an alternate source for the Unit Preferred System, and an alternate for RPS Busses 2A and 2B. Due to transformer loading restrictions, only the Unit Preferred System or RPS Bus 2A or RPS Bus 2B shall be supplied from the transformer at any time. Due to the magnetic properties of this transformer, an unloaded "BURN-IN" (energized) period may be required to enable the output voltage to stabilize within specification (+/- 2% of 120V). Without the "BURN-IN" period the output voltage may initially be elevated, which could initiate an RPS Over Voltage Alarm. Therefore, if voltages are elevated, the "BURN-IN" period ensures that the output voltage of the transformer stabilizes to within output voltage specifications to avoid the RPS Over Voltage Alarm.

I. [NRC/C] Time an RPS Bus is energized from Unit Preferred Transformer (alternate source) should be minimized.

J. When transferring the 4kV power supply to 480V Shutdown Bd 2B, with the RPS Bus energized from Unit Preferred Transformer, the probability of tripping RPS Circuit Protectors due to voltage fluctuations is raised. [LER 259/87003, 260/88006]

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
212000 (SF7 RPS) Reactor Protection **K6.01** (10CFR 55.41.7)	Tier #	2	------
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Reactor Protection System: • AC electrical distribution	Group #	1	------
	K/A #	212000K6.01	
	Importance Rating	3.9	------

Proposed Question: **# 15**

Unit 2 was operating at 100% RTP when 480V RMOV Board 2B is de-energized.

Which **ONE** of the following completes the statement below regarding the impact to the Reactor Protection System (RPS)?

RPS Motor Generator (MG) Set **(1)** and circuit protectors **(2)** are de-energized.

A. (1) 2A
(2) A1 and A2 **ONLY**

B. (1) 2A
(2) B1 and B2 **ONLY**

C. (1) 2B
(2) A1 and A2, **AND** the alternate source circuit protectors C1 and C2

D. (1) 2B
(2) B1 and B2, **AND** the alternate source circuit protectors C1 and C2

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the BFN Electrical Distribution System alignment and loads, both AC and DC, are easily and often confused especially between 4KV and 480V Shutdown Boards and RMOV Boards including 120V loads. Second part is incorrect but plausible if the candidate does not accurately recall power supplies to the normal and alternate source circuit protectors associated with RPS.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that B1 and B2 circuit protectors are correct, but C1 and C2 circuit protectors are also lost when 480V RMOV 2B is lost.

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible if the candidate does not accurately recall power supplies to the normal source circuit protectors associated with RPS.

D **CORRECT:** *(See attached)* In accordance with 2-ARP-9-8C, (2-9-5B, Window 10) which is not given in the stem, 480V RMOV Board 2B provides normal and alternate power to the RPS MG SET 2B. The Unit 2 alternate power is supplied through a transformer shared with the Unit Preferred System (UPS) from 480V RMOV Board 2B. This is also illustrated on PIP-02-03, AC ELECTRICAL DISTRIBUTION SYSTEM BFN. For second part, REACTOR PROTECTION 120V POWER SYSTEM ABNORMAL, (2-9-5B, Window 10), alarms due to an abnormal RPS power lineup. It is actuated when one of the normal supplies is transferred to alternate or when a circuit protector has tripped. This Alarm Response Procedure refers to 2-AOI-99-1, Loss of Power to One RPS Bus, related to any one circuit protector tripping (2A1, 2A2, 2B1, 2B2, 2C1, 2C2). The outputs of 2A(2B) RPS MG Sets and alternate power transformer are monitored by circuit protectors with two protectors in series from each source. 2-AOI-99-1 contains resetting circuit protectors B1 and B2, and the alternate source circuit protectors C1 and C2 for RPS MG Set 2B.

RO Level Justification: Tests the candidate's knowledge of the effect that an AC Electrical Distribution System component malfunction has on the Reactor Protection System (RPS). This question is rated as Memory due to the requirement to strictly recall facts related to RPS power supply alignment.

Technical Reference(s): 2-ARP-9-5B, Rev. 33 (Attach if not previously provided)
2-ARP-9-8C, Rev. 22
2-AOI-99-1, Rev. 31
OPL171.028, Rev. 26U2

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.028 Obj. 3c (As available)

Question Source:

Bank #	**ILT EXAM BANK OPL171.028-11 001 #1051**
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Form 4.2-1 **Written Examination Question Worksheet**

Question Cognitive Level:	Memory or Fundamental Knowledge	**X**
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

Copy of Bank Question:

1051. OPL171.028-11 001

Which **ONE** of the following identifies which RPS MG Set and circuit protectors trip on a loss of 480 VAC RMOV Board B?

RPS MG Set__(1)__ Circuit Protectors __(2)__.

A. (1) B
(2) B1 and B2 ONLY

B✔ (1) B
(2) B1 and B2, AND the alternate source circuit protectors C1 and C2

C. (1) A
(2) A1 and A2, AND the alternate source circuit protectors C1 and C2

D. (1) A
(2) A1 and A2 ONLY

Excerpt from 2-ARP-9-8C:

BFN Unit 2	Panel 9-8 2-XA-55-8C	2-ARP-9-8C Rev. 0022 Page 31 of 47

480V REACTOR
MOV BD 2B OR 2E
UV
2-EA-57-109

24

(Page 1 of 2)

Sensor/Trip Point:

Any or all of the following: Loss of voltage

Bd 2B-(27NX-2) 2-27X-268B/2ANX-2
(27B) 2-27-268-B/16A-B
Bd 2E-27X-1A, (27B) 2-27-268-E/5A-B

52E, 52N Bell Alarm Overcurrent

Sensor Location:

480V Rx MOV Bd 2B
El 593', R-14 R-LINE
Shutdown Bd B Rm

480V Rx MOV Bd 2E
El 621', R-8 U-LINE

Probable Cause:

A. Normal and alternate breakers open due to overcurrent or loss of normal and alternate supply.
B. Fuse failure (metering PTs, control power, etc.).
C. Relay malfunction.

Automatic Action:

Undervoltage, loss of the following:

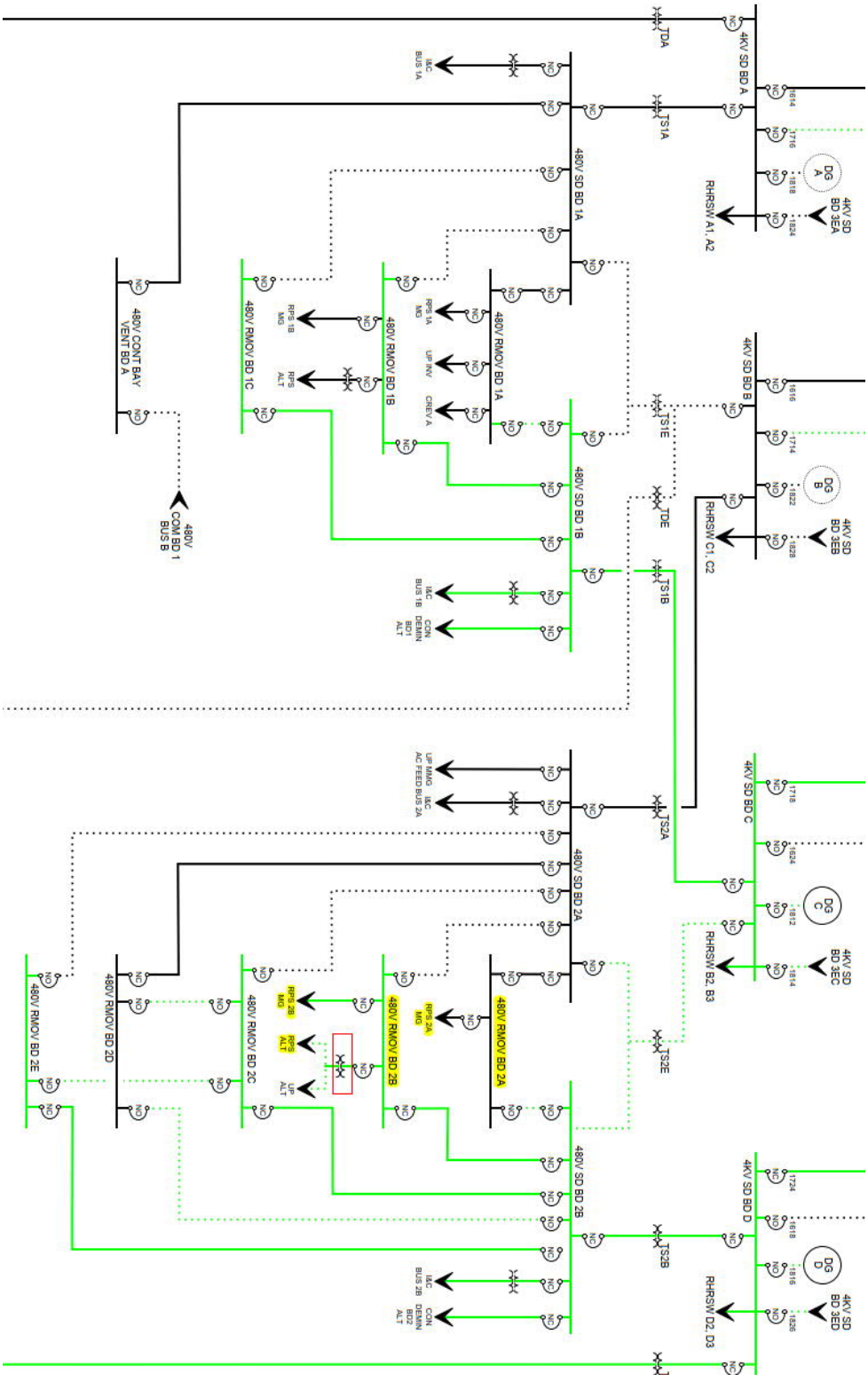
A. 480V Rx MOV Bd 2B.
   - RPS MG Set 2B.
   - DW Equip and Floor Drain sump pumps 2B.
   - DW blowers 2A-4 and 2B-4.

B. 480V Rx MOV Bd 2E.
   - Reactor Recirc Pump 2A discharge valve (2-FCV-68-3).
   - RHR inboard valve (2-FCV-74-67).

C. Complete listing on reference drawing.

Continued on Next Page

Excerpt from PIP-02-03:



AC ELECTRICAL DISTRIBUTION SYSTEM
BROWNS FERRY NUCLEAR PLANT

PIP-02-03 07/30/2019

Excerpt from 2-ARP-9-5B:

BFN Unit 2	Panel 9-5 2-XA-55-5B	2-ARP-9-5B Rev. 0033 Page 13 of 43

REACTOR PROT
120V PWR SYS
ABNORMAL

10

(Page 1 of 1)

Sensor/Trip Point:	
Bus A	42T-A circuit breaker
Bus B	42T-B circuit breaker
Bus A	2A1 circuit protector
Bus A	2A2 circuit protector
Bus B	2B1 circuit protector
Bus B	2B2 circuit protector
Unit Preferred TUP-2	2C1 circuit protector
Unit Preferred TUP-2	2C2 circuit protector

Sensor Location: Battery Board 2, Panel 9A, Bus A and B
A11 circuit protectors Battery Board Room 2.

Probable Cause:
A. Circuit breaker open.
B. RPS MG Set A or B abnormal.
C. One RPS channel on the alternate power supply.
D. Circuit protector tripped on over voltage, under voltage or under frequency. (Placing the Unit Preferred XFMR2 (TUP2) in service to an RPS bus without a burn-in period to stabilize transformer output voltage, can result in elevated output voltage.)
E. Sensor malfunction.
F. Cleared fuse for control power to any circuit protector.

Automatic Action: De-energizing one RPS channel actuates every scram-related alarm for the affected channel and actuates PCIS inboard or outboard logic.

Operator Action:
A. **REFER TO** 2-AOI-99-1.
B. **REFER TO** Tech Spec Sect 3.3.8.2.

References: 2-45E620-6 2-45E641-4 2-AOI-99-1
Technical Specifications

Excerpts from 2-AOI-99-1:

BFN Unit 2	Loss of Power to One RPS Bus	2-AOI-99-1 Rev. 0031 Page 4 of 19

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic action and operator action for the loss of power to one of the two RPS buses.

2.0 SYMPTOMS

2.1 Annunciators in alarm:

A. REACTOR PROT 120V PWR SYS ABNORMAL (2-XA-55-5B, Window 10).

1. Alternate power supply breaker (42T-A or 42T-B) CLOSED, or

2. Any one Circuit Protector tripped (2A1, 2A2, 2B1, 2B2, 2C1, 2C2).

B. All Reactor Scram and Primary Containment Isolation alarms associated with de-energized RPS bus.

C. RX BLDG VENTILATION ABNORMAL (2-XA-55-3D, Window 3).

1. Trip of Reactor and Refuel Zone supply/exhaust fans.

D. PANEL 25-32 BACKUP SW IN EMER POSN (2-XA-55-3E, Window 10) ref. TMOD BFN-2-2021-001-002

2.2 Control Room Indication:

A. Loss of Quadruple Low Voltage Power Supplies (QLVPS) as indicated by the extinguished green lights on the front of the QLVPS.

B. The Affected APRM/LPRM will have a "FAULT" indicated in Inverse Video on the top bar of the display panel. This indication will clear just after RPS is restored.

4.2 Subsequent Actions (continued)

[8] **RESTORE** Alternate Power to RPS Bus A as follows:

[8.1] **VERIFY** memory lights inside RPS Circuit Protector cabinets are RESET.

- RPS CIRCUIT PROTECTOR RESET PB CAB 2C1, 2-HS-099-0002C1
- RPS CIRCUIT PROTECTOR RESET PB CAB 2C2, 2-HS-099-0002C2

[8.2] **VERIFY** Circuit Protectors 2C1 and 2C2 are RESET.

- RPS CIRCUIT PROTECTOR 2C2, 2-HS-099-0002C1/1
- RPS CIRCUIT PROTECTOR 2C2, 2-HS-099-0002C2/1

[8.3] **VERIFY** ALTERNATE SOURCE AVAILABLE amber light ILLUMINATED.

[8.4] In Battery Board Room No. 2

PLACE RPS BUS 2A NORMAL/ALTERNATE TRANSFER SWITCH, 2-XS-099-0002A, to ALT.

[8.5] **NOTIFY** Unit 2 Operator that RPS 2A is on Alternate power supply.

[8.6] **PROCEED** to Step 4.2[12].

4.2 Subsequent Actions (continued)

[9] **RESTORE** Alternate Power to RPS Bus B as follows:

[9.1] **VERIFY** memory lights inside RPS Circuit Protector cabinets are RESET.

- RPS CIRCUIT PROTECTOR RESET PB CAB 2C1, 2-HS-099-0002C1
- RPS CIRCUIT PROTECTOR RESET PB CAB 2C2, 2-HS-099-0002C2

[9.2] **VERIFY** Circuit Protectors 2C1 and 2C2 are RESET.

- RPS CIRCUIT PROTECTOR 2C2, 2-HS-099-0002C1/1
- RPS CIRCUIT PROTECTOR 2C2, 2-HS-099-0002C2/1

[9.3] **VERIFY** ALTERNATE SOURCE AVAILABLE amber light ILLUMINATED.

[9.4] In Battery Board Room No. 2

PLACE RPS BUS B NORMAL/ALTERNATE TRANSFER SWITCH, 2-XS-099-0002B, to ALT.

[9.5] **NOTIFY** Unit 2 Operator that RPS 2B is on Alternate power supply.

[9.6] **PROCEED** to Step 4.2[12].

4.2 Subsequent Actions (continued)

[10] **STARTUP** and Load RPS MG Set 2A as follows:

[10.1] **VERIFY** RPS MG Set 2A available to be placed in service.

[10.2] **IF** RPS MG Set 2A has tripped, **THEN**

PERFORM steps 4.2[10.4] thru4.2[10.14]. (Otherwise N/A)

[10.3] **IF** RPS MG Set 2A is still running and resetting of the circuit protectors is desired, **THEN**

PERFORM steps 4.2[10.7] thru 4.2[10.14]. (Otherwise N/A)

[10.4] **VERIFY** OPEN RPS MG SET 2A TIE TO BAT BD 2 U-2 Breaker, 0-BKR-280-0002/902.

[10.5] **CLOSE** or VERIFY CLOSED RPS MG SET 2A OUTPUT Breaker, 2-BKR-099-0002A/1, at RPS MG Set 2A.

[10.6] **PERFORM** the following at RPS Bus 2A control panel to start RPS MG Set 2A: (N/A if Voltage is already approximately 120 volts)

[10.6.1] **DEPRESS** and **HOLD,** RPS MG SET 2A START/RESET/VOLTAGE BUILDUP pushbutton, 2-HS-099-0002A/A .

[10.6.2] **WHEN** voltage stabilizes on RPS MG SET 2A VOLTMETER, 2-EI-099-0002A, **THEN**

RELEASE 2-HS-099-0002A/A pushbutton.

[10.6.3] **ADJUST** RPS MG SET 2A VOLTAGE ADJUST, 2-HS-099-0002A to 120 Volts as indicated on RPS MG SET 2A VOLTMETER, 2-EI-099-0002A.

4.2 Subsequent Actions (continued)

[10.7] **VERIFY** or **RESET** memory lights inside RPS Circuit Protector cabinets:

[10.7.1] **DEPRESS** RPS CKT PROTECTOR RESET PUSH-BUTTON CAB 2A1, 2-HS-099-0002A1.

[10.7.2] **DEPRESS** RPS CKT PROTECTOR RESET PUSH-BUTTON CAB 2A2, 2-HS-099-0002A2.

NOTE

Red ON light should illuminate and green OFF light should extinguish. Green OFF light indicates circuit protector **NOT** reset but control power available.

[10.8] **RESET** or VERIFY Reset or CIRCUIT PROTECTORS 2A1 and 2A2 as follows:

[10.8.1] **DEPRESS** RPS CIRCUIT PROTECTOR 2A1 RESET, 2-HS-099-0002A1/1.

[10.8.2] **DEPRESS** RPS CIRCUIT PROTECTOR 2A2 RESET,2-HS-099-0002A2/1.

[10.9] **PLACE** or VERIFY RPS Bus A NORMAL/ALTERNATE TRANSFER SWITCH, 2-XS-099-0002A to NORM.

[10.10] **VERIFY** 120 Volts indicated on RPS MG SET 2A VOLTMETER, 2-EI-099-0002A and ADJUST RPS MG SET 2A VOLTAGE, 2-HS-099-0002A if necessary.

4.2 Subsequent Actions (continued)

[11] **STARTUP** and load RPS MG Set 2B as follows:

[11.1] **VERIFY** RPS MG Set 2B available to be placed in service.

[11.2] **IF** RPS MG Set 2B has tripped, **THEN**

PERFORM steps 4.2[11.4] thru 4.2[11.14]. (Otherwise N/A)

[11.3] **IF** RPS MG Set 2B is still running and resetting of the circuit protectors is desired, **THEN**

PERFORM steps 4.2[11.7] thru 4.2[11.14]. (Otherwise N/A)

[11.4] **VERIFY OPEN** RPS MG SET B TIE TO BAT BD 2 Breaker, 0-BKR-280-0002/952.

[11.5] **CLOSE** or VERIFY CLOSED RPS MG SET 2B OUTPUT Breaker, 2-BKR-099-0002B/1, at RPS MG Set 2B.

[11.6] **PERFORM** the following at the RPS Bus 2B control panel to start RPS MG Set 2B:(N/A if Voltage is already approximately 120 volts)

[11.6.1] **DEPRESS** and **HOLD**, RPS MG SET 2B START/RESET/VOLTAGE BUILDUP pushbutton 2-HS-099-0002B/A.

[11.6.2] **WHEN** voltage stabilizes on RPS MG SET 2B VOLTMETER, 2-EI-099-0002B, **THEN**

RELEASE 2-HS-099-0002B/A pushbutton.

[11.6.3] **ADJUST** RPS MG SET 2B VOLTAGE ADJUST, 2-HS-099-0002B to 120 Volts as indicated on RPS MG SET 2B VOLTMETER, 2-EI-099-0002B.

4.2 Subsequent Actions (continued)

[11.7] **RESET** or VERIFY RESET memory lights inside RPS Circuit Protector cabinets:

[11.7.1] **DEPRESS** RPS CKT PROTECTOR RESET PUSH-BUTTON CAB 2B1, 2-HS-099-0002B1.

[11.7.2] **DEPRESS** RPS CKT PROTECTOR RESET PUSH-BUTTON CAB 2B2, 2-HS-099-0002B2.

NOTE

Red ON light should illuminate and green OFF light should extinguish. Green OFF light indicates circuit protector **NOT** reset but control power available.

[11.8] **RESET** or VERIFY RESET CIRCUIT PROTECTORS 2B1 and 2B2 as follows:

[11.8.1] **DEPRESS** RPS CIRCUIT PROTECTOR 2B1 RESET, 2-HS-099-0002B1/1.

[11.8.2] **DEPRES**S RPS CIRCUIT PROTECTOR 2B2 RESET, 2-HS-099-0002B2/1.

[11.9] **PLACE** or VERIFY RPS Bus 2B NORMAL/ALTERNATE TRANSFER SWITCH to NORM.

[11.10] **VERIFY** 120 Volts indicated on RPS MG SET 2B VOLTMETER, 2-EI-099-0002B and **ADJUST** RPS MG SET 2B VOLTAGE, 2-HS-099-0002B if necessary.

Excerpts from OPL171.028 Lesson Plan:

B. Component Description	NLOR/NLO 2
1. The Reactor Protection System includes the motor-generator power supplies with associated control and indicating equipment, sensors, relays, bypass circuitry, and switches that supply a signal to the Control Rod Drive (CRD) system to cause rapid insertion of control rods (SCRAM) to shut down the reactor. It also includes outputs to the process computer system and annunciators. The system includes two motor-generator power supplies with an alternate supply via a transformer and the sensors, relays, bypass circuitry and switches that operate valves to cause rapid insertion of all control rods.	LOR/ILT 2 IL-1
2. Normal power to RPS Buses A and B on each unit is supplied by two motor-generator (MG) sets / one per bus which powers two independent trip systems.	
a) Motor: (1) 480VAC (2) 3 phase (3) Powered from 480V RMOV Boards A and B on the respective unit.	NLOR/NLO 3a, (OF-5) LOR/ILT 3a (OF-5)
b) Generator: (1) 120VAC output (2) Single phase (3) 60 Hz (4) Motor-generator flywheel maintains voltage and frequency within 5 percent of rated values for at least 1.0 second following total loss of power to the drive motor. This allows for loss of power due to switching operations.	
3. Alternate Power to RPS is supplied from 480V RMOV Board B through a transformer.	
a) Unit 1 alternate power is supplied through a transformer from 480V RMOV Board 1B	NLOR/NLO 3b, OF-5
b) Unit 2 alternate power is supplied through a transformer shared with the Unit Preferred System (UPS) from 480V RMOV Board 2B. Unit Preferred transformer can only supply UPS or one of two RPS busses at any one time. This is due to the load restrictions of the transformer.	Unit Difference LOR/ILT 6
c) Unit 3 alternate power is supplied through a transformer from 480V RMOV Board 3B.	
d) Interlocked so that both RPS buses cannot be simultaneously fed from the alternate power supply.	NLOR/NLO 4c LOR/ILT 4c
e) Interlocked to prevent paralleling RPS MG Set with the alternate power supply.	

OPL171.028 , Reactor Protection System, Rev.# 26U2

4. Controls for the motor generator sets and for switching from normal to alternate power are located on the respective Unit Battery Board, Panel 9. OI-99 contains the procedures for:

   a) Startup and Loading of RPS MG Sets
   b) RPS Bus A/B Power Transfer
   c) Shutdown of RPS MG Sets

   NLOR/NLO 7

5. The outputs of the RPS MG sets and alternate power transformer are monitored by circuit protectors. The purpose of the circuit protectors is to protect the loads connected to the RPS Bus against adverse voltage and frequency conditions.

   Tech Spec B.3.3.8.2
   NLOR/NLO 3c, OF-5

   a) There are two circuit protectors in series from each source. The circuit protectors trip to de-energize the associated power source if certain electrical parameters are exceeded.

   b) The circuit protectors trip on high voltage (≤132V), low voltage (≥108.5V), or low frequency (≥56 Hz). All the trips have a 3 second time delay. If the setpoint is cleared before the time delay expires, the circuit protector will not trip.

   IL-1

   c) If tripped, the circuit protectors must be manually RESET in the respective Unit battery board room. Control power to operate the circuit protectors is supplied from the output of the MG set or alternate supply. If the green light on the circuit protector is off when the circuit protector is tripped, there is no control power available to reset the circuit protector.

   d) Inside each circuit protector there are indicating lights the operator can use to determine the cause of the trip. A reset button below the lights is used to clear them.

   Reset P.B. outside on front of cabinet
   Memory P.B. inside cabinet

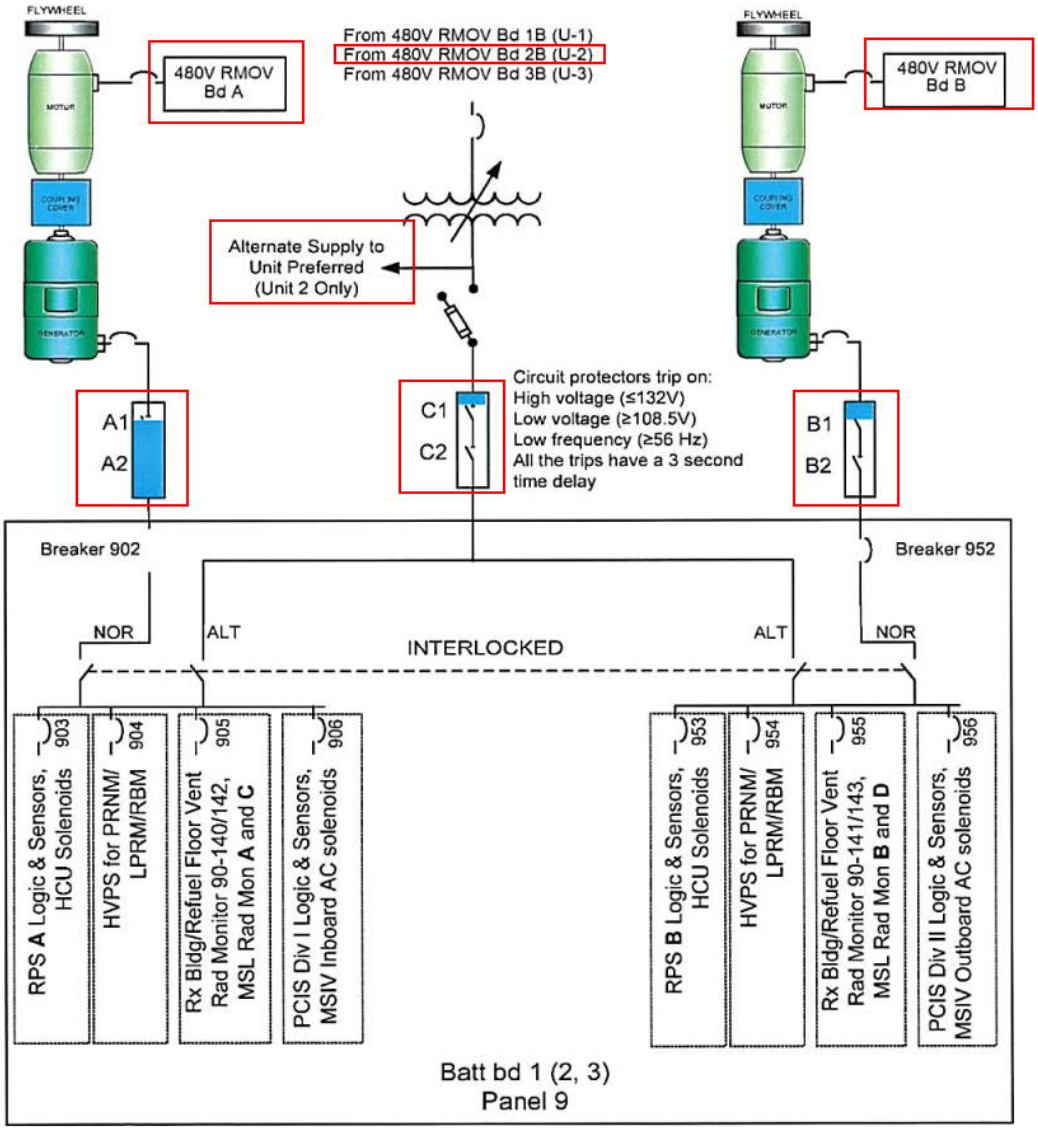
   e) A loss or malfunction of an RPS power supply will result in the de-energization of all RPS Channels powered from that power supply. This may result in a half or full scram depending on the power supplies lost and a loss of all loads off that power supply.

   LOR/ILT 10a

3. Manual Bypasses of SCRAMs	
a) SCRAM discharge volume high level is bypassed provided:	
(1) The Mode Switch is in SHUTDOWN or REFUEL, and	
(2) A keylock switch on Panel 9-5 is placed in the BYPASS position.	2-730E915RF-11 2-730E915RF-12
(3) Note: both A trip system and B trip system relays must be energized for this bypass to work in either trip system.	
(4) EOI Appendix 1F places jumpers that defeat all SCRAMs except the manual, Mode Switch in SHUTDOWN and individual rod SCRAM switches.	NLOR/NLO 7
(a) Jumpers are installed from relays 5A-K10 to 5A-K12 in each channel.	
(b) This allows resetting the SCRAM without clearing the initiating signal.	
(5) POI-200.4 defeats RPS, PCIS, and ECCS logic to minimize any unplanned actuations when there is no fuel in the vessel. This procedure de-energizes (trips) RPS logic.	
J. Technical Specifications	
1. TS 3.3.1.1, RPS Instrumentation	Discuss Appendix D & OI-99 Illustration 3
2. TS 3.3.2.1, Control Rod Block Instrumentation	
3. TS 3.3.6.1, PCIS Instrumentation	LOR/ILT 7 LOR/ILT 8
4. TS 3.3.6.2, Sec Cont. Isolation Instrumentation	
5. TS 3.3.8.2, RPS Electric Power Monitoring	
6. TRM 3.3.1, RPS Instrumentation	
7. TRM 3.3.4, Control Rod Block Instrumentation	
K. Alarms **	
1. The first out annunciator panel on 9-7 provides the operator with a visual indication of some of the parameters that may have initiated the reactor SCRAM. Refer to Appendix D for a listing of the first out annunciators.	
2. Other Alarms	LOR/ILT 5e
a) REACTOR PROT 120V PWR SYS ABNORMAL alarm indicates an abnormal RPS power lineup. It is actuated one of the normal supplies is transferred to alternate or when a circuit protector has tripped.	Pnl-9-5B-window 10

OPL171.028 , Reactor Protection System, Rev.# 26U2

IL-1: RPS POWER SUPPLIES



Examination Outline Cross-reference:

295031 (EPE 8) Reactor Low Water Level / 2

EA2.03 (10CFR55.41.10)

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:

- Reactor Pressure

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295031EA2.03	
Importance Rating	4.2	------

Proposed Question: **# 16**

A Unit 2 Loss of Coolant Accident (LOCA) has occurred and resulted in the following conditions:

- 2-PI-3-207A, REACTOR PRESSURE, indicates 400 psig
- 2-LI-3-52 and 2-LI-3-62, REACTOR WATER LEVEL ACCIDENT RANGE, indicates (-) 190 inches

Given the current conditions, which **ONE** of the following completes the statements below?

The Top of Active Fuel (TAF) **(1)** submerged at this time.

To help ensure accurate Reactor Water Level can be reported, 2-LI-3-52 and 2-LI-3-62 require the use of correction curves due to being calibrated at **(2)**.

[REFERENCE PROVIDED]

A. **(1)** is
(2) 0 psig / 212 °F

B. **(1)** is
(2) Normal Operating Pressure / Normal Operating Temperature

C. **(1)** is NOT
(2) 0 psig / 212 °F

D. **(1)** is NOT
(2) Normal Operating Pressure / Normal Operating Temperature

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* To ensure Adequate Core Cooling via Core Submergence in accordance with RPV Control (2-EOI-1) Table L-3, PIP 95-64 for 2-LI-3-52 / 2-LI-3-62 and BFN-ODM-4.20, Reactor Water Level must be above TAF (-) 162 inches. For the given Accident Conditions, the Operator is required to use the correction curves from the Operator aid on Panel 9-3 to determine that the given indicated Reactor Water Level at (-) 190 inches and the provided Reactor Pressure results Reactor Water Level above TAF (-) 162 inches, therefore the Reactor Core is submerged at this time. For second part, these two Post Accident Range Instruments (LI-3-52 / LI-3-62) are calibrated to indicate from (-) 268 inches to (+) 32 inches, a Reactor Water Level range. These Post Accident Monitoring (PAM) conditions assume the Reactor is depressurized to 0 psig (212 °F) in Accident Conditions and are NOT temperature compensated. Per Safety Analysis on Water Level instruments, the conclusion is that the Accident range instruments adequately indicate Water Level provided they are corrected for off-calibration conditions for RPV Pressure utilizing the Operator aid (PIP 95-64) on Panel 9-3 for Reactor Water Level Correction.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that other (Narrow Range) Level Instruments are calibrated for Normal Operating Pressure (1035 psig) / Normal Operating Temperatures (550) °F as indicated in 2-SR-2 (*attached*), NOT Accident Conditions and are Temperature Compensated by a pressure signal.

C INCORRECT: First part is incorrect but plausible if the candidate does not correlate the determined Reactor Water Level from the correction curves to be below actual Top of Active Fuel or (-) 162 inches. Any of the lines on the chart could be the Top of Active Fuel, as the lines are not labeled. The lines have the respective Reactor Water Level that they correspond to but it requires the candidate to know what each numerical Reactor Water Level is since the chart's legend is not given to the candidate. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability, utilizing Accident Condition parameters, to determine actual Reactor Water Level by correctly interpreting the required Reactor Water Correction Curves. This will help to ensure the status of Adequate Core Cooling via Core Submergence. This question is rated as C/A due to the requirement to assemble and integrate parameters using a graph to predict an outcome.

In reference to Operating Licensing Program Feedback, 401.55, Tier 1, Emergency and Abnormal Plant Evolutions, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) Assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems and/or safety functions.

Technical Reference(s): EOIPM 0-V-C, Rev. 3 (Attach if not previously provided)
0-TI-414, Rev. 19 (PIP 95-64)
2-SR-2, Rev. 92
OPL171.003, Rev. 26

Proposed references to be provided to applicants during examination: **2-LI-3-52 & 62 CORRECTION CURVES, PIP 95-64 (without legend)**

Form 4.2-1 **Written Examination Question Worksheet**

Learning Objective: OPL171.003 Obj. 15 (As available)
OPL171.202 Obj. 7

Question Source:

Bank #		
Modified Bank #	BFN 2104 NRC #23	(Note changes or attach parent)
New		

Question History:

Last NRC Exam	2021

Question Cognitive Level:

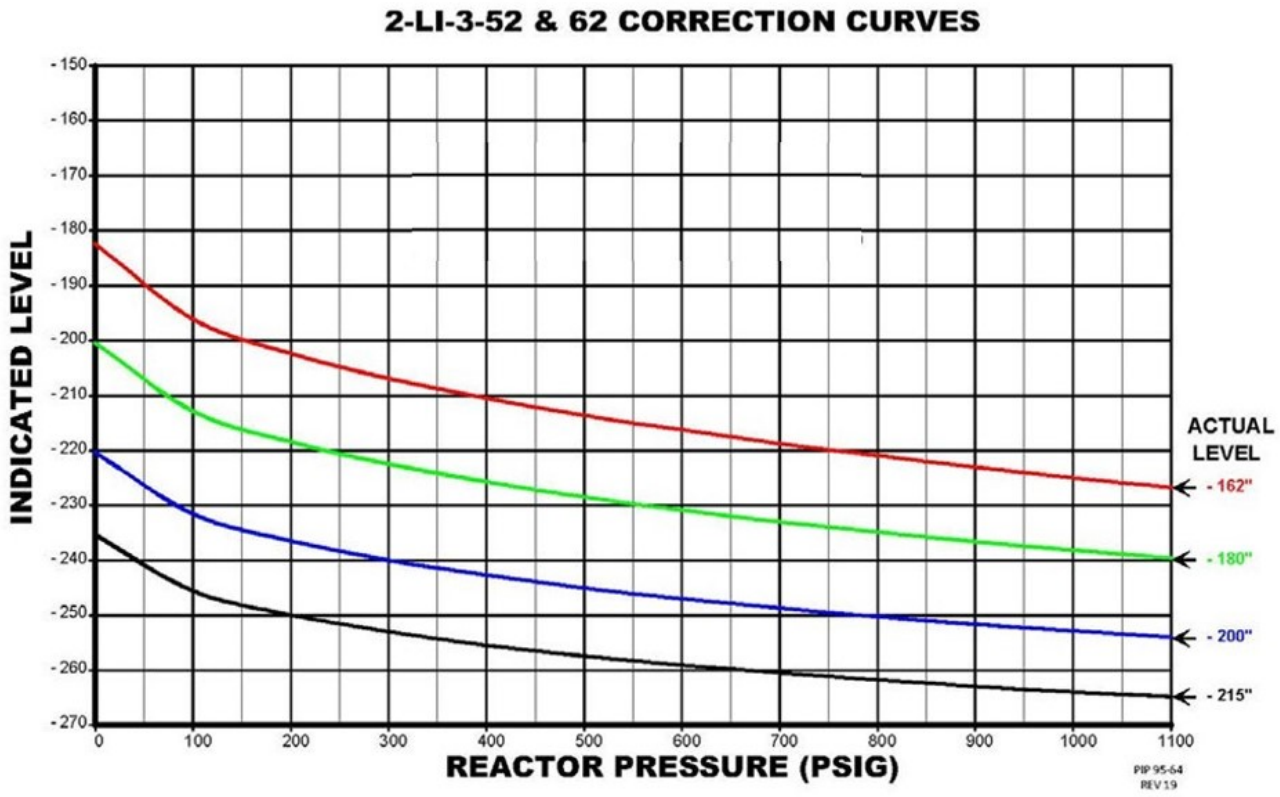
Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

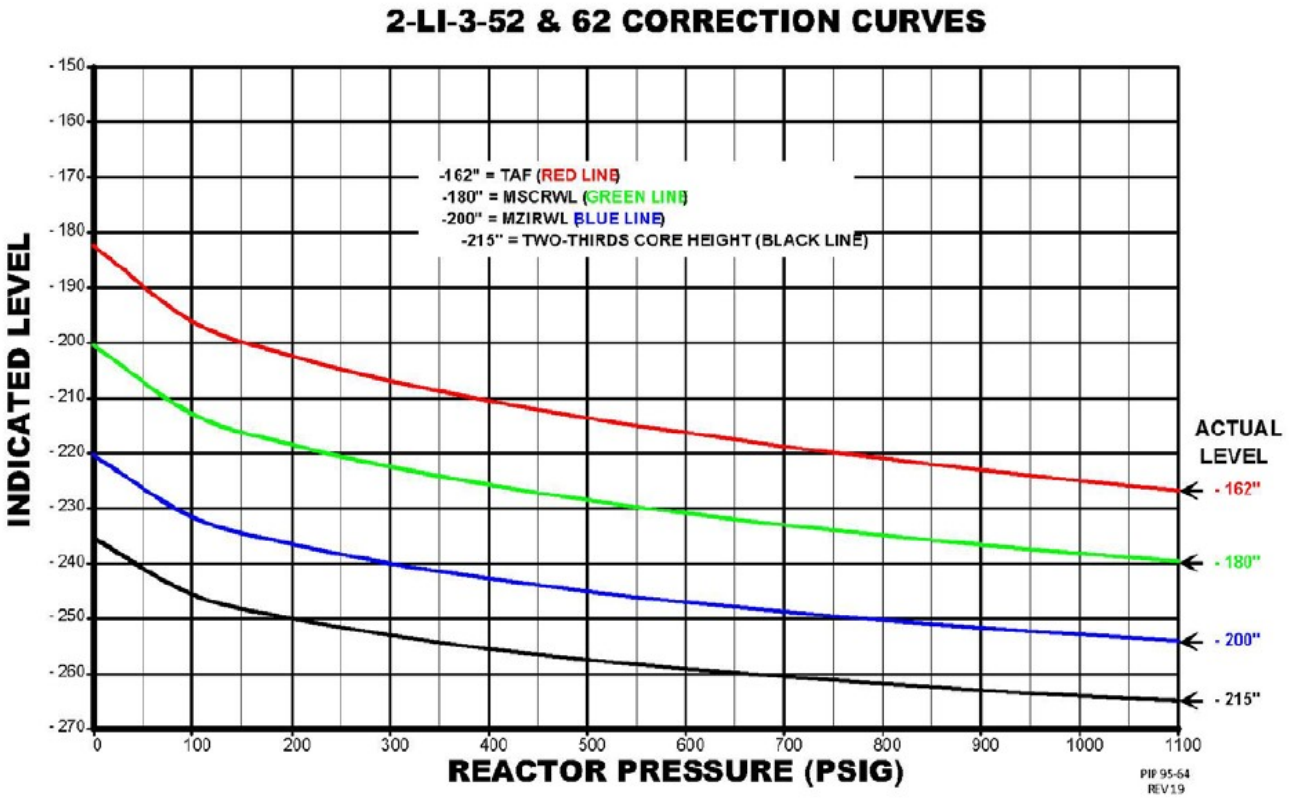
55.41	X
55.43	

Comments:

REFERENCE PROVIDED to candidate: PIP 95-64 without legend



PIP 95-64 with legend:



Copy of Bank Question:

Proposed Question: **# 23**

A Unit 2 LOCA has occurred and resulted in the following conditions:

- 2-PI-3-207A, REACTOR PRESSURE indicates 500 psig
- 2-LI-3-52 and 2-LI-3-62, RX WTR LEVEL ACCIDENT RANGE indicates (-) 220 inches

Given the conditions above, which **ONE** of the following completes the statements below?

Actual Reactor Water Level is **(1)** Top of Active Fuel (TAF).

To help ensure accurate Reactor Water Level can be reported, 2-LI-3-52 and 2-LI-3-62 require the use of correction curves due to being calibrated at **(2)**.

[REFERENCE PROVIDED]

A. (1) above
(2) 0 psig / 212 °F

B. (1) above
(2) Normal Operating Pressure / Normal Operating Temperature

C. (1) below
(2) 0 psig / 212 °F

D. (1) below
(2) Normal Operating Pressure / Normal Operating Temperature

Proposed Answer: **C**

Excerpts from EOIPM Section 0-V-C:

BFN Unit 0	EOI-1, RPV Control Bases	EOIPM Section 0-V-C Rev. 0003 Page 23 of 141

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L flowpath

The RPV water level control flowpath establishes and maintains adequate core cooling through core submergence. A list of the preferred injection systems for use in doing so is provided; along with a list of alternate injection subsystems which may be used to augment RPV water level control (refer to the definition of alternate injection subsystems in EOIPM Section 0-I-C).

BFN Unit 0	EOI-1, RPV Control Bases	EOIPM Section 0-V-C Rev. 0003 Page 75 of 141

1.0 EOI-1, RPV CONTROL BASES (continued)

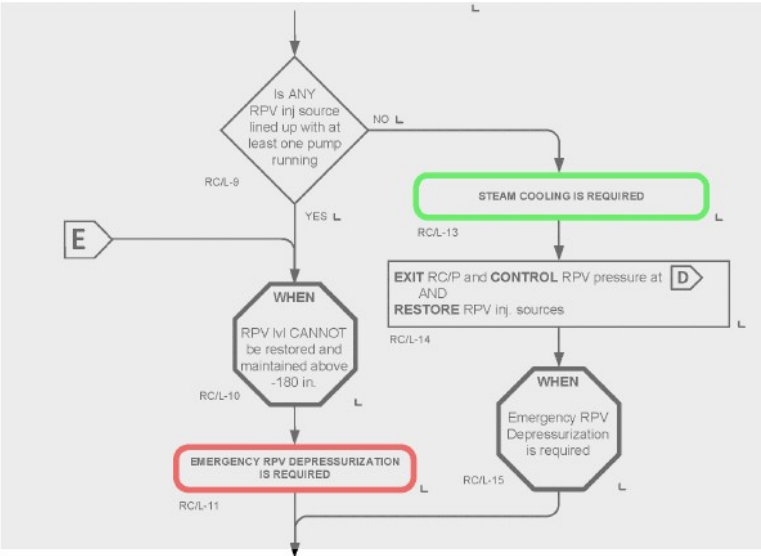
DISCUSSION: RC/L-12

Adequate core cooling with injection is ensured following emergency RPV depressurization as long as one of three conditions exists (Table L-3):

- The core is completely submerged in which case override RC/L-4 requires return to Step RC/L-3.
- RPV water level can be restored and maintained above **A.71** (Minimum Steam Cooling RPV Water Level). The core is then cooled by a combination of submergence and steam cooling even with no core spray flow.
- Design core spray flow requirements are satisfied and RPV water level can be restored and maintained at or above **A.49** (Two-Thirds Core Height Water Level).

1.0 EOI-1, RPV CONTROL BASES (continued)

RC/L-12 (cont'd)



RESTORE and **MAINTAIN** RPV water lvl above -180 in. using one or more Preferred Injection Systems (Table L-1)

- OK to use ANY of the Alternate Injection Subsystems (Table L-2)

IF	THEN
RPV water lvl cannot be restored and maintained above -180 in. AND Spray Cooling cannot be restored and maintained (Table L-3)	**MAXIMIZE** injection with ALL available sources
Maximizing injection with ALL available sources cannot restore and maintain Spray Cooling or RPV water lvl above -180 in. (Table L-3)	**WHEN** the TSC team assumes command and control **THEN** **EXIT** all EOIs and **ENTER** all SAMGs

RC/L-12

Table L-3
Adequate Core Cooling

Adequate core cooling exists if one of the following is met:

- Core submergence: RPV water lvl above -162 in.
- Steam Cooling with injection: RPV water lvl above -180 in.
- Spray cooling: Either CS subsystem operating with at least 6,250 gpm to the RPV
  AND
  RPV water lvl above -215 in.
- Steam cooling without injection: RPV water lv above -200 in.

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L-13, RC/L-14, RC/L-15

If the decreasing RPV water level trend has not been reversed before RPV water level drops to the top of the active fuel and no source of injection into the RPV is available (NO path from RC/L-12), the only mechanism able to provide adequate core cooling is steam cooling.

Flow path D, Steam Cooling is an RPV pressure control action and is thus entered by way of the third override in RC/P-4. While in flow path D, Step RC/L-14 continues attempts to restore RPV injection capability. Steam cooling is not initiated until RPV water level decreases to the top of the active fuel because:

- Adequate core cooling exists so long as RPV water level remains above the top of the active fuel.

Excerpt from 2-SR-2:

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0092 Page 153 of 159

Attachment 4
(Page 1 of 2)

Reactor Water Level Indication Correction

The Reactor Water Level Instrumentation tables in Attachment 2 are arranged such that only instruments in the same compensation group are compared. However, determination of corrected level indication may be required during operation at off-normal conditions or if desired to compare instruments from different compensation groups.

Corrected level indication may be used for satisfying MAX DEV criteria provided the following are observed:

- The parameter correction is appropriately applied to all instruments being compared.
- Both the indicated and corrected level indications are recorded and annotated in Attachment 2 along with the bases for the corrected level indication.

Corrected level indication can be determined from the following table which provides commonly needed corrections or from Technical Instruction 2-TI-149. The following table presents Reactor Water Level as: indicated, corrected for Reactor Vessel Temperature 100°F, and corrected for Reactor Vessel Temperature 212°F for various water level instruments. Enter the indicated Rx water level and find the correct instrument column and use the closest Rx vessel temperature. (Matching corrected levels between instruments and subtracting the associated indicated levels will yield an approximate deviation value in inches between those instruments. i.e., If the Narrow Range Compensated Instrument is reading 38", the corrected level would be 32". Also, using a corrected value of 32" in the Narrow Range Uncompensated column shows the instrument should be reading 47". Therefore, a deviation of approximately 9" would be expected between the Narrow Range Compensated instrument and the Narrow Range Uncompensated instrument.)

Attachment 4
(Page 2 of 2)

Reactor Water Level Indication Correction

INDICATED LEVEL	CORRECTED LEVEL										
Indicated Reactor Water Level	Narrow Range Compensated 2-LIS-3-53(60) (206) (253) Level		Narrow Range Uncompensated 2-LIS-3-184, 185,203(A-D), 208(A-D) Level		Wide Range 2-LI-3-58A(B), 2-LIS-3-56A(D) Level		Post Accident 2-LR-3-62 2-LI/LIS-3-52 2-LI/LIS-3-62A Level		Floodup 2-LI-3-55 Level		Wide Range 2-LI-3-46A(B) Level
	100°	212°	100°	212°	100°	212°	100°	212°	100°	212°	
50	40.5	42.5	34.5	36	Note 1	Note 2	No Calculated Correction Value		48.5	50	No Calculated Correction Value
48	39	41	33	35					46.5	48	
46	37.5	39.5	31.5	33					44.5	46	
44	36	38	30	32					42.5	44	
42	35	36.5	28.5	30.5					40.5	42	
40	33.5	35	27	29					38.5	40	
38	32	34	26	27.5					36.5	38	
36	30.5	32.5	24.5	26					34.5	35.5	
34	29	31	23	24.5					32.5	33.5	
32	28	29.5	21.5	23					30.5	31.5	
30	26.5	28	20	21.5					28.5	29.5	
28	25	26.5	19	20					26.5	27.5	

(1) Indicates > 60" if actual Water Level is > 5".

(2) Indicates > 60" if actual Water Level is > 11.5".

Excerpts from OPL171.003 Lesson Plan:

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

c. Definitions 1) Reactor vessel zero: a) Reactor pressure vessel bottom head invert (the inside of the bottom head) b) Provides reference for all in core components and vessel nozzle taps. 2) Instrument zero: a) 528 inches above vessel zero	Obj. ILT-4, NLO/NLOR-1
d. Four ranges of level indication 1) Normal Control Range (Narrow Range) a) 0 to +60 inch range covering the normal operating range (analog) with +60" up to +70" digital and 0" down to – 10" digital readings. b) Referenced to instrument zero c) Four of these instruments are used by Feedwater Level Control System (FWLCS). (1) The level signal utilized by the FWLCS is NOT directed through the Analog Trip System. (2) Temperature compensated by a pressure signal (3) Most accurate level indication available to the operator (4) Calibrated for normal operating pressure and temperature d) These indicators and a recorder point (average of the four) are located on Panel 9-5 NOTE: An air bubble or leak in the reference leg can cause inaccurate readings in a nonconservative direction resulting in a mismatch between level indicators. This problem is particularly prevalent after extended outages when starting up from cold shutdown conditions and at low reactor pressures e) Four other narrow range instruments are located in the control room, two above the FWLCS level indicators on panel 9-5 (3-208A & D), one above HPCI (3-208B) and one above RCIC (3-208C) on panel 9-3. f) Calibration Conditions: (1) Reactor Pressure ~ 1035 psig (2) Reactor Temperature ~ 550 F (3) DW Temperature 135 F (4) Reactor Bldg Temp 80 F (5) No Recirc flow impact	Obj. ILT-5 Obj. NLO/NLOR-2 Obj. ILT-6 Obj. NLO-4/ NLOR-3 Obj. ILT-11.i/ LOR-7.i, ILT-13. LOR-8,NLO-6

4) Post-accident Flood Range

a) -268" to +32" range covering active core area and overlapping the lower portion of the Normal Control Range.

b) Referenced to instrument zero

c) Calibration Conditions:
(1) Accident conditions
(2) Reactor pressure at 0 psig (212°F)
(3) DW temperature 212°F
(4) RB Temperature 212°F
(5) Recirculation pumps tripped.

d) Variable leg tap is from diffuser of jet pumps 1 and 6 (or 11 and 16).

e) Per Safety Analysis on water level instruments the conclusion is that the accident range instruments adequately indicate water level-- provided they are corrected for off-calibration conditions of RPV pressure utilizing the operator aid on Panel 9-3 for level correction.

f) An interlock associated with this range will prevent using the RHR System for containment de-pressurization when it is needed to flood the core region.

Obj.ILT-15./ LOR-9/ NLO-7

Obj.ILT-15./LOR-9 /NLO-7

Obj. ILT-11.d/ LOR-7.d

e. Level instrumentation and piping layout

1. The major portions of the reference legs for the normal control range (A and B) and the Emergency Systems Range (A and B) are outside the drywell, thereby greatly reducing the drywell temperature effects on indicated water level. The reference leg for level transmitter LT-3-55 is still in the drywell.
2. Reference leg piping penetrates the containment on the south side through the penetration for old head spray piping. On the north side the reference leg piping penetrates the primary containment via the drywell purge line.
3. Each reactor instrument line (inside the drywell) connects to a condensing chamber, which is just an enlarged volume in the piping.
4. Piping run is uphill from the vessel to the condensing chamber, and then it is sloped downward to the transmitters, making the condensing chamber the highest point of the instrument line.

b) In order for drains from the interior separators to flow outward, a hydraulic gradient is required.

c) At 100 percent power, the "top" of the dome is 4 inches higher than the "outside," providing the hydraulic gradient. The degree of hydraulic gradient is variable, ranging from 0 inches at 0 percent power to 4 inches at 100 % power.

h. Density effects on reactor vessel level ranges — Obj. ILT-9./ LOR-5

1) All vessel level indicators are calibrated at the reactor conditions for which they are to be used.

a) Narrow and Wide ranges are calibrated hot. Rated Operating pressure/ temperature

b) Flood-up and Post-accident ranges are calibrated cold.

2) Unless some density compensation is used, any level instrument actual measurement span will change with vessel pressure and temperature. This is due to changes in reactor water density with little or no change in reference column density.

The variable legs passing through the RWCU pump room may be at elevated temperatures introducing additional level mismatches between the A and B side instruments.

3) Plant transients can cause errors in the sensed level primarily because of the vessel pressure and drywell temperature variations.

a) Reference leg flashing is still possible during rapid de-pressurization transients; however, the effect is minimal since most of the reference legs (vertical runs) are located outside the drywell and are now much cooler than before (during reactor power operations). Additionally "RVLIS" minimize this effect. — Obj. ILT-10/ LOR-6

Reference leg flashing results in erroneous high level indications. Level setpoints are shifted in a non-conservative direction.

Transient flashing effects can cause indicated level to oscillate or be erratic. As the reference leg refills, the indicated level approaches a more accurate water level indication. The RVLIS mod decreases the time necessary for this refill to occur

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

i. Normal Control Range (Narrow Range) and Emergency Systems Range (Wide Range) Level Discrepancies	Obj. ILT-9/ LOR-5 Obj. ILT-15/LOR-9 Obj. NLO-7
1) Narrow Range level instrumentation is calibrated to be most accurate at rated temperature and pressure (particularly the instruments for FWLCS, since they are temperature compensated). At cold conditions the non-FWLCS instruments read high (not temperature compensated).	
2) Wide Range instruments are also calibrated for rated temperature and pressure	
a) The indicated level on the Wide Range (9-5) is also affected by changes in the subcooling of recirculation water and the amount of flow at the lower (variable leg) tap.	
b) At rated conditions with minimum recirculation flow the Wide Range instruments are accurate. As recirculation flow is increased past the lower tap it has a significant velocity head and some friction loss which reduces the pressure on the variable leg to the differential pressure instrument, resulting in an indicated level lower than actual. This could be as much as 10-15 inches error when at rated flow and power.	
c) Due to calibration for rated conditions and no density compensation at cold conditions these instruments read high.	
3) Operational effects of recirculation flow	Obj.ILT-9/ LOR-5
a) Recirculation flow effect on the Wide Range does not affect low pressure ECCS initiation setpoints because the recirc pumps are automatically tripped before their initiation setpoint is reached. The containment spray permissive interlock is not affected by recirculation flow for the same reason.	
b) Scram setpoints are not affected because they are taken from the Narrow Range instruments whose lower taps are above the influence of recirculation flow.	
c) Setpoints for HPCI/RCIC initiation have been established taking recirculation flow effects into consideration.	
4) Mismatch of reactor water level indication	LER 85-006-02, Mismatch of Reactor Water level indicators
a) During the Unit 3 reactor startup on 2/13/85, it was determined that LI-3-53 and 3-206 were reading high and LI-3-60 was reading 17" low.	

OPL171.003 REACTOR VESSEL PROCESS INSTRUMENTATION REV 26

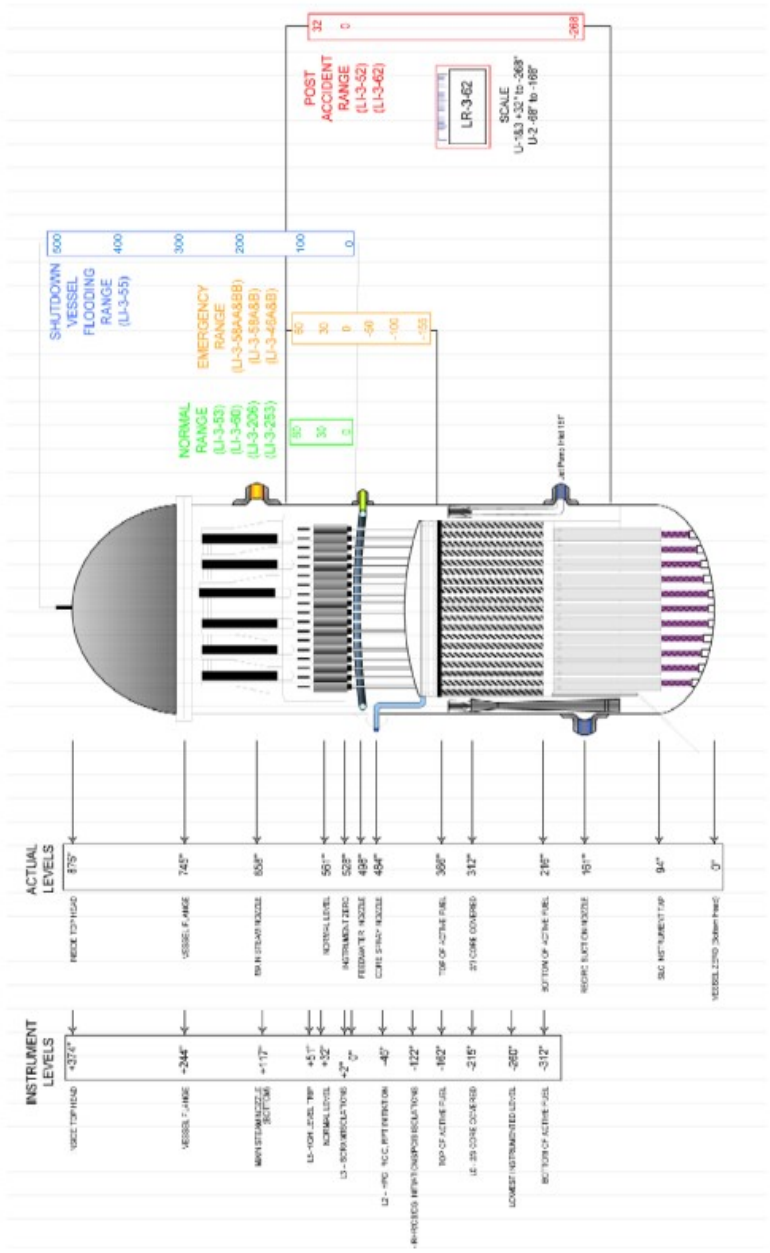


Figure - 3 Vessel Level Instrument Ranges

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295037 (EPE 14) SCRAM Condition Present and Reactor Power above APRM Downscale or Unknown / 1	Tier #	1	------
G2.1.30 (10CFR 55.41.7)	Group #	1	------
Ability to locate and operate components, including local controls.	K/A #	295037G2.1.30	
	Importance Rating	4.4	------

Proposed Question: **# 17**

Unit 2 was operating at 100% RTP when events occurred resulting in the following conditions:

- A fire and heavy smoke forces the crew to abandon the Unit 2 Control Room
- During AOI IMMEDIATE ACTIONS, an electrical ATWS occurs
- Reactor Power is 100%

In accordance with 2-AOI-100-2, Control Room Abandonment, the crew will insert a SCRAM at Panel ___**(1)**___ by ___**(2)**___.

A. (1) 2-25-32
(2) pulling fuses

B. (1) 2-25-32
(2) individually SCRAMMING Control Rods

C. (1) 2-9-15 and 2-9-17
(2) pulling fuses

D. (1) 2-9-15 and 2-9-17
(2) individually SCRAMMING Control Rods

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the last Immediate Action of 2-AOI-100-2 is to proceed to Panel 2-25-32, Backup Control Panel where numerous controls and indications exist, just not Control Rod insertion. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that individually SCRAMMING Control Rods can be performed using 2-AOI-85-4(5)(6), Loss of RPIS (Rod Drift In) (Rod Drift In), however not using 2-AOI-100-2.

C **CORRECT**: (*See attached*) In accordance with 2-AOI-100-2, Control Room Abandonment, if ALL Control Rods were not fully inserted and RPS failed to deenergize, then an Operator is directed to Unit 2's Control Bay Auxiliary Room to perform Attachment 11. Given that an electrical ATWS occurred, the crew is required to perform an alternate method at the backup location outside of the Control Room. The Auxiliary Room's Panel 2-9-15 and 2-9-17 provides that backup location to insert the Control Rods during an electrical ATWS. For second part, at Panel 2-9-15 and 2-9-17 each RPS Bus A(B) will be deenergized to insert the Control Rods by removing/replacing the respective RPS SCRAM solenoid fuses.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability to locate and operate components, including local controls as it relates to ATWS conditions. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution in relation to ATWS conditions during Control Room abandonment.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-AOI-100-2, Rev. 61 (Attach if not previously provided)
2-AOI-85-4, Rev. 21
2-AOI-85-5, Rev. 24
2-AOI-85-6, Rev. 23

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.208 Obj. 6 (As available)

Question Source:

Bank #	
Modified Bank #	
New	**X**

(Note changes or attach parent)

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpts from 2-AOI-100-2:

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0061 Page 8 of 95

4.2 Unit 2 Subsequent Actions

[1] **IF** ALL control rods were **NOT** fully inserted **AND** RPS failed to deenergize, **THEN:** (Otherwise N/A)

DIRECT an operator to Unit 2 Auxiliary Instrument Room to perform Attachment 11.

NOTES

1) The following transfers Reactor Pressure Control to Panel 2-25-32 to allow for pressure control while completing the Panel Checklist.
2) Attachment 9, Alarm Response Procedure Panel 2-25-32, provides for any alarms associated with this instruction.

CAUTION

1) Failure to place control switch in desired position prior to transferring to emergency position may result in inadvertent actuation of the component.
2) [NER/C] Operation from Panel 2-25-32 bypasses logic and interlocks normally associated with the components. [GE SIL 326,

[2] **PLACE** the following MSRV control switches in CLOSE/AUTO at Panel 2-25-32:

Switch No.	Description	
2-HS-1-22C	MAIN STM LINE B RELIEF VALVE	☐
2-HS-1-5C	MAIN STM LINE A RELIEF VALVE	☐
2-HS-1-30C	MAIN STM LINE C RELIEF VALVE	☐
2-HS-1-34C	MAIN STM LINE C RELIEF VALVE	☐

Attachment 11
(Page 1 of 2)

Removal and Replacement of RPS Scram Solenoid Fuses

Date _____

Control Bay - Unit 2 Auxiliary Instrument Room - El 593'

Initials

[1] **OBTAIN** fuse pullers. _____

[2] **LOCATE** Terminal Strip CC inside Panel 2-9-15, Bay 2 (Rear). _____

[3] **REMOVE** the following fuses (located at bottom of terminal strip CC):

RPS BUS A (Panel 2-9-15, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	REMOVED
CC	FOUR (4)	2-FU1-085-0037AA	_____
CC	FIVE (5)	2-FU1-085-0039A/2	_____
CC	SIX (6)	2-FU1-085-0039A/3	_____
CC	SEVEN (7)	2-FU1-085-0039A/4	_____

[4] **LOCATE** Terminal Strip CC inside Panel 2-9-17, Bay 2 (Rear). _____

[5] **REMOVE** the following fuses (located at bottom of terminal strip CC):

RPS BUS B (Panel 2-9-17, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	REMOVED
CC	FOUR (4)	2-FU1-085-0037BA	_____
CC	FIVE (5)	2-FU1-085-0039B/2	_____
CC	SIX (6)	2-FU1-085-0039B/3	_____
CC	SEVEN (7)	2-FU1-085-0039B/4	_____

Attachment 11
(Page 2 of 2)

Removal and Replacement of RPS Scram Solenoid Fuses

Date _____

Control Bay - Unit 2 Auxiliary Instrument Room - El 593'

Initials 1st/CV

[6] **NOTIFY** UO at Panel 2-25-32 upon completion of fuse removal. ____/____

[7] **WHEN** directed by UO at Panel 2-25-32, **THEN**

PROCEED with the remainder of this Attachment. ____/____

[8] **REPLACE** the following fuses (located at bottom of terminal strip CC):

RPS BUS A (panel 2-9-15, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	INSTALLED
CC	FOUR (4)	2-FU1-085-0037AA	____/____
CC	FIVE (5)	2-FU1-085-0039A/2	____/____
CC	SIX (6)	2-FU1-085-0039A/3	____/____
CC	SEVEN (7)	2-FU1-085-0039A/4	____/____

[9] **REPLACE** the following fuses (located at bottom of terminal strip CC):

RPS BUS B (Panel 2-9-17, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	INSTALLED
CC	FOUR (4)	2-FU1-085-0037BA	____/____
CC	FIVE (5)	2-FU1-085-0039B/2	____/____
CC	SIX (6)	2-FU1-085-0039B/3	____/____
CC	SEVEN (7)	2-FU1-085-0039B/4	____/____

[10] **NOTIFY** UO at Panel 2-25-32 upon completion of fuse replacement. ____/____

Supports Distractors A1,B1: (related to Panel 2-25-32)

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0061 Page 7 of 95

4.1 Immediate Action (continued)

[12] **ANNOUNCE** to all plant personnel:

- Units 1/2 Main Control Room is being evacuated.
- All Operations personnel report to your assigned backup control stations.

[13] **OBTAIN** hand-held radios from Unit 2 Control Room.

[14] **PROCEED TO** Backup Control Panel 2-25-32.

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0061 Page 33 of 95

Attachment 2
(Page 1 of 4)

Unit 2 UO Panel Checklist

Date _____

NOTE

Sound-powered phone (Alternate line) is mounted North end of Panel 2-25-32.
Sound-powered phone (Primary line) is mounted South end of Panel 2-25-32.

1.0 PART A

Switch/ Breaker Number	Component Description	Required Position	Initials
	Control Bay - Pnl 2-25-32 - Backup Control Panel - El 621'		
2-HS-55-32B	ANNUNCIATOR POWER DISCONNECT	ON	_____
2-FIC-71-36B	RCIC SYSTEM FLOW/CONTROL	AUTO & SET for 620 gpm	_____
2-XS-71-6	RCIC STM LINE TRAP DRAIN VLV XFR	DISCT	_____
2-XS-71-7	RCIC CNDS PUMP CRW DISCH VLV XFR	DISCT	_____
2-XS-71-5	RCIC STM LINE TRAP BYPASS VLV XFR	DISCT	_____
2-HS-71-9D	RCIC TURB TRIP/THROT VALVE RESET	NORM	_____
2-XS-71-9A	RCIC TURBINE TRIP TRANSFER	EMERG	_____
2-XS-71-9B	RCIC TURB TRIP/THROT VALVE RESET XFR	EMERG	_____
	STOP here until directed to perform Part B.		_____

Attachment 2
(Page 2 of 4)

Unit 2 UO Panel Checklist

2.0 PART B

Switch/ Breaker Number	Component Description	Required Position	Initials
	Control Bay - Pnl 2-25-32 - U-2 Backup Control Panel - El 621'		
2-XS-64-54	SUPPR POOL LVL TRANSFER	EMERG	_____
2-XS-64-51	SUPPR CHMBR PRESS TRANSFER	EMERG	_____
2-XS-64-55B	SUPPR POOL WTR TEMP SEL	1 or 2	_____
2-XS-64-52	DRYWELL AIR TEMP XFR	2	_____
2-XS-256-1	ATU BUS MANUAL TRANSFER	NORM or ALT	_____
2-XS-57-170	UNIT PREFERRED MANUAL TRANSFER	NORM or ALT	_____
2-HS-67-50A	EECW N HDR SPLY TO RBCCW HX	AUTO	_____
2-HS-67-51A	EECW S HDR SPLY TO RBCCW HX	AUTO	_____
2-HS-67-53A	EECW N HDR SPLY TO AIR COMP	AUTO	_____
XS-85-36-B	CRD SCRAM DISCH VOL TEST DR & VT VLV TRANS	EMERGENCY	_____
XS-85-36-A	CRD SCRAM DISCH VOL TEST DR & VT VLV TRANS	EMERGENCY	_____
HS-85-36C	CRD SCRAM DISCH VOL ISOL TEST VALVE	DEPRESSED	_____
2-HS-70-16B	DRYWELL COOLER 2A1 VLV AND DAMPER	OPEN	_____
2-HS-70-18B	DRYWELL COOLER 2A2 VLV AND DAMPER	OPEN	_____
2-HS-70-20B	DRYWELL COOLER 2A3 VLV AND DAMPER	OPEN	_____

Excerpt from 2-AOI-85-4: Supports Distractors B2, D2

4.4 Alternate Methods of Determining Control Rod Position Of A Single Control Rod (continued)

[1.2] **IF** any of the three bullets in 4.4[1] cannot be met, **THEN**

Control Rods may be moved to an Operable Position Indication as a means of position verification (**REFER TO** Tech Spec Bases SR 3.1.3.1).

[2] **IF** the "ONES" Digit for an individual reed switch position indicator is out-of-service on Full Core Display and 4-Rod Display for the Control Rod, **THEN**

Control Rods may be moved to an Operable Position Indication as a means of position verification (**REFER TO** Tech Spec Bases SR 3.1.3.1).

NOTE

The following actions need to be completed within 3 and 4 hours per Tech Spec Action: 3.1.3.C.1 & 2.

[3] **IF** Notching the CRD in as directed by the Reactor Engineer and step 4.4[2] failed to restore the RPIS indication, **THEN**

FULLY INSERT the CRD.

A. Indications of rod insertion include:

- Monitoring the RBM LPRM's for lowering values for the selected CRD.
- CRD drive water flow @ 3-5 gpm which will drop to 0 when motion stops.
- Reactor power reduction as predicted by Reactor Engineer.
- Normal drive time for a CRD from [48] to [00] is ≈ 60 seconds.

B. **WHEN** the above indications show the CRD is fully inserted and RPIS has not restored, Individually SCRAM the CRD @ 1-PNLA-009-0016 (a key is required) by placing the individual Scram Test Switch to the "DOWN" position for 10 seconds, **THEN**

RETURN Test Switch to the normal "UP" position.

Excerpt from 2-AOI-85-5: Supports Distractors B2, D2

BFN Unit 2	Rod Drift In	2-AOI-85-5 Rev. 0024 Page 8 of 11

4.2 Subsequent Actions (continued)

NOTES

1) If either the Scram Inlet or Outlet valve is found to be leaking, the preferred action is to scram the affected rod at Panel 2-9-16, and initiate corrective maintenance if necessary.
2) Individual control rod scram to attempt to reset scram valves may be performed at the discretion of Unit SRO. Reactor Engineer will determine impact on core thermal limits and preconditioning envelope.

[12] **IF** either Scram Inlet or Outlet Valve is found to be leaking, **THEN**

ATTEMPT to reseat the scram valves as follows: (Otherwise N/A)

[12.1] **OBTAIN** Key from Unit 2 Control Room key cabinet for Rod Scram Switches on Panel 9-16.

[12.2] **ESTABLISH** communications between Control Room and the following locations:

- Hydraulic control unit affected, elevation 565, Reactor Building.
- Panel 2-9-16 in Auxiliary Instrument Room, elevation 593, Control Bay.

[12.3] **ENSURE CLOSED** CHARGING WATER SOV, 2-SHV-085-588.

[12.4] **INITIATE** individual rod scram by actuating selected rod scram switch on Panel 2-9-16 to its scram (DOWN) position.

[12.5] **ENSURE** control rod has reached FULL IN on Panel 2-9-5.

[12.6] **RETURN** rod scram switch at Panel 2-9-16, for associated control rod to its normal (UP) position.

Excerpt from 2-AOI-85-6: Supports Distractors B2, D2

4.2 Subsequent Actions (continued)

NOTE

1) A control rod that will not latch could be indicative of either of the following:
   - A failed open withdraw directional control valve (2-FCV-85-40C),
   - Foreign material from the reactor entering and lodging in the CRD collet area wedging the collet open,
   - Collet retainer tube cracking failure.
2) If control rod does not latch at position 00, Step 4.2[8] may be skipped to avoid control rod drift to full out position when removed from service.

[7] **IF** the control rod will not latch into position "00" and continues to demonstrate occurrences of inadvertent withdrawal, **THEN**

PERFORM the following: (Otherwise N/A)

[7.1] **MAINTAIN** an insert signal on the control rod to keep the control rod at position 00 or full in overtravel.

[7.2] **PLACE** the appropriate Rod Scram Switch, on 2-PNLA-009-0016 in the Auxiliary Instrument Room, in the DOWN position.

[7.3] **RELEASE** the control switch placing an insert signal on the control rod.

[7.4] **PLACE** the appropriate Rod Scram Switch, on 2-PNLA-009-0016 in the Auxiliary Instrument Room, in the UP position.

[7.5] **IF** the control rod will not latch into position "00" and continues to demonstrate occurrences of inadvertent withdrawal, **THEN**

PERFORM the following as directed by the Unit SRO: (Otherwise N/A)

[7.5.1] **PLACE** the appropriate Rod Scram Switch, on 2-PNLA-009-0016 in the DOWN position.

[7.5.2] **REDUCE** Core Flow to 50 -60%. (N/A if not desired or less than 50% core flow.)

[7.5.3] **INSERT** a Manual scram and enter 2-AOI-100-1.

[8] **IF** the control rod is latched into position "00", **THEN**

REMOVE associated HCU from service per 2-OI-85.
(N/A if Control will not latch at "00".)

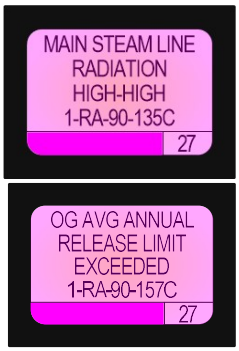
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9	Tier #	1	------
EA1.08 (10CFR 55.41.10)	Group #	1	------
Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RADIOACTIVITY RELEASE RATE:	K/A #	295038EA1.08	
• MSIV leakage control	Importance Rating	3.1	------

Proposed Question: **# 18**

Unit 1 was operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual SCRAM was inserted, **ALL** Control Rods are in
- MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27) is sealed in
- OFFGAS AVERAGE ANNUAL RELEASE LIMIT EXCEEDED (1-9-4C, Window 27) is sealed in



Given the conditions above, Main Steam Isolation Valves (MSIVs) require closure based on actions contained within Alarm Response Procedure (ARP) **(1)**, Window 27.

The MAIN STEAM LINE RADIATION HIGH-HIGH alarms at radiation levels of **(2)** normal full power background.

A. (1) 1-9-3A
(2) 1.5 x

B. (1) 1-9-3A
(2) 3.0 x

C. (1) 1-9-4C
(2) 1.5 x

D. (1) 1-9-4C
(2) 3.0 x

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that 1.5 x normal full power background radiation is the alarm setpoint for MAIN STEAM LINE RADIATION HIGH, (1-9-3A, Window 7) which is not given in this question.

B **CORRECT**: *See attached)* In accordance with the given sealed in (meaning valid) alarm MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27), a core flow runback is to be inserted followed by a manual SCRAM. Additionally, the same ARP specifies that, if SLC injection per RC/Q of EOI-1 is NOT required, then MSIVs must be closed. Since all Control Rods are inserted (as given), this is specifically indicating that the Reactor is shut down under all conditions therefore closing the MSIVs is required. For second part, in accordance with the given MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27), the alarm setpoint is 3.0 x normal full power background radiation levels.

C INCORRECT: First part is incorrect but plausible in that candidates must determine the specifics between MAIN STEAM LINE RADIATION HIGH, (1-9-3A, Window 7) which is not given versus the given MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27). Window 7 alarms at 1.5 x normal full power background and does NOT require MSIV closure versus Window 27 alarming at 3.0 x normal full power background and does require MSIV closure when SLC injection is not required. Additionally, the given OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED, (1-9-4C, Window 27) does NOT require MSIV closure.

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's ability to operate equipment and monitor plant conditions to control releases of offsite radioactivity associated with procedural requirements related to MSIV operation. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-ARP-9-3A, Rev. 55 (Attach if not previously provided)
1-ARP-9-4C, Rev. 31

Proposed references to be provided to applicants during examination: **MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27), OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED (1-9-4C, Window 27)**

Learning Objective: OPL171.033, Obj 3.a (As available)

Form 4.2-1 **Written Examination Question Worksheet**

Question Source:	Bank #	BFN 2204 NRC #34	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2022	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295038 (EPE 15) High Offsite Radioactivity Release Rate / 9
EA1.08 (10CFR 55.41.7)
Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RADIOACTIVITY RELEASE RATE:

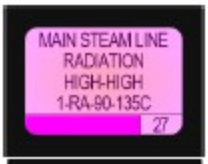
- MSIV leakage control

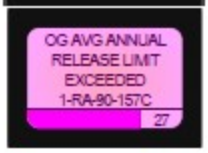
Level	RO	SRO
Tier #	1	-----
Group #	1	-----
K/A #	295038EA1.08	
Importance Rating	3.1	-----

Proposed Question: **# 34**

Unit 1 was operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual SCRAM was inserted, **ALL** Control Rods are in
- MAIN STEAM LINE RADIATION HIGH-HIGH, (1-9-3A, Window 27) is sealed in
- OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED, (1-9-4C, Window 27) is sealed in





Given the conditions above, Main Steam Isolation Valves (MSIVs) require closure based on actions contained within Alarm Response Procedure (ARP) **(1)**, Window 27.

The MAIN STEAM LINE RADIATION HIGH-HIGH alarms at radiation levels of **(2)** normal full power background.

A. (1) 1-9-3A
(2) 1.5 x

B. (1) 1-9-3A
(2) 3.0 x

C. (1) 1-9-4C
(2) 1.5 x

D. (1) 1-9-4C
(2) 3.0 x

Proposed Answer: B

Excerpts from 1-ARP-3A:

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0055 Page 47 of 60

MAIN STEAM LINE
RADIATION
HIGH-HIGH
1-RA-90-135C

27

(Page 1 of 2)

Sensor/Trip Point:

1-RM-90-136 1-RM-90-137	3.0 x normal full power background including N-16 contribution and HWC System injection.

Sensor Location: Radiation monitor drawers are on Panel 1-9-10 in the control room.

Probable Cause:
A. Radiation is three times the normal full power background.
B. Sensor malfunctions.
C. SI (SR) in progress.
D. Additional causes as listed in 1-ARP-9-3A Window 7

Automatic Action:
A. Mechanical vacuum pumps trip.
B. Vacuum pump suction valves 1-FCV-066-0036 and 1-FCV-066-0040 close.

Operator Action:

A. **CHECK** the alarm on 1-RM-90-136 and 1-RM-90-137 on Panel 1-9-10.

B. **CONFIRM** main steam line radiation level on recorder 1-RR-90-135, Panel 1-9-2.

C. **IF** alarm is valid and Reactor Scram has not occurred, **THEN** **PERFORM** the following:
1. **IF** core flow is above 60%, **THEN** **LOWER** core flow to between 50-60%.
2. **MANUALLY SCRAM** the Reactor.
3. **REFER TO** 1-AOI-100-1.

D. **IF** SLC injection per ARC/Q of EOI-1A is NOT required, **THEN**
ENSURE the MSIVs closed.

E. **NOTIFY** RAD PRO.

F. **ENSURE** actions of 1-ARP-9-3A Window 7 have been completed.

G. **IF** Technical Specifications limits are exceeded, **THEN** **REFER TO** EPIP-1.

Continued on Next Page

:

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0055 Page 48 of 60

MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C, Window 27
(Page 2 of 2)

Operator Action: (Continued)

H. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function.
REFER TO NPG-SPP-18.3.5.

References: 1-47E610-90-1 730E915-9, 10 1-45E620-5

Excerpts from 1-ARP-3A, WINDOW 7: Supports Distractors A2, C2

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0055 Page 17 of 60

MAIN STEAM LINE RADIATION HIGH
1-RA-90-135A
7

(Page 1 of 2)

Sensor/Trip Point:

1-RM-90-136 Channel A
1-RM-90-137 Channel C

[NRC/C] Setpoint is 1.5 X normal full power background including N-16 contribution and HWC System injection[nco 940247001]

Sensor Location: Panel 1-9-10

Probable Cause:

A. Raising power following or during reapplication of noble metal to the vessel and its internals using the Noble Metal Chemical Addition (NMCA) process. The new noble metal coating on the fuel from NMCA reapplication causes a temporary increase in the amount of N16 and N13 released from the core, which increases main steam line radiation.
B. SI/SR in progress.
C. Air injection from placing standby cond demin in service.
D. Resin trap failure (RWCU or Cond Demin).
E. Fuel damage.
F. Sensor malfunction.
G. RCIC in service.
H. Placing HWC in service.
I. Recent Noble Metal Chemical Addition
J. Hydrocarbon intrusion (turbine oil, glycol, EHC fluid)
K. Raw water intrusion (condenser tube leak)

Automatic Action: None

Operator Action:

NOTE

For High MSL rad alarms due to recent Noble Metals applications, industry experience indicates that the only corrective action is continued reactor operation. The normal build up of iron oxide crud on the fuel surface and system cleanup by the Reactor Water Cleanup system will reduce the amount of N16 and N13 released from the core, which will decrease main steam line radiation over time.

A. **CHECK** following radiation recorders on Panel 1-9-2:
   1. MAIN STEAM LINE RADIATION monitor, 1-RR-90-135.
   2. OFFGAS PRETREATMENT RADIATION, 1-RR-90-157.
   3. OFFGAS POST-TREATMENT RADIATION, 1-RR-90-265.
   4. STACK GAS/CONT RM RADIATION, 0-RR-90-147.

Continued on Next Page

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0055 Page 18 of 60

MAIN STEAM LINE RADIATION HIGH, 2-XA-55-135A, Window 7
(Page 2 of 2)

Operator Action: (Continued)

B. **NOTIFY** RAD PRO.

C. [NRC/C] **REQUEST** Chemistry to perform radiochemical and ionic analysis of primary coolant. [NCO 940247001]

D. **IF** off-gas PRETREATMENT RADIATION, 1-RR-90-157, has risen significantly (30% above previous hour average), **THEN**
REQUEST Chemistry to perform analysis of pretreatment off-gas.

E. **SHUTDOWN** Hydrogen Water Chemistry. **REFER TO** 1-OI-4.

F. **REFER TO** 0-SI-4.8.B.1.A.1 for ODCM compliance and to determine if power level reduction is required.

G. **IF** Power Ascension is in progress, **THEN**

HALT power ascension until there is a decrease in Steam Line Radiation.

H. [NRC/C] **LOWER** reactor power to maintain off-gas radiation within ODCM limits as directed by Unit SRO. [NCO 940247001]

I. IF ODCM limits are exceeded, **THEN**
REFER TO EPIP-1.

J. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function.
REFER TO NPG-SPP-18.3.5.

References:

1-47E610-90-1	47W600-11	1-45E620-3
1-729E814-1	OE23734	

Excerpts from 1-ARP-4C: Supports Distractors C1, D1

OG AVG ANNUAL RELEASE LIMIT EXCEEDED

1-RA-90-157C

27

(Page 1 of 2)

Sensor/Trip Point:

1-RE-090-0157 1.37E+4 mR/Hr (Alarm from recorder)

Sensor Location: Elevation 565
Turbine Building
Column B-T3
Recorder is on Panel 1-9-2.

Probable Cause:
A. Abnormal flow in the off gas system.
B. Resin trap failure (RWCU or Condensate Demins).
C. Fuel damage.

Automatic Action: None

Operator Action:
A. **DETERMINE** if the Off Gas Annual Release Rate Limit is exceeded, **THEN PERFORM** the following:
1. **CHECK** alarm condition on the following:
   a. OFFGAS RADIATION, 1-RR-90-266 on Panel 1-9-2.
   b. OG PRETREATMENT RADIATION Recorder, 1-RR-90-266, Panel 1-9-2.
   c. OG PRETREATMENT RAD MON RTMR, 1-RM-90-157 on Panel 1-9-10.

B. **NOTIFY** Radiation Protection.

NOTE

High Off-Gas flow can sweep settled particulates into flow stream and cause momentary rise in monitor reading. Low Off-Gas flow can result in improper dilution and cause monitor reading to rise.

C. **ENSURE** Off-Gas flow normal and proper sample flow to the monitor.

Continued on Next Page

OG AVG ANNUAL RELEASE LIMIT EXCEEDED, Window 27
(Page 2 of 2)

Operator Action: (Continued)

NOTE
Load reduction may be required to keep Off-Gas within ODCM limits.

D. **REQUEST** Chemistry perform radiochemical analysis to determine source.
E. **WITH** OPS MGT and Shift Manager's permission, **PLACE** charcoal beds in parallel with another unit. **REFER TO** 1-OI-66.
F. **IF** fuel damage is suspected, **THEN**
REFER TO 1-SR-3.4.6.1 for dose equivalent iodine-131 determination.
G. **REFER TO** 0-SI-4.8.B.1.a.1 and SR-3.4.6.1-a for ODCM compliance and to determine if power level reduction is required.
H. **IF** directed by Shift Manager or Unit SRO, **THEN**
REDUCE reactor power to maintain off-gas radiation within ODCM limits.
I. **REFER TO** EPIP-1.

References: GE 729E814 Series 1-47E610-90-1 ODCM 5.5.1
FSAR Sections 1.6.4.4.6, 7.12.2.2, and 13.6.2
Technical Specifications 4.6.B.6 and 4.8.B.1.a.1
Technical Requirements Manual 3.3.9.1, 3.3.5.1, 3.7.2.1

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
600000 (APE 24) Plant Fire On Site / 8	Tier #	1	------
AK3.04 (10CFR 55.41.10)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE:	K/A #	600000AK3.04	
• Actions contained in the abnormal procedure for plant fire on site	Importance Rating	3.6	------

Proposed Question: **# 19**

A fire has occurred in the Unit 3 Reactor Building.

Which **ONE** of the following completes the statement below?

In accordance with the **NOTES and CAUTIONS** of 0-AOI-26-1, Fire Response, the reason Assistant Unit Operators (AUOs) report to their assigned Control Room is to _____.

Note: Self-Contained Breathing Apparatus (SCBA)
Fire Safe Shutdown (FSS)

A. perform FSS Recovery Actions

B. retrieve the necessary SCBA kits

C. perform Personnel Accountability

D. retrieve the Hard Hat head lamps

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 0-AOI-26-1, Fire Response, NOTES state that the AUOs are assembled in the Control Room to ensure FSS Recovery Actions can be completed with maximum efficiency.

B INCORRECT: Incorrect but plausible in that SCBAs may be required by the Emergency Responders (an AUO position). 0-FSS-001 states that SCBAs are stored in Electrical Board Rooms 1A, 2A, and 3A, for required plant access.

C INCORRECT: Incorrect but plausible in that a plant fire may result in an Emergency Classification and, as a result, possibly the need for Personnel Accountability, however with the given information, no Emergency Declaration has been declared.

D INCORRECT**:** Incorrect but plausible in that the Unit Supervisor may direct the AUOs assembled in the Control Room to obtain (if required for FSS-16-2) hard hat head lamps.

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's knowledge of the actions contained in the Abnormal Operating Instructions concerning the actions to be taken by the AUOs related to a plant fire on site. This question is rated as Memory due to the fact that it requires the strict recall of facts related to plant fire on site.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-FSS-1, Rev. 10 (Attach if not previously provided)
0-AOI-26-1, Rev. 25

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.074 Obj. 2 (As available)

Question Source:

Bank #	BFN NRC 1909 #26	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2019

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Copy of Bank Question:

A fire has occurred in the Unit 3 Reactor Building.

Which **ONE** of the following completes the statement below?

In accordance with the **NOTES and CAUTIONS** of 0-AOI-26-1, Fire Response, the reason Assistant Unit Operators (AUOs) report to their assigned Control Room is to _____.

Note: Self-Contained Breathing Apparatus (SCBA)
Fire Safe Shutdown (FSS)

A. retrieve necessary SCBA kits

B. perform FSS Recovery Actions

C. perform Personnel Accountability

D. retrieve the Hard Hat headlamps

Proposed Answer: **B**

Excerpt from 0-AOI-26-1:

BFN Unit 0	Fire Response	0-AOI-26-1 Rev. 0023 Page 7 of 77

4.2 Subsequent Actions (continued)

NOTES

1) The Shift Manager will remain in communication with the Incident Commander and reference 0-FSS-001 for applicability based on the severity of the fire.
2) Fire Safe Shutdown procedures contains Tables which assist in depicting the credited plant/unit equipment and instrumentation for that specific Fire Area.
3) The AUOs are assembled in the Control Rooms to ensure FSS Recovery Actions can be completed with maximum efficiency (the exception would be FSS-16-2, Main Control Room Abandonment, which requires the ERs to assemble at the backup control panels for their respective Unit).
4) To ensure that in the event of a severe fire, containment pressure is not vented below that which is needed to maintain RHR pump NPSH, maintain 1(2,3)-FIC-84-19 in normal position of Manual and "0" scfm.

[8] **IF** directed by the Unit SRO, **THEN**

PERFORM the following:

[8.1] **NOTIFY** AUOs to report to their assigned Control Room(s), all other AUOs will report to Unit 2 MCR (ERs will report to the Backup Control Room Panels for MCR Abandonment).

[8.2] **REVIEW** applicable FSS for the fire area.

[8.3] **DISTRIBUTE** FSS Attachment/Sections (Recovery Actions for the Fire Area) to assigned Emergency Responders, as directed by the SROs for either the EOIs or the FSSs.

[8.4] **NOTIFY** the AUOs to:

- **OBTAIN** a NFPA-805 Radio from the Main Control Room and, If required for FSS-16-2, Hard Hat head lamps from the REP Locker located on El 3C outside the TSC.

  AND

- **STANDBY** until fire is out OR determination is made to enter applicable FSS.

Excerpt from 0-FSS-1:

BFN	Fire Safe Shutdown	0-FSS-001 Rev. 0008 Page 10 of 135

NOTE

The sub-instructions require precise communication and close coordination of operations department personnel. Portable radios set to channel F2, F4 and "Ops Group" on Harris Radios are common Fire Safe Shutdown channels for communication. Other communications systems (i.e. sound powered phones) can be used, if available.

4.0 PRECAUTIONS AND LIMITATIONS

A. The Shift Manager, or designee, will be responsible for the implementation and control of the FSS and will assign personnel as necessary.

B. The initiation of the RHR system shall be a high priority from the loss of all high pressure make-up to the reactor to minimize the consequences of uncovering the fuel.

C. When monitoring parameters such as reactor water level and reactor pressure, it is preferred to use the credited instrumentation directed by the procedure since other instrumentation may be unreliable due to the fire.

D. Spurious operation of plant equipment may be encountered due to fire induced damage to control circuits and instrumentation. Frequent observation of the control boards by the Operations personnel is required to ensure that spurious equipment operation does not adversely affect nuclear safety.

E. For FSSs, EOIs will be used concurrently if EOI entry conditions have been met. The exception is Fire Area 16 when the Main Control Room is required to be evacuated. There are no EOI procedures at the back up control panel 9-32.

F. Tasks in the fire affected area or adjacent/above the fire affected area may contain smoke/fumes after 60 minutes and the Fire Captain/Incident Commander will be contacted prior to entry to determine if breathing apparatus is still required. SCBA units are stored in Electrical Board Rooms 1A, 2A and 3A. These are stored in the electrical board rooms since Emergency Lights for ingress/egress and equipment operation are not installed in the turbine buildings. Entry to Reactor buildings on each unit is through the Electrical Board Rooms.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6	Tier #	1	------
AK1.03 (10CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Generator Voltage and Electric Grid Disturbances:	K/A #	700000AK1.03	
• Under-excitation	Importance Rating	3.2	------

Proposed Question: **# 20**

Unit 1 is operating at 100% RTP with the following conditions:

- Severe Storms are impacting the electric grid
- The NUSO has entered 0-AOI-57-1E, Grid Instability

Generator **INCOMING** MVARs are limited to **(1)** MVAR to prevent **(2)**.

A. **(1)** 300
(2) slipping a pole

B. **(1)** 300
(2) rotor overheating

C. **(1)** 175
(2) slipping a pole

D. **(1)** 175
(2) rotor overheating

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the previous maximum outgoing (lagging) MVAR limit was 300 MVAR in accordance with the previous revision of 1-OI-47. First part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* First part is incorrect in that the 1-OI-47 Capability Curves are heating limit curves based on hydrogen cooling availability (pressure) for Main Generator cooling medium. One of the curves represents heating limits on the rotor due to high field current.

C **CORRECT:** *(See attached)* In accordance with 1-OI-47, when operated in automatic, the Generator Voltage Regulator has an electronic limit which limits incoming reactive, at full load, to approximately 175 MVAR. When operated in Manual, there is an administrative limit of 175 MVAR. For second part, these Generator Voltage Regulator operation administrative limits are intended to prevent the possibility of the Generator slipping a pole.

D INCORRECT: First part is correct *(See C).* First part is incorrect but plausible *(See B).*

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's knowledge of the operational implications and/or cause and effect relationships of under-excitation as it applies to Generator Voltage and Electric Grid Disturbances. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to generator voltage and electrical grid disturbances.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases document.

Technical Reference(s): 1-OI-47, Rev. 74 (Attach if not previously provided)
1-OI-47, Rev. 52
0-AOI-57-1E, Rev. 13
OPL171.134, Rev. 4

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.134, Obj. 3 (As available)

Question Source: Bank # ILT EXAM BANK #1260 OPL171.036-13 018 (Note changes or attach parent)
Modified Bank #
New
Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1260. OPL171.036-13 018

Given the following conditions:

- Unit 1 is operating at 80% with 100 VARS incoming
- The crew has entered 0-AOI-57-1E, Grid Instability
- System voltage is 508 kV
- System frequency is 60.03 hz

Which ONE of the following completes the statements below?

Under these conditions, the Main Generator Under-excited Reactive Ampere Limit (URAL) circuit is intended to prevent __(1)__.

The operators will coordinate between the units and, as directed by 0-AOI-57-1E, Grid Instability, will __(2)__ power.

A. (1) the generator slipping a pole
(2) lower reactor

B✔ (1) the generator slipping a pole
(2) raise reactive

C. (1) rotor overheating
(2) lower reactor

D. (1) rotor overheating
(2) raise reactive

Excerpt from 1-OI-47:

5.5 Generator Synchronization and Loading (continued)

NOTES

1) Unless otherwise directed, the normal band for Generator MVARS should be 0 + or - 50 MVARs while regulation is in manual.
2) When operated in Automatic, the Generator Voltage Regulator has an electronic limit which limits incoming reactive, at full load, to approximately 175 MVAR. When operated in Manual, there is an administrative limit of 175 MVAR. These limits are intended to prevent the possibility of the Generator slipping a pole.
3) See Section 8.10.1 for operation with the voltage regulator in Manual.

[18] **OPERATE** VOLTAGE REGULATOR LOWER/RAISE ADJUST, 1-HS-57-26 as required to maintain generator parameters.

NOTES

1) Steps 5.5[19] through 5.5[24.2] are performed from Panel 1-9-7 unless indicated otherwise.
2) When the exhaust hood temperature is above 125°F, turbine load should be raised slowly until the temperature falls below 125°F. Plant operating history indicates that maintaining load at about 80 MWe will aid in lowering the Exhaust temperatures due to more steam flow.

[19] **IF** necessary, **THEN**

DEPRESS Load Set RAISE pushbutton, 1-HS-47-75C, or LOWER pushbutton, 1-HS-47-75B, to maintain desired generator output between 60 MWe to 80 MWe. (Otherwise N/A)

[20] **HOLD** generator output between 60 MWe to 80 MWe **UNTIL** turbine first stage and valve chest temperatures stabilize and exhaust hood temperature lowers to less than 125°F, as indicated on 1-XR-47-20.

Excerpt from 0-AOI-57-1E:

BFN Unit 0	Grid Instability	0-AOI-57-1E Rev. 0013 Page 15 of 18

Attachment 1
(Page 2 of 4)

Grid Stability Background and Basis

2.0 STABILITY

A. Grid instability can be caused by a failure in breaker protection schemes that result in loss of transmission lines and an inability of the transmission system to maintain either the required voltage or required system frequency due to current system loading and generating/distributing capacities. During off-normal grid system conditions a reduction in system frequency will occur concurrent with a reduction in system voltage. Conversely a rise in system frequency will result in a coincident rise in grid system voltage.

B. Unit stability is very sensitive to the gross MVAR output level of the generators. The less reactive power they generate the more the stability margin is reduced.

C. Following a system perturbation, if the BFN units are stable, the power swings will be sufficiently damped and the BFN turbine generators return to synchronous speed. If the BFN units are unstable, the power swings are **NOT** sufficiently damped to prevent the uncontrolled acceleration of the BFN turbine generators and their pulling out of step with the system. When this occurs the BFN units will be automatically tripped by protective relaying. During lightly loaded conditions, the stability margins of the turbine generators are reduced because there is less load for damping power swings due to disturbances. Also, the BFN units may be generating at low gross MVAR levels and very likely may be absorbing MVARs in order to control system voltages. This causes the BFN units to be less stable due to low magnetic flux levels in the field windings for damping.

D. To assist in maintaining grid stability, the Load Coordinator may request BFN to raise the incoming reactive power (-MVAR/leading) for each operating unit. Incoming MVARs inherently reduce the individual stability of the turbine generator system whereas outgoing reactive power (+MVAR/lagging) raises it. Normal turbine generator system incoming reactive power is limited to -175 MVARs in automatic voltage regulator operation. Manual voltage regulator operation beyond this limit is allowed only as directed by the Load Coordinator during transmission system disruption contingencies.

Excerpt from 1-OI-47(previous revision): (Supports A1, B1)

8.11 Transferring the Voltage Regulator

NOTES

1) Each BFN Unit has a maximum outgoing (lagging) Mvar limit of 300 Mvar. This limit applies to both 500 kV and 161 kV offsite power source qualifications. If the outgoing Mvar limit is exceeded and is not corrected with 15 minutes, then both offsite power sources are disqualified for the unit that is exceeding the limit.
2) When operated in automatic the Generator Voltage Regulator has an electronic limit which limits incoming (leading) reactive, at full load, to approximately 150 MVAR. When operated in manual, an administrative limit of 150 Incoming MVAR is in effect. This limit is intended to prevent the possibility of the Generator slipping a pole.
3) Unless otherwise directed, the normal band for Generator MVARS should be 0 + or - 50 Mvars while regulation is in manual.
4) The Transmission Operator (TOp) is required to be notified of any status change (Manual/Auto/Loss of Function) of the Voltage Regulator, as soon as practical but within 30 minutes. [NERC VAR-002-2] [SOER 99-1]
5) The Transmission Operator is required to be notified prior to a planned Voltage Regulator transfer from Manual or Automatic. [NERC VAR-002] [SOER 99-1]
6) Operation of the Main Generator without automatic control could impact grid voltage requirements Refer to Sections , 6.1[12], 8.11.1[5]
7) When the Main Generator is online, the Voltage Regulator is required to be operated in Automatic unless coordinated with the Transmission Operator.
8) All position changes (Auto or Manual) of the Voltage Regulator is required to be entered into the Narrative long along with the date, time of position change, reasons, anticipated duration, and notifications made. [NERC VAR-002-3] [SOER 99-1]

CAUTION

Frequent monitoring of GENERATOR MVARS, 1-EI-57-51 will be required while the VOLTAGE REGULATOR MAN/AUTO SEL, 1-HS-57-27, switch is in MAN.

8.11.1 Transfer of the Voltage Regulator to Manual

[1] **IF** Voltage Regulator transfer is of a non-emergency nature, and time permits, **THEN:**

NOTIFY the Transmission Operator of the planned transfer of the Voltage Regulator from Automatic to Manual. [SOER 99-1] ☐

Excerpt from 1-OI-47: (Supports B2, D2)

Attachment 7
(Page 1 of 1)

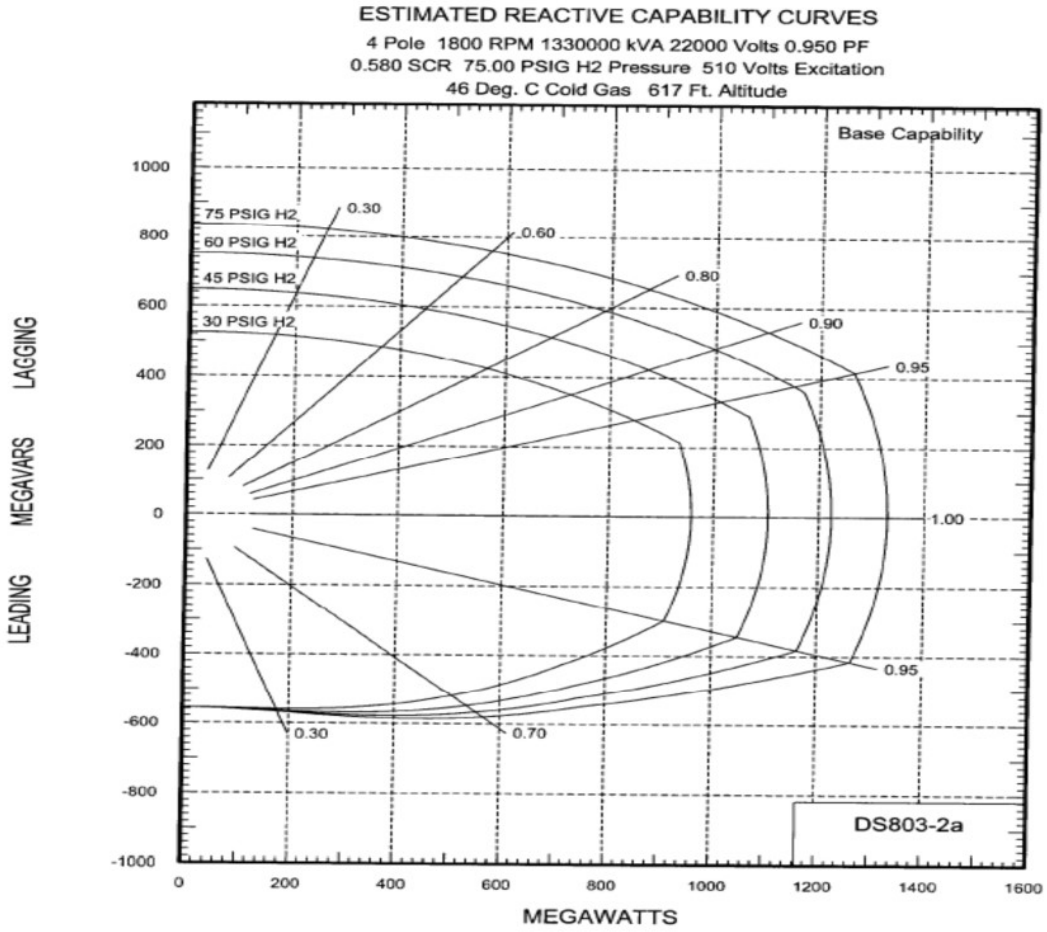
Generator Kilovar Limitations (Capability Curve)

NOTES

1) A 175 MVAR maximum outgoing (lagging) limit applies to all three units for both 500kV and 161kV offsite power source qualification (based on unit MVAR capability limits provided by BFN and used in grid studies) 0-GOI-300-4.

2) The AVR has a dynamic limiter when operating in Automatic. In Manual, the limiter is not applicable. There is a dynamic protection setpoint that will trip the exciter field breaker in Manual or Automatic.

3) Operation with MVARs below 175 MVARs incoming is administratively limited unless calibration or testing is being performed. Under no circumstances should the capability curve be exceeded.



Excerpt from OPL171.134 Lesson Plan: (Also supports B2, D2)

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
Main Generator, as soon as practical but within 30 minutes. 2. The Transmission Operator is required to be notified prior to a planned Voltage Regulator transfer from Manual or automatic. 3. All position changes (Auto or Manual) of the Voltage Regulator are required to be entered into the Narrative Log along with the date, time of position change, reasons, anticipated duration, and notifications made.	(**FUND-5** - Having a solid understanding of plant design, engineering principles, and sciences.)
4. Review OI-47 section 5.5, Generator Synchronization and Loading.	
5. Review OI-47 section 8.11.2 (U-1 & U-2), 8.10.2 (U-3), Transfer of the Voltage Regulator to Automatic	Obj. ILT-9 CR1685782 Give students a set of parameters (Hydrogen pressure, MVARS and Megawatt output) and have them determine effects and corrective action, if necessary
6. Review OI-47 Illustration 7. It specifies separate limits for turbine and generator load limitations, They are: a. A 175 MVAR maximum outgoing (lagging) limit applies to all three units for both 500kV and 161kV offsite power source qualification (based on unit MVAR capability limits provided by BFN and used in grid studies).. b. When operated in Automatic, the Generator Voltage Regulator has an electronic limit which limits incoming reactive, at full load, to approximately 175 MVAR. c. When operated in Manual, there is an administrative limit of 175 MVAR. These limits are intended to prevent the possibility of the Generator slipping a pole. d. Under no circumstances should the capability curve be exceeded.	If the outgoing MVAR limit is exceeded and is not corrected with 15 minutes, then both offsite power sources are disqualified for the unit that is exceeding the limit.
7. The more limiting of these 0-GOI-300-4 limits apply: a. Capability curve is family of heating limit curves based on hydrogen cooling availability (pressure). b. Curve AB represents heating limits on the rotor due to high field current. c. Curve BC represents heating limit on the stator due to armature current d. Curve CD represents the heating limit on the armature (stator) core end due to magnetic field distortion e. Operation is within heating limits when the generator operating point is inside the hydrogen pressure curve.	Unless otherwise directed, the normal band for Generator MVARS should be 0 + or - 50 MVARs while regulation is in manual. .

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295002 (APE 2) Loss of Main Condenser Vacuum / 3

AK3.11 (10CFR 55.41.5)

Knowledge of the reasons for the following responses or actions as they apply to Loss of Main Condenser Vacuum:

- Bypass augmented offgas system

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295002AK3.11	
Importance Rating	3.1	------

Proposed Question: **# 21**

Which **ONE** of the following completes the statements below regarding the Off-Gas System mechanical vacuum pumps?

The mechanical vacuum pumps' discharge flow path is routed through the **(1)**.

If Main Condenser vacuum began degrading at 20% RTP, a mechanical vacuum pump **(2)** to be placed in service.

A. (1) 1.75 minute holdup volume then to the main stack
(2) is required

B. (1) 1.75 minute holdup volume then to the main stack
(2) is NOT allowed

C. (1) six hour holdup volume then to the charcoal absorbers
(2) is required

D. (1) six hour holdup volume then to the charcoal absorbers
(2) is NOT allowed

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct (*See B*). The second part is incorrect but plausible in that in accordance with 1-OI-66, Off-Gas System, the mechanical vacuum pumps are used to purge the Main Condenser. Additionally, during shutdown and/or low Power operations where Main Condenser vacuum is maintained, non-condensable gases are removed using the mechanical vacuum pumps.

B **CORRECT:** *(See attached)* The mechanical vacuum pumps are designed to discharge through the 1.75 minute holdup volume to the stack of the Off-Gas System to remove air and non-condensable gases from the Main Condenser during startup and shutdown activities. Conversely, the 6 hour holdup volume is the normal at power Off-Gas System flow path to delay travel of non-condensable gases, primarily Krypton, Xenon and air.
For second part, in accordance with 1-OI-66, Off-Gas System, the mechanical vacuum pumps can NOT be used when Reactor Power is greater than 5% unless being electrically rotated for preventative maintenance. This is due to possible hydrogen explosive concentrations. Mechanical vacuum pump operation is not even allowed to purge the Main Condensers when hydrogen concentration is suspected. Given that the question asks at 20% RTP, the mechanical vacuum pumps have an automatic trip signal and the associated suction valves have an automatic closure signal. These automatic signals occur when Hotwell Pressure is ≤ 26 inches Mercury or when ≤ 22 inches Mercury with Reactor Pressure ≥ 600 psig.

C INCORRECT: The first part is incorrect but plausible in that the Off-Gas System normal at Power flow path is through the steam jet air ejector discharge path to the 6 hour holdup volume then the charcoal absorbers. This is designed to delay travel of non-condensable gases, primarily Krypton, Xenon and air to provide decay time prior to entering the charcoal adsorbers. The second part is incorrect but plausible (*See A*).

D INCORRECT: The first part is incorrect but plausible (*See C*). The second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the reasons for utilizing a bypass augmented system associated with the Off-Gas System as it relates to a Loss of Main Condenser Vacuum condition. This question is rated as Memory due to the requirement to strictly recall facts related to operation of the Off-Gas System.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-OI-66, Rev. 53 (Attach if not previously provided)
OPL171.030, Rev. 21

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.030, Obj. 9 (As available)

Question Source:

Bank #	
Modified Bank #	(Note changes or attach parent)
New	**X**

Question History:

Last NRC Exam	

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

Form 4.2-1	Written Examination Question Worksheet

10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

Excerpts from 1-OI-66:

3.0 PRECAUTIONS AND LIMITATIONS (continued)

G. Following startup, while still at low power, recombiner performance and hydrogen concentration should be closely monitored.

H. Off-Gas System valves are potentially spark-producing when operated; therefore, when hydrogen concentration is suspected of being greater than 4%, NO action is allowed to be taken that will change off-gas valve positions until after the unit is shut down, except for the SJAEs which may be started following an isolation, and alternated if required with greater than 4% hydrogen. SJAEs have non-sparking valve seats, and hydrogen flammability lower limit is **NOT** a concern in a saturated steam environment.

I. The Mechanical Vacuum Pumps are not to be used to purge the main condenser if hydrogen concentration is suspected of being present.

J. The Mechanical Vacuum Pumps are not to be used when reactor power is greater than 5% unless being electrically rotated for Preventive Maintenance.

The Mechanical Vacuum Pump(s) may be electrically rotated for Preventive Maintenance if the suction valve(s) is closed and seal water in-service to prevent seizing. This requires the automatic trip to be defeated by a step text Work Order. [BFPER 00-003819-000] [BFPER 02-014849-000]

K. Charcoal bed re-alignment during power operation requires Shift Manager approval. Any major change in off-gas flow will disturb bed equilibrium and result in a temporary (8 to 12 day) rise in stack discharge activity. Offsite release rates should be reviewed to ensure compliance with ODCM limits.

L. Charcoal bed prefilter and afterfilter differential pressure is not to exceed 10" H_2O. Switching to standby filters is recommended when filter differential pressure reaches 8" H_2O.

M. A Mechanical Vacuum Pump will auto trip under any of the following conditions:

1. Hotwell pressure is equal to or below -26" HG, or
2. Hotwell pressure is equal to or below -22" HG, with reactor pressure greater than or equal to 600 psig (vacuum pumps suction valves also auto close), or
3. Main Steam Line radiation is greater than or equal to 3 times normal background at full load (vacuum pumps suction valves also close), or
4. Seal water pump trips, or
5. Undervoltage

3.0 PRECAUTIONS AND LIMITATIONS (continued)

G. Following startup, while still at low power, recombiner performance and hydrogen concentration should be closely monitored.

H. Off-Gas System valves are potentially spark-producing when operated; therefore, when hydrogen concentration is suspected of being greater than 4%, NO action is allowed to be taken that will change off-gas valve positions until after the unit is shut down, except for the SJAEs which may be started following an isolation, and alternated if required with greater than 4% hydrogen. SJAEs have non-sparking valve seats, and hydrogen flammability lower limit is **NOT** a concern in a saturated steam environment.

I. The Mechanical Vacuum Pumps are not to be used to purge the main condenser if hydrogen concentration is suspected of being present.

J. The Mechanical Vacuum Pumps are not to be used when reactor power is greater than 5% unless being electrically rotated for Preventive Maintenance.

The Mechanical Vacuum Pump(s) may be electrically rotated for Preventive Maintenance if the suction valve(s) is closed and seal water in-service to prevent seizing. This requires the automatic trip to be defeated by a step text Work Order. [BFPER 00-003819-000] [BFPER 02-014849-000]

K. Charcoal bed re-alignment during power operation requires Shift Manager approval. Any major change in off-gas flow will disturb bed equilibrium and result in a temporary (8 to 12 day) rise in stack discharge activity. Offsite release rates should be reviewed to ensure compliance with ODCM limits.

L. Charcoal bed prefilter and afterfilter differential pressure is not to exceed 10" H_2O. Switching to standby filters is recommended when filter differential pressure reaches 8" H_2O.

M. A Mechanical Vacuum Pump will auto trip under any of the following conditions:

1. Hotwell pressure is equal to or below -26" HG, or
2. Hotwell pressure is equal to or below -22" HG, with reactor pressure greater than or equal to 600 psig (vacuum pumps suction valves also auto close), or
3. Main Steam Line radiation is greater than or equal to 3 times normal background at full load (vacuum pumps suction valves also close), or
4. Seal water pump trips, or
5. Undervoltage

Excerpts from OPL171.030:

OPL171.030, Off-Gas, Rev.21

19. FCV-66-28 Offgas Isolation Valve

a) Located on the ground elevation of the stack. One for each unit.

b) FCV-66-28 is a air operated valve with a handwheel that can be used to hold the valve open on a loss of air. The power supply is auctioneered from 120V Plant Preferred and 120V Unit Non-Preferred to the solenoid that operates the valve.

FCV-66-28 will fail closed on a loss of air, a diaphragm rupture, OR on a loss of power.

c) FCV-66-28 will auto close if OG Post-treatment Rad Monitor detects HI-HI-HI radiation, downscale, or INOP in both channels (any combination) simultaneously after a 5 second time delay.

MCR handswitch on panel 9-53.

If the MCR handswitch is left in the open position after the valve closes, once the condition clears, the valve will reopen.

Reference 45E614-2 for RM logic to the 66-28.

20. Dilution Fans and Chamber

a) Each unit has two 100% capacity Dilution Fans. These are located on the second level of the Stack (elevation 580) and discharge into the common dilution chamber located above. Fresh air is mixed with the remaining offgas mixture for dilution. Each fan has a filter on the intake side to limit particulates in the stack discharge. Dilution fans also provide adequate flow for the Wide Range Gaseous Effluent Radiation Monitor (WRGERMS) of 7000 scfm and provide the final motive force for exit from the stack.

b) Fans can be controlled locally or from panel 9-8 in the MCR. The lights above the handswitch in the MCR are for outlet damper position and not the fan motor.

Normally one dilution fan is in service at a time on each unit. A selector switch between the two handswitches is placed in the position of the standby fan. This enables the logic of the standby fan to "see" low flow from the fan that is in service and auto start if a low flow condition is sensed from the in-service fan..

An alarm for low flow is in the MCR. Both fans must indicate low flow for this alarm to come in.

c) "A" Fans are normally powered from 480V Diesel Aux Board A and "B" fans are normally powered from 480V Diesel Aux Board B for all three units. Backup power supply is from Control Bay Vent Board A for all fans.

Rated flow is 3000 SCFM at 15" Hg and 1850 SCFM at 7" Hg.

21. Mechanical Vacuum Pumps

a) Located on elevation 604 above heater alley in TB. These pumps are powered from the respective 480V Unit Board.

b) Removes air and non-condensables from the condenser and establishes sufficient vacuum for condenser operation during startup and shutdown activities. Discharge is via the 1.75 minute hold-up volume to the stack.

Per OI-66 Mechanical Vacuum pumps are not to be used >5% reactor power.

ILT Objective 9

LOR Objective 7

OPL171.030, Off-Gas, Rev.21

c) Seal water is required for the pump to operate. The seal water pump gets a start signal from the Mechanical Vacuum pump handswitch in the MCR. There are also local individual controls for each of the seal water pumps and Mechanical Vacuum pumps.

ILT NLOR Objective 3

NLO Objective 8

d) Start Interlocks:

(1) Seal water pump running

(2) MVP suction pressure is < 26" Hg vacuum.

Reference 45E777-2 for start and trip interlocks of the MVPs.

MVP suction pressure and condenser pressure are sensed from different pressure switches.

e) MVP will Auto Trip on any of the following signals:

(1) MVP suction pressure is >26" Hg vacuum

(2) Main Steam Line Radiation is greater than 3 x NFLB (16A-K45 relay)

(3) Seal water pump trips

(4) Condenser vacuum is >22" Hg AND reactor pressure is > 600 psig.

(5) Undervoltage on the Bus (27X relay)

The signal for 3xNFLB comes from PCIS.

Reference 45E777-11 for MVP suction valve closure interlocks.

f) MVP suction valve will close on either of the following signals:

(1) Main Steam line radiation is >3 x NFLB (16A-K46 relay)

(2) Condenser vacuum >22" Hg and reactor pressure is >600 psig.

22. Steam Packing Exhauster

a) Located on elevation 586 TB.

b) Condenses Main turbine and Reactor Feedpump turbine seal steam and routes any remaining non-condensables to the stack through 1.75 minute hold-up volume.

c) In service during power operations.

Interlocks and operational information for SPE have been moved to OPL171.027 Steam Seals.

Attachment 4.

Stack Damper arrangement

23. Stack Dampers

a) There are 18 redundant dampers located in 9 flowpaths to the stack chamber. These are mechanical backdraft dampers (check valves) designed with a weighted counter-balance which requires air flow to hold them open.

b) This damper arrangement provides for each units independent auto closure of dampers due to backdraft or no-flow conditions. These dampers also alleviate the concern that during operation of SBGT, the BFN stack could allow release of radioactive effluents at ground level via reverse flow through common discharge paths.

NLO Objective 9

LOR Objective 5

ILT Objective 7

the radiolytic decomposition of reactor water. Shielding is provided as necessary for piping and equipment.

4. During normal power operations, non-condensable gases from the condenser are routed through the Steam Jet Air Ejector flow path and non-condensable gases from the Steam Packing Exhauster are routed through the 1.75 minute holdup volume.
5. During shutdown and/or low power operations where condenser vacuum is maintained, non-condensable gases from the condenser are routed through the 1.75 minute holdup volume by the Mechanical Vacuum Pumps.

B. Components and Descriptions

Reference Attachment 1 or 1/2/3-47E809-2

ILT Objective 1 and 2

NLOR Objective 1 and 2

NLO Objective 2 and 3

OF-5 Understanding Plant Design

1. Main Condenser Vacuum Breakers (PCV-66-1A and PCV-66-1B)
   a) 14-inch throttle valves which tap into the A and C condensers. They are used to break condenser vacuum.
   b) Seal water is applied to each valve to prevent air leakage from the main condenser.
   c) Both valves are controlled by one switch in the Main Control Room on panel 9-8 or locally by individual controls.
   d) There is one indication in the control room for both valves. The indication is fed from parallel limit switches (one from each valve).

2. Offgas Inlet Isolation Valves (FCV-66-2, 3, 4, 5, 6, 7)
   a) Motor-operated butterfly valves that also have seal water applied to prevent air inleakage.
   b) Located near the condenser in the low pressure feedwater heater area.
   c) Valves can be controlled from the Main Control Room on panel 9-8.

ILT Objective 3

LOR Objective 1

3. Offgas Precooler
   a) Located on TB elevation 604.
   b) Raw Cooling water is the cooling medium through the tubes.
   c) Cools the non-condensable gases and routes any condensate to the main condenser through a loop seal.
   d) FCV-66-10 is provided to bypass the precooler. Controls are in the Main Control Room on panel 9-8.

4. Steam Jet Air Ejectors

OPL171.030, Off-Gas, Rev.21

Gas Select Switch for the respective sample train is in the NITROGEN position.

Setpoints and possible actions for alarms associated with this RM will be discussed in the Instrumentation and Alarms Section.

11. Offgas Pretreatment Radiation Monitor
    a) RM-90-157 and 160 take suction prior to the 6 hour hold-up volume and discharges back to the suction of the SJAEs through the Offgas Analyzer return line.
    b) There are no automatic trips or isolations associated with this alarm.

12. 6-hour hold-up volume
    a) A 54 inch diameter underground pipe located between the turbine building and offgas building.
    b) The purpose of the 6-hour hold-up volume is to delay travel of the remaining non-condensables (primarily Krypton, Xenon and air) to provide decay time prior to entering the charcoal adsorbers.
    c) The holdup volume slopes toward the middle for drainage of any condensate that may collect. This drainage is routed to the offgas sump in radwaste. FSV-66-23 is normally open with the line having a loop seal to prevent escape of offgas noncondensables to radwaste. FSV-66-23 will close on a loss of power and high pressure in the offgas system as sensed at the entrance to the 6-hour hold-up volume.

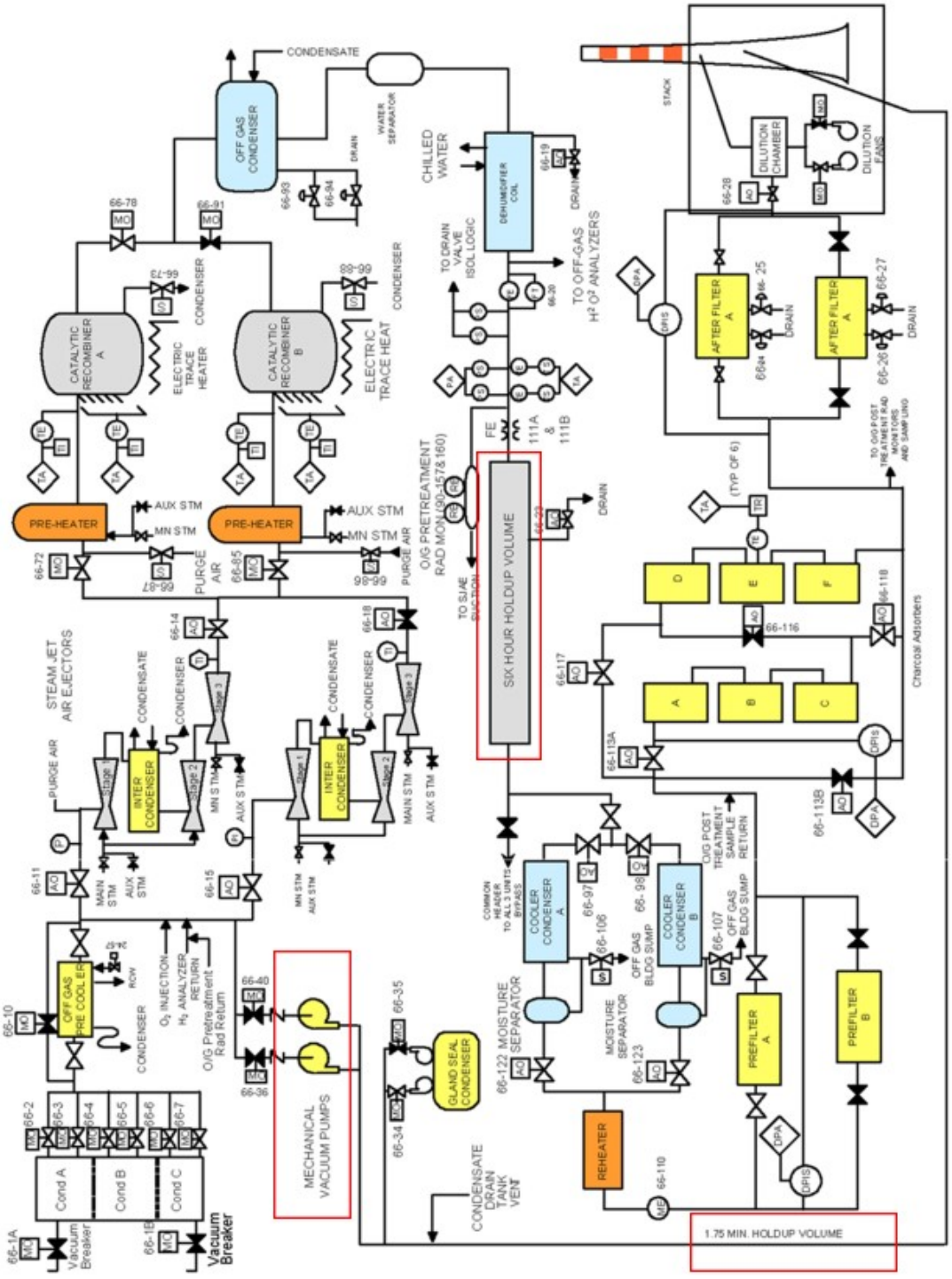
13. Cooler Condensers
    a) Two located in the Offgas Treatment Building. One is normally in service with one in standby. These can be operated in parallel.

480V Offgas MCC supplies power to this equipment.

    b) U-tube type heat exchanger. Offgas mixture passes over the tubes which contain ethylene glycol.

       The Glycol storage and equipment in located in the lower elevation of the Offgas Treatment Building. A Conventional Freon refrigeration unit cools the glycol. The glycol system consists of a storage tank with agitator, two pumps and refrigeration units.
    c) Further cools the offgas mixture exiting the 6-hour hold-up volume to 45 ^{0}F. This helps to remove moisture before entering the charcoal beds.
    d) A moisture separator is located at the effluent of each of the cooler condensers. This retains any moisture carryover and drains any condensate through the FSV-66-106 or 107 drain

OPL171.030, Off-Gas, Rev.21



Attachment 1

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295007 (APE 7) High Reactor Pressure / 3 **AA2.02** (10CFR 55.41.10)	Tier #	1	------
Ability to determine and/or interpret the following as they apply to High Reactor Pressure: • Reactor Power	Group #	2	------
	K/A #	295007AA2.02	
	Importance Rating	4.5	------

Proposed Question: **# 22**

Unit 3 is operating at 80% RTP when the Main Turbine tripped resulting in the following conditions:

- ATWS actions have been completed
- Reactor Power is 9%
- Reactor Pressure is 1030 psig and rising

Given the conditions above, in accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, a **MINIMUM** of _____ opened Main Turbine Bypass Valves would be required to accurately control Reactor Pressure in the normally prescribed band.

A. 2

B. 3

C. 6

D. 9

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that in accordance with BFN-ODM-4.20, one Main Steam Relief Valve (MSRV) is equivalent to approximately 5 – 7% Reactor Power near rated Reactor Pressure.

B **CORRECT:** *(See attached)* In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, one Main Turbine Bypass (MTBPV) Valve fully open is equivalent to about 2 – 3.5% Reactor Power near rated Reactor Pressure. Given that Reactor Power is 9%, a minimum of 3 opened MTBPVs would be required to accurately control Reactor Pressure in the normally prescribed band of 800 – 1000 psig.

C INCORRECT: Incorrect but plausible in that in accordance with 0-TI-394, Technical Support Guidelines for Use with EOIs and SAMGs, at 1000 psig, the MSRVs are equivalent to approximately 6.2 – 7.1% Reactor Power.

D INCORRECT: Incorrect but plausible in that in accordance with 0-TI-394, Technical Support Guidelines for Use with EOIs and SAMGs, the table of approximate ATWS Power levels and Injection Steaming Rates to Maintain Level list various Power levels. The candidate could misinterpret these parameters related to Reactor Power.

RO Level Justification: Tests the candidate's ability to determine and/or interpret Reactor Power as it relates to High Reactor Pressure. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): BFN-ODM-4.20, Rev. 9 (Attach if not previously provided)
0-TI-394, Rev. 14

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.276 Obj. 2 (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK OPL171.009-16 003 #531 (Note changes or attach parent)
New	

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

531. OPL171.009-16 003

Unit 3 has experienced a Main Turbine Trip from 80% power, but RPS did not deenergize and ARI did not work. Reactor water level has been lowered to control power. Present conditions are as follows:

Reactor power	25%
Reactor level	-80"
Mode switch in	Shutdown
Turbine bypass valves open	8
Reactor pressure	962 psig
Torus temperature	102 deg F
Drywell pressure	1.05 psig
SLC	NOT Injected

If EHC problems were encountered and Main Turbine Bypass Valves close, which **ONE** of the following indicates approximately how many MSRVs would be required to control pressure when the Unit Supervisor directed pressure control using the MSRVs?

A. 2 to 3

B. 4 to 5

C. 6 to 7

D. 7 to 8

Excerpt from BFN-ODM-4.20:

4.2.3 Crew Communications During Transients

Communications among crew members that are not briefings or updates are expected to follow the guidance of OPDP-1 for 3-way communications. Other discussions are permissible without utilizing 3-way communications, the NUSO and STA collaborating over a technical specification determination, for example. All crew members bear responsibility for the formality of communications and are expected to offer coaching and correction when needed.

4.2.4 Plant Announcements

The plant announcing system may be used by operators without specific direction from the NUSO. Plant announcements provide a means of informing station personnel of pertinent information in much the same manner that an update aligns the control room operating crew. Plant announcements should be made when significant changes to area radiological or temperature conditions, like the receipt of a valid area radiation monitor or area temperature alarm, or a reactor scram. Operators should consider that plant ambient noise levels may preclude some personnel from understanding an announcement when the announcement is spoken too rapidly.

4.2.5 Miscellaneous - Communication and Expectations

When providing directions, crew members and particularly the NUSO must avoid using polite or soft phrases such as "*please,*" "*listen up for a brief,*" "*I need you to,*" or "*keep an eye on.*" Communications during plant transients must be focused, precise, and succinct.

Abnormal Annunciator Response - During Off Normal conditions, i.e. AOI or EOI entry or as directed by the NUSO, the standards for annunciator response as described in OPDP-1 are still in effect. Critical alarms are identified and communicated to the NUSO. Alarms that are not critical in nature should be silenced and addressed as time permits so as not to interfere with abnormal or emergency procedure performance. Normal annunciator response should resume at the direction of the SM/NUSO after the abnormal or emergency procedures are exited.

4.2.6 Thumb Rules

Within 3 minutes following a Shutdown, Decay heat removal should be within the capacity of the RCIC System.

One Turbine BPV fully open is equivalent to about 2-3.5% Reactor Power near rated pressure.

One MSRV is equivalent to approximately 5-7% Reactor Power near rated pressure.

1% Reactor Power is equivalent to ~240 gpm injection.

Sustained injection flow rates of >1 Mlbm/hr of cold injection will result in a swell of approximately 2.5 times the observed reactor level rise due to the injection. (e.g. if HPCI is used at full flow to raised level from -20" to 0", level will swell to approx. 50"). This does not take into account inventory losses due to decay heat.

Excerpt from 0-TI-394:

6.1.4 Reactor May Not Remain Shutdown Under All Conditions Without Boron; RPV Water Level CAN Be Determined (Ref. EOI-1A) (continued)

[2] Estimate reactor power using

A. Attachment 1 Control Parameter Assessment section 2.3, Reactor Power, and

B. determine if power is below 5% (SPDS screen OVIEW has APRM power) using

- Attachment 2 Plant Status Assessment section 2.10,Reactor Will Remain Shutdown Under All Conditions without Boron Injection, and
- Section 2.16, Reactor Power above Downscale.

C. APRM power is found on SPDS screens “OVIEW” and “NHB”. SRV positions are found on “STSRV” and turbine bypass valve positions are found on “HPTURB”. Injection rates are found on “NSYS”.

D. At ~1000 psig the SRVs are equivalent to ~6.2 to 7.1% power. The TBVs are equivalent to 2.25 to 3.5%. The first value is when injection is from the warmer suppression pool; the second value is for injection from the CST.

E. Table of ATWS Power Level and Injection Steaming Rates to Maintain Level- Assumes suction of 100°F and pressure of 1135 psig

Power level	Volumetric Flow Rate	Mass Flow Rate
%	gpm	lbm/hr
1.0	240	119,700
2.5	600 (RCIC)	299,300
4.1	1,000	498,800
9.1	2,200	1,100,000
12.4	3,000	1,496,400
16.5	4,000	1,995,200
20.6	5,000 (HPCI)	2,494,000
21.9	5,300 (HPCI)	2,644,000
23.2	5,600	2,793,200
24.4	5,920 (HPCI and RCIC)	2,952,800

Examination Outline Cross-reference:

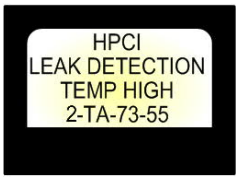
295032 (EPE 9) High Secondary Containment Area Temperature / 5
G2.4.50 (10CFR 55.41.10)
Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295032G2.4.50	
Importance Rating	4.2	------

Proposed Question: **# 24**

Unit 2 was operating at 100% RTP when an event occurred resulting in the following condition:

- HPCI LEAK DETECTION TEMP HIGH, (2-9-3F, Window 10) alarms
- HPCI AREA TEMPERATURE is 150 °F



Given the conditions above, HPCI AUTO-ISOL LOGIC A and HPCI AUTO-ISOL LOGIC B (see below), **(1)** illuminated and 2-EOI-3, Secondary Containment Control, **(2)** required to be entered.

Note: Provided as reference only, answer (1) based on given conditions above.



A. **(1)** are
(2) is

B. **(1)** are
(2) is **NOT**

C. **(1)** are **NOT**
(2) is

D. **(1)** are **NOT**
(2) is **NOT**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the automatic trip and isolation signals of HPCI and RCIC are easily and often confused. Second part is correct (*See C).*

B INCORRECT: First part is correct (*See A)*. Second part is incorrect but plausible in that

C **CORRECT**: (*See attached)* In accordance with 2-OI-73, HPCI, the System will automatically isolate when steamline space temperature reaches 165 °F Torus area or 185 °F HPCI Pump Room. With the given HPCI Area Temperature at 150 °F, the HPCI AUTO-ISOL LOGIC A(B) lights are not illuminated. For second part, in accordance with the Alarm Response Procedure for the given HPCI LEAK DETECTION TEMP HIGH, 2-9-3F, Window 10, if high temperature is confirmed, then enter 2-EOI-3, Secondary Containment Control. This is based on meeting the entry condition of 'Any Secondary Containment Area Temperature above Max Normal value of Table SC-1'.

D INCORRECT: First part is correct (*See)*. Second part is incorrect but plausible (*See B).*

RO Level Justification: Tests the candidate's ability to verify system alarm setpoints and operate controls identified in the alarm response procedure as it relates to High Secondary Containment Area Temperature. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-EOI-3, Rev. 19
2-ARP-9-3F, Rev. 42
2-OI-73, Rev. 103
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: **HPCI LEAK DETECTION TEMP HIGH, (2-9-3F, Window 10) HPCI AUTO-ISOL LOGIC A/B light indications on Panel 2-9-3**

Learning Objective: OPL171.042 Obj. 3b (As available)

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Question Source:

Bank #		
Modified Bank #		(Note changes or attach parent)
New	X	

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

Excerpt from 2-ARP-9-3F:

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0042 Page 14 of 41

HPCI LEAK DETECTION TEMP HIGH

2-TA-73-55

10

(Page 1 of 1)

Sensor/Trip Point:

2-TR-69-29	
TE-73-55A (point 16)	150°F
TE-73-55B (point 17)	150°F
TE-73-55C (point 18)	150°F
TE-73-55D (point 19)	150°F

Sensor Location: Panel 2-9-22, Main Control Rm El 617'.

Probable Cause:
A. HPCI steam line leak.
B. Sensor malfunction.

Automatic Action: Continued rise will cause the following valves to isolate (at Steamline Space Temperature of 165°F Torus Area or 185°F HPCI Pump Room):
- HPCI STEAMLINE INBD ISOL VALVE, 2-FCV-73-2
- HPCI STEAM LINE OUTBD ISOL VALVE, 2-FCV-73-3
- HPCI STEAMLINE WARM-UP VALVE, 2-FCV-73-81.
- HPCI SUPP POOL INBD SUCT VLV, 2-FCV-73-26.
- HPCI SUPP POOL OUTBD SUCT VLV, 2-FCV-73-27.
- HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30
- HPCI TURBINE STOP VALVE, 2-FCV-73-18

Operator Action:
A. **CHECK** HPCI temperature elements on LEAK DETECTION SYSTEM TEMPERATURE recorder, 2-TR-69-29 on Panel 2-9-22.
B. **IF** high temperature is confirmed, **THEN ENTER** 2-EOI-3 Flowchart.
C. **CHECK** the following on Panel 2-9-11 and **NOTIFY** RADCON if rising radiation levels are observed:
   1. HPCI ROOM El 519 RX BLDG radiation indicator, 2-RI-90-24A.
   2. RHR EAST ROOM El 519 RX BLDG radiation indicator, 2-RI-90-28A.

D. **DISPATCH** personnel to investigate for leaks consistent with ALARA considerations in HPCI Turbine Area (El 519) and HPCI Steam Supply Area (El 550).
E. **EVALUATE** equipment associated with this alarm that is out of service to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References: 2-47E610-73-1 2-47E610-69-1 920D351
2-45E620-1

Excerpt from 2-EOI-3:

Any Secondary Cntmt area temp above Max Normal value of Table SC-1

NOTE

(3) Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Cntmt (emergency depressurization will reduce discharge)

Table SC-1
Secondary Cntmt Area Temp

Area	Panel 9-3 Alarm Window (unless noted)	Panel 9-21 Temp Element (unless noted)	Max Normal Value °F	Max Safe Value °F	Potential Isolation Sources
RHR sys I pumps	XA-55-3E-4	74-95A	Alarmed	150	FCV-74-47, 48
RHR sys II pumps	XA-55-3E-4	74-95B	Alarmed	210	FCV-74-47, 48
HPCI room	XA-55-3F-10	73-55A	Alarmed	270	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	XA-55-3D-10	71-41A	Alarmed	190	FCV-71-2, 3, 39
Top of torus	XA-55-3D-10 XA-55-3F-10 XA-55-3E-4	71-41B, C, D 73-55B, C, D 74-95G, H	Alarmed Alarmed Alarmed	200 240 240	FCV-71-2, 3 FCV-73-2, 3, 81 FCV-74-47, 48
Steam tunnel (RB)	XA-55-3D-24	1-60A (Panel 9-3)	Alarmed	315	MSIVs FCV-71-2, 3, FCV-69-1, 2, 12
DW access	XA-55-3E-4	74-95E	Alarmed	170	FCV-74-47, 48
RB el 565 W (RWCU pipe trench)	XA-55-5B-32 (Panel 9-5) XA-55-5B-33 (Panel 9-5)	69-835A, B, C, D (Aux Inst room)	Alarmed	170	FCV-69-1, 2, 12
RWCU hx room	XA-55-3D-17	69-29F, G, H	Alarmed	220	FCV-69-1, 2, 12
RWCU pump A	XA-55-3D-17	69-29D	Alarmed	215	FCV-69-1, 2, 12
RWCU pump B	XA-55-3D-17	69-29E	Alarmed	215	FCV-69-1, 2, 12
RB el 593	XA-55-3E-4	74-95C, D	Alarmed	195	FCV-74-47, 48
RB el 621	XA-55-3E-4	74-95F	Alarmed	155	FCV-43-13, 14

Excerpt from 2-OI-73:

3.3 Equipment (continued)

H. The HPCI Injection valve, 2-FCV-73-44, is a 14 inch, Crane, Class 900, flex wedge gate valve. Flex wedge valves are potentially susceptible to pressure locking. DCN 69896 has been implemented to eliminate the potential for pressure locking of 2-FCV-73-44 by drilling a 1/4" hole in the downstream side of the disc.

3.4 Initiation

A. When any of the following signals are received, the HPCI System automatically initiates:

1. Low RPV water level at -45".
2. High drywell pressure at 2.45 psig.

B. The HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, will automatically open when system flow is at or below 900 gpm (lowering) if a system initiation signal is present, and will automatically close when system flow is at or above 1255 gpm (rising) regardless of presence of initiation signal.

C. HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, will open on receipt of an initiation signal even with HPCI Auxiliary Oil pump in PULL-TO-LOCK position resulting in slowly draining CST to Suppression Chamber.

3.5 Isolation

A. When any of the following signals are received, the HPCI System automatically isolates: (REFER TO 2-AOI-64-2b, Group 4 HPCI Isolation.)

1. High steamline flow at 85 psid(approximately 200%) of rated (3 sec time delay).
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room.
3. Low RPV pressure at 110 psig (does not seal-in).
4. High pressure between rupture diaphragms at 10 psig.
5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 2-HS-73-61, if automatic initiation signal is present.

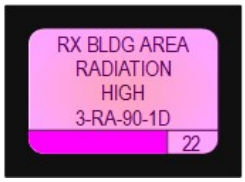
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9	Tier #	1	------
EK2.02 (10CFR 55.41.7)	Group #	2	------
Knowledge of the relationship between High Secondary Containment Area Radiation Levels and the following systems or components:	K/A #	295033EK2.02	
• Process radiation monitoring system	Importance Rating	3.5	------

Proposed Question: **# 25**

Regarding High Secondary Containment Area Radiation Levels, the **VALID** Annunciator **(1)** requires entry into 3-EOI-3, Secondary Containment Control.

One radiation detector/monitor that triggers this annunciator is **(2)**.





A. (1) REACTOR BUILDING AREA RADIATION HIGH (3-9-3A, Window 22)
(2) 3-RM-90-142, REACTOR ZONE RADIATION MONITOR

B. (1) REACTOR BUILDING AREA RADIATION HIGH (3-9-3A, Window 22)
(2) 3-RE-90-20, CRD-HCU WEST AREA ELEVATION 565 REACTOR BUILDING

C. (1) REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 4)
(2) 3-RM-90-142, REACTOR ZONE RADIATION MONITOR

D. (1) REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 4)
(2) 3-RE-90-20, CRD-HCU WEST AREA ELEVATION 565 REACTOR BUILDING

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B)*. Second part is incorrect but plausible in that 3-RM-90-142, REACTOR ZONE RADIATION MONITOR, is a radiation monitor that will initiate REACTOR ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 21).

B **CORRECT**: *(See attached)* In accordance with REACTOR BUILDING AREA RADIATION HIGH (3-9-3A, Window 22), 3-EOI-3, Secondary Containment Control is required to be entered based on Table SC-2, Secondary Containment Area Radiation. For second part, in accordance REACTOR BUILDING AREA RADIATION HIGH (3-9-3A, Window 22), 3-RE-90-20, CRD-HCU WEST AREA ELEVATION 565 REACTOR BUILDING, is listed as one of numerous sensors that will initiate this alarm. If valid, then 3-EOI-3 entry is required, referencing Table SC-2, Secondary Containment Area Radiation, list 3-RE-90-20 as an applicable radiation indicator.

C INCORRECT**:** First part is incorrect but plausible in that REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 4), alarms when high radiation is detected in the applicable exhaust ventilation ducts requiring the use of Abnormal Operating Instructions, but 3-EOI-3 entry is not required. Second part is incorrect but plausible in that REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 4) alarms based on 3-RM-90-250, REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST VENT CONTINUOUS AIR MONITOR.

D INCORRECT: First part is incorrect but plausible *(See C).* First part is correct *(See B).*

RO Level Justification: Tests the candidate's knowledge of the relationship between High Secondary Containment Area Radiation Levels and the Process Radiation Monitoring System. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to High Secondary Containment Area Radiation Levels.

Technical Reference(s): 3-EOI-3, Rev. 17 (Attach if not previously provided)
3-ARP-9-3A, Rev. 60

Proposed references to be provided to applicants during examination: **REACTOR BUILDING AREA RADIATION HIGH (3-9-3A, Window 22) REACTOR BUILDING, TURBINE BUILDING, REFUEL ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 4)**

Learning Objective: OPL171.033 Obj. 3e (As available)

Question Source:

Bank #	BFN NRC 1108 #25	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2011

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

25. 295033 EK2.02 NEW/L

Which ONE of the following completes the statements below?

An VALID annunciator that requires entry into 3-EOI-3, Secondary Containment Control is __(1)__.

One radiation detector/monitor that triggers this annunciator is __(2)__.

A. (1) RX BLDG AREA RADIATION HIGH (3-9-3A, Window 22)
(2) Rx Zone HVAC Radiation Monitor 142

B. (1) RX BLDG AREA RADIATION HIGH (3-9-3A, Window 22);
(2) area radiation monitor on the 565' elevation of the Reactor Bldg

C. (1) RX BLDG, TURB BLDG, RF ZONE EXH RADIATION HIGH (3-9-3A, Window 4)
(2) Rx Zone HVAC Radiation Monitor 142

D. (1) RX BLDG, TURB BLDG, RF ZONE EXH RADIATION HIGH (3-9-3A, Window 4)
(2) area radiation monitor on the 565' elevation of the Reactor Bldg

CORRECT ANSWER B

Excerpts from 3-ARP-9-3A:

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 38 of 59

RX BLDG AREA
RADIATION
HIGH
3-RA-90-1D

22

(Page 1 of 3)

Sensor/Trip Point:

RI-90-4A	RI-90-23A
RI-90-8A	RI-90-24A
RI-90-9A	RI-90-25A
RI-90-13A	RI-90-26A
RI-90-14A	RI-90-27A
RI-90-20A	RI-90-28A
RI-90-21A	RI-90-29A
RI-90-22A	

For setpoints **REFER TO** 3-SIMI-90B.

Sensor Location:

RE-90-4	MG set area	Rx Bldg El. 639	R-17 Q-LINE
RE-90-8	Main Control Room	Rx Bldg El. 617	R-16 R-LINE
RE-90-9	Clean-up System	Rx Bldg El. 621	R-16 T-LINE
RE-90-13	North Clean-up Sys.	Rx Bldg El. 593	R-16 P-LINE
RE-90-14	South Clean-up Sys.	Rx Bldg El. 593	R-16 S-LINE
RE-90-20	CRD-HCU West	Rx Bldg El. 565	R-16 R-LINE
RE-90-21	CRD-HCU East	Rx Bldg El. 565	R-20 R-LINE
RE-90-22	Tip Room	Rx Bldg El. 565	R-19 P-LINE
RE-90-23	Tip Drive	Rx Bldg El. 565	R-19 P-LINE
RE-90-24	HPCI Room*	Rx Bldg El. 519	R-21 U-LINE
RE-90-25	RHR West	Rx Bldg El. 519	R-16 U-LINE
RE-90-26	Core Spray-RCIC	Rx Bldg El. 519	R-16 N-LINE
RE-90-27	Core Spray	Rx Bldg El. 519	R-20 N-LINE
RE-90-28	RHR East	Rx Bldg El. 519	R-20 U-LINE
RE-90-29	Suppression Pool	Rx Bldg El. 519	R-19 U-LINE

* Due to the location of the Rad Monitor in relation to the Test line in the HPCI Quad, the HPCI Room Rad Alarm may be received when the HPCI Flow test is in progress.

Probable Cause: Radiation levels have risen above alarm set point. HPCI Flow Rate Surveillance in Progress.

Automatic Action: None

Continued on Next Page

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 39 of 59

RX BLDG AREA RADIATION HIGH 3-RA-90-1D, Window 22
(Page 2 of 3)

Operator Action:

A. **DETERMINE** area with high radiation level on Panel 3-9-11. (Alarm on Panel 3-9-11 will automatically reset if radiation level lowers below setpoint.)

B. **IF** the alarm is from the HPCI Room while Flow testing is being performed, **THEN**
REQUEST personnel at the HPCI Quad to validate conditions.

C. **NOTIFY** RAD PRO.

D. **IF** the TSC is **NOT** manned and a "VALID" radiological condition exists., **THEN**
USE public address system to evacuate area where high radiological conditions exist

E. **IF** the TSC is manned and a "VALID" radiological condition exists, **THEN**
REQUEST the TSC to evacuate non-essential personnel from affected areas.

F. **MONITOR** other parameters providing input to this annunciator frequently as these parameters will be masked from alarming while this alarm is sealed in.

G. **IF** a CREV initiation is received, **THEN**
1. **CHECK** CREV A(B) Flow is ≥ 2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213) within 5 hours of the CREV initiation. [BFPER 03-017922]
2. **IF** CREV A(B) Flow is **NOT** ≥ 2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213) **THEN**
PERFORM the following: (Otherwise **N/A**)
[BFPER 03-017922]
   a. **STOP** the operating CREV per 0-OI-31.
   b. **START** the standby CREV per 0-OI-31.

H. **IF** alarm is due to malfunction, **THEN**
REFER TO 0-OI-55.

Continued on Next Page

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 40 of 59

RX BLDG AREA RADIATION HIGH 3-RA-90-1D, Window 22
(Page 3 of 3)

Operator Action: (Continued)

I. **ENTER** 3-EOI-3 Flowchart.

J. **REFER TO** 3-AOI-79-1 or 3-A01-79-2 if applicable.

K. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function.
REFER TO NPG-SPP-18.3.5.

References: 3-45E620-3 3-45E610-90-1 GE 730E356-1

Supports Distractors C1, D1:

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 11 of 59

RX BLDG, TURB BLDG,
RF ZONE EXH
RADIATION HIGH
3-RA-90-250A

4

(Page 1 of 2)

Sensor/Trip Point:

RM-90-250

Gas

HIGH ALARM - 2145 CPM
ALERT - 1073 CPM

Sensor Location: El 664' Refuel floor, Col R-17 P-LINE

Probable Cause:
A. Daily source check.
B. High radiation in the Reactor Building, Turbine Building, or Refuel Zone exhaust ventilation ducts.
C. Dry Cask storage activities in progress.

Automatic Action: None

Operator Action:
A. **REQUEST** Unit 1 Operator assist in determining source of alarm.

B. **IF** high activity is confirmed, **THEN**
NOTIFY RAD PRO.

C. **REQUEST** Chemistry perform radiochemical analysis to determine source.

D. **IF** Dry Cask storage activities are in progress, **THEN**
NOTIFY CASK Supervisor.

E. **IF** the TSC is **NOT** manned, **THEN**
EVACUATE personnel from affected areas.

F. **IF** the TSC is manned, **THEN**
REQUEST the TSC to evacuate non-essential personnel from affected areas.

G. **REFER TO** 3-AOI-79-1 or 3-AOI-79-2 if applicable.

H. **MONITOR** release rate for ODCM compliance. **IF** ODCM limits are exceeded, **THEN**
REFER TO EPIP-1.

Continued on Next Page

RX BLDG, TURB BLDG, RF ZONE EXH RADIATION HIGH 3-RA-90-250A, Window 4
(Page 2 of 2)

Operator Action: (Continued)

I. **IF** Air Particulate Monitor Controller is operable, **THEN** **REFER TO** 3-OI-90, to reset alarms.

J. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References:

47W600-80	3-47E610-90-1	3-45E620-3
3-SIMI-90B	TVA Calc NDQ00902005008/EDC63693	

Supports Distractors A2, C2:

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 36 of 59

REACTOR ZONE EXHAUST RADIATION HIGH
3-RA-90-142A
21

(Page 1 of 2)

Sensor/Trip Point:

3-RE-90-142A 72 mR/HR
3-RE-90-142B 72 mR/HR
3-RE-90-143B 72 mR/HR
3-RE-90-143B 72 mR/HR

Technical Specifications require a setting at ≤ 100 mR/HR.

Sensor Location:
Panel 9-42
Rx Bldg, El 664'
R-18 P-LINE

Probable Cause:
A. High radiation in the exhaust duct.
B. Dry Cask loading/unloading activities in progress.
C. Primary system leak in Secondary Containment.
D. Loss of power to NUMAC drawer.

Automatic Action:
A. Control Room, Reactor Zone, and Refueling Zone ventilation isolate.
B. Control Room Emergency Pressurization units start.
C. SGTS initiates.
D. PCIS Group 6 isolation initiates.

Operator Action:
A. **CHECK** alarm condition on the following:
1. REACTOR & REFUEL ZONE EXHAUST RADIATION recorder, 3-RR-90-144 points 1 and 2 on Panel 3-9-2.
2. RX & REFUEL ZONE EXH CH A RAD MON RTMR, 3-RM-90-140/142 on Panel 3-9-10.
3. RX & REFUEL ZONE EXH CH B RAD MON RTMR, 3-RM-90-141/143 on Panel 3-9-10.

B. **IF** Dry Cask loading/unloading activities are in progress, **THEN NOTIFY** the Cask Supervisor to **PLACE** the MPC in a safe condition per one of the following methods:
1. MSI-0-079-DCS400.1
2. MSI-0-079-DCS400.1FW
3. As directed by Radiation Protection.

Continued on Next Page

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 37 of 59

REACTOR ZONE EXHAUST RADIATION HIGH 3-RA-90-142A, Window 21
(Page 2 of 2)

Operator Action: (Continued)

C. **ENTER** 3-EOI-3 Flowchart.

D. **REFER TO** 3-AOI-64-2d and, for loss of power, 3-OI-90.

E. **REFER TO** 3-AOI-79-1 or 3-AOI-79-2 as applicable.

F. **REFER TO** EPIP-1.

G. **REFER TO** Technical Specification Section 3.3.6.2 and 3.3.7.1.

References:

3-45E620-3	3-47E610-90-1	GE 3-730E927-21
0-47E201-1	0-47E202-1	
Technical Specifications 3.3.6.2 and 3.3.7.1		

Excerpt from 3-EOI-3:

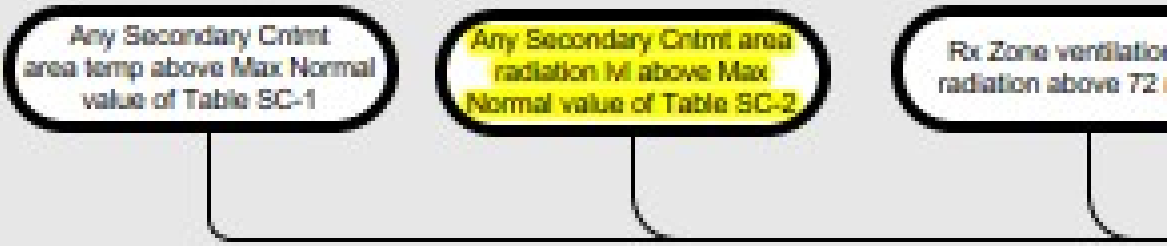


Table SC-2
Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5

EA1.03 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to Secondary Containment High Sump/Area Water Level:

- Radwaste system

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295036EA1.03	
Importance Rating	2.7	------

Proposed Question: **# 26**

Which **ONE** of the following completes the statement below?

The indication that **ONLY** exists in the Radwaste (RW) Control Room which requires the RW Operator to immediately notify the affected Main Control Room of required entry into EOI-3, Secondary Containment Control, is the Unit 1, 2 or 3 Reactor Building _____.

A. EQUIPMENT DRAIN SUMP ABNORMAL LEVEL at 20 inches

B. FLOOR DRAIN SUMP 'A' OR 'B' ABNORMAL LEVEL at 20 inches

C. EQUIPMENT DRAIN SUMP ABNORMAL LEVEL at 66 inches

D. FLOOR DRAIN SUMP 'A' OR 'B' ABNORMAL LEVEL at 66 inches

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible (*See B and C*).

B INCORRECT: Incorrect but plausible (*See D*). The 20 inches is based on the EOI-3 Table SC-3 Secondary Containment Water Level. This is the MAX SAFE value of Secondary Containment **AREA** Water Level not from the Floor Drain Sumps MAX NORMAL value of 66 inches (also in Table SC-3). The 20 inches requires action in EOI-3 based on severity to either Shutdown the Reactor or SCRAM with possibly Emergency Depressurizing.

C INCORRECT: Incorrect but plausible in that the Reactor Building **Equipment** Drain alarms at 75 inches (not 66 inches) from the bottom of the sump on the Panel 25-17A ONLY located in the RW Control Room. The ARP directs the RW Operator to notify the affected Unit to assist in determining cause of the alarm, but this is NOT an EOI-3 entry condition. Both Reactor Building Floor and Equipment Drain sump level alarm conditions will also activate the Panel 25-17A affected Unit alarm 'REACTOR UNIT ONE (TWO or THREE)'

D **CORRECT**: (*See attached*) In accordance with 0-ARP-25-17A, the Reactor Building **Floor** Drain Sump 'A' or 'B' Abnormal level alarms at **66 inches** from the bottom of the sump. This alarm Panel 25-17A is ONLY located in the RW Control Room therefore the ARP directs the RW Operator to immediately notify the affected Unit of a sealed in alarm requiring an EOI-3 entry condition. Secondary Containment floor drain sump water level above 66 inches is an EOI-3 entry condition and listed in Table SC-3.

RO Level Justification: Tests the candidate's ability to operate and/or monitor the Radwaste System as it relates to Secondary Containment High Sump Levels/Area Water Levels as required EOI entry conditions. This question is rated as Memory due to the requirement to strictly recall facts.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-EOI-3, Rev. 7 (Attach if not previously provided)
OPL171.084, Rev. 10U4
0-ARP-25-17A, Rev. 18

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.084 Obj. 3 (As available)

Question Source:

Bank #	BFN 1804 NRC #27
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2018

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 27**

Which ONE of the following completes the statement below?

The indication that **ONLY** exists in the Radwaste (RW) Control Room which requires the RW Operator to notify the affected Main Control Room immediately of required entry into EOI-3, Secondary Containment Control is the Unit 1, 2 or 3 Reactor Building _____.

A. EQUIPMENT DRAIN SUMP ABNORMAL LEVEL (20 inches)

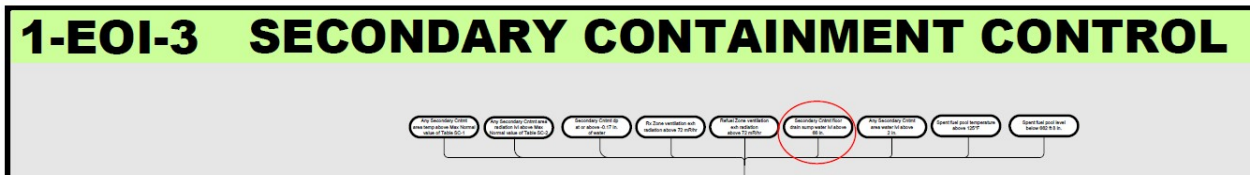
B. FLOOR DRAIN SUMP 'A' OR 'B' ABNORMAL LEVEL (20 inches)

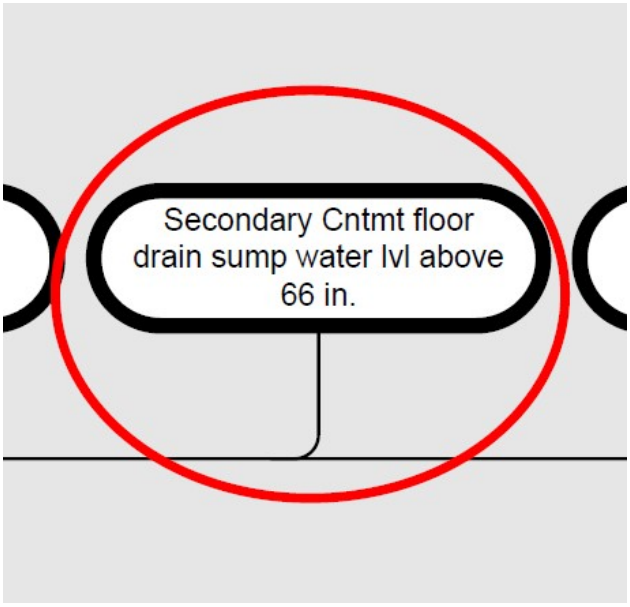
C. EQUIPMENT DRAIN SUMP ABNORMAL LEVEL (66 inches)

D. FLOOR DRAIN SUMP 'A' OR 'B' ABNORMAL LEVEL (66 inches)

Proposed Answer: **D**

Excerpts from 1-EOI-3 illustrating entry conditions and Table SC-3:





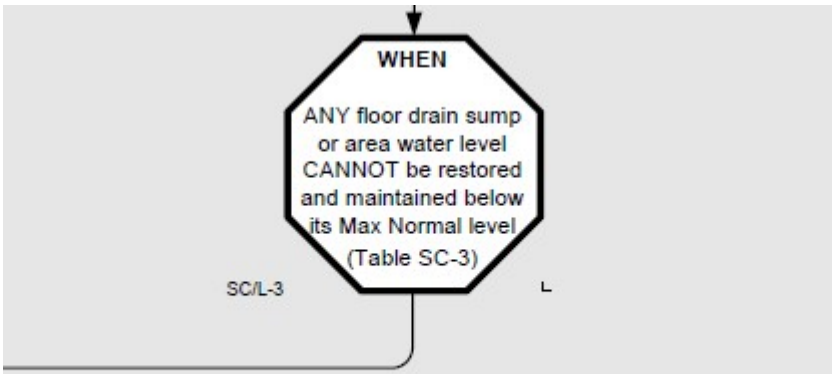
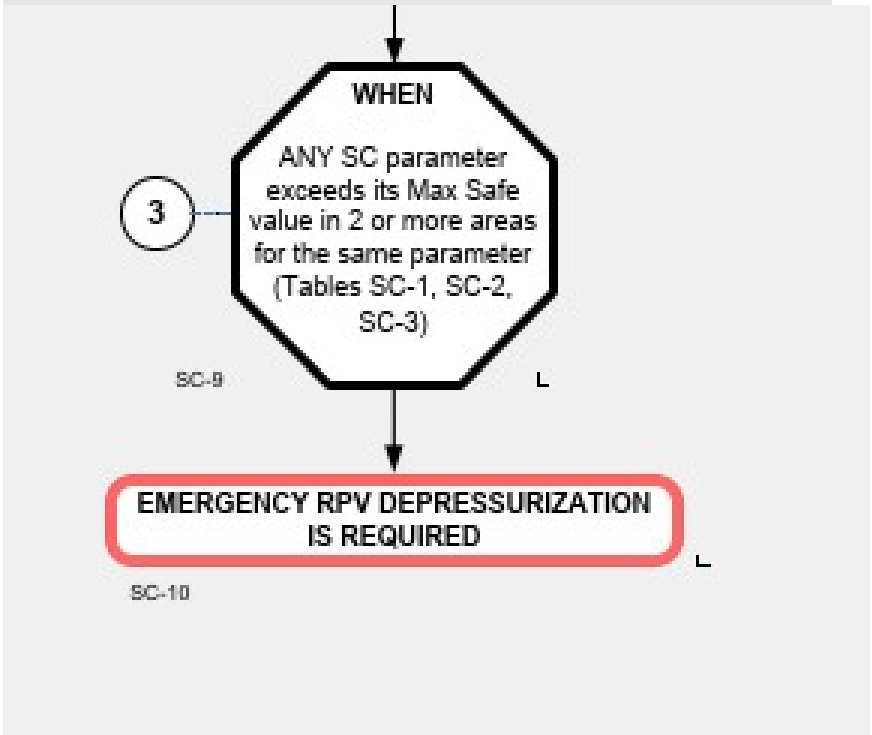


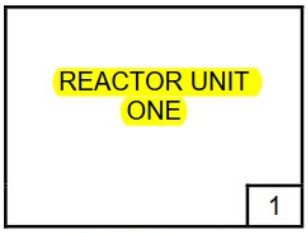
Table SC-3
Secondary Cntmt Water Level

Water Level	Max Normal Value in.	Max Safe Value in.
Floor drain sumps	66	None
Areas	2	20



Excerpts from Radwaste Panel 0-ARP-25-17A:

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 4 of 52



(Page 1 of 1)

Sensor/Trip Point:

Window number one has multiple inputs for reflash. These inputs represent unit specific alarms associated with radwaste on 0-LPNL-925-0017, XA-55-17A.

Sensor Location: 0-LPNL-925-0017
Elevation 565
Radwaste Control Room

Probable Cause:

A. Actuation of any of the following abnormal parameters concerning unit one:
   1. Drywell floor and equipment sump level.
   2. Reactor bldg floor and equipment drain sump level.
   3. Drywell and reactor bldg equipment drain sump temperature.
   4. Turbine bldg floor and equipment drain sump level.
   5. Backwash receiver pit and turbine bldg condensate pump pit floor drain sump level.
   6. If in service, floor drain filter and waste collector filter turbidity.
   7. Condensate pump pit equipment drain sump level.

B. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** for other alarms on 0-LPNL-925-0017.
B. **REFER TO** the appropriate alarm response procedure.
C. **IF** no other annunciator on 0-LPNL-925-0017, XA-55-17A is in alarm, **THEN**
   1. **PERFORM** an alarm test.
   2. **REQUEST** unit one UO to check for possible duplicate alarms.
   3. **NOTIFY** SRO and initiate CR for any failed annunciators or sensors.
D. **LOG** valid events and actions taken in NOMS narrative log.

References: 45N799-4 730E934-16 791E492-10

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 8 of 52

LA-77-17
RX BLDG
EQUIP DR SUMP
ABNORMAL
4

(Page 1 of 1)

Sensor/Trip Point:

1-LT-077-0017	Elevation 514'
2-LT-077-0017	75" from bottom of sump'
3-LT-077-0017	
1/2/3-PLC-077-0017	Excessive leakage

Sensor Location: Rx Bldg Equip Sump, Elevation 519
Reactor Bldg

Probable Cause:
A. Excessive in-leakage.
B. Pump(s) and/or valve malfunction.
C. Valve misalignment or discharge check valve of idle pump stuck open.
D. Level Transmitter malfunction.

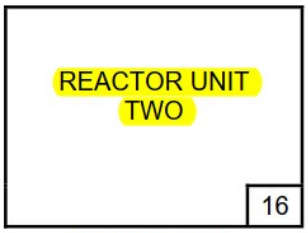
Automatic Action: Second sump pump starts if due to HI LEVEL, if due to EXCESSIVE INLEAKAGE no auto actions.

Operator Action:
A. **IDENTIFY** which unit's sump is in alarm by use of PLC HMI Work Station.
B. **IDENTIFY** which condition is causing alarm (EXCESSIVE INLEAKAGE) or (HI LEVEL).
C. **IF** EXCESSIVE INLEAKAGE, **THEN** **INVESTIGATE** the change in leak rate to sump.
D. **IF** HI LEVEL, **THEN** **CHECK OPEN** valve 1(2)(3)-FCV-077-0017B.
   1. **CHECK** RUNNING or **START** both A and B sump pumps.
   2. **CHECK** for any associated alarm(s) or abnormal indications:
      - Sump high temp alarm REACTOR BLDG EQUIP SUMP TEMP HIGH (XA-55-17A, Window 3).
      - Loss of power indication(s) (pump/valve red & green light extinguished).
   3. **REQUEST** operator assistance from the appropriate UO to determine cause of alarm by checking:
      - Pump coupling and sump level on 1(2)(3)-LI-77-17B.
      - Sump pump A 480V RMOV BD C breaker 4B position.
      - Sump pump B 480V RMOV BD B breaker 18B position.
   4. **NOTIFY** SRO and **LOCATE/STOP** source of excess water (see print 1-47E852-2, 2-47E852-2, 3-47E852-2).
   5. **LOG** valid events and actions taken in NOMS narrative log.

References:

0-45E779-16	0-47E610-77-1	791E492-10	1-47E852-2
2-47E852-2	3-47E852-2	730E934-20,21,22	

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 21 of 52



(Page 1 of 1)

Sensor/Trip Point:

Window number sixteen has multiple inputs for reflash. These inputs represent unit specific alarms associated with radwaste on 0-LPNL-925-0017, XA-55-17A.

Sensor Location: 0-LPNL-925-0017
Elevation 565
Radwaste Control Room

Probable Cause:

A. Actuation of any of the following abnormal parameters concerning unit two:
   1. Drywell floor and equipment sump level.
   2. Reactor bldg floor and equipment drain sump level.
   3. Drywell and reactor bldg equipment drain sump temperature.
   4. Turbine bldg floor and equipment drain sump level.
   5. Backwash receiver pit and turbine bldg condensate pump pit floor drain sump level.
   6. Condensate pump pit equipment drain sump level.

B. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** for other alarms on 0-LPNL-925-0017.
B. **REFER TO** the appropriate alarm response procedure.
C. **IF** no other annunciator on 0-LPNL-925-0017, XA-55-17A is in alarm, **THEN**
   1. **PERFORM** an alarm test.
   2. **REQUEST** unit two UO to check for possible duplicate alarms.
   3. **NOTIFY** SRO and initiate CR for any failed annunciators or sensors.

D. **LOG** valid events and actions taken in NOMS narrative log.

References: 45N799-4 730E934-16 791E492-10

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 23 of 52

LA-77-8A
RX BLDG
FLR DR SUMP A
ABNORMAL
18

(Page 1 of 2)

Sensor/Trip Point:

1-LT-077-0008A	Elevation 516' 6"
2-LT-077-0008A	66" from bottom of sump
3-LT-077-0008A	
1/2/3-PLC-077-0008A	Excessive leakage

Sensor Location: Rx Bldg FD Sump
Elevation 519 near RHR Pump RM A & C
Reactor Bldg

Probable Cause:
A. Excessive in-leakage.
B. Pump(s) and/or valve malfunction.
C. Valve misalignment or discharge check valve on idle pump stuck open.
D. Level transmitter malfunction

Automatic Action: Sump pump starts at Elevation 515

Operator Action:
A. **IDENTIFY** which unit's sump is in alarm and which condition is responsible for alarm (Excessive leakage or High-High level) by use of PLC HMI Work Station.
B. **IF** high inleakage is the cause of the alarm, **THEN** **NOTIFY** appropriate Unit SRO and request assistance to **LOCATE/STOP** source of excess water (**REFER TO** print 1-47E852-1, 2-47E852-1, 3-47E852-1).
C. **IF** high level in the sump is the cause of the alarm, **THEN** **PERFORM** the following:
   1. **CHECK** RUNNING or **START** Pump A.
   2. **NOTIFY** affected unit Unit SRO of an EOI-3 entry condition alarm being sealed in on Radwaste Control Panel (High-High level is an EOI-3 entry condition).
   3. **REQUEST** operator assistance from the appropriate Unit Operator to determine cause of alarm by checking the following: (steps a through c can be performed in any logical order and may be suspended when the cause of the High Level is discovered):
      a. Sump local level indicator LI-077-0008AB and pump coupling.
      b. Sump pump A breaker (480V RMOV BD C breaker 7A).
      c. Valve alignment. **REFER TO** OI-77.

Continued on Next Page

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 24 of 52

LA-77-8A REACTOR BLDG FD SUMP A LEVEL HIGH HIGH, Window 18
(Page 2 of 2)

Operator Action: (Continued)

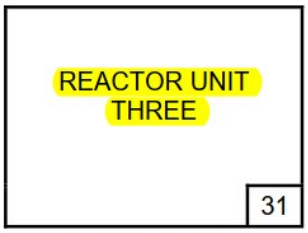
4. **RECORD** entry in narrative log including the name of the notified Unit SRO.

D. **CHECK** for any associated alarm(s) or abnormal indications:
1. Sump B high level high alarm RX BLDG FLR DR SUMP B ABNORMAL (XA-55-17A, Window 33).
2. Loss of power indications (pump red & green light extinguished).

References:

0-45E779-16	791E492-10	0-47E610-77-1
1-47E852-1	730E934-20, 21, 22	2-47E852-1
3-47E852-1	45N1711-6	2-45E2751-7
3-45E3751-7		

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 37 of 52



(Page 1 of 1)

Sensor/Trip Point:

Window number thirty-one has multiple inputs for reflash. These inputs represent unit specific alarms associated with radwaste on 0-LPNL-925-0017, XA-55-17A.

Sensor Location: 0-LPNL-925-0017
Elevation 565
Radwaste Control Room

Probable Cause:

A. Actuation of any of the following abnormal parameters concerning unit three:
   1. Drywell floor and equipment drain sump level.
   2. Reactor bldg floor and equipment drain sump level.
   3. Drywell and reactor bldg equipment drain sump temperature.
   4. Turbine bldg floor and equipment drain sump level.
   5. Backwash receiver pit and turbine bldg condensate pump pit floor drain sump level.
   6. Condensate pump it equipment drain sump level.

B. Level Switch Malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** for other alarms on 0-LPNL-925-0017.
B. **REFER TO** the appropriate alarm response procedure.
C. **IF** no other annunciator on 0-LPNL-925-0017, XA-55-17A is in alarm:
   1. **PERFORM** an alarm test.
   2. **REQUEST** unit three UO to check for possible duplicate alarms.
   3. **NOTIFY** SRO and initiate CR for any failed annunciators or sensors.

D. **LOG** valid events and actions taken in NOMS narrative log.

References: 45N799-4 730E934-16 791E492-10

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 39 of 52

LA-77-8B
RX BLDG
FLR DR SUMP B
ABNORMAL
33

(Page 1 of 2)

Sensor/Trip Point:

1-LT-077-0008B	Elevation 516' 6"
2-LT-077-0008B	66" from bottom of sump
3-LT-077-0008B	
1/2/3-PLC-077-0008B	Excessive leakage

Sensor Location: Rx Bldg Floor Drain Sump
Elevation 519 near RHR pump Rm B & D
Reactor Bldg

Probable Cause:
A. Excessive in-leakage.
B. Pump(s) and/or valve malfunction.
C. Valve misalignment or discharge check valve on idle pump stuck open.
D. Level transmitter Malfunction.

Automatic Action: Sump pump starts at Elevation 515.

Operator Action:
A. **IDENTIFY** which unit's sump is in alarm and which condition is responsible for alarm (Excessive leakage or High-High level) by use of PLC HMI Work Station.
B. **IF** high inleakage is the cause of the alarm, **THEN** **NOTIFY** appropriate Unit SRO and request assistance to **LOCATE/STOP** source of excess water. (**REFER TO** print 1-47E852-1, 2-47E852-1, 3-47E852-1).
C. **IF** high level in the sump is the cause of the alarm, **THEN** **PERFORM** the following:
   1. **CHECK** RUNNING or **START** Pump B.
   2. **NOTIFY** affected unit Unit SRO of an EOI-3 entry condition alarm being sealed in on Radwaste Control Panel (High-High level is an EOI-3 entry condition).
   3. **REQUEST** operator assistance from the appropriate Unit Operator to determine cause of alarm by checking the following: (steps a through c can be performed in any logical order and may be suspended when the cause of the High Level is discovered):
      a. Sump local level indicator LI-077-0008BB and pump coupling.
      b. Sump pump B breaker (480V RMOV BD B breaker 1B).
      c. Valve alignment. **REFER TO** OI-77.

Continued on Next Page

BFN Unit 0	0-LPNL-925-0017 XA-55-17A	0-ARP-25-17A Rev. 0018 Page 40 of 52

LA-77-8B REACTOR BLDG FD SUMP B LEVEL HIGH HIGH, Window 33
(Page 2 of 2)

Operator Action: (Continued)

4. **RECORD** entry in narrative log including the name of the notified Unit SRO.

D. **CHECK** for any associated alarm(s) or abnormal indications:
   1. Sump AB level high alarm RX BLDG FLR DR SUMP A ABNORMAL (XA-55-17A, Window 18).
   2. Loss of power indications (pump red & green light extinguished).

References:

0-45E779-16	0-47E610-77-1	791E492-10
1-47E852-1	2-47E852-1	3-47E852-1
730E934-20,21,22	45N1750-13	3-45N3750-13
2-45N2750-13		

Excerpt from Radwaste Systems Lesson Plan:

Outline of Instruction	Instructor Notes and Methods (Optional)
2) Equipment Access Lock Water seal level low alarm could indicate loss of Secondary Containment if outside Equipment doors (229A, B, or C) are open	0-ARP-25-17A/32
3) The Reactor Building floor drain sump (RBFDS) High High level (66") is an EOI-3 entry condition. This alarm is on Radwaste control panel 25-17A only. For any unit there is a common alarm RBFD sump A (B) Abnormal. Concurrently, the alarm Reactor Unit One (Two or Three) will alarm to specify which unit has abnormalities. Per the ARP, the RW operator will verify the EOI-3 entry condition and MUST notify the affected Unit's MCR immediately.	Objective 3 (OF-4) Note: RB equipment drain (RBED) sump High/High is not an EOI-3 entry condition
4. Sump locations that pump into radwaste.	
a. Turbine building floor and equipment drain sumps located north end of condenser room (U3 South end).	
b. Condensate pump pit floor and equipment drain sumps located west side of each condensate pump pit.	U1/U2 Share CBWRP FD.
c. Condensate Backwash receiver pit floor drain (CBWRPFD) sump located in receiver pit. Entered from EL 557, Con Demin area.	U3 has own pit.
d. Reactor building equipment drain sump located north-east quadrant basement reactor building.	
e. Reactor building floor drain sumps.	
1) 2 per unit, (one pump per sump).	
2) Cross connected by 8 inch line at overflow and 6 inch line at normal level.	
3) Located in south-east and south-west quads.	
f. Drywell floor and equipment drain sumps (DWFD and DWED) located north end of drywell basement. DWED and RBED sumps are designed with RBCCW cooling water supplied heat exchangers to cool sump water. U1 has been upgraded such that the DWED sump HX is physically located within the sump volume. The recirc valve and logic have been deleted.	
g. Radwaste equipment and floor drain (RWED and RWFD) sumps located north wall in basement of Radwaste.	
h. Off-Gas Condenser sump located northwest end of Radwaste Basement.	
i. SBGT building sumps located in standby gas treatment building.	
j. Off-Gas Building sump located in basement of off-gas building.	

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode **K4.13** (10CFR 55.41.7) Knowledge of RHR/LPCI: Injection Mode design features and/or interlocks that provide for the following: • Prevention of leakage to the environment through LPCI/RHR heat exchanger	Tier #	2	------
	Group #	1	------
	K/A #	203000K4.13	
	Importance Rating	3.0	------

Proposed Question: **# 27**

Complete the statements below as it relates to the design and interlocks of the RHR System Shutdown Cooling mode of operation.



- RHR SYS I/II DISCHARGE OR SHUTDOWN COOLING HEADER PRESSURE HIGH (3-9-3E, Window 32) alarms

3-FCV-74-53/67, RHR LPCI INBOARD INJECTION VALVES, **(1)** automatically close when 3-FCV-74-47 (48), RHR SHUTDOWN COOLING SUCTION OUTBOARD (INBOARD) ISOLATION VALVES, are open when a PCIS Group 2 Isolation signal is received.

In accordance with the given ARP, 3-FCV-74-47 (48), RHR SHUTDOWN COOLING SUCTION OUTBOARD (INBOARD) ISOLATION VALVE, automatically close when Reactor Pressure rises to a **MINIMUM** of **(2)** to protect low pressure components from high Reactor Pressure.

A. (1) will
(2) 100 psig

B. (1) will
(2) 110 psig

C. (1) will NOT
(2) 100 psig

D. (1) will NOT
(2) 110 psig

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 3-OI-74, RHR System, when a PCIS Group 2 Isolation occurs while in Shutdown Cooling (which means that 3-FCV-74-47 (48), RHR SHUTDOWN COOLING SUCTION OUTBOARD (INBOARD) ISOLATION VALVES are both open) then 3-FCV-74-53/67, RHR LPCI INBOARD INJECTION VALVES will close. For second part, in accordance with the Alarm Response Procedure for the given RHR SYS I/II DISCHARGE OR SHUTDOWN COOLING HEADER PRESSURE HIGH (3-9-3E, Window 32), 3-FCV-74-47 (48), RHR SHUTDOWN COOLING SUCTION OUTBOARD (INBOARD) ISOLATION VALVE, automatically close when Reactor Pressure rises to a minimum of 100 psig. This is to protect low pressure components from high Reactor Pressure.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that the HPCI System automatically isolates when Reactor Pressure lowers to 110 psig.

C INCORRECT: First part is incorrect but plausible if the candidate fails to associate 3-FCV-74-53/67, RHR LPCI INBOARD INJECTION VALVES, as PCIS Group 2 isolation valves. Second part is correct *(See A).*

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's knowledge of RHR/LPCI Injection Mode design features and/or interlocks as it relates to prevention of leakage to the environment through the RHR low pressure components. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate them across multiple systems utilizing extensive knowledge of various modes of RHR and Primary Containment Isolation signals.

Technical Reference(s): 3-OI-74, Rev. 142 (Attach if not previously provided)
3-ARP-9-3E, Rev. 35
OPL171.044, Rev. 23

Proposed references to be provided to applicants during examination: **RHR SYS I/II DISCHARGE OR SHUTDOWN COOLING HEADER PRESSURE HIGH (3-9-3E, Window 32)**

Learning Objective: OPL171.044 Obj. 4 i, 4f (As available)

Question Source: Bank # ____
Modified Bank # BFN ILT 2204 #54 (Note changes or attach parent)
New ____

Question History: Last NRC Exam 2022

Form 4.2-1 **Written Examination Question Worksheet**

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	**X**
10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

Copy of Bank Question:

Proposed Question: **# 54**

Unit 3 is in MODE 3 with Loop I RHR in Shutdown Cooling when the following conditions occur:

- A Drywell leak occurs
- Drywell Pressure rises to 2.7 psig
- RHR SYS I/II DISCHARGE OR SHUTDOWN COOLING HEADER PRESSURE HIGH (3-9-3E, Window 32) alarms
- Reactor Pressure is 105 psig



In accordance with the appropriate ARP, **(1)** automatically closed when Reactor Pressure reached **(2)**.

Note: 3-FCV-74-47, RHR SHUTDOWN COOLING SUCTION OUTBOARD ISOLATION VALVE
3-FCV-74-48, RHR SHUTDOWN COOLING SUCTION INBOARD ISOLATION VALVE

A. (1) **ONLY** 3-FCV-74-47
(2) 95 psig

B. (1) **ONLY** 3-FCV-74-47
(2) 100 psig

C. (1) 3-FCV-74-47 **AND** 3-FCV-74-48
(2) 95 psig

D. (1) 3-FCV-74-47 **AND** 3-FCV-74-48
(2) 100 psig

Proposed Answer: **D**

Excerpt from 3-ARP-9-3E:

BFN Unit 3	Panel 9-3 3-XA-55-3E	3-ARP-9-3E Rev. 0035 Page 38 of 41

RHR SYS I/II
DISCH OR SD CLG
HDR PRESS HIGH

3-PA-74-51

32

(Page 1 of 1)

Sensor/Trip Point:

PS-74-51 for System I	400 psig
PS-74-65 for System II	400 psig
PS-74-93 for Shutdown Cooling Suction	100 psig

Sensor Location:	PS-74-51	Rx Bldg PS-74-65	Rx Bldg, PS-74-93
	Panel 25-59	Panel 25-62	Panel 25-59
	Rx Bldg, El 519'	Rx Bldg, El 519'	Rx Bldg, El 519'
	R-15 T-LINE	R-20 T-LINE	R-15 T-LINE

Probable Cause:

A. RHR inboard injection valve and RHR testable check valve leaking from Reactor to RHR loop.
B. RHR shutdown cooling suction isolation valves, leaking from Reactor to RHR shutdown cooling suction piping.
C. RHR pump start for testing or torus cooling.

Automatic Action:

3-FCV-74-47 and -48 will close at 100 psig in Shutdown Cooling.

Operator Action:

A. **CHECK** RHR system pressures, 3-PI-74-51 (SYS I) and 3-PI-74-65 (SYS II) on Panel 3-9-3.

B. **REFER TO** Tech Spec 3.5.1, 3.5.2, 3.6.2.3, 3.6.2.4, 3.6.2.5) Technical Requirements Manual 3.5.

C. **REFER TO** 3-OI-74, for returning header high pressure to normal.

References:

3-45E620-1 3-47E610-74-1 3-47E811-1 3-730E938-12
Technical Specifications 3.5.1, 3.5.2, 3.6.2.3, 3.6.2.4, 3.6.2.5, 5.4 and 5.5
Technical Requirements Manual 3.5
FSAR Section 13.0
3-SIMI-74B

Excerpt from 3-OI-74:

3.6 Interlocks (continued)

6. The RHR outboard LPCI injection valves, 3-FCV-74-52(66), have throttling capability. They receive an auto open signal in the presence of a LPCI initiation signal when Reactor pressure is ≤ 450 psig and are interlocked open under these conditions until the appropriate LPCI SYS I (SYS II) OUTBD INJ VLV BYPASS SEL keylock Switch, 3-HS-74-155A(155B), is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is >450 psig, if its in-line companion valve 3-FCV-74-53(67) is not fully closed.

   The circuitry of 3-FCV-74-66 has been modified so that a fire can not energize the closing coil and shut the valve (any close signal with the control room handswitch in NORMAL will short out the closing coil and clear the control power fuses).

7. If Unit 3 reactor pressure exceeds 100 psig, or a Group II isolation occurs on Unit 3 while Shutdown Cooling is in operation, the following will occur for the given condition:

   (100 psig) RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 3-FCV-74-47 and 3-FCV-74-48, close, thus tripping operating Unit 3 RHR Pumps.

   (Group II) RHR SYS I and II LPCI INBD INJECT VALVEs, 3-FCV-74-53 and 3-FCV-74-67, close and RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 3-FCV-74-47 and 3-FCV-74-48, close, thus tripping operating Unit 3 RHR Pumps.

8. To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 3-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton, 3-HS-74-126(132), must be depressed after either of the following occur:

   a. Isolation signal has been reset **OR**

   b. 3-FCV-74-47 or 3-FCV-74-48 are fully closed.

Excerpt from OPL171.044 Lesson Plan:

(b) Prevents operation from anywhere but the breaker.

(4) Can be electrically opened with reactor pressure below 450 psig, regardless of the in-line Outboard Injection Valve position (10A-K35A/B)

(a) 450 psig interlock can be overridden using SSI Bypass or OI Bypass position of HS-74-180/181)

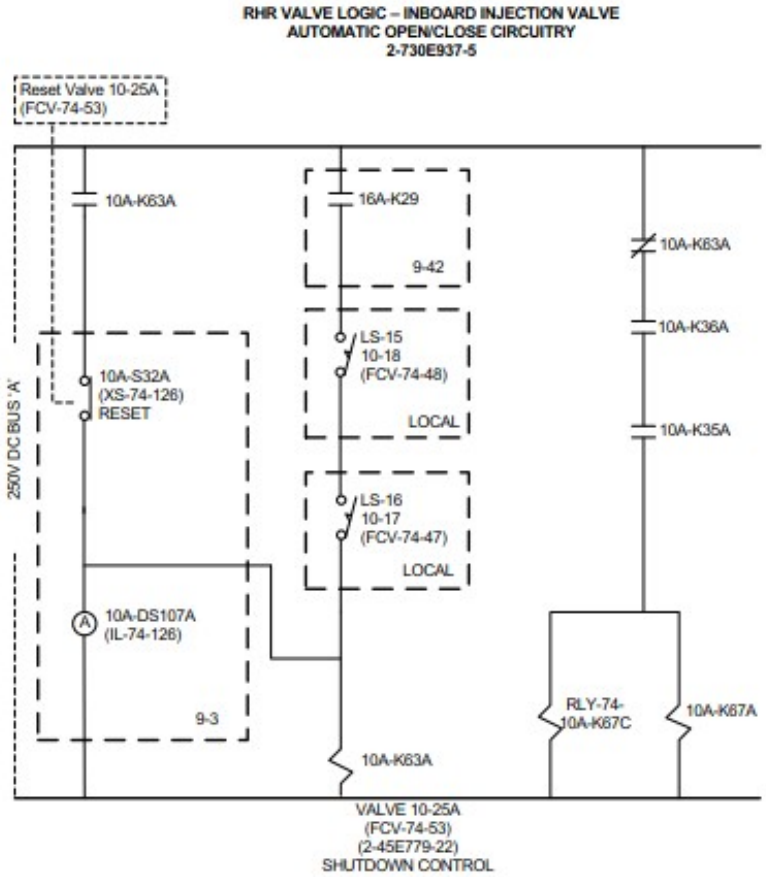


Figure 40 – RHR LPCI Inboard Injection Valve Auto Open/Close Logic

(5) Automatically opens with the following:

(a) Reactor pressure below 450 psig (10A-K35A/B)

(b) LPCI Initiation Signal (10A-K36A)

(6) With LPCI signal sealed-in, LPCI Inboard Injection Valve will remain open until signal is cleared and reset.

(7) Inboard Injection Valves (74-53/67) automatically close on the following:

(a) FCV-74-47 and FCV-74-48 NOT full closed (open) AND a Group 2 Isolation Signal (16A-K29) present. Group 2

Excerpt from 3-OI-73: (Supports Distractors B2, D2)

BFN Unit 3	High Pressure Coolant Injection System	3-OI-73 Rev. 0066 Page 9 of 98

3.0 PRECAUTIONS AND LIMITATIONS

A. The HPCI turbine automatically trips on any of the following:

1. RPV water level high at +51 inches
2. Low pump suction pressure at 19.3" HG Vacuum (4.7 sec time delay)
3. Turbine high exhaust pressure at 140 psig
4. Any isolation signal
5. Remote Manual HPCI TURBINE TRIP pushbutton, 3-HS-73-18A

B. HPCI turbine overspeed at 122% (~5000 rpm) of rated speed (~4100 rpm) results in a hydraulic trip. The hydraulic trip occurs when operating oil is ported from the HPCI TURBINE STOP VALVE, 3-FCV-073-0018, causing the stop valve to close under spring force. Once the stop valve is closed, the piston of the hydraulic trip resets. With the HPCI turbine under load, the field-adjusted reset should occur between 2500 and 3000 rpm, and the startup sequence should commence. Since the overspeed trip condition does not result in any automatic trip signals in the HPCI control circuit, the HPCI PUMP MIN FLOW VALVE, 3-FCV-73-0030 does not close as a direct result of the turbine overspeed.

C. The HPCI System automatically isolates on any of the following: (Refer to 3-AOI-64-2b, Group 4 HPCI Isolation.)

1. High steamline flow at 85 psid (~200% of rated) (3 sec time delay)
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room
3. Low RPV pressure at 110 psig (does not seal-in)
4. High pressure between rupture diaphragms at 10 psig
5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 3-HS-73-61, if automatic initiation signal is present

D. HPCI System automatically initiates from one-out-of-two-taken twice logic from either: (**REFER TO** Section 5.1).

1. Low RPV water level at -45" (3-LIS-3-58A through D), OR
2. High drywell pressure at 2.45 psig (3-PS-64-58A through D).

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295026 (EPE 3) Suppression Pool High Water Temperature / 5	Tier #	1	------
EK2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of the relationship between Suppression Pool High Water Temperature and the following systems or components:	K/A #	295026EK2.01	
• RHR/LPCI	Importance Rating	3.8	------

Proposed Question: **# 28**

Unit 1 is operating at 100% RTP when an event occurred resulting in heat added to the Suppression Pool with the following conditions:

- Suppression Pool Temperature is 90 °F and rising
- The NUSO has directed Suppression Pool Cooling to be placed in service

In accordance with 1-OI-74, Residual Heat Removal System, it is preferable to operate **(1)** while in Suppression Pool Cooling.

In accordance with 1-EOI-2, Primary Containment Control, when Suppression Pool Temperature **CANNOT** be maintained below 95 °F, **(2)** Suppression Pool Cooling is required to be placed in service.

A. (1) 1 RHR Pump in each loop
(2) **ONLY** 1 loop of

B. (1) 1 RHR Pump in each loop
(2) **ALL** available

C. (1) 2 RHR Pumps in one loop
(2) **ONLY** 1 loop of

D. (1) 2 RHR Pumps in one loop
(2) **ALL** available

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that if 1 RHR pump in each loop is being operated to support Suppression Pool Cooling, then 2 RHR heat exchangers would be in use. Second part is incorrect but plausible in that 1-EOI-2, Primary Containment Control, (SP/T-1) states to monitor and control Suppression Pool Temperature below 95 °F using available Suppression Pool Cooling which could lead the candidate to select only 1 loop is required to be placed in service.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See attached)* In accordance with 1-OI-74, RHR System, use two RHR pumps per loop when needed to maximize Suppression Pool Cooling in order to minimize total time spent in Suppression Pool Cooling. Two loops of RHR in Suppression Pool Cooling should be minimized.
For second part, in accordance with 1-EOI-2, Primary Containment Control, (SP/T-3) when Suppression Pool Temperature cannot be maintained below 95 °F, operate all available Suppression Pool Cooling using RHR pumps not required for continuous injection.

RO Level Justification: Tests the candidate's knowledge of the relationship between Suppression Pool High Water Temperature and the RHR System. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 1-OI-74, Rev.122 (Attach if not previously provided)
1-EOI-2, Rev. 9

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203 Obj. 13 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.044-10 018 #1766
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

1766. OPL171.044-10 018

Unit 1 entered 1-AOI-1-1, Relief Valve Stuck Open, due to a partially open Main Steam Relief Valve.

The crew is placing Suppression Pool Cooling in service in accordance with 1-OI-74, Residual Heat Removal System.

Which one of the following completes both statements below?

In accordance with 1-OI-74, it is preferable to operate **(1)** while in Suppression Pool Cooling.

In accordance with 1-EOI-2, when Suppression Pool Temperature exceeds **(2)**, all available Suppression Pool Cooling is required to be operated.

A. (1) 2 RHR Pumps in one loop
(2) 95 °F

B. (1) 2 RHR Pumps in one loop
(2) 105 °F

C. (1) 1 RHR Pump in each loop
(2) 95 °F

D. (1) 1 RHR Pump in each loop
(2) 105 °F

Excerpt from 1-OI-74:

BFN Unit 1	Residual Heat Removal System	1-OI-74 Rev. 0122 Page 19 of 449

3.5 Suppression Pool (continued)

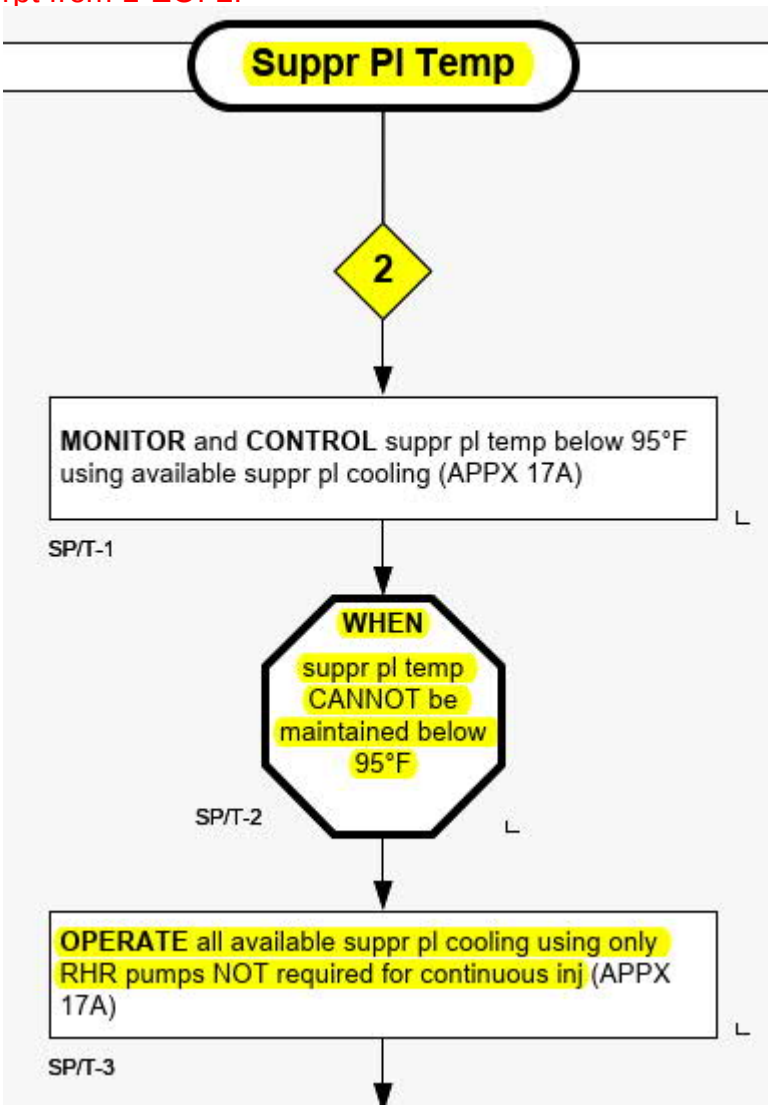
D. [NRC/C] PSA concerns with RHR in Suppression Pool Cooling Mode with a LOCA and a LOSP identify that severe water hammer may occur during the pump restart. Therefore, the following guidelines should be used to try and maintain the system below the PSA Risk Assessment goals:

1. RHR in Suppression Pool Cooling should be minimized.
2. Two Loops of RHR in Suppression Pool Cooling should be minimized.
3. Use two pumps per loop if needed to maximize Suppression Pool Cooling in order to minimize total time spent in Suppression Pool Cooling. [NRC IN 87-10]
4. Suppression Pool Cooling run times are tracked in 1-SR-2 to ensure risk assessment goals are not exceeded.

E. EDC 70724 incorporates valves 2-FCV-074-0001,0012, 0024 and 0035 into the GL 89-10 program. This EDC also requires that RHR the torus suction valves are not to be opened with pressure against the valve seats. The minimum flow valve 2-FCV-074-0007 and 0030 will be open prior to opening valves 2-FCV-074-0001,0012, 0024 and 0035 which will equalize pressure across the valves.

F. EDC 70723 incorporates valves 1-FCV-074-0001,0012, 0024 and 0035 into the GL 89-10 program. This EDC also requires that RHR the torus suction valves are not to be opened with pressure against the valve seats. The minimum flow valve 1-FCV-074-0007 and 0030 will be open prior to opening valves 1-FCV-074-0001,0012, 0024 and 0035 which will equalize pressure across the valves.

Excerpt from 1-EOI-2:



Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

206000 (SF2, SF4 RHR/HPCIS) High-Pressure Coolant Injection
K4.01(10CFR 55.41.7)
Knowledge of High-Pressure Coolant Injection System design features and/or interlocks that provide the following:

- Turbine trips

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	206000K4.01	
Importance Rating	3.8	------

Proposed Question: **# 29**

During Emergency Depressurization on Unit 3, High Pressure Coolant Injection (HPCI) will automatically isolate when Reactor Pressure reaches ___**(1)**___ which ___**(2)**___ result in a HPCI turbine trip signal.

A. (1) 110 psig
(2) will

B. (1) 110 psig
(2) will **NOT**

C. (1) 230 psig
(2) will

D. (1) 230 psig
(2) will **NOT**

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 3-OI-73, HPCI System, the HPCI System automatically isolates when Reactor Pressure lowers to 110 psig. For second part, in accordance with 3-OI-73, the HPCI Turbine automatically trips on any PCIS Group 4 Isolation signal. HPCI TURBINE TRIPPED (3-9-3F, WINDOW 11) alarms when HPCI trips and/or isolates.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible in that other steam driven pumps such as the Reactor Feedwater Pump Turbines (RFPTs) will trip, but they do not have isolations in the sense of PCIS Group Isolations. RFPTs will lose steam supply from a PCIS Group I Isolation when the MSIVs close.

C INCORRECT: First part is incorrect in that Recirc Pump discharge valves automatically close if Reactor Pressure lowers to less than 230 psig with a LPCI initiation signal present. Second part is correct *(See A)*.

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is incorrect but plausible *(See B)*.

RO Level Justification: Tests the candidate's knowledge of the HPCI System design features and/or interlocks as it relates to HPCI Turbine trips. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to HPCI Turbine trips.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): 3-OI-3, Rev. 115 (Attach if not previously provided)
3-OI-68, Rev. 106
3-OI-73, Rev. 66
3-ARP-9-3F, Rev. 38

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.042 Obj. 3b-c (As available)

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

Question History:
New **X**
Last NRC Exam

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpt from 3-OI-73:

BFN Unit 3	High Pressure Coolant Injection System	3-OI-73 Rev. 0066 Page 9 of 98

3.0 PRECAUTIONS AND LIMITATIONS

A. The HPCI turbine automatically trips on any of the following:

1. RPV water level high at +51 inches
2. Low pump suction pressure at 19.3" HG Vacuum (4.7 sec time delay)
3. Turbine high exhaust pressure at 140 psig
4. Any isolation signal
5. Remote Manual HPCI TURBINE TRIP pushbutton, 3-HS-73-18A

B. HPCI turbine overspeed at 122% (~5000 rpm) of rated speed (~4100 rpm) results in a hydraulic trip. The hydraulic trip occurs when operating oil is ported from the HPCI TURBINE STOP VALVE, 3-FCV-073-0018, causing the stop valve to close under spring force. Once the stop valve is closed, the piston of the hydraulic trip resets. With the HPCI turbine under load, the field-adjusted reset should occur between 2500 and 3000 rpm, and the startup sequence should commence. Since the overspeed trip condition does not result in any automatic trip signals in the HPCI control circuit, the HPCI PUMP MIN FLOW VALVE, 3-FCV-73-0030 does not close as a direct result of the turbine overspeed.

C. The HPCI System automatically isolates on any of the following: (Refer to 3-AOI-64-2b, Group 4 HPCI Isolation.)

1. High steamline flow at 85 psid (~200% of rated) (3 sec time delay)
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room
3. Low RPV pressure at 110 psig (does not seal-in)
4. High pressure between rupture diaphragms at 10 psig
5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 3-HS-73-61, if automatic initiation signal is present

D. HPCI System automatically initiates from one-out-of-two-taken twice logic from either: (**REFER TO** Section 5.1).

1. Low RPV water level at -45" (3-LIS-3-58A through D), OR
2. High drywell pressure at 2.45 psig (3-PS-64-58A through D).

Excerpt from 3-ARP-9-3F:

BFN Unit 3	Panel 9-3 3-XA-55-3F	3-ARP-9-3F Rev. 0038 Page 16 of 42

HPCI TURBINE TRIPPED

3-ZA-73-18

11

(Page 1 of 1)

Sensor/Trip Point:

ZS-73-18	Valve full closed ZS-73-18 made up
23A-K31	3-FCV-073-0018 shut AND
23A-K18	3-FCV-073-0016 NOT fully shut (15 second time delay)

Sensor Location:

ZS-73-18	Relay 23A-K31
HPCI Room	Panel 9-39
Rx Bldg, El 519'	Aux Instr Rm, El 593'

Probable Cause:

A. HPCI turb trip and/or isolation.
B. Manual trip.
C. Sensor malfunction.

Automatic Action:

A. HPCI turbine trip (without isolation).
   1. HPCI TURBINE STOP VALVE, 3-FCV-73-18, closes.
   2. HPCI PUMP MIN FLOW VALVE, 3-FCV-73-30, closes.

Operator Action:

A. **DETERMINE** cause of trip.

B. **IF** valid alarm, **THEN**
REFER TO 3-OI-73.

C. **REFER TO** Tech Spec 3.5.1 and 3.5.2.

References:

3-45E620-1 3-47E610-73-1 3-730E928 -3 and 4
Technical Specifications 3.5.1 and 3.5.2

Excerpt from 3-OI-3: (Supports Distractors B2, D2)

BFN Unit 3	Reactor Feedwater System	3-OI-3 Rev. 0115 Page 329 of 331

Attachment 11
(Page 2 of 2)

Reactor Vessel Levels

LEVEL	SETPOINT (in inches)	AUTOMATIC ACTIONS
Level 8	+55 +51	• Main Turbine and RFPT trip • HPCI and RCIC turbine trip
Level 7	+39	High Water Level Alarm (REACTOR WATER LEVEL ABNORMAL annunciation 3-9-5A window 8)
Level 6	< +39	Top end of programmed level control range
Level 5	> +27	Low end of programmed level control range
Level 4	+27	Low Water Level Alarm (REACTOR WATER LEVEL ABNORMAL annunciation 3-9-5A window 8)
Level 3	+2	• Reactor Scram • PCIS Groups 2,3,6,8 isolations • SGT trains auto start • CREV System initiates • SD Cooling suction valves close • ADS confirmatory low level • EOI-1 entry condition
Level 2	-45	• HPCI initiates • RCIC initiates • ATWS/ARI/RPT trip
Level 1	-122	• RHR(LPCI) Pumps auto start (injection valves open <450 psig) • Core Spray Pumps auto start (injection valves open <450 psig) • Group I isolation (MSIVs, 1-55, 1-56 auto close) • ADS initiation setpoint (also contingent on logic timers, drywell pressure, RHR/Core Spray pumps running) • CASA/PASA (D/G auto start) initiation.
Level 0	-215	Containment Spray permissive interlock (2/3 core height)

Excerpt from 3-OI-68: (Supports Distractors C1, D1)

BFN Unit 3	Reactor Recirculation System	3-OI-68 Rev. 0106 Page 26 of 213

3.6 VFD's (continued)

8. Protective trips associated with the VFD (excessive drive losses, input one cycle trip, input under and over voltage, input phase unbalance, etc.) are still functional, but will only trip the VFD and NOT the input breaker. Medium Voltage will still be applied to the VFD and is to be removed by manually tripping the 4kV Recirc VFD supply breaker. Failure to remove Medium Voltage can damage the VFD due to overheating.

9. With a failed RECIRC DRIVE I/O MODULE (WAGO) module, the input breaker will only trip under the following conditions:

   a. Loss of coolant flow

   OR

   b. Transformer high temperature.

10. Vents and drains (both RCW and VFD cooling water) have been added to each heat exchanger to allow on line maintenance.

3.7 Valves

A. The following valve limitations should be observed:

1. Recirc Pump discharge valve should not be opened with more than 200 psid across the valve to prevent potential valve or actuator damage.

2. Only when the jogging circuit is functioning during a pump start will the pump operate with the discharge valve less than 90% open.

3. Recirc Pump discharge valves auto close if reactor pressure drops to less than 230 psig with a LPCI initiation signal present.

B. The Recirc Pump suction valve motors 3-MTR-068-0001 and 0077 have the following motor start limitations:

1. May have up to ten consecutive starts within a one hour period followed by a required three hour cooldown period.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295021 (APE 21) Loss of Shutdown Cooling / 4	Tier #	1	------
AK3.04 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses or actions as they apply to Loss of Shutdown Cooling:	K/A #	295021AK3.04	
• Maximizing Reactor Water Cleanup flow	Importance Rating	3.5	------

Proposed Question: **# 30**

Unit 1 is in MODE 4 for a Refueling Outage when the following conditions occur:

- Shutdown Cooling is lost
- 1-AOI-74-1, Loss of Shutdown Cooling, has been entered

The reason 1-AOI-74-1 directs maximizing Reactor Water Cleanup System (RWCU) blowdown flowrate is to **(1)**.

In accordance with 1-OI-69, RWCU System, with 2 pumps in operation, maximum system flow is limited to **(2)**.

A. (1) establish natural circulation
(2) 100 gpm

B. (1) establish natural circulation
(2) 200 gpm

C. (1) maintain Reactor Coolant Temperature < 200 °F
(2) 100 gpm

D. (1) maintain Reactor Coolant Temperature < 200 °F
(2) 200 gpm

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with 1-AOI-74-1, one of the alternate methods of decay heat removal is ambient losses with natural or forced circulation. Second part is incorrect but plausible in that 100 gpm is the maximum flow limit for one RWCU Pump operation.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See A*).

D **CORRECT**: (*See attached*) In accordance 1-AOI-74-1, Loss of Shutdown Cooling, RWCU flow rate is required to be raised to maximum and RWCU blowdown (1-OI-69) as required to maintain Reactor Coolant Temperatures < 200 °F on all indications. For second part, in accordance with 1-OI-69, Reactor Water Cleanup System, any time the Unit is in MODE 4 with two RWCU Pumps in operation, then maximum flow is limited to 100 gpm per pump (200 gpm total).

RO Level Justification: Tests the candidate's knowledge of the reasons for maximizing Reactor Water Cleanup Flow as it relates to a Loss of Shutdown Cooling. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 1-AOI-74-11, Rev.10 (Attach if not previously provided)
1-OI-69, Rev. 87

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.044 Obj. 11d (As available)

Question Source:

Bank #	BFN 1006 AUDIT #49
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

9. 295021 AK3.04 /NEW/H

Unit 1 is in Mode 4. A loss of shutdown cooling has occurred.

The crew has placed RWCU in service in accordance with 1-AOI-74-1, Loss of Shutdown Cooling, with two demineralizers in service.

Which ONE of the following completes the statements below?

The reason why 1-AOI-74-1 directs maximizing RWCU blowdown flow is to ____.

The maximum allowed flow with two-pump operation in accordance with 1-OI-69, RWCU System is ____.

A. maintain reactor coolant temperature < 200 °F;
320 gpm

B. establish natural circulation;
320 gpm

C. maintain reactor coolant temperature < 200 °F;
200 gpm

D. establish natural circulation;
200 gpm

CORRECT ANSWER C

Excerpts from 1-AOI-74-1:

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

NOTES

1) Performance of this instruction , in conjunction with a Station Blackout (0-AOI-57-1A), will also require implementation of 1-AOI-30B-1,(Reactor Building Ventilation Failure)
2) The following systems, if available, may be used as alternate methods of decay heat removal. **REFER TO** the applicable Tec Spec Bases B 3.4.7, B 3.4.8, B 3.9.7, B 3.9.8
   - ADHR System- (0-OI-72) (attempt to place in service prior to Supplemental Fuel Pool Cooling)
   - Fuel Pool Cooling System- (1-OI-78)
   - RWCU System- (1-OI-69)
   - Ambient losses with natural or forced circulation
   - Compensatory Decay Heat Removal During Refuel Operation using Condensate Transfer Pumps through the Core Spray Injection Line, 1-POI-78-2.

CAUTIONS

1) Reactor vessel stratification may occur until Shutdown Cooling is restored or a Reactor Recirculation Pump is placed in service.
2) Loss of Shutdown Cooling during the first 24 hours is most critical due to massive decay heat and limitations on the RHRSW piping. If Shutdown Cooling is lost during the first 24 hours post reactor shutdown, priorities shall be placed on the recovery of shutdown cooling in an expeditious manner [BFN PER 02-003140-000].

[1] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). (Otherwise **N/A**)

[2] **NOTIFY** the Unit SRO.

[3] **IF** Refueling is in progress, **THEN**

NOTIFY the Refueling Floor SRO. (Otherwise **N/A**)

4.2 Subsequent Actions (continued)

[12.9] **IF** the tripped pump has been determined to be in operating condition and with Unit SRO permission, **THEN:** (Otherwise N/A)

RESTART RHR PUMP 1A(1C)(1B)(1D) using 1-HS-74-5A(16A)(28A)(39A).

[12.10] **THROTTLE** RHR SYS I(II) LPCI OUTBD INJECT VALVE, 1-FCV-74-52(66), to establish and maintain RHR flow as indicated by 1-FI-74-50(64), RHR SYS I(II) FLOW, as follows:

RHR Pumps in Operation	1	2
Loop Flow	7,000 to 10,000 gpm	14,000 to 20,000 gpm
Loop Flow (1 or more fuel bundles removed from core)	6,000 to 6,500 gpm	N/A

[12.11] **WHEN** time permits after RHR pump is started, **THEN**

CHECK RHR Pump Breaker charging spring recharged by observing amber breaker spring charged light is on and closing spring target indicates charged.

[12.12] **ENSURE** inservice RHRSW pump for the appropriate header. **REFER TO** 0-OI-23.

[12.13] **SLOWLY THROTTLE** RHR HX 1A(1C)(1B)(1D) RHRSW OUTLET VLV, 1-FCV-23-34(40)(46)(52), to obtain desired cooldown rate.

[13] **PERFORM** the following as required:

[13.1] **RAISE** RWCU flow rate to maximum AND **MAXIMIZE** RWCU blowdown in accordance with 1-OI-69 as required to maintain reactor coolant temperatures less than 200°F on all indications.

[13.2] **RAISE** CRD flow in accordance with CRD Pump Operation at Elevated Flow section of 1-OI-85.

Excerpt from 1-OI-69:

BFN Unit 1	Reactor Water Cleanup System	1-OI-69 Rev. 0087 Page 14 of 155

3.6 Pumps (continued)

D. The Reactor Water Cleanup Pumps may be operated with the Reactor Building Closed Cooling Water System out of service provided the following conditions are met:

1. Reactor water temperature is ≤ 120°F and the reactor is de-fueled.
2. Reactor Water Cleanup Pump room temperature is ≤ 95°F and logged once per shift in the Narrative Log.
3. Reactor Water Cleanup Pump oil temperature is ≤ 180°F and logged once per shift in the Narrative Log.

E. RWCU is required to be operated within the following restrictions with reactor pressure ≤ 50 psig (Modes 2 or 3) or any time the unit is in Mode 4, Mode 5, or defueled:

1. One pump in operation - pump can be operated to its maximum flow.
2. Two pumps in operation - maximum flow limited to 100 gpm per pump (200 gpm total).

F. If pumps trip during normal operation, they should not be restarted until pump suction temperature is greater than 2°F below saturation temperature. Table 1, 1-SR-3.4.9.3 & 4, Reactor Recirculation Pump Start Limitations, may be used in determining adequate pump suction temperature.

G. When warming a RWCU Pump, maintain as close to 10°F/minute heatup rate as possible to prevent seal damage.

H. When RWCU is being operated during normal or hot shutdown reactor condition, CRD seal purge should be maintained at all times for 1A pump in order to provide seal flushing and cooling to maximize seal life and preserve seal integrity. Condensate backup is provided only for cool down of the RWCU pump when CRD is unavailable. Following a hot shut down of the 1A RWCU pump, seal purge should be maintained during cool down of the pump in order to prevent overheating the seal. Seal purge also provides 1A pump pressurization with cold water following pump isolation; thus preventing depressurization and rapid deterioration of the seal faces due to steam cutting.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
211000 (SF1 SLCS) Standby Liquid Control **G2.2.3** (10CFR 55.41.7) (Multi-unit license) Knowledge of the design, procedural, or operational differences between units	Tier #	2	------
	Group #	1	------
	K/A #	211000G2.2.3	
	Importance Rating	3.8	------

Proposed Question: **# 31**

In accordance with 2-EOI-Appendix-3B, Alternate SLC Injection, the Unit _____ Standby Liquid Control System (SLC) can be lined up to provide SLC Injection to Unit 2.

A. 1 ONLY

B. 2 ONLY

C. 3 ONLY

D. 1 OR 3

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-EOI-Appendix-3B, Alternate SLC Injection, the lineup directs hoses and equipment stored on Unit 2 and Unit 1. Hose connections are made from the Unit 1 SLC storage tank drain and routed to Unit 1's 1B CRD Pump (spare pump for both Unit 1 and Unit 2) suction filter which can be aligned to Unit 2.

B INCORRECT: Incorrect but plausible if that candidate does not recall the specific SLC System design with unique cross Unit capabilities concluding that all operations and equipment are specific from and to Unit 2 only.

C INCORRECT**:** Incorrect but plausible if that candidate does not recall the specific SLC System design with unique cross Unit capabilities concluding that all operations and equipment are specific to Unit 3 only supporting Unit 2 as well. For Unit 3 specifics, hoses and equipment are stored on Unit 3 and Unit 2. Hose connections are made from Unit 2 and routed to Unit 3. However, Unit 1 is also able to cross support Unit 2.

D INCORRECT**:** Incorrect but plausible if that candidate does not recall the specific SLC System design with unique cross Unit capabilities. This would lead them to conclude that cross Unit support can come from either Unit 1 or 3. However, only Unit 1 provides cross Unit support to Unit 2 *(See A and B)*.

RO Level Justification: Tests the candidate's knowledge of the Unit 1, 2 and 3 design, procedural, or operational differences as it relates to the SLC System. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to Unit differences.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): 1-EOI APPENDIX 3B, Rev. 1 (Attach if not previously provided)
2-EOI APPENDIX 3B, Rev. 6
3-EOI APPENDIX 3B, Rev. 5
OPL171.039, Rev. 24

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.039 Obj. 5a (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

New	**X**
Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Excerpt from 1-EOI APPENDIX-3B:

BFN UNIT 1	ALTERNATE SLC INJECTION	1-EOI APPENDIX-3B Rev. 1 Page 1 of 9

LOCATION: Units 1 and 2 Reactor Building

ATTACHMENTS:
1. Tools and Equipment
2. Unit 1 Reactor Building and Hose Routing
3. Unit 2 Reactor Building and Hose Routing (√)

1. **NOTIFY** Unit 1 and Unit 2 Operators and **CONTINUE** in this procedure. _____

2. **VERIFY** CRD Pump 1B is available for use. _____

NOTE

Steps 3 through 6 may be performed concurrently with Steps 7 through 12.

3. **REFER TO** Attachment 1 and **REMOVE** 1½-in. hose from EOI Equipment Storage Boxes on Unit 1 El 565 ft, and Unit 2 El 621 ft. _____

4. **REFER TO** Attachment 3 and **ROUTE** three sections of hose from Unit 2 El 621 ft EOI box, from Unit 2 SLC Tank Drain (El 621 ft NE) across south side of El 621 ft Unit 2, down U-1/U-2 stairwell to El 565 ft. _____

5. **REFER TO** Attachment 2, **ROUTE** three sections of hose from Unit 1 El 565ft. EOI box, from U-1/U-2 stairwell, north across Unit 1 RB east side and down stairwell to Unit 1 CRD pumps on El 541 ft. _____

6. **COUPLE** ALL sections of the hose and **TIGHTEN** couplings. _____

7. **ISOLATE** CRD Pump 1B as follows:

   a. **RACK OUT** 1-BKR-085-0002, CONTROL ROD DRIVE HYDRAULIC FEED PUMP 1B (4KV Shutdown Board A, Compartment 13). _____

   b. **NOTIFY** Unit 1 Operator to close 1-FCV-85-8, CRD PUMP 1B DISCH TO U1. _____

   c. **NOTIFY** Unit 2 Operator to close 2-FCV-85-8, CRD PUMP 1B DISCH TO U2. _____

Excerpt from 2-EOI APPENDIX-3B:

BFN UNIT 2	ALTERNATE SLC INJECTION	2-EOI APPENDIX-3B Rev. 6 Page 1 of 9

LOCATION: Units 1 and 2 Reactor Building

ATTACHMENTS:
1. Tools and Equipment
2. Unit 1 Reactor Building and Hose Routing
3. Unit 1 Reactor Building and Hose Routing (√)

1. **NOTIFY** Unit 1 and Unit 2 Operators and **CONTINUE** in this procedure. _____

2. **VERIFY** CRD pump 1B is shut down but available for use on Unit 2. _____

NOTE

Steps 3 through 6 may be performed concurrently with Steps 7 through 12.

3. **REFER TO** Attachment 1 and **REMOVE** 1½-in. red rubber hose from EOI Equipment Storage Boxes on El 565 ft and El 621 ft. _____

4. **REFER TO** Attachment 3 and **ROUTE** BOTH sections of hose in El 621 ft box from Unit 1 SLC Tank Drain (El 621 ft NW) across West side of El 621 ft Unit 1, down SW stairwell to El 565 ft. _____

5. **REFER TO** Attachment 2, **OBTAIN** ALL three sections of hose from the El 565 ft EOI box, and **ROUTE** from SW stairwell across Unit 1 RB South, around to NE corner, and down stairwell to Unit 1 CRD pumps on El 541 ft. _____

6. **COUPLE** ALL sections of the hose and **TIGHTEN** couplings. _____

7. **ISOLATE** CRD Pump 1B as follows:

   a. **RACK OUT** 1-BKR-085-0002, CONTROL ROD DRIVE HYDRAULIC FEED PUMP 1B (4KV Shutdown Board A, Compartment 13). _____

   b. **NOTIFY** Unit 1 Operator to close 1-FCV-85-8, CRD PUMP 1B DISCH TO U1. _____

   c. **NOTIFY** Unit 2 Operator to close 2-FCV-85-8, CRD PUMP 1B DISCH TO U2. _____

Excerpt from 3-EOI APPENDIX-3B:

BFN UNIT 2	ALTERNATE SLC INJECTION	2-EOI APPENDIX-3B Rev. 6 Page 1 of 9

LOCATION: Units 1 and 2 Reactor Building

ATTACHMENTS:
1. Tools and Equipment
2. Unit 1 Reactor Building and Hose Routing
3. Unit 1 Reactor Building and Hose Routing (√)

1. **NOTIFY** Unit 1 and Unit 2 Operators and **CONTINUE** in this procedure. _____

2. **VERIFY** CRD pump 1B is shut down but available for use on Unit 2. _____

NOTE

Steps 3 through 6 may be performed concurrently with Steps 7 through 12.

3. **REFER TO** Attachment 1 and **REMOVE** 1½-in. red rubber hose from EOI Equipment Storage Boxes on El 565 ft and El 621 ft. _____

4. **REFER TO** Attachment 3 and **ROUTE** BOTH sections of hose in El 621 ft box from Unit 1 SLC Tank Drain (El 621 ft NW) across West side of El 621 ft Unit 1, down SW stairwell to El 565 ft. _____

5. **REFER TO** Attachment 2, **OBTAIN** ALL three sections of hose from the El 565 ft EOI box, and **ROUTE** from SW stairwell across Unit 1 RB South, around to NE corner, and down stairwell to Unit 1 CRD pumps on El 541 ft. _____

6. **COUPLE** ALL sections of the hose and **TIGHTEN** couplings. _____

7. **ISOLATE** CRD Pump 1B as follows:

   a. **RACK OUT** 1-BKR-085-0002, CONTROL ROD DRIVE HYDRAULIC FEED PUMP 1B (4KV Shutdown Board A, Compartment 13). _____

   b. **NOTIFY** Unit 1 Operator to close 1-FCV-85-8, CRD PUMP 1B DISCH TO U1. _____

   c. **NOTIFY** Unit 2 Operator to close 2-FCV-85-8, CRD PUMP 1B DISCH TO U2. _____

Excerpt from OPL171.039 Lesson Plan:

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

B. **Injection into the RPV under ATWS Conditions with SLC System Failure**

1. If SLC Injection is required, and the Unit's SLC System has a demand failure (i.e., SQUIB VALVES fail to fire, SLC PUMPS will not start), then an alternate path is available to inject boron. This evolution is governed by EOI Appendix 3B, "Alternate SLC Injection." Another Unit's SLC STORAGE TANK drain line is aligned, via temporary hose connections, to an operable Control Rod Drive (CRD) Hydraulic System Pump's suction filter on the respective, or other, Unit. The hose is connected to a prefabricated strainer cover, which requires installation. Hoses and equipment are stored in staged EOI Equipment Storage Boxes on the Units.

NLO Obj. 6b / d / e, 7

NLOR Obj. 7a / b / c / e / f / g

ILT Obj. 5a, 6, 9, 10b / d / e, 12a / b / c / e / f / g / h

LOR Obj. 5a, 6, 9, 10b / d / e, 12a / b / c / e / f / g / h

The CRD Pump is then utilized to inject boron into the Reactor via its normal path.

Unit 1 specifics: 1-EOI Appendix 3B, "Alternate SLC Injection," directs this lineup. Hoses and equipment are stored on Unit 1 Reactor Building EL 565 ft and Unit 2 EL 621 ft. Hose connections are made from Unit 2's SLC STORAGE TANK drain and routed to Unit 1's CRD PUMP 1B suction filter.

HU TOOL: Procedure Use and Adherence

Unit 2 specifics: 2-EOI Appendix 3B, "Alternate SLC Injection," directs this lineup. Hoses and equipment are stored on Unit 2 Reactor Building EL 565 ft and Unit 1 EL 621 ft. Hose connections are made from Unit 1's SLC STORAGE TANK drain and routed to Unit 1's CRD PUMP 1B suction filter; which can be aligned to Unit 2.



Unit 3 specifics: 3-EOI Appendix 3B, "Alternate SLC Injection," directs this lineup. Hoses and equipment are stored on Unit 3 Reactor Building EL 565 ft and Unit 2 EL 621 ft. Hose connections are made from Unit 2's SLC STORAGE TANK drain and routed to Unit 3's CRD PUMP 3B suction filter.

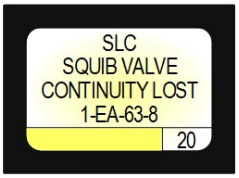
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
211000 (SF1 SLCS) Standby Liquid Control	Tier #	2	------
K5.03 (10CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Standby Liquid Control System:	K/A #	211000K5.03	
• Shutdown margin	Importance Rating	3.7	------

Proposed Question: **# 32**

Unit 1 Anticipated Transient Without SCRAM (ATWS) conditions exist with the following conditions:

- 1-EOI-1A, ATWS RPV Control, is in progress
- SLC SQUIB VALVE CONTINUITY LOST (1-9-5B, Window 20) alarms





Squib Valve ___**(1)**___ has fired.

The time to inject Hot Shutdown Boron Weight (HSBW) is ___**(2)**___ compared to the time with both Squib Valves open.

A. (1) A
(2) longer

B. (1) A
(2) the same

C. (1) B
(2) longer

D. (1) B
(2) the same

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible if the candidate concludes that one squib valve not firing open provides less boron flowrate to the Reactor.

B **CORRECT**: *(See attached)* In accordance 1-EOI Appendix-3A, SLC Injection, once either SLC Pump is started, both squib valves are supposed to fire. By design, the SLC System has a continuity circuit that provides squib valve status. The blue 1-ZI-63-8A(B), SQUIB VALVE A(B) CONTINUITY, indicating lights on Control Room Panel 1-9-5 are lit when SLC is in normal shutdown standby alignment (squib valves have not fired). Once either SLC Pump is started from the Control Room, both respective squib valves fire indicated by the extinguished blue indication light. Checking the blue indicating lights on Panel 1-9-5 is also directed in the given Alarm Response Procedure for SLC SQUIB VALVE CONTINUITY LOST (1-9-5B, Window 20). Therefore, for the question above, only Squib Valve 'A' has fired. For second part, the boron solution flows from the SLC storage tank to the combined pump (either A or B) suction lines into the selected pump. The pump discharges into the common header that splits into two parallel lines. The boron solution then passes through the fired open squib valve(s) (1 or 2) through the Drywell penetration into the Reactor. With each SLC Pump being a positive-displacement pump, they are designed to inject boron solution into the Reactor in 50 – 125 minutes at approximately 50 gpm. With each squib valve being 100% capacity valve, the time to inject HSBW is the same compared to the time with both squib valves open.

C INCORRECT: First part is incorrect but plausible if the candidate fails to recall how the SLC System continuity monitoring circuitry works and concludes that if the blue light is lit, then the squib valve must have fired. Second part is incorrect but plausible *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C).* Second part is correct *(See B).*

RO Level Justification: Tests the candidate's knowledge of the operational implications or cause and effect relationships of the Standby Liquid Control System as it relates to Shutdown Margin. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

Technical Reference(s): 1-ARP-9-5B, Rev. 25 (Attach if not previously provided)
OPL171.039, Rev. 24
1-EOI Appendix-3A, Rev. 0

Proposed references to be provided to applicants during examination: **SLC SQUIB VALVE CONTINUITY LOST (1-9-5B, Window 20), Panel 1-9-5B, 1-ZI-63-8A(B), SQUIB VALVE A(B) CONTINUITY, indicating lights**

Learning Objective: OPL171.039 Obj. 5c (As available)

Form 4.2-1	Written Examination Question Worksheet		
Question Source:	Bank #		
	Modified Bank #	BFN 2104 #3	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

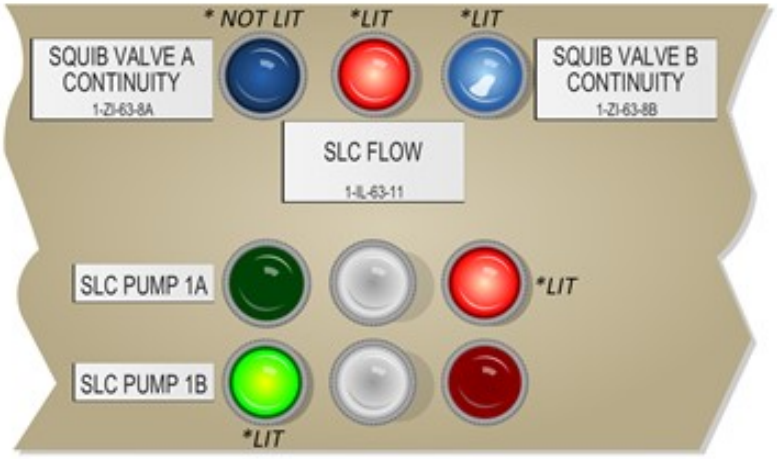
Copy of Bank Question:



Unit 1 Anticipated Transient Without a SCRAM (ATWS) conditions exist with the following conditions:

- 1-EOI-1A, ATWS RPV Control is in progress
- Reactor Pressure is 950 psig
- 1-EOI Appendix-3A, SLC INJECTION in progress as indicated
- SLC SQUIB VALVE CONTINUITY LOST, (1-9-5B, WINDOW 20) alarms





Given the conditions above, which **ONE** of the following completes the statement below regarding the status of the Standby Liquid Control (SLC) System?

SQUIB VALVE ___**(1)**___ has fired and SLC ___**(2)**___ injecting to the Reactor.

A. (1) A
(2) is

B. (1) A
(2) is NOT

C. (1) B
(2) is

D. (1) B
(2) is NOT

Proposed Answer: **A**

Excerpt from 1-ARP-9-5B:

BFN Unit 1	Panel 9-5 1-XA-55-5B	1-ARP-9-5B Rev. 0025 Page 23 of 42

SLC
SQUIB VALVE
CONTINUITY LOST
1-EA-63-8

20

(Page 1 of 1)

Sensor/Trip Point:

1-XM-63-8A and 8B	fall 3.0 ma rise 7.0 ma

Sensor Location: Behind Panel 1-9-5
Main Control Room

Probable Cause:
A. Cleared fuse.
B. Loss of power supply.
C. Blown Photohelic Bulb.
D. Sensor malfunction.
E. SLC pump start from the control room.
F. Corrosion buildup inside electrical connector

Automatic Action: None

Operator Action:
A. **IF** SLC has been initiated, **THEN**
REFER TO 1-EOI-1 or 1-AOI-79-2.
B. **IF** SLC has **NOT** been initiated, **THEN**
PERFORM the following:
1. **CHECK** blue indicating lights on Panel 1-9-5 to determine which valve ignition circuit failed.
2. **CHECK** sensor and amp meter in back of Panel 1-9-5.
3. **DISPATCH** personnel to the SLC tank, RB El. 639', to investigate squib valve wiring connection.

C. **REFER TO** Tech Spec 3.1.7.

References: 1-47E610-63-1 1-45E620-6-2 1-729E854-1

Excerpt from 1-EOI APPENDIX-3A:

BFN UNIT 1	SLC INJECTION	1-EOI APPENDIX-3A Rev. 0 Page 1 of 2

LOCATION:	Unit 1 Control Room
ATTACHMENTS:	None (√)

1. **UNLOCK** and **PLACE** 1-HS-63-6A, SLC PUMP 1A/1B, control switch in START-A or START-B position. _____

2. **CHECK** SLC System for injection by observing the following:

   - Selected pump starts, as indicated by red light illuminated above pump control switch. _____
   - Squib valves fire, as indicated by SQUIB VALVE A and B CONTINUITY blue lights extinguished, _____
   - SLC SQUIB VALVE CONTINUITY LOST 1-EA-63-8 Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 20). _____
   - 1-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. _____
   - System flow, as indicated by 1-IL-63-11, SLC FLOW, red light illuminated on Panel 1-9-5, _____
   - SLC INJECTION FLOW TO REACTOR 1-FA-63-11, Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 14). _____

3. IF ..... Proper system operation CANNOT be verified,

   THEN..... **RETURN** to Step 1 and **START** other SLC pump. _____

4. **VERIFY** RWCU isolation by observing the following:

   - RWCU Pumps 1A and 1B tripped _____
   - 1-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed _____
   - 1-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed _____
   - 1-FCV-69-12, RWCU RETURN ISOLATION VALVE closed. _____

5. **VERIFY** ADS inhibited. _____

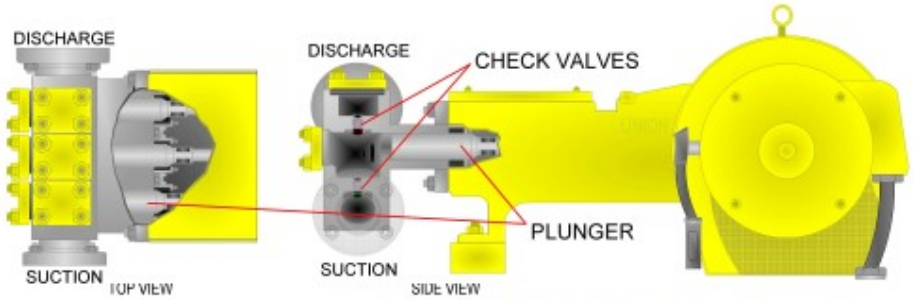
6. **MONITOR** reactor power for downward trend. _____

Excerpts from OPL171.039 Lesson Plan:

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

There are two 100% capacity, triplex, positive-displacement, piston SLC Pumps (A / B) installed in parallel. Each positive-displacement pump is sized to inject the solution into the Reactor in 50 to 125 minutes (approximately 50 gpm), depending on the amount of solution in the tank, from zero psig up to the Reactor Vessel's maximum operating pressure. To provide margin, the SLC Pumps and system have a design discharge pressure of 1500 psig. The pumps are required, by BFN Tech Specs, to be able to pump ≥39 gpm at a discharge pressure of ≥1325 psig . Either pump is capable of supplying a system flow of approximately 50 gpm at a system pressure of 1150 psig. Each pump contains internal suction and discharge check valves, which open at approximately 5 psid, allowing only forward flow through an idle pump.

Electrical power is provided to SLC Pump A by its



respective Unit's 480V Shutdown Board A. SLC Pump B receives power from its Unit's 480V Shutdown Board B. The motors have installed space heaters that will remain on as long as the motor is off. Once the motor is energized, the heater will de-energize via 'B' contacts off of the motor-start relay. Motor heaters receive power from 480V Reactor MOV Boards A & B (SLC Pump A / B). (Obj. 10f)

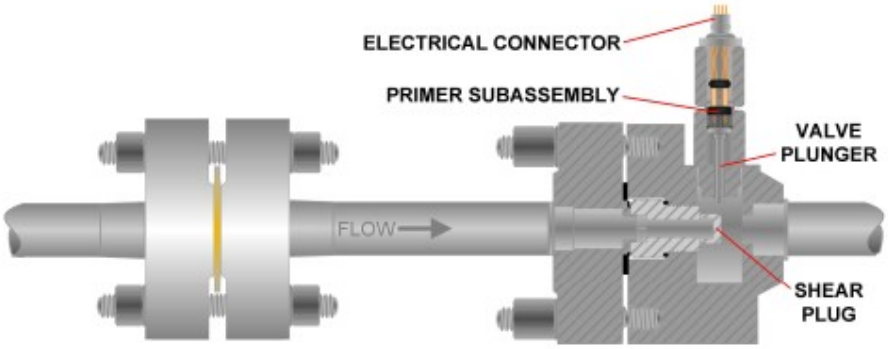
Instructor Note:

Revisit Mechanical Fundamentals of Pump Design and Operation

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

5. Explosive-actuated Squib Valves

a. Each Squib Valve is a 100% capacity, dual-squib (2 primers), shear-plug, designed for zero-leakage. The zero leakage design ensures boron will not leak into the reactor when the pumps are being tested. (Objective 4a, OF5 Sound knowledge of fundamentals) Two firing squibs are installed in each valve for high reliability in that either squib firing will shear open the valve. Power for each squib firing circuit is via their respective 250V DC Control Power on the 480V Shutdown Boards A and B.



When a significant magnitude of current (>2 amps) is supplied to the squib circuitry, bridge wiring embedded in the explosive primer becomes energized causing the primer to explode. Ignition creates a shock wave that is directed at the valve plunger, driving it downward and into forceful contact with the inlet chamber plug (sometimes referred to as a shear plug). The plunger breaks off the cap of the inlet chamber plug, circumscribed with a deep groove so the end will easily shear. Both the plunger and cap lodge in the space provided at the bottom of the valve body, out of the system flow path. Once initiated, the entire firing sequence occurs in approximately 2 milliseconds. All explosive by-products are retained in the explosive chamber.

To service the valve after firing, a 6-inch length of pipe (spool piece) immediately upstream of the valve must be removed to gain access to the shear plug.

NLO Obj. 3e / f, 4a, 5b

NLOR Obj. 2e / f, 4a

ILT Obj. 3e / f, 4a, 5d, 9

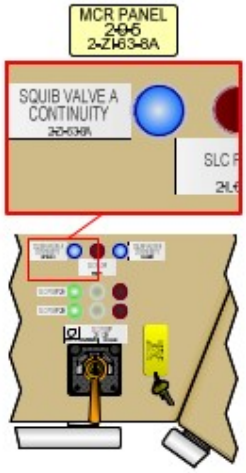
LOR Obj. 5d, 9

OF5 SOLID UNDERSTANDING OF PLANT DESIGN AND SYSTEM INTER-RELATIONSHIPS

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

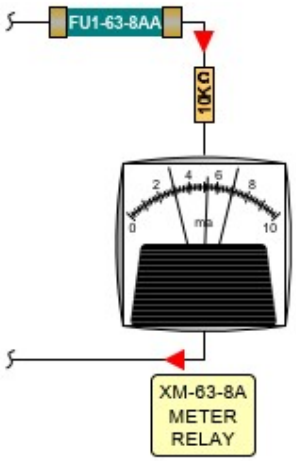
6. SLC Vave Control and Isolation Logic

a. A loss of circuit continuity through the bridge wire would render the respective valve incapable of being fired. Therefore, monitoring continuity is important to ensure proper operation of the system. When the system is in the normal shutdown, standby alignment, a small electric current is passed through a continuity monitor to ensure electrical continuity of the bridge wiring within the explosive charge.



A series of current-limiting resistors in the circuit hold the magnitude of continuity current below that necessary for squib ignition. This continuity current, maintained within the specified band of 3–7 milliamps (ma), enables, via normally closed contacts, a blue SQUIB VALVE A(B) CONTINUITY indicating light (ZI-63-8A and 8B) on Control Room Panel 9-5 to remain lit for each valve. The power supply for both blue lights is 120V AC I&C A.

In addition, indication is available (in ma) on the A(B) SLC SQUIB VALVE CONTINUITY METER (XM-63-8A and 8B) in the back of Control Room Panel 9-5. The power supply for these Analog Optical Meter Relays is via their respective 120V AC I&C A Bus; and they operate on the photohelic principle to control between a high and low setpoint.



NLO Obj. 3e / f, 4d, 5b, 6e

NLOR Obj. 2e / f, 4b, 7a / g

ILT Obj. 3e / f, 4d, 5d, 10e, 12a / g

LOR Obj. 5d, 10e, 12a / g

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

When current changes, reaching either setpoint, the infrared light to the limiting phototransistor is cut off by the helix-driven light shutter. The resulting phototransistor signal is electronically amplified to actuate its slave relay and switching occurs.

Thus, a failure or loss of continuity through either primer circuit, as sensed by a low current of 3 ma (lowering) or a high current of 7 ma (rising), will be indicated by an extinguished blue SQUIB VALVE A (B) CONTINUITY indicating light and Annunciator SLC SQUIB VALVE CONTINUITY LOSS, (9-5B Window 20). These indications also occur when either SLC PUMP is started from the Control Room.



7. SLC Pump and Squib Valve Circuitry

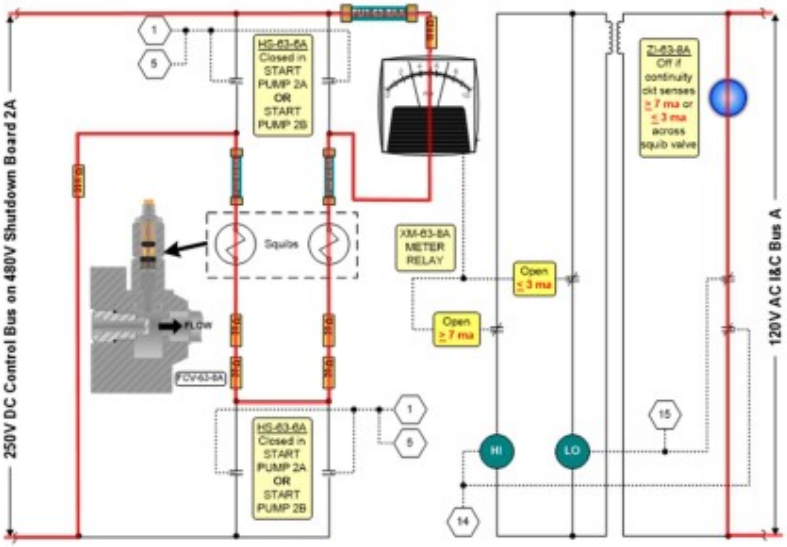
a. With the SLC system in standby, continuity current is passed within the circuit through a 10K Ω current-limiting resistor, the continuity meter/relay, through one of the squib primers, through 20 Ω current-limiting resistors, back through the second squib primer, and then through the last 39K Ω current-limiting resistor. As stated previously, closed contacts enable the blue continuity light to remain lit.

NLO Obj. 3e / f, 4d, 5b, 6e

NLOR Obj. 2e / f, 4b, 7a / g

ILT Obj. 3e / f, 4d, 5d, 9, 10e, 12a / g

LOR Obj. 5d, 9, 10e, 12a / g



OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

b. When starting a SLC Pump locally, a STOP - START spring-return-to-normal switch is utilized after placing the test switch in "TEST." The START position completes the circuit with the starting relay, thereby shutting the starting contacts and energizing the motor. Contacts labeled 1, 2, 3, and 4 above remain open; and therefore the only current supplied to the Squib Valves is that associated with the continuity circuit. As a result, the pump is allowed to start, but the squibs are prevented from igniting.

In the event that system operation is required, the Control Room operator must place key-locked SLC PUMP A/B switch (HS-63-6A on Panel 9-5) to the START PUMP A or START PUMP B position. This switch, in either START PUMP A / B position, acts to close contacts in the Squib Valve Firing Circuit . As a result, the 10K Ω and 39K Ω current-limiting resistors (in each pump circuit) are bypassed. With the current limiting resistors bypassed, sufficient current is provided to BOTH Squib Valves to enable them to ignite and sodium pentaborate is delivered to the reactor. The pumps will continue to run until secured by the Control Room operator.

NLO Obj. 3e / f, 4b, 5c

NLOR Obj. 2e / f, 3a, 4b

ILT Obj. 3e / f, 4b, 5d, 9

LOR Obj. 5d, 9

HU TOOL:

Pre-Job Brief

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

e. SLC INJECTION FLOW TO REACTOR, (9-5B Window 14) Annunciator activates on Panel 9-5.

Reactor Water Cleanup System INBOARD AND OUTBOARD SUCTION ISOLATION VALVES (FCV-69-1 and 2) and RETURN ISOLATION VALVE (FCV-69-12) automatically close to prevent removal of the poison solution from the reactor vessel. The RWCU PUMPS also trip on interlock.

2. The boron solution flows from the SLC STORAGE TANK to the combined pump suction lines and into the selected SLC PUMP. The pump discharges into the common discharge header that splits into two parallel lines. The solution then passes through the SQUIB VALVES, through the Drywell penetration, and into the REACTOR VESSEL lower plenum via the SLC INJECTION SPARGER.

NLO Obj. 4a
NLOR Obj. 6a
ILT Obj. 4a, 9, 11a
LOR Obj. 9, 11a

At this point, the operator should monitor reactor power for a downward trend and SLC TANK level is lowering at approximately 1% per minute. Depending upon the injection rate, it is expected that the SLC System will take 12-30 minutes to shut down the reactor (achieve Hot / Cold Shutdown Boron Weights). Once initiated, the SLC System will remain in service until the SLC STORAGE TANK level drops to 0%, or securing is directed by the Unit Supervisor (SRO).

Anytime the SQUIB VALVES have been fired and the Reactor is at atmospheric pressure, it is possible to create a gravity-drain path from the SLC STORAGE TANK to the RPV. Therefore, the system should be manually isolated as soon as practical when it is no longer required. Once the pump is secured, this is accomplished by unlocking and closing the manually-operated SLC DRYWELL OUTBOARD ISOLATION VALVE (ISV-63-524).

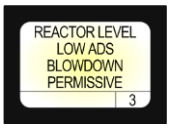
Form 4.2-1 **Written Examination Question Worksheet**

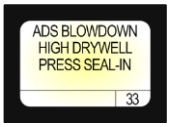
Examination Outline Cross-reference:	Level	RO	SRO
218000 (SF3 ADS) Automatic Depressurization **G2.2.44** (10CFR 55.41.5) Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions	Tier #	2	------
	Group #	1	------
	K/A #	218000G2.2.44	
	Importance Rating	4.2	------

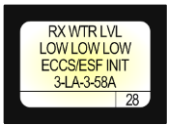
Proposed Question: **# 33**

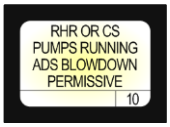
Unit 3 has SCRAMMED due to a LOCA with the following conditions:

- REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 3) alarms
- ADS BLOWDOWN HIGH DRYWELL PRESSURE SEAL-IN (3-9-3C, Window 33) alarms
- REACTOR WATER LOW-LOW-LOW ECCS/ESF INITIATE (3-9-3C, Window 28) alarms
- RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 10) alarms









Given the conditions above, in accordance with the given Alarm Response Procedures, the Automatic Depressurization System (ADS) initiation logic for High Drywell Pressure is bypassed when Window **(1)** is in alarm for a **MINIMUM** of **(2)**.

A. (1) 10
(2) 95 seconds

B. (1) 10
(2) 265 seconds

C. (1) 28
(2) 95 seconds

D. (1) 28
(2) 265 seconds

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 10), alarms when one RHR (or Core Spray) Pump is running with greater than or equal to 100 (185) psig discharge pressure. Second part is incorrect but plausible in that 95 seconds is a timer in the ADS blowdown logic string, however the timer does not bypass High Drywell Pressure. This 95 second allows time for high pressure injection and HPCI systems to recover Reactor Water Level to possibly prevent ADS from automatically blowdown the Reactor without Operator action.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible (*See A*).

D **CORRECT:** (*See attached*) In accordance with REACTOR WATER LOW-LOW-LOW ECCS/ESF INITIATE (3-9-3C, Window 28), when Reactor Water Level lowers to (-) 122 inches, automatic start signals are generated for Core Spray, RHR LPCI Mode, EDGs and RHRSW (EECW) Pumps and the ADS blowdown logic input signal is received. When Reactor Water Level is lowering to this point, it is an indication that high pressure injection it likely not adequate. In turn, once (-) 122 inches is reached, the low pressure ECCS Pumps automatically start in preparation for automatic Reactor Vessel blowdown to allow re-injection. For second part, in accordance with 3-OI-1, Main Steam System, ADS automatically initiates when all logic conditions are met. The given ADS BLOWDOWN HIGH DRYWELL PRESSURE SEAL-IN (3-9-3C, Window 33), alarms when Drywell Pressure is greater than or equal to 2.45 psig. This Drywell Pressure of 2.45 psig is an ADS blowdown logic input which seals in along the logic string to energize the 95 second timer. This 95 second allows time for high pressure injection and HPCI systems to recover Reactor Water Level to possibly prevent ADS from automatically blowdown the Reactor without Operator action. However, in the event that High Drywell Pressure existed for a minimum of 265 seconds and ADS hadn't automatically blowdown, a separate timer contact would allow the bypassing of the (+) 2.45 Drywell Pressure contact in the ADS logic string.

RO Level Justification: Tests the candidate's 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 as it relates to the Automatic Depressurization System (ADS). This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution related to the ADS logic.

Technical Reference(s): 3-OI-1, Rev. 48 (Attach if not previously provided)

3-ARP-9-3C, Rev. 34

2-730E929-2, Rev. 24

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 3)**
ADS BLOWDOWN HIGH DRYWELL PRESSURE SEAL-IN (3-9-3C, Window 33)
REACTOR WATER LOW-LOW-LOW ECCS/ESF INITIATE (3-9-3C, Window 28)
RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 10)

Learning Objective: OPL171.043 Obj. 3 (As available)

Question Source:

	Bank #	
	Modified Bank #	BFN NRC 2104 #24
	New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2021

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

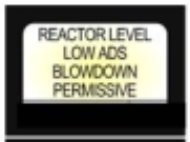
10 CFR Part 55 Content:

55.41	**X**
55.43	

Copy of Bank Question:

Proposed Question: **# 24**

Unit 3 is operating at 100% RTP, when a LOCA occurs with the following conditions:









- REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE (3-9-3C, Window 3) alarms
- DRYWELL PRESSURE APPROACHING SCRAM (3-9-3B, Window 30) alarms
- Manual Reactor SCRAM was inserted
- ADS BLOWDOWN HIGH DRYWELL PRESSURE SEAL-IN (3-9-3C, Window 33) alarms
- REACTOR WATER LOW-LOW-LOW ECCS/ESF INITIATE (3-9-3C, Window 28) alarms
- RHR Pumps are **NOT** available
- Core Sprays Pumps failed to **AUTOMATICALLY** start

Given the conditions above, which **ONE** of the following completes the statement below?

In accordance with the given Alarm Response Procedures, the Automatic Depressurization System (ADS) will automatically initiate, when the Operator has manually started ___**(1)**___ Core Spray Pumps and a **MINIMUM** of ___**(2)**___ has elapsed.

A. (1) 3B **AND** 3C
   (2) 95 seconds

B. (1) 3B **AND** 3C
   (2) 265 seconds

C. (1) 3A **AND** 3B
   (2) 95 seconds

D. (1) 3A **AND** 3B
   (2) 265 seconds

Excerpt from 3-OI-1:

3.4 Main Steam Relief Valve (MSRV / ADS) (continued)

F. ADS will initiate when ALL of the following conditions are met:

1. A confirmatory low reactor water level signal (+2.0 inches), REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE, 3-9-3C Window 3,

2. Two coincident signals for each of the following parameters:

   a. high drywell pressure (+2.45 psig) in conjunction with low-low-low reactor water level (-122 inches), ADS BLOWDOWN HIGH DRYWELL PRESS SEAL-IN, 3-9-3C Window 33 and RX WTR LVL LOW LOW LOW ECCS/ESF INIT 3-LA-3-58A, 3-XA-55-9-3C Window 28

   OR

   b. ow-low-low reactor water level (-122 inches), RX WTR LVL LOW LOW LOW ECCS/ESF INIT 3-LA-3-58A, 3-XA-55-9-3C Window 28, for 265 seconds (High drywell pressure bypass)

   c. When the above logic is satisfied, the 95 second timer starts (ADS BLOWDOWN TIMERS INITIATED, 3-XA-55-9-3C, Window 11).

3. ADS 95 second timer timed out.

4. One RHR pump OR two Core Spray pumps (A or B and C or D) running, RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 3-XA-55-9--3C Window 10.

G. Depressing 3-XS-1-159 and -161 on Panel 3-9-3 resets the ADS Blowdown Timers. They also reset the ADS initiation, if the timers have timed out. ADS will re-initiate upon subsequent timing out of the timer provided the low level and pump logic signals still exist. The timer setpoint is 95 seconds; however setpoint tolerance allows it to be as low 77 seconds.

H. ADS LOGIC A (B) INHIBIT Switches 3-XS-1-159A (161A) will prevent the actuation of the ADS timers or will also reset an ADS initiation, if the timers have timed out when placed in INHIBIT.

1. ADS Logic Bus Inhibited alarms are received on two alarm windows. Logic Bus A annunciates on 3-XA-55-9-3C, Window 18, and Logic Bus B annunciates on 3-XA-55-9-3C, Window 31 when the associated Inhibit switch is placed in INHIBIT.

2. These switches are not to be manipulated unless required to do so per EOIs or required by testing in accordance with approved testing procedures.

Excerpts from 3-ARP-9-3C:

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0034 Page 41 of 43

ADS BLOWDOWN
HIGH DRYWELL
PRESS SEAL-IN

33

(Page 1 of 1)

Sensor/Trip Point:

PIS-64-57A
PIS-64-57B
PIS-64-57C
PIS-64-57D

Drywell Press ≥ 2.45 psig

Sensor Location:	PIS-64-57 A and C Panel 9-82 Aux Inst Rm El 593'	PIS-64-57 B and D Panel 9-81 Aux Inst Rm El 593'

Probable Cause:

A. Drywell press ≥ 2.45 psig.
B. SI/SR in progress.
C. Sensor malfunction.

Automatic Action: ADS high Drywell press signal seals in.

Operator Action:

A. **CHECK** Drywell pressure by multiple indications.
B. **IF** alarm is valid, **THEN**
ENTER 3-EOI-1 Flowchart and 3-EOI-2 Flowchart.
C. **IF** alarm is **NOT** valid, **THEN**
RESET high Drywell pressure seal-ins, 3-XS-1-158 and -160, on Panel 3-9-3.
D. if alarm will **NOT** reset, **THEN**
DISPATCH personnel to Aux Instrument Room to investigate the following:
1. Panel 3-9-30 Relays 2E-K2 or 2E-K3 energized.

NOTE
Indicates sensor malfunction or loss of power supply.

2. Panel 3-9-33 Relays 2E-K13 or 2E-K14 energized.

E. **REFER TO** Tech Spec 3.5.1.

References: 3-45E620-2 3-47E610-64-1 47W600-57A and -58 3-730E929 -1 and -2
Technical Specification 3.5.1

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0034 Page 6 of 43

REACTOR
LEVEL LOW
ADS BLOWDOWN
PERMISSIVE

3

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-184	RPV level
LIS-3-185	≤ +2.0 inches

Sensor Location:	LIS-3-184 Panel 9-81 Aux Inst. Rm	LIS-3-185 Panel 9-82 Aux Inst. Rm

Probable Cause:

A. SI/SR is in progress.
B. Low reactor water level (Level 3).
C. Sensor malfunction.

Automatic Action: None

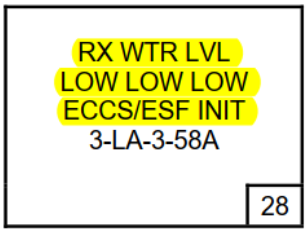
Operator Action:

A. **CHECK** Reactor water level by multiple indications.
B. **DISPATCH** personnel to Aux Instrument Rm El 593, to check relays energized:
   1. Panel 9-30, Relay 2E-K29
   2. Panel 9-33, Relay 2E-K24

C. **REFER TO** Tech Spec Section 3.3.5.1 and 3.5.1.
D. **EVALUATE** equipment associated with this alarm that is out of service to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References: 3-47E610-3-1 3-45E620-2 47W600-57 and 58 3-730E929 -1 and 2
Technical Specifications 3.5.1 and 3.3.5.1
Technical Specifications Bases 3.3.5.1

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0034 Page 35 of 43



(Page 1 of 1)

Sensor/Trip Point:

LIS-3-58A, B, C and D ≤ -122 inches (RPV low-low-low level)(Level 1)

Sensor Location: Panel 9-81, 82 Aux Instrument Room

Probable Cause:
A. Reactor Water Low Level
B. SI/SR in progress.

Automatic Action: (One out-of-two taken twice logic)

A. The following receive auto start signals:
- Core Spray System
- RHR System LPCI Mode
- Diesel Generators
- RHRSW (EECW) pump

B. ADS blowdown logic input.

Operator Action:
A. **CHECK** RPV water level using multiple indications.
B. **REFER TO** the EOIs.
C. **EVALUATE** equipment associated with this alarm that is out of service to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References: 3-45E620-2, 3-47E610-3-1, 47W600-57 and 58, GE 730E930-13, -14 and 19, 45E670-13 and 19

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0034 Page 14 of 43

RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE

10

(Page 1 of 2)

Sensor/Trip Point:

Core Spray

Pump	Instrument	Panel	Relay	Setpoint
A	PS-75-7	25-1	14A-K27A	≥ 185 psig
B	PS-75-35	25-60	14A-K27B	≥ 185 psig
C	PS-75-16	25-1	14A-K28A	≥ 185 psig
D	PS-75-44	25-60	14A-K28B	≥ 185 psig

RHR Pump	Instrument	Panel	Setpoint
A	PS-74-8A & 8B	25-59	≥ 100 psig
B	PS-74-31A & 31B	25-62	≥ 100 psig
C	PS-74-19A & 19B	25-59	≥ 100 psig
D	PS-74-42A & 42B	25-62	≥ 100 psig

Panel 25-1	Rx Bldg, El 519'	R-16 N-LINE
Panel 25-59	Rx Bldg, El 519'	R-15 T-LINE
Panel 25-60	Rx Bldg, El 519'	R-20 N-LINE
Panel 25-62	Rx Bldg, El 519'	R-21 T-LINE

Probable Cause:

A. One RHR pump running with ≥ 100 psig discharge pressure or one CS pump running with ≥ 185 psig discharge pressure.

NOTE

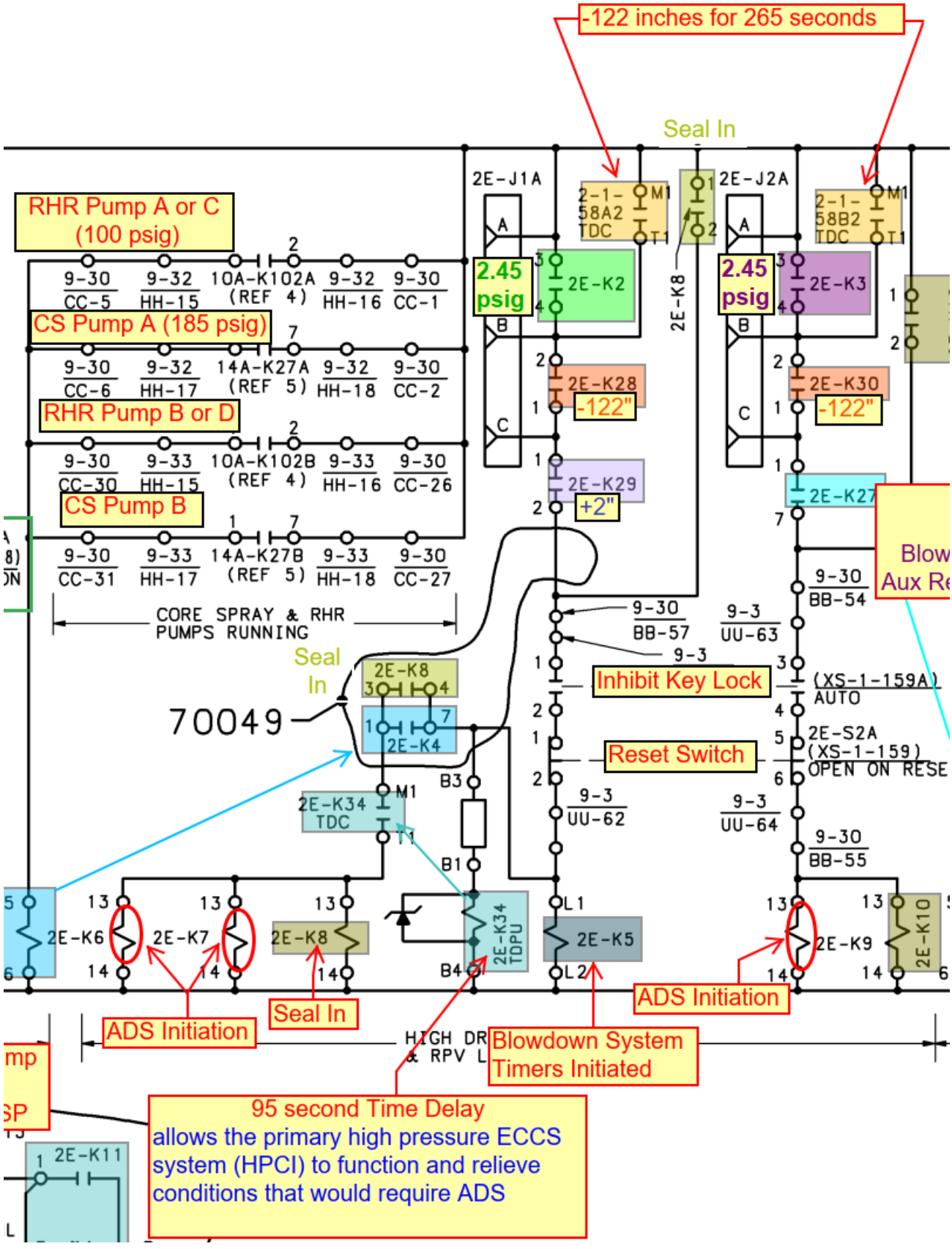
Logic for ADS requires only one RHR pump or two CS pumps, (3A or 3B) and (3C or 3D) to initiate either Logic Bus if all other initiation requirements are met. Either Logic Bus initiation will cause the ADS System to actuate after all relays have timed out.

RHR Pumps Switches	Relays
PS-74-8A and PS-74-19A	Relay 10A-K102A
PS-74-8B and PS-74-19B	Relay 10A-K103A
PS-74-31A and PS-74-42A	Relay 10A-K102B
PS-74-31B and PS-74-42B	Relay 10A-K103B

B. Sensor malfunction.

Continued on Next Page

Excerpt from 2-730E929-2:



Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295030 (EPE 7) Low Suppression Pool Water Level / 5

EA2.03 (10CFR 55.41.10)

Ability to determine and/or interpret the following as they apply to Low Suppression Pool Water Level:

- Reactor pressure

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295030EA2.03	
Importance Rating	4.1	------

Proposed Question: **# 34**

Unit 3 is operating at 100% RTP when Suppression Pool Water Level begins to lower.

In accordance with 3-EOI-2, Primary Containment Control, step SP/L-18, **WHEN** Suppression Pool Water Level **CANNOT** be maintained above **(1)**, **(2)** is required.

A. (1) 11.5 ft
(2) SCRAM

B. (1) 11.5 ft
(2) Emergency RPV Depressurization

C. (1) 12.75 ft
(2) SCRAM

D. (1) 12.75 ft
(2) Emergency RPV Depressurization

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that before Suppression Pool Water Level drops to 11.5 ft, a Reactor SCRAM is required.

B **CORRECT**: *(See attached)* In accordance with 3-EOI-2, Primary Containment Control, when Suppression Pool Water Level cannot be maintained above 11.5 ft then Operator action is required. For second part, when Suppression Pool Water Level cannot be maintained above 11.5 ft, then Emergency RPV Depressurization is required.

C INCORRECT: First part is incorrect but plausible in that if Suppression Pool Water Level cannot be maintained above 12.75 ft then only if adequate core cooling is assured, secure HPCI. Second part is incorrect but plausible *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C).* Second part is correct *(See B).*

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's ability to determine and/or interpret Reactor Pressure as it applies to Low Suppression Pool Water Level. This question is rated as C/A due to the requirement to correctly determine the required plant actions in accordance with the Emergency Operating Instruction Flow Chart steps based on trending parameter analysis and meeting critical threshold decision points, which is not RO level of knowledge.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 3-EOI-2, Rev. 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

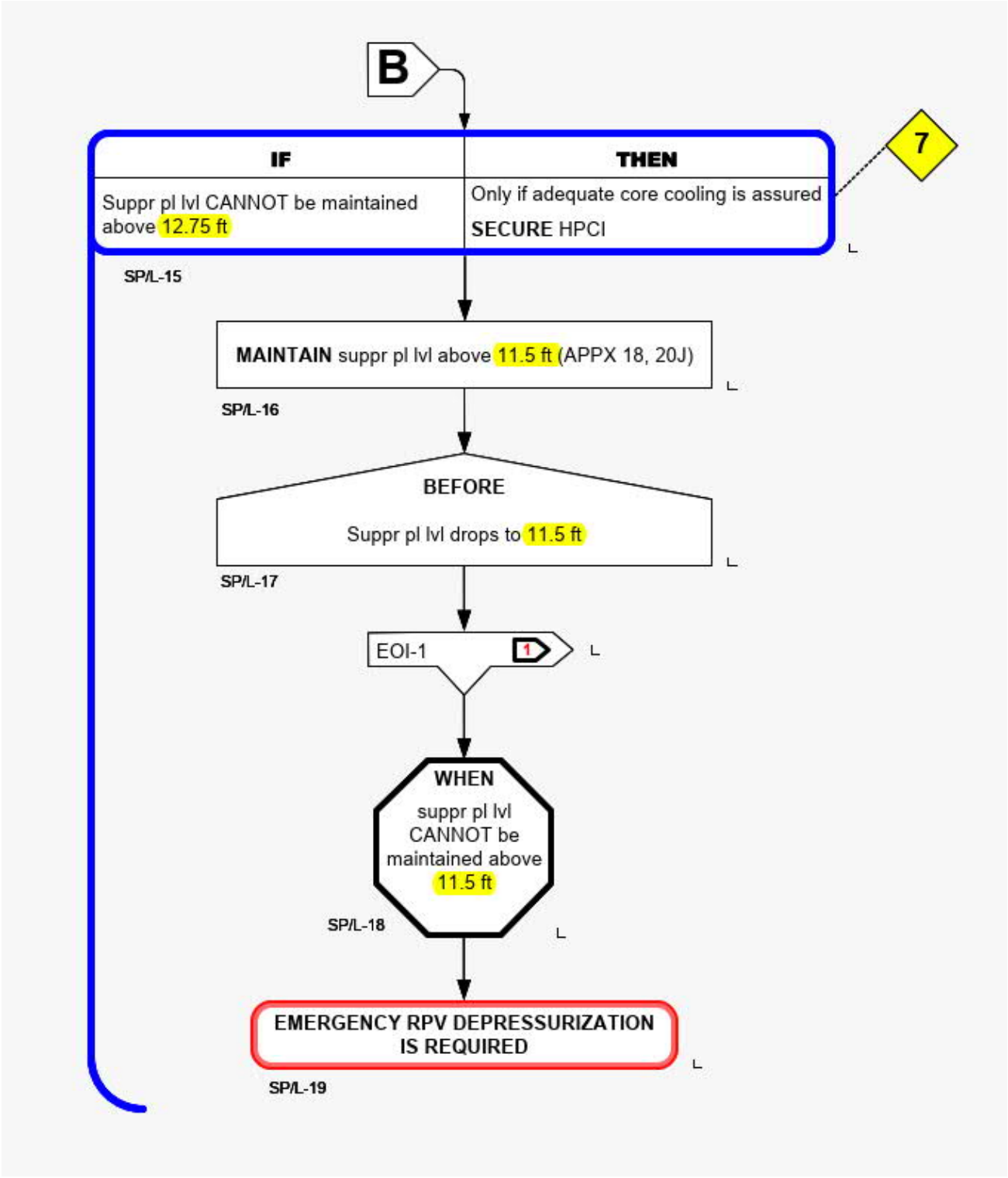
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Excerpt from 3-EOI-2:



Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5	Tier #	1	------
EK3.01 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses or actions as they apply to High Drywell Temperature:	K/A #	295028EK3.01	
• Emergency depressurization	Importance Rating	4.1	------

Proposed Question: **# 35**

Unit 3 is operating at 100% RTP when the following occurs:

- A large break Loss of Coolant Accident (LOCA) in the Drywell
- Drywell Sprays are **NOT** available

In accordance with 3-EOI-2, Primary Containment Control, when Drywell Temperature **CANNOT** be restored and maintained below __(1)__, Emergency Depressurization is required to __(2)__.

A. (1) 280 ºF
(2) ensure the MSRVs can be opened

B. (1) 280 ºF
(2) prevent evaporative cooling in the Drywell

C. (1) 350 ºF
(2) ensure the MSRVs can be opened

D. (1) 350 ºF
(2) prevent evaporative cooling in the Drywell

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 3-EOI-2 requires a Reactor SCRAM before Drywell Temperature rises to 280 ºF. Second part is correct *See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in accordance with EOIPM Section 0-V(G), the Drywell Spray Initiation Limit is the highest Drywell Temperature at which initiation of Drywell Sprays will not result in an evaporative cooling pressure drop to below high Drywell Pressure SCRAM setpoint.

C **CORRECT**: *(See attached)* In accordance with 3-EOI-2, Primary Containment Control, when Drywell Temperature cannot be restored and maintained below 350 ºF then Emergency Depressurization is required. For second part, in accordance with EOIPM Section 0-V(G), Emergency Depressurization is performed to minimize any continuing direct energy release to the Drywell through a Primary System break and ensures that the MSRVs are opened while still OPERABLE.

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's knowledge of the reasons for Emergency Depressurization as it applies to High Drywell Temperature. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 3-EOI-2, Rev. 15 (Attach if not previously provided)
EOIPM Section 0-V(G), Rev. 2

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203, Obj. 5c (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

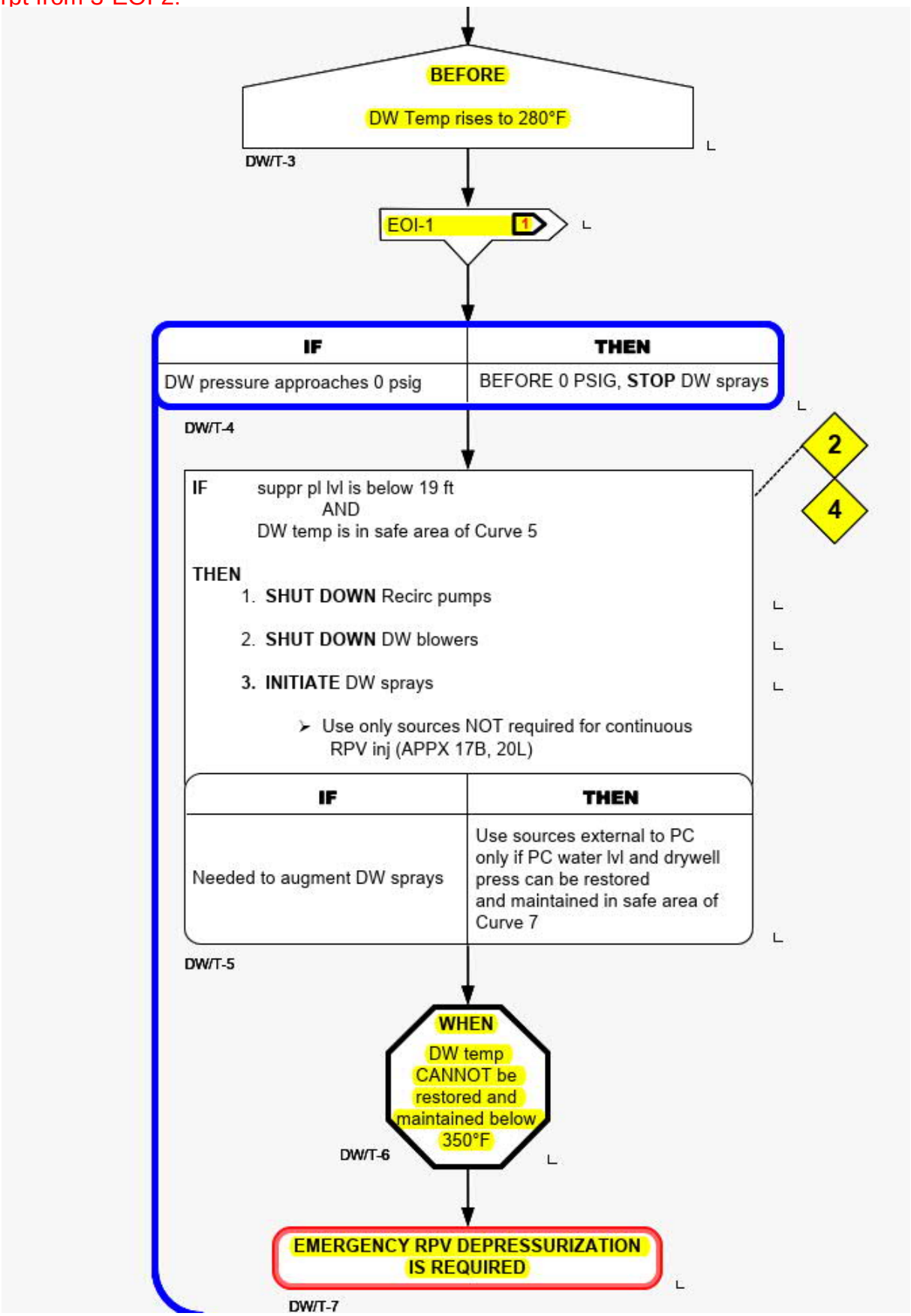
New	X
Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpt from 3-EOI-2:



Excerpts from EOIPM 0-V(G):

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: DW/T-6, DW/T-7

If drywell temperature cannot be restored and maintained below the ADS qualification temperature, emergency RPV depressurization is performed. This action minimizes any continuing direct energy release to the drywell through a primary system break and ensures that the MSRVs are opened while still operable.

Consistent with the definition of "restore," emergency RPV depressurization is not required until it has been determined that drywell sprays (initiated in Step DW/T-5) are ineffective in reducing drywell temperature. It is not expected that MSRV operability will be immediately challenged when the ADS qualification temperature is reached. If drywell temperature is already above the specified value when Steps DW/T-6 and DW/T-7 are reached, drywell sprays may still be used, if available, in preference to emergency depressurization. If sprays are effective in reducing drywell temperature, emergency depressurization need not be performed. Extended operation above the ADS qualification temperature is not permitted, however.

A determination that drywell temperature cannot be restored and maintained below the ADS qualification temperature may be made when, before, or after temperature actually reaches the specified value.

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: Step PC/P-7 (cont'd)

- Drywell temperature must be below the Drywell Spray Initiation Limit (DWSIL, Curve 5). The DWSIL is the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below the high drywell pressure scram setpoint. The final pressure following evaporative cooling is limited to the scram setpoint to ensure that the operator has time to terminate sprays before convective cooling reduces pressure below 0 psig. The scram setpoint is used as a predefined, relatively low action level with sufficient margin to atmospheric pressure.

  The DWSIL is a function of drywell pressure. It is utilized to preclude containment failure or de-inertion following initiation of drywell sprays. Refer to EOIPM Section 0-V(B) for discussion of the DWSIL.

The instructions to shut down recirculation pumps and drywell blowers are performed before initiation of drywell sprays because continuing to operate this equipment in a spray environment is more damaging than shutting it down.

When drywell spray is supplied by pumps capable of injecting into the RPV, drywell spray is permitted only if the pumps to be used do not have to be operated continuously in the injection mode to accomplish the objectives of RPV injection strategies. Restoring and maintaining adequate core cooling takes precedence over initiating drywell spray since catastrophic failure of the primary containment is not expected under the conditions for which spray initiation is required. The wording of this step does, however, permit alternating between RPV injection and drywell spray modes as the need for each occurs, provided adequate core cooling can be maintained.

ECCS initiation logic may preclude drywell spray operation even when adequate core cooling is assured. Since this step requires an evaluation as to whether the spray pumps are needed for core cooling, the automatic logic is unnecessary and may be defeated. EOI Appendix-17B provides instructions for bypassing these interlocks.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295003 (APE 3) Partial or Complete Loss of A.C. Power / 6

AK2.05 (10CFR 55.41.7)

Knowledge of the relationship between Partial or Complete Loss of A.C. Power and the following systems or components:

- Decay heat removal systems

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295003AK2.05	
Importance Rating	4.2	------

Proposed Question: **# 36**

Unit 2 is in MODE 3, with the following plant conditions:

- Loop II of RHR in Shutdown Cooling (SDC) Mode
- RHR SDC Hardening is **NOT** completed
- Diesel Generator (EDG) 'B' is tagged out of service
- 4KV Shutdown Board 'B' trips and locks out due to an electrical fault
- Reactor Water Level lowers to 0 inches

Given the conditions above, 2-FCV-74-47, SHUTDOWN COOLING OUTBOARD SUCTION ISOLATION, will **(1)** and 2-FCV-74-48, SHUTDOWN COOLING INBOARD SUCTION ISOLATION, will **(2)**.

A. (1) close
(2) close

B. (1) close
(2) remain open

C. (1) remain open
(2) close

D. (1) remain open
(2) remain open

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that both the AC and DC Electrical Distribution Systems at BFN are quite complex and easily and often confused. Candidate could believe that 2-FCV-74-48 is also powered from the same DC source as 2-FCV-74-47 and conclude that it still has DC power to close.

B **CORRECT**: (*See attached*) With the given Reactor Water Level lowering to 0 inches, a PCIS Group 2 Isolation occurs for specific Shutdown Cooling valves at (+) 2 inches. This results in both 2-FCV-74-47(48) getting isolation signals to close if they are still energized since they are motor operated valves. For first part, in accordance with 2-OI-74, RHR System, 2-FCV-74-47 is powered by 250VDC RMOV 2A with the NORMAL source of Battery Board 2 (BB-2) (requires manual transfer to alternate from BB-3). Given that DC power is still available to 2-FCV-74-47, it will close from the PCIS Group 2 Isolation signal. For second part, in accordance with PIP-02-03, AC ELECTRICAL DISTRIBUTION SYSTEM-BFN, EDG 'B' supplies 4KV Shutdown Board 'B'. With this AC Board given as de-energized, 480V Shutdown Board 2A is de-energized (requires manual transfer to alternate), therefore 480V RMOV Board 2A is also de-energized. In accordance with 2-OI-74, RHR System, 480V RMOV Board 2A supplies 2-FCV-74-48, therefore it is now de-energized leaving it to remain open.

C INCORRECT: First part is incorrect but plausible in that both the AC and DC Electrical Distribution Systems at BFN are quite complex and easily and often confused. Candidate could believe that 2-FCV-74-47 is also powered from the same AC source as 2-FCV-74-48 and conclude that it is de-energize and will remain open. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the relationship between partial and complete loss of AC Power as it relates to decay heat removal systems, specifically RHR Shutdown Cooling System. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution especially since this question is integrated with numerous systems.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 2-OI-74, Rev. 195 (Attach if not previously provided)

PIP-02-03, Rev. 04062022

OPL171.044, Rev. 23

OPL171.037, Rev. 18U1

OPL171.036, Rev. 26

OPL171.017, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Form 4.2-1 **Written Examination Question Worksheet**

Learning Objective: OPL171.044 Obj. 3b_ (As available)

Question Source:

Bank #	BFN 1102 AUDIT #2
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:

Given the following UNIT 2 plant conditions:

- RHR Loop II is in Shutdown Cooling (SDC) mode.
- D/G 'B' is tagged Out of Service.
- Reactor Pressure is 15 psig.
- Reactor Water Level lowers to (-) 2 inches
- 4kv Shutdown Board B is de-energized

Which ONE of the following describes the effect on the SHUTDOWN COOLING INBD SUCT ISOL, 2-FCV-74-48, **AND** SHUTDOWN COOLING OTBD SUCT ISOL, 2-FCV-74-47?

(ASSUME NO OPERATOR ACTION)

A. 2-FCV-74-47 **AND** 2-FCV-74-48 will shut.

B. 2-FCV-74-47 **AND** 2-FCV-74-48 will **NOT** shut.

C. 2-FCV-74-47 will shut, 2-FCV-74-48 will **NOT** shut.

D. 2-FCV-74-47 will **NOT** shut, 2-FCV-74-48 will shut.

Proposed Answer: **C**

Excerpts from OPL171.017 Lesson Plan: (Illustrates the PCIS Group 2 RHR Shutdown Cooling valves and isolation signals)

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 22

Isolation Signals	Group 2	Valve type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)	Re-Open Manipulation
Level 3; Rx low level: +2" OR Hi Drywell Press +2.45 psig OR Hi Reactor pressure <100 psig. **SDC only.**	RHR shutdown cooling supply (FCV 74-47)	MO Gate	Outside	DC	DC	(9)
	RHR shutdown cooling supply (FCV 74-48)	MO Gate	Inside	AC	AC	(9)
	RHR LPCI to Reactor (FCV 74-53 &67)	MO Gate	Outside	AC	AC	(10)
	Drain pump A Isolation valve (FCV 75-57 & 58)	AO Gate	Outside	Air/AC	Spring	(9)
	Drywell equipment drain discharge Isolation valves (FCV 77-15A &-15B)	AO Gate	Outside	Air/AC	Spring	(9)
	Drywell floor drain discharge Isolation valves (FCV 77-2A &-2B)	AO Gate	Outside	Air/AC	Spring	(9)

ILT- 2g,k,l
LOR- 2g,k,l

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 22

Isolation Signals	Group 8	Valve type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)	Re-Open Manipulation
Hi drywell press +2.45 psig OR Level 3; Rx low level:+2"	TIP guide tubes (5 Total)	SO Ball Valve	Outside	AC	AC	(13)

ILT- 2j,l
LOR- 2j,l

NOTES FOR TABLE 1
These notes are keyed by number to correspond to numbers, in parentheses, in Table 1.

1. Main steam Isolation valves require that both solenoid pilots be deenergized to close valves. Accumulator air pressure plus spring pressure act together to close valves when both pilots are deenergized. Voltage failure at only one pilot does not cause valve closure. The valves are designed to fully close in 3-5 seconds.

   - The basis for 3 seconds is to allow rod insertion (i.e. scram) to complete prior to the resulting pressure transient.
   - The basis for 5 seconds is to minimize inventory loss and resulting release to the environment.

2. These valves only isolate on Rx low, low, low level. (Level 1)

3. AC motor operated valves shall be powered from the ac standby power busses. DC operated Isolation valves are powered from the plant batteries.

4. All motor operated Isolation valves remain in the last position upon failure of valve power. All air operated valves close on motive air failure.

5. Tech Spec -130"

6. Reset on 9-4, Depress Pushbuttons on 9-54, 9-55.

7. These valves do not isolate on high refuel zone radiation.

8. Handswitches in Close, Reset on 9-4, Reopen

9. Reset on 9-4, Reopen

Excerpt from OPL171.044 Lesson Plan:

7. Major Valves and Associated Interlocks — NLO Initial Objective 6

a) Power Supplies

(1) All RHR motor-operated valves are powered from the 480V RMOV Boards, except as noted:

480V RMOV Board	Normal Power	Division	Alternate Power	Division	Valves
'A'	480V S/D Board 'A'	I	480V S/D Board 'B'	II	RHR System I Valves (except as noted).
'B'	480V S/D Board 'B'	II	480V S/D Board 'A'	I	RHR System II Valves (except as noted).
'C'	480V S/D Board 'B'	II	480V S/D Board 'A'	I	none
'D' **	480V S/D Board 'A'	I	480V S/D Board 'B'	II	74-7 (Minimum Flow) 74-53 (Inbd. Inj. Valve)
'E'**	480V S/D Board 'B'	II	480V S/D Board 'A'	I	74-30 (Minimum Flow) 74-67 (Inbd. Inj. Valve)

**Unit 1 does not have RMOV Board 'D' and 'E' (retired in place). — LOR Objective 2, ILT Objective 6

(a) The loads for the 1D 480V RMOV Board are fed from 1A 480V RMOV Board

(b) The loads for the 1E 480V RMOV Board are fed from 1B 480V RMOV Board

(2) 480V RMOV boards 2D, 2E, 3D, and 3E have had their LPCI Motor Generator Sets retired

(a) Loads are powered directly from the 480V Shutdown Boards

(b) There is no longer an auto transfer capability (load transfer must be performed manually).

(3) OUTBOARD SHUTDOWN COOLING SUCTION ISOLATION VALVE FCV-74-47 is powered from 250V RMOV Board 'A'

(a) Valve is normally closed with its associated breaker open except for testing or when RHR is in the Shutdown Cooling mode of operation for NFPA 805 concerns.

(4) FCV-74-108, DRAIN PUMP DISCHARGE ISOLATION VALVE, is powered from 250V RMOV Board 'B'.

b) Generic Valve Interlock Scheme

(1) Assure all RHR Pump flow is directed to the LPCI injection flowpath during ECCS initiation.

(2) Protect low pressure RHR piping from high reactor pressures

(3) Assure that the RHR pumps have a source of water at their suction before starting.

(4) Prevent inadvertent vessel drain-down from occurring.

NLOR Obj 8
ILT Objective 12g
LOR Objective 8g

c) Generic information concerning valves in the LPCI injection paths

(1) Inboard LPCI Injection Valves have a small hole in the valve wedge to prevent pressure locking.

Excerpt from OPL171.037 Lesson Plan:

OPL171.037, DC Systems, Rev: 18U1

250V RMOV POWER SUPPLIES

U1			U2			U3		
250V RMOV BOARD	NORMAL	ALTERNATE	250V RMOV BOARD	NORMAL	ALTERNATE	250V RMOV BOARD	NORMAL	ALTERNATE
1A	BB1	BB2	2A	BB2	BB3	3A	BB3	BB2
1B	BB3	BB1	2B	BB3	BB1	3B	BB1	BB3
1C	BB2	BB1	2C	BB1	BB2	3C	BB2	BB3

0-45E701-1/0-45E702-1/0-45E703-1/0-OI-57-D

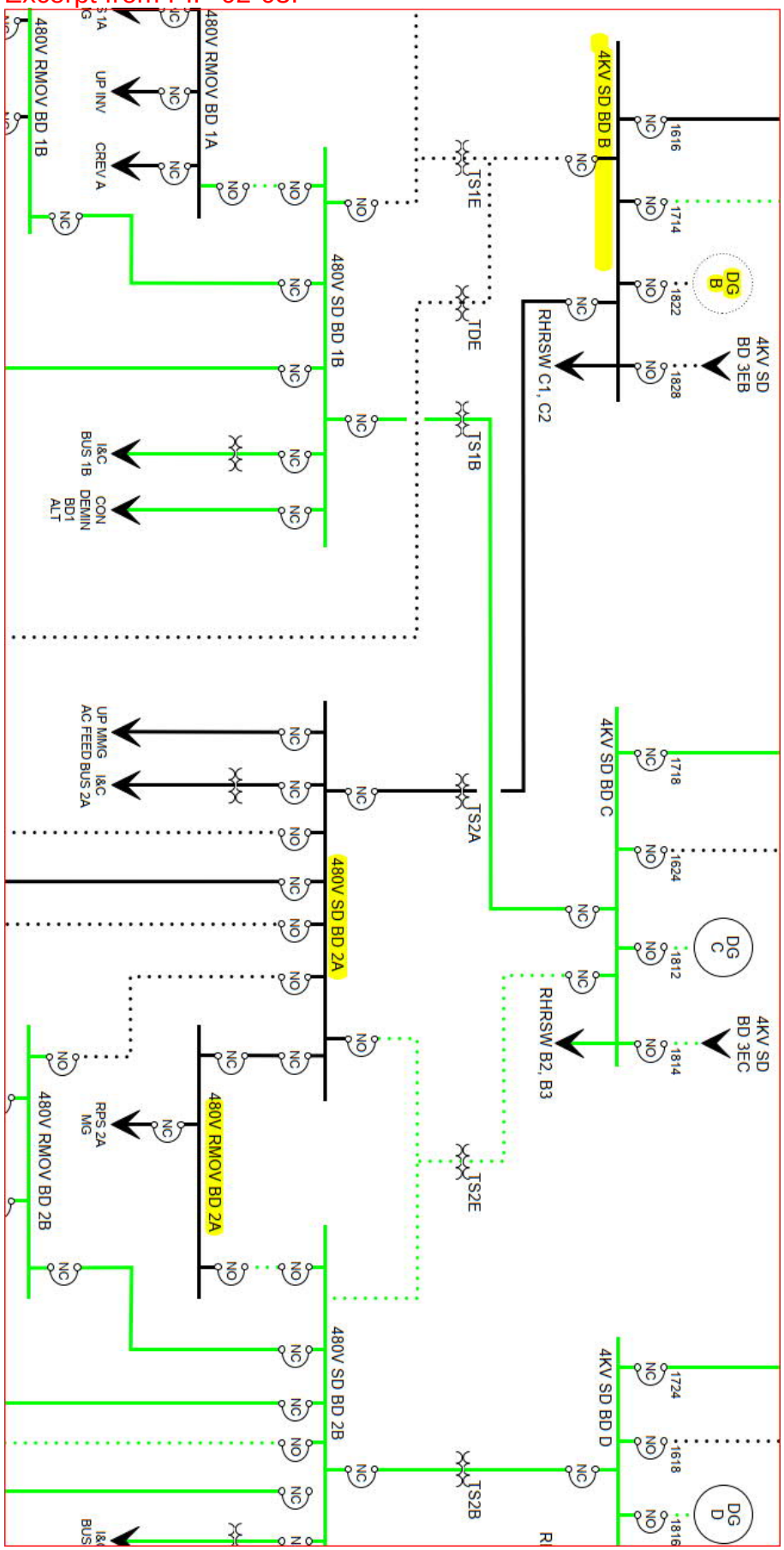
TYPICAL LOADS ON 250V RMOV BOARDS (U1 AND U3 SIMILAR)

250V RMOV 2A	250V RMOV BD 2B	250V RMOV BD 2C
2-FCV-73-34 HPCI DISCH VALVE 2-FCV-73-3 HPCI STEAM LINE ISOL 2-FCV-73-16 HPCI STM SUP VLV 2-FCV-73-40 HPCI PMP CST SUCT VLV 2-FCV-73-26 HPCI SC SUCT VLV 2-FCV-73-27 HPCI SC SUCT VLV 2-FCV-73-30 HPCI MIN FLOW VLV 2-FCV-73-44 HPCI PMP DISCH VLV 2-FCV-73-35 HPCI BYP TO CST VLV HPCI TURB AUX OIL PMP HPCI GLAND SEAL COND BLOWER HPCI GLAND SEAL COND CNDS PMP HPCI SYS LOGIC DIV II PNL 9-39 ECCS DIV II ATU INV CS SYS LOGIC DIV II-2 PNL 9-33 RHR SYS LOGIC II-2 PNL 9-33 RCIC SYS AND ADS LOGIC DIV II-2 PNL 9-33 VARIOUS MSRV POWER 4KV RPT BD 2-II NORM CONTROL PWR BU SCRAM VALVES PNL 9-15 MSIV OUTBD ISLN VLV DIV II PNL 9-43 MSIV OUTBD ISLN VLVS DIV II-2 PNL 25-32 2-FCV-74-47 RHR SDC SUCTION OUTBD 2-FCV-64-222 HARDENED SC OUTBD ISOL VLV RECIRC VFD 2A PNL 25-23	2-FCV-73-36 HPCI SHUTOFF TO CST VLV 2-FCV-71-3 RCIC STM LINE ISOL VLV 2-FCV-71-34 RCIC MIN FLOW VLV RCIC SYS LOGIC DIV I PNL 25-31 ECCS DIV I ATU INV CS SYS LOGIC DIV I PNL 9-32 RHR SYS LOGIC DIV I PNL 9-32 HPCI SYS LOGIC DIV I PNL 9-32 ADS SYS LOGIC BUS A DIV I PNL 9-30 ADS DIV I BUS B PNL 9-30 VARIOUS MSRV POWER MSIV INBD ISLN VLVS DIV I PNL 9-43 MSIV INBD ISLN VLVS DIV I PNL 25-32 2-FCV-69-2 RWCU ISOL VLV U2 HCVS INST TRANSFER SW 2-FCV-64-221 HARDENED SC OUTBD ISOL VLV FCV-74-108 RHR FLUSH DISCH VLV 2-FCV-1-56 MSL WARMING ISOLATION VLV RECIRC VFD 2B PNL 25-24	2-FCV-71-P RCIC TURB TRIP/THROTTLE VLV 2-FCV-71-37 RCIC PUMP DISCH VLV 2-FCV-71-39 RCIC PUMP INJ VLV 2-FCV-71-8 RCIC TURB STM SUP VLV 2-FCV-71-19 RCIC CST SUCT VLV 2-FCV-71-18 RCIC SC SUCT VLV 2-FCV-71-17 RCIC SC SUCT VLV 2-FCV-71-38 RCIC TEST BYP VLV RCIC GLAND SEAL VAC TK COND PMP 2FCV-71-25 RCIC LUBO COOLING WTR VLV RCIC TURB TRIP SOLENOID CKT VARIOUS MSRV POWER

2-45E712-1/-2/-3

Fig- 5 250V RMOV Board Loads

Excerpt from PIP-02-03:



Excerpt from OPL171.036 Lesson Plan:

OPL171.036, AC Power Distribution, Rev# 26

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
3. Major MCC Boards a) Reactor MOV Boards (A through E, Unit 2 & 3), Unit 1 is A thru C b) Reactor Building Vent Boards (A and B, each unit) c) Turbine MOV Boards (A, B, and C, each unit) d) Turbine Building Vent Boards (A and B, each unit) e) Control Bay Vent Boards (A and B) f) Radwaste Boards (1, 2, and 3) g) Condensate Demineralizer Boards (1, 2 and 3) h) Auxiliary Boiler Board (A and B buses)	Unit Difference
4. 480V Reactor MOV Boards Loads on Unit 1 D & E boards were moved to A & B boards. Unit 1 does not use D & E boards. U2 and U3 have D and E boards, but no LPCI MG Sets.	45E749 1/2/3-45E751 1/2/3-45E779
a) Reactor MOV Boards serve the smaller 480V loads that are important to plant safety. Each MOV board has two incoming sources, one from each 480V shutdown board. Reactor MOV Boards A and D feed normally from 480V Shutdown Board A and alternately from 480V Shutdown Board B. The normal supply for Reactor MOV Boards B, C, and E is 480V Shutdown Board B with A being the alternate.	Unit Difference ILT/NLO Obj. 6.g LOR Obj. 1.g
b) Boards 2(3)D and 2(3)E, the "LPCI Valve Boards," normal and alternate power supplies are from the 480V Shutdown Boards. All three Units LPCI MG sets have been removed. Loads that were on U-1 D/E board are now on A & B boards (Unit difference). All 480V RMOV boards have manual transfer of power supplies.	Unit Difference U2 DCN 67409 ILT/NLO Obj. 6.g, 8.g LOR Obj. 1.g, 2.g NLOR Obj. 1.g OF-5
c) The potential of propagating a fault to both divisions is not a concern since the position of the ALTERNATE feeder breakers to all the 480V RMOV Boards at the 480v Shutdown Boards is changed from normally CLOSED to normally OPEN.	
d) Control power is AC. It is provided by transformers connected to the line side of either the incoming power circuits or the individual breakers and contactors. Supply voltage and ground indications are provided. Selected loads have normal/emergency selector switches at the breakers to allow local operation.	ILT Obj. 15.f 1-45E751-1

Excerpts from 2-OI-74:

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0195 Page 501 of 533

Attachment 12
(Page 3 of 5)
Loop II Shutdown Cooling Protected Equipment List

COMPONENT	DESCRIPTION	LOCATION
0-HS-23/27A/1	RHRSW PUMP D2	Panel 1-9-3
0-HS-23/27A/2	RHRSW PUMP D2	Panel 2-9-3
0-HS-23/27A/3	RHRSW PUMP D2	Panel 3-9-3

NOTE

When placing Shutdown Cooling protected equipment information cards / placards on breakers, Tensa Barriers should be used if practical.

COMPONENT	DESCRIPTION	LOCATION
2-BKR-074-0047	RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV FCV-74-47	250VDC RMOV Bd 2A (R1A)
2-HS-74-157	MODE SELECTOR SWITCH, 2-HS-74-157	480V RMOV Bd 2A (5B)
2-BKR-074-0048	RHR SHUTDOWN COOLING SUCTION ISOLATION VLV FCV-74-48	480V RMOV Bd 2A (8C)
2-BKR-074-0066	RHR OUTBOARD VALVE FCV-74-66	480V RMOV Bd 2B (3A)
2-BKR-074-0024	RHR PUMP 2B SUPPR POOL SUCTION VALVE FCV 74-24	480V RMOV Bd 2B (5E)
2-BKR-074-0025	RHR PUMP 2B SHUTDOWN COOLING SUCTION VALVE FCV-74-25	480V RMOV Bd 2B (6C2)
2-BKR-074-0035	RHR PUMP 2D SUPPR POOL SUCTION VALVE FCV-74-35	480V RMOV Bd 2B (6E)
2-BKR-074-0036	RHR PUMP 2D SHUTDOWN COOLING SUCTION VLV FCV-74-36	480V RMOV Bd 2B (7C)
2-BKR-023-0046	RHR HEAT EXCHANGER 2B SERVICE WATER DISCHARGE VALVE (FCV-23-46)	480V RMOV Bd 2B (7E)

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215005 (SF7 PRMS) Average Power Range Monitor / Local Power Range Monitor A4.04 (10CFR 55.41.7)	Tier #	2	------
Ability to manually operate and/or monitor in the control room:	Group #	1	------
• LPRM back panel switches, meters, and indicating lights	K/A #	215005A4.04	
	Importance Rating	3.4	------

Proposed Question: **# 37**

In accordance with 2-OI-92B, Average Power Range Monitoring, Local Power Range Monitors (LPRMs) can only be **MANUALLY** bypassed at Panel ___**(1)**___ using a ___**(2)**___ switch.

A. (1) 2-9-5
(2) pushbutton

B. (1) 2-9-5
(2) keylock

C. (1) 2-9-14
(2) pushbutton

D. (1) 2-9-14
(2) keylock

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that APRMs, IRMs and SRMs all can be manually bypassed on Panel 2-9-5. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that if ARPM/OPRM chassis keylock switch is in the INOP position, an automatic inoperative trip occurs.

C **CORRECT:** (*See attached*) In accordance with 2-OI-92B, APRMs, LPRMs can only be manually bypassed at Panel 2-9-14. For second part, LPRMs can only be manually bypassed at Panel 2-9-14 by depressing the BYPASS/HIGH VOLTS ON or BYPASS/HIGH VOLTS OFF softkey to bypass the selected LPRM.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability to locate and operate and/or monitor LPRM back panel switches, meters, and indicating lights in the control room. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to LPRM operation.

Technical Reference(s): 2-OI-92B, Rev. 44 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Form 4.2-1 **Written Examination Question Worksheet**

Learning Objective: OPL171.148 Obj. 7a (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

New	X
Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

Excerpts from 2-OI-92B:

3.0 PRECAUTIONS AND LIMITATIONS (continued)

X. Bypassed LPRMs are not used by PRNM system for calibration, flux, or trip points. LPRMs are normally bypassed from Panel 2-9-14 using "BYP/HV ON" or "BYP/HV OFF".

BYP/HV ON	LPRM is manually bypassed with the voltage on the detector. Indication of the detector output is available, but the signal is not included for any input to the APRM/OPRM/RBM functions.
BYP/HV OFF	LPRM is manually bypassed with the voltage off. No detector output.

In addition, LPRMs may be bypassed as indicated below:

BYP/IV	LPRM is automatically bypassed as a remotely initiated I/V (current to Voltage check)process is in progress.
BYP/CAL	LPRM is automatically bypassed while the LPRM is being calibrated (CALIBRATE) or that the calibration is being checked(CAL CHECK).
BYP/SUB'D	LPRM is bypassed while undergoing TRIP CHECK
BYP/FAULT	LPRM is automatically bypassed as a LPRM self-test fault is detected

Y. EPU will allow operating along the 100.7% Load line and 99% core flow to achieve 100% power.

Z. Each APRM chassis receives 22 LPRM inputs and the LPRM chassis receives 21 LPRM inputs for a total of 43 LPRM inputs to each APRM channel. The average of these 43 LPRM inputs is identified as the APRM flux and with a 6 second time constant applied, it is referred to as Simulated Thermal Power (STP). APRM STP approximates thermal power being transferred to the coolant. The STP trip level is varied as a function of recirc flow.

AA. Recirculation Flow signals are used to produce the flow biasing used in the APRMs only. To operate with a single recirculation flow loop, the Δ flow value in the flow biased flow equation will be different for single loop operation verses two loop operation. This function is only manually initiated by the operator. This ΔFlow takes into account any backflow that might exist in the idle loop during single loop operation. When operating with two loops there is no ΔFlow to offset the flow biased setpoints. During operation, if a recirculation pump trips, the operator can operate in compliance with Technical Specifications and apply the Δ flow factor into the flow biased setpoints. Flow biased Equation is:

SETPOINT = (.55 (Flow - ΔFlow)) + OFFSET.

6.3 Bypassing LPRM detectors (continued)

[9] **SELECT** the desired LPRM to be bypassed by using the left or right arrows on the softkey board until the inverse video illuminates the correct LPRM.

[10] **DEPRESS** the "BYPASS/HV ON" or the "BYPASS/HV OFF" softkey to bypass the selected LPRM.

[11] **CHECK** "BYP/HV ON" or "BYP/HV OFF" is indicated under the selected LPRM.

[12] **DEPRESS** "EXIT" softkey to return display to the desired bargraph.

[13] **CHECK**, as a result of bypassing the LPRM, that no alarms are received on Panel 2-9-5 or on the APRM/LPRM channel.

[14] **ENSURE** Work Order(s) written for bypassed LPRMs AND **INCLUDE** the position of the HV Bypass.

[15] For Bypassed LPRM(s), **ENTER** the following in NOMS:

- The LPRM number,
- The HV Bypass position,
- Remaining number of Operable LPRMs on the APRM Chassis (Total and number at the level bypassed),
- The Work Orders generated.

6.3 Bypassing LPRM detectors

NOTE

All operations are performed on Panel 2-9-14 unless specifically stated otherwise.

CAUTION

Bypassing LPRMs such that there are less than 20 detectors per channel, or, less than 3 per any given level, will yield the associated APRM channel instrument INOP CONDITION (i.e. Rod Block). This condition results in the associated APRM being INOP but does not yield an Automatic APRM INOP TRIP. Tech Spec Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation, and associated basis, addresses APRM operability requirements.

[1] **REVIEW** all precautions and limitations.
REFER TO Section 3.0.

[2] **REFER TO** Attachment 4 to find the APRM/LPRM Channel associated with the desired LPRM to be bypassed.

[3] **DEPRESS** any softkey to illuminate the display on the desired APRM/LPRM channel chassis.

[4] **CHECK** that more than 20 LPRMs in average exist by observing "number of LPRMs in average" on the selected APRM chassis.

[5] **CHECK** more than 3 LPRMs are operable in the LPRM strings at the level in which the LPRM is to be bypassed for the selected APRM chassis.

[6] **IF** either Step 6.3[4] **OR** 6.3[5] checks are not met, **THEN** (otherwise N/A)

REFER TO Tech Spec Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation, and associated basis.

[7] **DEPRESS** the "ETC" softkey until "BYPASS SELECTIONS" illuminates on the bottom row of the display.

[8] **DEPRESS** "BYPASS SELECTIONS" softkey, enter the password, and **DEPRESS** "ENT".

Supports Distractors A1, B1

6.0 SYSTEM OPERATIONS

NOTES

1) Only one APRM/OPRM can be bypassed at a time.
2) All operations are performed on Panel 2-9-5 unless specifically stated otherwise.
3) In order to prevent inadvertent rod withdrawal block or Reactor scram while operating APRM BYPASS selector switch, always ensure the previously bypassed channel returns to normal status by observing the BLUE bypassed lights on Panel 2-9-14 Voters are extinguished prior to selecting any other channel to be bypassed. After bypassing a channel, the applicable BLUE BYPASSED status lights on Panel 2-9-14 Voters should be illuminated prior to testing, operating, or working on that channel.

6.1 Bypassing APRM/OPRM Channel

CAUTION

[QA/C] NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with APRMs bypassed unless bypassing is specifically allowed within approved procedures. [ISE-NPS-92-R01]

[1] **REVIEW** all precautions and limitations.
REFER TO Section 3.0.

[2] **PLACE** APRM BYPASS, 2-HS-92-7B/S3, to desired channel to be bypassed.

[3] **CHECK** BLUE BYPASSED lights illuminated on Panel 2-9-14, Voters.

[4] **CHECK** white bypass light on Panel 2-9-5 is illuminated.

6.2 Returning APRM/OPRM to Service from Bypassed Condition

[1] **REVIEW** all precautions and limitations.
REFER TO Section 3.0.

[2] **PLACE** APRM BYPASS, 2-HS-92-7B/S3, in neutral (off).

[3] **CHECK** BLUE BYPASSED lights extinguished on Panel 2-9-14, Voters.

[4] **CHECK** the white bypass light on Panel 2-9-5 is extinguished.

Supports Distractors B2, D2

3.0 PRECAUTIONS AND LIMITATIONS (continued)

BB. The APRM/OPRM system is designed to provide an automatic Inoperative trip when any one or more of the following occur:

1. A critical self-test fault is detected in the APRM Instrument.
2. The APRM/OPRM chassis keylock switch is in the "INOP" Position.
3. The firmware/software watchdog timer has timed out.
4. Loss of input power to the APRM.

CC. The APRM/OPRM system is designed to provide an alarm (Inverse Video) when any one or more of the following conditions occur:

1. Any self-test fault (critical fault results in an INOP trip, as stated above, and the Inverse Video alarm. A non-critical fault results in an Inverse video alarm only).
2. The quantity of LPRM detectors in the average Neutron Flux Level calculation is less than the required minimum (20). This also results in a Rod Block.
3. The quantity of LPRM detectors in any given level (in the reactor) in the average Neutron Flux Level calculation is less than the required minimum (3). This also results in a Rod Block.

DD. The Reactor CAN BE operated indefinitely with one Recirc loop out of service, provided the requirements of T.S. 3.4.1 are implemented within 24 hours of entering single loop operations.

EE. Single recirculation loop operation (SLO) is prohibited in the MELLLA+ operating domain. **REFER TO** Tech Spec 3.4.1.

FF. Per Tech Specs 3.3.1.1-I and Table 3.3.1.1-1, if more than 1 OPRM channel is not capable of performing its trip function, then alternate Thermal-Hydraulic Instability monitoring and required actions are required to be initiated within 12 hours. 2-SR-3.3.1.1.I, Core Thermal Hydraulic Stability, is the procedure for performing this function.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295004 (APE 4) Partial or Complete Loss of D.C. Power / 6	Tier #	1	------
AA2.03 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of DC Power:	K/A #	295004AA2.03	
• Battery parameters	Importance Rating	3.8	------

Proposed Question: **# 38**

All Units are operating at 100% RTP when the following occurs:

- A fire was reported in Battery Board 1
- Battery Board 1 was de-energized and the fire is out

Given the conditions above, 4KV Shutdown Board **___(1)___** has lost its **NORMAL** **___(2)___** control power.

A. **(1)** 3EA
(2) 125V DC

B. **(1)** 3EA
(2) 250V DC

C. **(1)** A
(2) 125V DC

D. **(1)** A
(2) 250V DC

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that EDG 3A associated with 4KV Shutdown Board 3EA does have a 125V DC Distribution Panel that if lost, Alarm Response, 3-ARP-9-23A, Window 14, would alarm.

B **CORRECT**: (*See attached*) In accordance with 0-OI-57D, DC Electrical System, 4KV Shutdown Board 3EA's normal control power is 250V DC Battery Board 1. Although not given in the question, Alarm response procedure 3-ARP-9-23A, Windows 14 and 27, would be in alarm.
For second part, in accordance with both 0-OI-57D and 3-ARP-9-23A, 250V DC Battery Board 1 provides normal 250V DC control power to 4KV Shutdown Board 3EA.

C INCORRECT: First part is incorrect but plausible in that the DC Electrical System at BFN is quite complex and often confused. The candidate could easily conclude that 4KV Shutdown Board 'A' that supports Unit 1 and 2 gets its control power from Battery Board 1. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's ability to determine and/or interpret a partial or complete loss of DC Power as it relates to Battery Boards and AC powered 4KV Shutdown Boards. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 0-OI-57D, Rev. 196 (Attach if not previously provided)
3-ARP-9-23A, Rev. 30

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.037 Obj. 1d (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK OPL171.037-11 008 #12 (Note changes or attach parent)
New	

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Copy of Bank Question:

ILT 23-04 System 2 Exam

12. OPL171.037-11 008

All units are operating at 100% power when a fire is reported in Battery Board 1.

Battery Board 1 has been de-energized and the fire is out.

4kV SD BD ___(1)___ has lost control power at this time.

___(2)___ is expected to be in alarm at this time.

A. 1) 3EA
   2) 3-XA-55-23A Window 14, DIESEL GEN 3A CONTROL POWER OFF,

B. 1) 3EA
   2) 3-XA-55-23A Window 24, 4160V BKR 1334 FEEDER UNDERVOLTAGE,

C. 1) 3EC
   2) 3-XA-55-23C Window 14, DIESEL GEN 3C CONTROL POWER OFF,

D. 1) 3EC
   2) 3-XA-55-23C Window 24, 4160V BKR 1338 FEEDER UNDERVOLTAGE,

Excerpt from 0-OI-57D:

3.0 PRECAUTIONS AND LIMITATIONS (continued)

EE. If 250VDC control power is lost to SWGR "E" or "F" the red strobe light 0-STRB-248-0007/DCM located outside the SWGR Bldg will be flashing and 0-IL-248-0007/DCM located on 0-LPNL-925-0007/ DCM will be extinguished.

Shutdown Board	Norm Control Power	Transfer Switch	Drawing
4160V SD BD A	250V Battery SB-A	0-XSW-211-A	0-45E724-1
4160V SD BD B	250V Battery SB-B	0-XSW-211-B	0-45E724-2
4160V SD BD C	250V Battery SB-C	0-XSW-211-C	0-45E724-3
4160V SD BD D	250V Battery SB-D	0-XSW-211-D	0-45E724-4
4160V SD BD 3EA	250V Battery BD 1	3-XSW-211-3EA	3-45E724-6
4160V SD BD 3EB	250V Battery SB-3EB	3-XSW-211-3EB	3-45E724-7
4160V SD BD 3EC	250V Battery BD 3	3-XSW-211-3EC	3-45E724-8
4160V SD BD 3ED	250V Battery BD 2	3-XSW-211-3ED	3-45E724-9
480V SD BD 1A	250V Battery SB-A	1-XSW-231-1A	1-45E749-1
480V SD BD 1B	250V Battery SB-C	1-XSW-231-1B	1-45E749-2
480V SD BD 2A	250V Battery SB-B	2-XSW-231-2A	2-45E749-3
480V SD BD 2B	250V Battery SB-D	2-XSW-231-2B	2-45E749-4
480V SD BD 3A	250V Battery BD 1	3-XSW-231-3A/A	3-45E749-5
480V SD BD 3B	250V Battery BD 3	3-XSW-231-3B/A	3-45E749-6

FF. Relays will be considered ENERGIZED if the movable contact fingers (metal plate) are pushed back away from the relay case glass front cover with movable fingers and stationary fingers making contact, and DE-ENERGIZED if the contact fingers (metal plate) are in the forward position towards the relay case glass front cover against the rest bar (movable fingers and stationary fingers are NOT touching).

GG. The ideal ambient temperature in Battery Rooms is 77°F to 80°F. While there is not a maximum ambient room temperature limit, battery life decreases at temperatures above this range. Therefore, consideration should be given to establishing temporary cooling when temperatures are above 80°F."

Excerpts from 3-ARP-9-23A:

4160V SD BD
3EA 250V CONT
PWR UV

27

(Page 1 of 1)

Sensor/Trip Point:

Relays DC-S3A or DC-S3AE Failure of 250V control power

Sensor Location:
4160V Shutdown Bd. 3EA
Elevation 583
4kV Shutdown Bd. Rm. 3EC

Probable Cause:
A. Loss of normal supply (Battery Board, bkr 703):
B. Loss of alternate supply (Battery Board 2, bkr 703).
C. Cleared fuse.
D. Sensor malfunction.

Automatic Action:
A. Loss of Shutdown Bd. 3EA logic (degraded voltage, etc.)
B. Loss of diesel generator "3EA" shutdown bd. logic.

Operator Action:
A. **OBSERVE** for any other alarms on panel 9-8 or 9-23 which may indicate problem area.
B. **CHECK** for indication of failure of normal 250V DC supply by observing lights extinguished for equipment fed from Shutdown Bd. 3EA.
C. **CHECK** parameters of effected equipment and **REFER TO** appropriate OI as conditions dictate to maintain unit in stable condition.
D. **IF** 250V DC normal control power is lost, **THEN** **DISPATCH** personnel to:
   1. 4kV Shutdown Bd. 3EA to transfer to alternate DC supply with NORMAL/ALTERNATE knifeswitch in compartment 8.
   2. Battery Bd. Rm. 1 to check 250V normal supply for abnormal conditions:
      a. Battery and battery charger.
      b. Battery Bd. 1, panel 7, bkr. 703.
      c. Fuses, relays, smoke, etc.

E. **REFER TO** TS 3.8.7 and 3.8.8.

References:
45E766-17 thru 29 7-45E702-1 3-45E724-6
Technical Specifications 3.8.7 and 3.8.8

Supports Distractors A2 and C2:

BFN Unit 3	Panel 3-9-23A, 3-XA-55-23A	3-ARP-9-23A Rev. 0030 Page 19 of 41

DIESEL GEN 3A CONTROL POWER OFF

14

(Page 1 of 1)

Sensor/Trip Point:

Any of following relays:

(74) 3-74-082-02547A, 3-RLY-082-2547AFAR
(NVR) 3-RLY-082-A/NVR,
3-RLY-211-DC-A1,
3-RLY-211-DC-A2, or
3-RLY-211-EADCPFA3

Loss of DG control power.

3-RLY-211-EADCPFA3
4kV SD BD 3EA
Elevation 583
4kV SD BD Rm. 3EC

Sensor Location:	3-74-082-2547A, Relay (74) 3-RLY-082-2547AFAR Protective Relay Cab. Elevation 565 Diesel Generator Bldg.	3-RLY-082-A/NVR, Relay (NVR) Engine Control Cab. Elevation 565 Diesel Generator Bldg.	Relay DC A1&A2 4kV SD BD 3EA Elevation 583 4kV SD BD Rm. 3EC

Probable Cause:

A. Loss of power from 125V DC panel to diesel generator 3A.
B. Breaker(s) open at 125V DC distribution panel, electrical control cabinet or protective relay cabinet breakers.
C. Loss of 125 Vdc from diesel battery.
D. Breaker(s) off at the diesel 3A electrical control cabinet.
   - Alarm, start 1, start 2, AC control or DC control breakers.

E. Loss of 250 Vdc control pwr to 4160V Shutdown Bd. 3EA.
F. Cleared fuse or sensor malfunction.

Automatic Action: None: Possible loss of diesel generator 3A logic.

Operator Action:

A. **OBSERVE** any other alarms on panels 9-8 or 9-23 which may indicate problem area.
B. **CHECK** panels, breakers and batteries. **IF** necessary, **THEN CHECK** fuses and relays.
C. **IF** loss of normal power has occurred, **THEN TRANSFER** to alternate power source. **REFER TO** 0-OI-57D.
D. **REFER TO** TS 3.8.1, 3.8.2, 3.8.4, and 3.8.5.

References: 45E768-6 45E767-6&8 45E766-29
Technical Specifications 3.8.1, 3.8.2, 3.8.4, 3.8.5

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling System
K2.02 (10CFR 55.41.7)
Knowledge of electrical power supplies to the following:

- Initiation/isolation logic

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	217000K2.02	
Importance Rating	3.7	------

Proposed Question: **# 39**

A fault causes a loss of 250VDC RMOV Board 3B.

If a Unit 3 RCIC automatic start signal is received, RCIC **(1)** start.

Given the loss of power, RCIC **(2)** isolate on low Reactor Pressure.

A. (1) will
(2) will

B. (1) will
(2) will **NOT**

C. (1) will **NOT**
(2) will

D. (1) will **NOT**
(2) will **NOT**

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the initiation logic and power supplies for both HPCI and RCIC are easily and often confused especially between Division I(II) ECCS ATU and Inverter power as it relates to 250VDC RMOV Boards 3A(B). Second part is incorrect but plausible in that the isolation logic and power supplies for both HPCI and RCIC are easily and often confused. If 250VDC RMOV Board 3A had been lost, then RCIC as a Division 1 ECCS system, would automatically isolate on all isolation signals.

B INCORRECT: First part is incorrect but plausible (*See A).* Second part is correct (*See D).*

C INCORRECT: First part is correct (*See D).* Second part is incorrect but plausible (*See A).*

D **CORRECT**: (*See attached)* In accordance with 3-AOI-57-11, Loss of Power to An ECCS ATU Panel / ECCS Inverter, 250VDC RMOV Board 3B powers Division 1-ECCS ATU and Inverter which impacts RCIC operation and OPERABILITY. Given that 250VDC RMOV Board 3B is lost, RCIC's Logic Bus B (Div 1) is lost, therefore the automatic initiation circuit, turbine trip solenoid will not operate in accordance with RCIC RELAY LOGIC POWER FAILURE (3-ARP-9-3C, Window 1). This results in RCIC not being able to start. For second part, in accordance with 3-AOI-64-2C, RCIC Turbine Steam Line Low Pressure automatic isolation setpoint is approximately 70 psig Reactor Pressure. If the RCIC Low Reactor Pressure isolation signal is received with the loss of 250VDC RMOV Board 3B, then RCIC will only isolate on area high temperature and high steam line flow, but not on low Reactor Pressure.

RO Level Justification: Tests the candidate's knowledge of RCIC electrical power supplies as it relates to initiation and isolation logic. This question is rated as C/A due to the requirement to correctly assemble given parameter from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution related to system initiation and isolation logic.

Technical Reference(s): 3-OI-71, Rev. 73 (Attach if not previously provided)
3-AOI-57-11, Rev. 18
3-AOI-64-2C, Rev. 6
3-ARP-9-3C, Rev. 34

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.040, Obj. 4a(b) (As available)

Question Source:

Bank #	
Modified Bank #	BFN NRC 1404 #39
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2014

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Copy of Bank Question:

QUESTION 39

A fault causes a loss of 250V DC RMOV Board 3A.

Which ONE of the following completes both statements below?

If a Unit 3 RCIC automatic start signal is received, RCIC __(1)__ start.

The RCIC __(2)__ channel isolation logic is NOT functional.

A. (1) will
(2) 'A'

B. (1) will
(2) 'B'

C. (1) will NOT
(2) 'A'

D. (1) will NOT
(2) 'B'

ANSWER: **B**

Excerpts from 3-AOI-57-11:

BFN Unit 3	Loss of Power to An ECCS ATU Panel / ECCS Inverter	3-AOI-57-11 Rev. 0018 Page 4 of 31

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, operator actions, technical specification requirements, and reportability requirements resulting from a loss of power to ECCS ATU Panel 9-81 or 9-82, or loss of an ECCS inverter.

NOTES

1) Each Inverter provides electrical power to divisional logics plus one of the two redundant power supplies to its divisional ATU cabinet (**REFER TO** Attachment 5). The power supplies to ECCS ATU Panel 9-81 (Div. I), ECCS ATU Panel 9-82 (Div. II) and the ECCS inverters are as follows:

Panel 9-81

- Division I ECCS inverter — 250V RMOV Board 3B, compartment 8A
- Division I 250/24vdc converter — 250V RMOV Board 3B, compartment 1B1

Panel 9-82

- Division II ECCS inverter — 250V RMOV Board 3A, compartment 11A1
- Division II 250/24vdc converter — 250V RMOV Board 3A, compartment 9A1

Power will be lost to an ECCS ATU Panel due to the loss of the respective 250V RMOV board listed above, opening/loss of both of the breakers listed above, loss of ECCS ATU Panel internal fuses, or simultaneous loss of both redundant 24vdc power supplies in each ECCS ATU panel.

2) The total loss of power to the ECCS ATU panels results in power loss to all instrumentation on:

- Division I — Panel 9-81 (Aux Instrument Room)
- Division II — Panel 9-82 (Aux Instrument Room)

RHR System I(II) containment spray valve operation will require the use of the manual override switch 3-XS-74-122(130) upon a loss of Panel 9-81(82) due to a loss of the two-thirds core height level channel.

If the 3A 250V RMOV Bd is de-energized or 3-BKR-281-003A/9A1 is open the Alternate supply to ADS Logic "3B" is not available along with the High Drywell Pressure, Low Reactor Water Level inputs and the ADS High Drywell Pressure Bypass to the 3B ADS Logic.

2.0 SYMPTOMS (continued)

F. The following annunciators occur for a loss of the input power supply to the Division II 250/24vdc ECCS ATU converter (Attachment 5, Power Supply 3A):

1. ECCS ANALOG TRIP UNIT TROUBLE 3-XA-71-60 (3-XA-55-3E, Window 30)
2. RCIC RELAY LOGIC POWER FAILURE (3-XA-55-3C, Window 1)
3. ADS BLOWDOWN POWER FAILURE (3-XA-55-3C, Window 32)

NOTE

Review of annunciators received will enable the determination of which single component or multiple components are out of service and the extent of the loss of power.

G. The following control room indications fail downscale for a loss of both Division I ECCS ATU Panel (9-81) power supplies:

- RPV water level 3-LI-3-208A (1) Panel 9-5
- RPV water level 3-LI-3-58A Panel 9-5
- RPV water level 3-LI-3-58AA Panel 9-5
- RPV water level 3-LI-3-52 Panel 9-3
- RPV water level 3-LI-3-208C (1) Panel 9-3
- RPV pressure 3-PI-3-74A Panel 9-3

(1) RPV water level 3-LI-3-208A(C) indicators powered directly from Div I inverter.

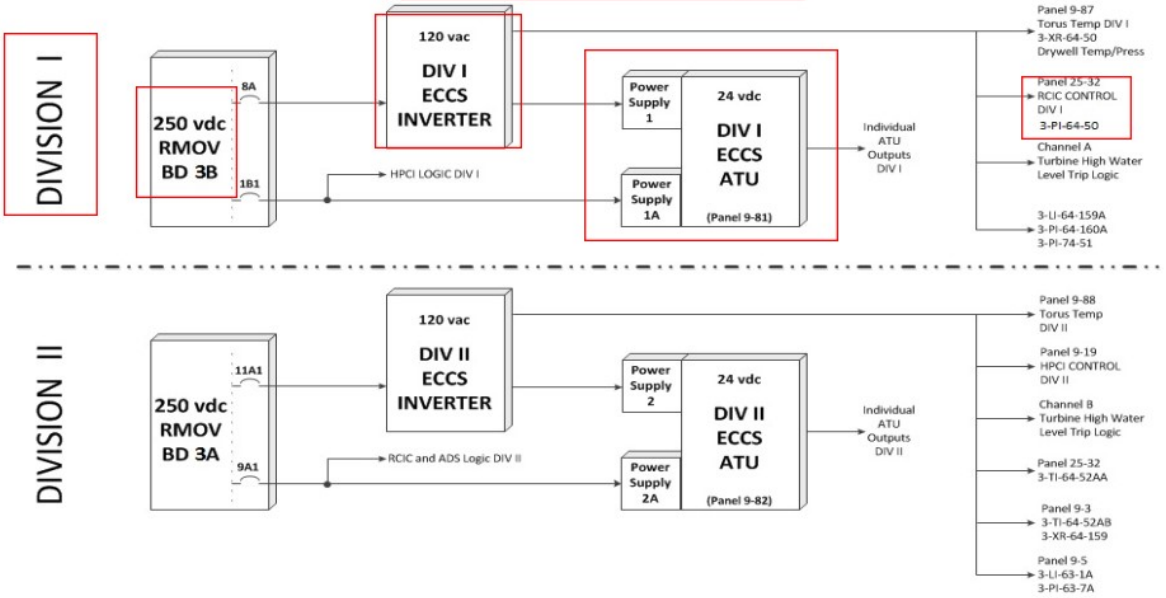
H. The following control room indications fail downscale for a loss of both Division II ECCS ATU Panel (9-82) power supplies:

- RPV water level 3-LI-3-208D(2) Panel 9-5
- RPV water level 3-LI-3-58B Panel 9-5
- RPV water level 3-LI-3-58BB Panel 9-5
- RPV water level 3-LI-3-62A Panel 9-3
- RPV water level 3-LI-3-208B(2) Panel 9-3
- RPV pressure 3-PI-3-74B Panel 9-3

(2) RPV water level 3-LI-3-208B(D) indicators powered directly from Div II inverter.

Attachment 5
(Page 1 of 1)

Division I and Division II Power Supply



Excerpt from 3-ARP-9-3C:

BFN Unit 3	Panel 9-3 3-XA-55-3C	3-ARP-9-3C Rev. 0034 Page 4 of 43

RCIC RELAY LOGIC POWER FAILURE

1

(Page 1 of 1)

Sensor/Trip Point:

Logic Bus A:	Relay 13A-K1, 13A-K24, or 13A-K40 deenergized.
Logic Bus B:	Relay 13A-K34 deenergized.

	Logic Bus A	Logic Bus B
Sensor Location:	Panel 25-31 Rx Bldg, El 621', R-20 Q-Line	Panel 9-33 Aux Instr Rm, El 593'

Probable Cause:

A. Cleared Fuse(s)
B. Loss of 250V DC power supply to panels.

Automatic Action:

A. If Bus A fails, the automatic initiation circuit and turbine trip solenoid will **NOT** operate. Channel A isolation logic circuit is lost.
B. If Bus B fails, B channel isolation logic will be inoperative.

Operator Action:

A. **DETERMINE** which logic bus (A or B) has failed and **DISPATCH** personnel(s) to investigate the following:
   1. Logic Bus A
      a. Power supply 250V DC Rx MOV Bd 3B, Compt 8EI. Loss of items listed in 2, 3, and 4.
      b. Fuses 3-FU1-071-0018D (13A-F9) and 3-FU1-071-0018E (13A-F10) (10 amp) - Panel 3-25-31, fuse block CC. Loss of initiation, trip, and Logic Bus A isolation logics.
      c. Fuses 3-FU2-071-0029D (3amp) - Panel 3-25-31, fuse block AA. Loss of isolation on Rupture Disc High Pressure.
      d. Fuses 3-FU1-071-0013AA (13A-F28) and 3-FU1-071-13AB (13A-F29) (10 amp) - Panel 3-25-31, fuse block CC. Loss of isolation on Turbine Exhaust Pressure High and Pump Suction Low Pressure.
   2. Logic Bus B
      a. Power supply - 250V DC Rx MOV Bd 3A, Compt 9A1.
      b. Fuses 3-FU2-71-13A-K30 (13A-F23) and 3-FU2-71-13A-K30 (13A-F24) (10 amp) - Panel3-9-33, fuse block GG. Loss of Logic Bus B isolation logic.

B. **REFER TO** Tech Spec Sections 3.3.5.3, 3.3.6.1,and 3.5.3.

References:

0-45E626-1, 2 3-45E620-2 3-45E626-2,3
Technical Specifications 3.3.5.3, 3.3.6.1, 3.5.3 TRM 3.3.3

Excerpt from 3-OI-71:

BFN Unit 3	Reactor Core Isolation Cooling System	3-OI-71 Rev. 0073 Page 9 of 85

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. Turbine controls provide for automatic shut down of the RCIC turbine upon receiving any of the following signals (**REFER TO** Section 8.4 for auto actions):

1. High RPV water level (+51 in.); 579 in. above vessel zero. The RCIC TURBINE STEAM SUPPLY VLV, 3-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, will close at +51 in. and will reopen when RCIC re-initiates at -45 in. RPV water level.
2. Turbine overspeed (Mechanical, 121% of rated speed).
3. Pump low suction pressure (10 inches HG vacuum).
4. Turbine high exhaust pressure (50 psig).
5. Any isolation signal.
6. Remote manual trip (RCIC TURBINE TRIP pushbutton, 3-HS-71-9A, depressed).

B. RCIC turbine steam supply will isolate from the following signals (**REFER TO** 3-AOI-64-2c for auto actions):

1. RCIC steamline space temperature ≤165°F Torus Area or ≤165°F RCIC Pump Room.
2. RCIC turbine high steam flow (150% flow, 3-second time delay).
3. RCIC turbine steam line low pressure - approximately 70 psig RX pressure. (86 psig pressure switch setpoint value with static head correction)
4. RCIC turbine exhaust diaphragms ruptured (10 psig).
5. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION pushbutton, 3-HS-71-54, depressed, only if RCIC initiation signal is present).

C. The RCIC turbine will auto initiate on RPV Low-Low Water Level, -45 in. (**REFER TO** Section 5.1 for auto actions.)

D. In the presence of a RCIC initiation signal, the RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve does **NOT** auto open on low flow if an initiation signal is **NOT** present.

Excerpt from 3-AOI-64-2C:

BFN Unit 3	Group 5 Reactor Core Isolation Cooling Isolation	3-AOI-64-2C Rev. 0006 Page 4 of 8

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 5 Reactor Core Isolation Cooling Isolation (RCIC).

NOTES

1) During a normal unit shutdown, this isolation will occur and is **NOT** considered abnormal.
2) Unless otherwise specified, all actions or indications are at Panel 3-9-3.

2.0 SYMPTOMS

- RCIC TURBINE TRIPPED (3-XA-55-3B, Window 14) in alarm.
- RCIC STEAM LINE FLOW EXCESSIVE (3-XA-55-3B, Window 21) in alarm.
- RCIC TURB EXH DISC RUPTURED (3-XA-55-3B, Window 35) in alarm.
- RCIC STEAM LINE LEAK DETECTION TEMP HIGH (3-XA-55-3D, Window 10) in alarm.
- RCIC Turbine Tripped and speed dropping on RCIC TURBINE SPEED, 3-SI-71-42A.
- RCIC steam line pressure less than 70 psig on RCIC STM LINE PRESS, 3-PI-71-4A.
- Amber RCIC AUTO-ISOL LOGIC A and B lights, 3-IL-71-51A and 51B, are illuminated.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling System	Tier #	2	------
K4.17 (10CFR 55.41.7)	Group #	1	------
Knowledge of Reactor Core Isolation Cooling System design features and/or interlocks that provide for the following:	K/A #	217000K4.17	
• Bypass trips and isolation logic	Importance Rating	3.7	------

Proposed Question: **# 40**

The Unit 2 Control Room has been abandoned, with the following conditions:

- 2-AOI-100-2, Control Room Abandonment, is in progress
- The Control Room Crew is stationed at Panel 2-25-32
- 2-XS-256-1, ATU BUS MANUAL TRANSFER, on Panel 2-25-32 is placed in ALTERNATE

Given the conditions above, the RCIC Flow Controller is powered by **(1)**, and RCIC **(2)** trip on High Reactor Water Level.

A. (1) Unit Preferred
(2) will

B. (1) Unit Preferred
(2) will NOT

C. (1) the Division I ECCS Inverter
(2) will

D. (1) the Division I ECCS Inverter
(2) will NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that some of the RCIC trips can still occur from Panel 2-25-32 such as low suction pressure, overspeed, manual pushbutton, high turbine exhaust pressure.

B **CORRECT:** (*See attached*) In accordance with 2-AOI-100-2, Control Room Abandonment, the Unit 2 Operator is required to complete Attachment 2–Part B at the Backup Control Panel 2-25-32. The Unit 2 Reactor Building AUO will complete Attachment 3-Part A at the RCIC Panel 2-25-31. When 2-XS-256-1, ATU BUS MANUAL TRANSFER, switch is placed in ALTERNATE from Panel 2-25-32, the RCIC Flow Controller will now be powered from Unit Preferred. For second part, in accordance with 2-AOI-100-2, once 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, transfer switch has been placed in EMERGENCY, with control established at the Backup Control Panel 2-25-32 as given, RCIC will not trip on High Reactor Water Level.

C INCORRECT: First part is incorrect but plausible in that the Division I ECCS Inverter is the normal power supply to the RCIC Flow Controller. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the RCIC design features and / or interlocks that provide bypass trips and isolation logic. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution as it relates to RCIC control.

Technical Reference(s): 0-OI-57C, Rev. 136 (Attach if not previously provided)
2-AOI-57-11, Rev. 18
2-AOI-57-4, Rev. 53
2-AOI-100-2, Rev. 61
OPL171.040, Rev.32

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.040 Obj. x (As available)

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

Question History: New **X**
Last NRC Exam

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpts from 2-AOI-100-2:

4.2 Unit 2 Subsequent Actions (continued)

NOTES

1) Attachment 1 provides normal backup control stations and available communications.
2) Attachment 10 provides PAX extensions and locations.

[7] **ESTABLISH** communication with the following personnel and **DIRECT** attachments be completed as follows:

- U-2 Unit Operator complete Attachment 2, Part A.
- U-2 Rx Bldg AUO complete Attachment 3, Part A.
- U-2 Turb Bldg AUO complete Attachment 4, Part A.

[8] Upon completion of attachments, **RE-ESTABLISH** communication using the best available means and continue procedure.

CAUTION

1) RCIC TURBINE STEAM SUPPLY VALVE, 2-FCV-71-8, transfer switch has been placed in EMERGENCY and will **NOT** trip on Reactor Water Level High (+51 inches). Failure to maintain level below this value may result in equipment damage.
2) RCIC will still trip on low suction pressure, high turbine exhaust pressure, mechanical overspeed, and trip push button on pnl 25-32.

[9] **INITIATE** RCIC as follows:

[9.1] **CHECK OPEN** 2-FCV-71-9 RCIC TURB TRIP/THROT VALVE RESET, 2-HS-71-9D At Panel 2-25-32. (Red Light above switch)

[9.2] **PLACE** RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, 2-HS-071-0034C, in OPEN at 250V DC RMOV Bd 2B, compt. 5D. (Unit 2 Turbine Building AUO)

[9.3] **PLACE** RCIC TURB STM SUPPLY VALVE EMER HAND SWITCH, 2-HS-071-0008C, in OPEN at 250V DC RMOV Bd 2C, compt. 4B. (Unit 2 Reactor Building AUO)

4.2 Unit 2 Subsequent Actions (continued)

NOTES

1) Attachment 1 provides normal backup control stations and available communications.
2) Attachment 10 provides PAX extensions and locations.

[12] **ESTABLISH** communication with the following personnel and **DIRECT** Attachments to be completed as follows:

- U-2 Unit Operator complete Attachment 2, Part B.
- U-2 Rx Bldg AUO complete Attachment 3, Part B.
- U-2 Turb Bldg AUO complete Attachment 4, Part B.
- U-2 Unit SRO (or designee) complete Attachment 5.
- U-1 Turb Bldg AUO complete Attachment 6.
- U-1 Unit Operator complete Attachment 7.

[13] Upon completion of attachments, **RE-ESTABLISH** communication using the best available means and AWAIT further instructions.

[14] **IF** CRD Pump 1B is to be aligned to Unit 2, **THEN**

PERFORM the following: (Otherwise N/A)

[14.1] **PLACE** CRD PUMP 1B DISCH TO UNIT 2, 2-HS-85-8C, in OPEN at 480V RMOV Bd 2C, compt. 11C, .

[14.2] **PLACE** CRD PUMP 1B MOTOR, 1-HS-085-0002C, in CLOSE to start CRD Pump 1B at 4160V Shutdown Bd A, compt. 13.

Attachment 2
(Page 2 of 4)

Unit 2 UO Panel Checklist

2.0 PART B

Switch/ Breaker Number	Component Description	Required Position	Initials
	Control Bay - Pnl 2-25-32 - U-2 Backup Control Panel - El 621'		
2-XS-64-54	SUPPR POOL LVL TRANSFER	EMERG	_____
2-XS-64-51	SUPPR CHMBR PRESS TRANSFER	EMERG	_____
2-XS-64-55B	SUPPR POOL WTR TEMP SEL	1 or 2	_____
2-XS-64-52	DRYWELL AIR TEMP XFR	2	_____
2-XS-256-1	ATU BUS MANUAL TRANSFER	NORM or ALT	_____
2-XS-57-170	UNIT PREFERRED MANUAL TRANSFER	NORM or ALT	_____
2-HS-67-50A	EECW N HDR SPLY TO RBCCW HX	AUTO	_____
2-HS-67-51A	EECW S HDR SPLY TO RBCCW HX	AUTO	_____
2-HS-67-53A	EECW N HDR SPLY TO AIR COMP	AUTO	_____
XS-85-36-B	CRD SCRAM DISCH VOL TEST DR & VT VLV TRANS	EMERGENCY	_____
XS-85-36-A	CRD SCRAM DISCH VOL TEST DR & VT VLV TRANS	EMERGENCY	_____
HS-85-36C	CRD SCRAM DISCH VOL ISOL TEST VALVE	DEPRESSED	_____
2-HS-70-16B	DRYWELL COOLER 2A1 VLV AND DAMPER	OPEN	_____
2-HS-70-18B	DRYWELL COOLER 2A2 VLV AND DAMPER	OPEN	_____
2-HS-70-20B	DRYWELL COOLER 2A3 VLV AND DAMPER	OPEN	_____

Attachment 3
(Page 1 of 9)

Unit 2 RB AUO Panel Checklist

Date _____

NOTE
PAX phone Ext. 2233 is located at Column R-12, P-line near stairs to SLC.

1.0 PART A

Switch/ Breaker Number	Component Description	Required Position	Initials
	Reactor Bldg. - RCIC System Aux Panel 2-LPNL-025-0031 - El 621'		
2-XS-071-0036B	RCIC PUMP DISCH FLOW EMER TRANS SWITCH	EMERG	_____
2-XS-071-0047	RCIC TURB GOV & CPLG END BRG HIGH TEMP EMER TRANS SWITCH	EMERG	_____
2-XS-071-0024	RCIC OIL CLR OIL OUTLET TEMP EMER TRANS SWITCH	EMERG	_____
	CONTINUE Part A of this Attachment.		_____

Attachment 3
(Page 3 of 9)

1.0 PART A (continued)

Switch/ Breaker Number	Component Description	Required Position	Initials
	Reactor Bldg. - 250V DC Reactor MOV Board 2C - El 565'		
3D	2-BKR-071-0039, RCIC PUMP INJECTION VALVE BREAKER (GE-13-21)		
	2-HS-071-0039C, RCIC PUMP INJECTION VALVE EMER HAND SWITCH	OPEN	_____
	2-XS-071-0039, RCIC PUMP INJECTION VALVE EMER TRANS SWITCH	EMERG	_____
4B	2-BKR-071-0008, RCIC TURBINE STM SUPPLY VALVE BREAKER (GE-13-131)		
	2-HS-071-0008C, RCIC TURB STM SUPPLY VALVE EMER HAND SWITCH	NOR	_____
	2-XS-071-0008, RCIC TURB STM SUPPLY EMER TRANS SWITCH	EMERG	_____
6D	2-BKR-071-0019, RCIC CST 2 SUCT VALVE BREAKER (GE-13-18)		
	2-HS-071-0019C, RCIC CST 2 SUCT VALVE EMER HAND SWITCH	OPEN	_____
	2-XS-071-0019, RCIC CST 2 SUCT VALVE EMER TRANS SWITCH	EMERG	_____

Excerpt from 0-OI-57C:

7.4 Removing Unit 2 ECCS Analog Trip Unit Inverters from Service

7.4.1 Removing Unit 2 Division I ECCS Analog Trip Unit Inverter from Service

[1] **ENSURE** the ECCS ATU INVERTER DIV I, 2-INVT-256-0001 is in service. REFER to Section 5.3.1.

[2] **REVIEW** all Precautions and Limitations. **REFER TO** Section 3.0.

[3] Following the removal or loss of ECCS ATU INVERTER DIV I, 2-INVT-256-0001, the following instruments/equipment are deenergized.

- ATU 2-PX-71-60-1 (no effect if 2-PX-71-60-1A is operating)
- CH. A RFPT & MAIN TURBINE HIGH WATER LEVEL TRIP LOGIC
- RCIC FLOW CONTROLLER and INSTRUMENTATION
- PANEL 2-25-32 RX WATER LEVEL A(B), 2-LI-3-46A(B)
- PANEL 2-25-32 RHR SYS I TOTAL FLOW 2-FI-74-79
- DIV I TORUS TEMPERATURE MONITORING PANEL 2-9-87
- RHR SYS I DISCH PRESSURE INDICATOR, 2-PI-74-51
- DRYWELL PRESSURE ACCIDENT RANGE, 2-PI-64-160A
- SUPPR POOL WATER LEVEL INDICATION, 2-LI-64-159A
- DRYWELL TEMPERATURE/PRESSURE, 2-XR-64-50
- DRYWELL PRESSURE, 2-PI-64-50

[3.1] **DETERMINE** plant impact and LCO applicability for the plant condition/Mode. (N/A if Div I not out of service)

NOTE

Steps 7.4.1[4] and 7.4.1[5] are performed at the ECCS ATU INVERTER DIV I, 2-INVT-256-0001 located in Electrical Board Room 2B.

[4] **OPEN** the ECCS ATU INVERTER DIV I AC OUTPUT, 2-BKR-256-0001B.

[5] **OPEN** the ECCS ATU INVERTER DIV I DC INPUT BKR, 2-BKR-256-0001A.

Excerpts from 2-AOI-57-11:

4.2 Subsequent Action (continued)

NOTES

1) Refer to the FPR/FPRM for entry into applicable LCOs due to loss of instrumentation and/or electrical breakers.
2) This procedure is based on no safety related systems out of service. Additional Tech Spec LCOs may be imposed if an active Tech Spec LCO was in effect prior to initiation of this AOI.
3) Upon a concurrent loss of the Division II Inverter and the ATU Panel 9-82, REFER TO Technical Specification 3.5.1 condition D for additional LCOs that will be applicable.

[3] **IF** Division I(II) ECCS ATU Panel 9-81(9-82) is lost AND Core Spray Loop I(II) Injection is required, **THEN**

ENSURE Reactor Pressure is Less Than 450 psig AND **DEFEAT** Rx Low Pressure Interlocks. REFER TO 2-OI-75.

[4] **IF** Division I ECCS ATU Panel 9-81 is lost REFER TO appropriate ARPs for Suppression Chamber Vacuum relief valves.

NOTE

Loss of the Div I Inverter will cause the RCIC Flow Controller to be INOP,but the alarm will not annunciate. Performance of the following step will restore RCIC controller function but will not allow RCIC to be declared operable.

[5] **IF** ECCS Div I inverter is lost AND it is desired to keep RCIC functional, **THEN**

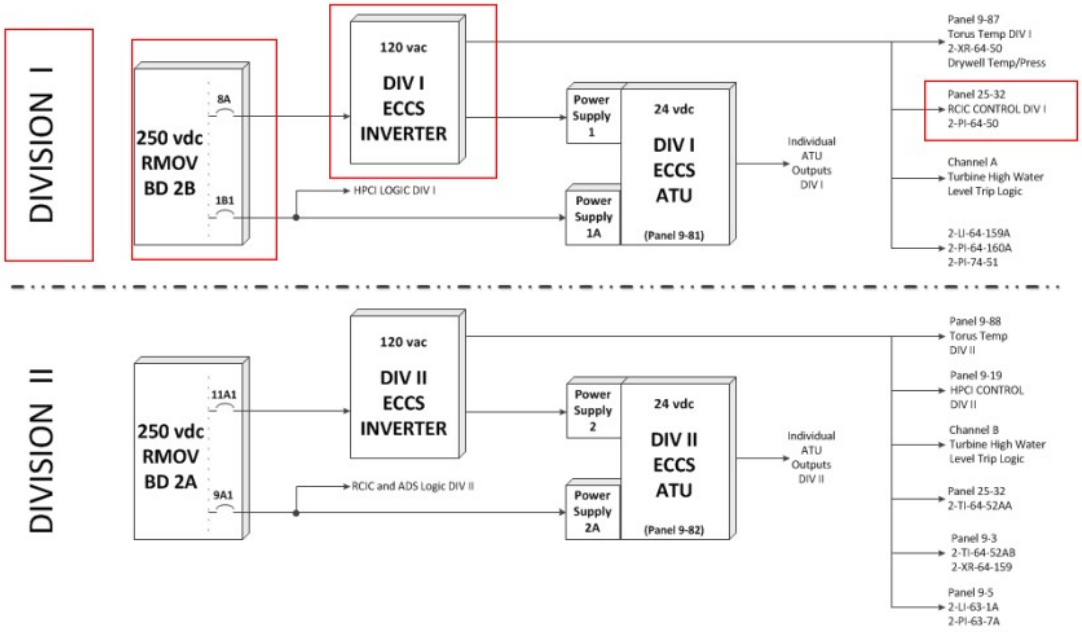
PLACE 2-XS-256-1 ATU BUS MANUAL TRANSFER on Panel 2-25-32 in ALT.

[6] **PERFORM** an inspection of the following for abnormalities:

- ECCS ATU Panel 9-81(82) in the Aux Instrument Room.
- ECCS Div I(II) inverter behind 250V RMOV board 2B(2A).
- 250V RMOV board 2B(2A) compartments 1B1 & 8A(9A1 & 11A1).
- Fuse 2-FU2-071-0060/1A, and 2-FU1-71-60/1, in Panel 9-81.
- Fuse 2-FU2-071-0060/2A, and 2-FU1-71-60/2, in Panel 9-82.

Attachment 5
(Page 1 of 1)

Redundant Power Supplies



Excerpt from 2-AOI-57-4:

Attachment 2
(Page 1 of 9)

Power Supplies

1.0 UNIT 2 BATTERY BOARD 2 PANEL 11, PROVIDES POWER TO THE FOLLOWING COMPONENTS

A. BKR No. 1171 UNIT 2 CKT BKR BD PNL 2-9-9 UPS NORM FDR

B. BKR No. 1172 UNIT 1 CKT BKR BD PNL 1-9-9 UPS ALT FDR

C. BKR No. 1173 SPARE

D. BKR No. 1174 SPARE

E. BKR No. 1175 UNIT 2 BACKUP CONT CENTER PNL 25-32 NORM FDR

F. BKR No. 1176 U-3 BACKUP CONT CENTER PNL 3-25-32 ALT FDR

G. BKR No. 1177 RADWASTE BD 1 & RADWASTE BD 2 (NORM FDR)

H. BKR No. 1178 UNIT 2 PASS CONTROL PANEL 2-25-366

I. BKR No. 1179 RFW HEATER LEVEL CONTROL SYSTEM NORMAL SUPPLY

J. BKR No. 1180 U-2 NORMAL SUPPLY CKT BKR BD 2-9-9 UNIT NONPREF AC SYS

2.0 UNIT 2 PANEL 9-9 CABINET 5 UNIT NON-PREFERRED

A. BKR 501, CONTROL AND SERVICE AIR VALVES

Secondary Containment Control Air isolation valves FSV-32-28B, FSV-32-29B, FSV-32-91B; Drywell Service Air valve FCV-33-10. Valves have parallel air supply solenoids, each solenoid having separate power supplies (Non-preferred and I&C Bus). Valve remains open if open.

B. BKR 502, PANEL 2-25-114 HYDROGEN GAS DRYER

Loss of power to 500W Hydrogen gas heater

Excerpts from OPL171.040 Lesson Plan:

OPL171.040, Reactor Isolation Cooling (RCIC), Rev.# 32

C. Instrumentation

1. Control Room Indications

Instrument Name	Instrument No.	Range	
RCIC FLOW CONTROLLER	FIC-71-36A	0-700 gpm	
RCIC PUMP DISCHARGE PRESURE	PI-71-35A	0-1500 psig	BFPER 00-011480-000
RCIC PUMP SUCTION PRESSURE	PI-71-20A	0-50 psig	
RCIC TURBINE STEAM LINE PRESSURE	PI-71-4A	0-1500 psig	
RCIC TURBINE EXHAUST PRESSURE	PI-71-12A	*0-75 psig-U1(U2) 0-200 psig(U3)	*DCN 70649(70675) U1(U2) RCIC Turbine Exhaust Pressure Indicator, 1(2)-PI-071-0012A, 0-75 psig BFPER971133 Indicator could read from 0-200 rpm in standby readiness due to non-linearity in low RPM range
RCIC TURBINE SPEED	SI-71-42A	0-6000 rpm	
RCIC TURBINE STEAM FLOW	FI-71-1A FI-71-1B	0-80 lbm x1000	

2. Flow Controller (FIC-71-36A & B)

   a. One located on Panel 9-3 and one on Panel 25-32 (Remote Shutdown Panel).

   b. Power Supply to the Pnl 9-3 controller (FIC-71-36A) is the Div I ECCS Inverter

   c. Power Supply to the Pnl 25-32 Controller (FIC-71-36B) is also the Div I ECCS Inverter.

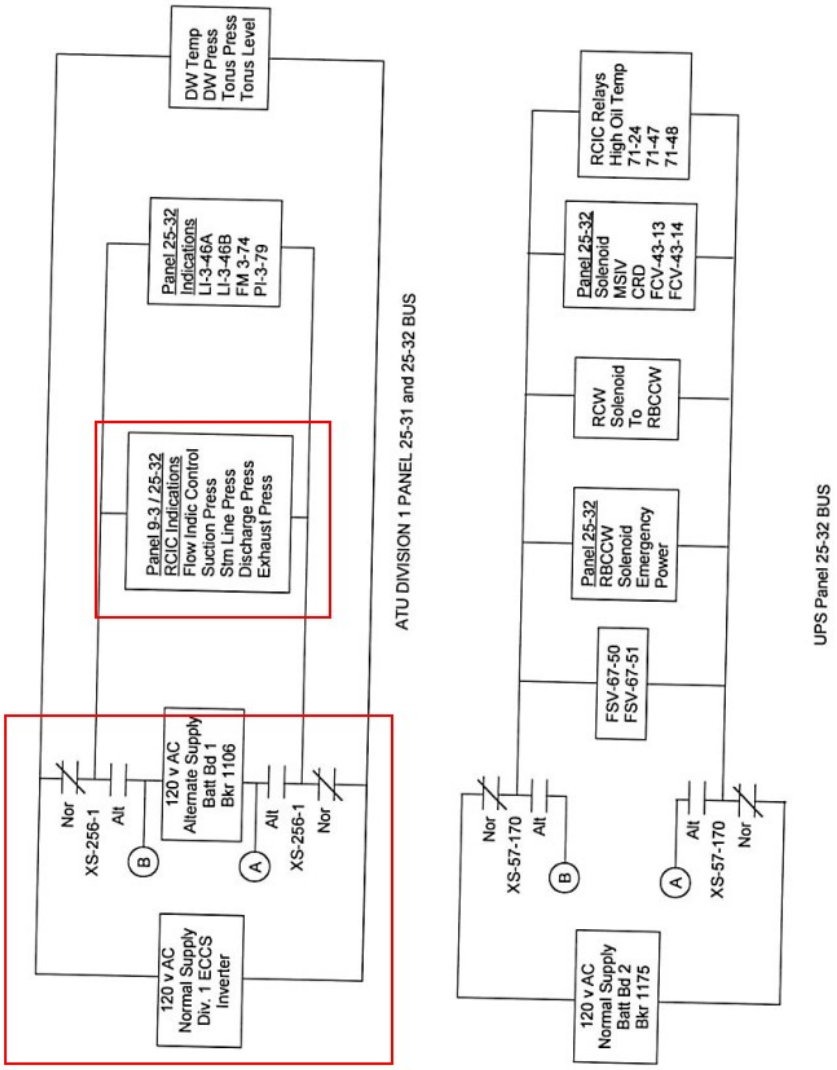
   d. AT Pnl 25-32, there is a power transfer switch (XS-256-1) which, if placed in the Alternate position, will transfer both (36A & B) Flow Controller power supplies from the Div I ECCS Inverter to the Unit Preferred 120VAC Power Supply.

   e. Current controller setpoint: **620** gpm

EDC 69932 changed FIC set point from 600 to 620gpm

IL-13

OPL171.040, Reactor Isolation Cooling (RCIC), Rev.# 32



IL-13 RCIC Normal and Alternate Power

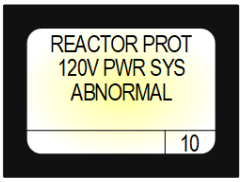
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
212000 (SF7 RPS) Reactor Protection **A1.04** (10CFR 55.41.5) Ability to predict and/or monitor changes in parameters associated with operation of the Reactor Protection System, including: • RPS bus status	Tier #	2	------
	Group #	1	------
	K/A #	212000A1.04	
	Importance Rating	3.7	------

Proposed Question: **# 41**

Unit 1 is at 100% RTP when the following plant conditions occur:

- REACTOR PROTECTION 120V POWER SYSTEM ABNORMAL, (1-9-5B, Window 10) alarms



Subsequently, the following valves travel CLOSED:

- 1-FCV-69-01, RWCU INBOARD SUCTION ISOLATION VALVE
- 1-FCV-69-02, RWCU OUTBOARD SUCTION VALVE

Reactor Protection System (RPS) Bus ___**(1)**___ has de-energized.

In accordance with 1-AOI-99-1, Loss of Power to One RPS Bus, Operators will **IMMEDIATELY** ___**(2)**___.

A. (1) 1A
(2) start up and load the respective RPS MG set

B. (1) 1A
(2) **STOP** all testing that could cause RPS half SCRAMS or PCIS Logic Isolation

C. (1) 1B
(2) start up and load the respective RPS MG set

D. (1) 1B
(2) **STOP** all testing that could cause RPS half SCRAMS or PCIS Logic Isolation

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that start up and load the respective RPS MG set is a subsequent action in 1-AOI-99-1.

B **CORRECT**: *(See attached)* In accordance with the given Alarm Response Procedure for REACTOR PROTECTION 120V POWER SYSTEM ABNORMAL, (1-9-5B, Window 10), RPS MG Set A or B has an abnormal condition and de-energizing one RPS channel will actuate the PCIS inboard/outboard logic, refer to 1-AOI-99-1, Loss of Power to One RPS Bus. 1-AOI-99-1 list the automatic action of inboard and outboard isolation of PCIS Group 3, RWCU on a loss of RPS A. Additionally, 1-OI-69, RWCU System, states RPS Bus A(B) impacts to RWCU valves. If RPS Bus 1A is being transferred, then both 1-FCV-69-01(02), RWCU INBOARD (OUTBOARD) SUCTION ISOLATION VALVEs will be CLOSED ahead of time. For second part, in accordance with 1-AOI-99-1, the Operators are required to immediately STOP all testing that could cause RPS half SCRAMs or PCIS logic isolation signals.

C INCORRECT: First part is incorrect but plausible in that 1-AOI-99-1 list the automatic action of outboard isolation of PCIS Group 3, RWCU on a loss of RPS B. Second part incorrect but plausible *(See A).*

D INCORRECT: First part is incorrect but plausible *(See A).* Second part correct (*See B).*

RO Level Justification: Tests the candidate's ability to predict and/or monitor changes in parameters associated with operation of the Reactor Protection System as it relates to RPS bus status. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution as it relates to both RPS and PCIS.

Technical Reference(s): 1-OI-69, Rev. 87 (Attach if not previously provided)
1-AOI-99-1, Rev. 23
1-ARP-9-5B, Rev. 25

Proposed references to be provided to applicants during examination: **REACTOR PROTECTION 120V POWER SYSTEM ABNORMAL, (1-9-5B, Window 10)**

Learning Objective: OPL171.028 Obj. 10a (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK OPL171.074-01 15 #2328
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

Form 4.2-1	**Written Examination Question Worksheet**	
10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

2328. OPL171.074-01 015

Unit 1 is at 100% Power when the following occur:
- REACTOR PROT 120V PWR SYS ABNORMAL (1-XA-55-5B, Window 10) is in alarm.
- REAC BLDG VENTILATION ABNORMAL (1-XA-55-3D, Window 3) is in alarm.

The OATC notes the following:
- 1-FCV-069-0002, RWCU OUTBD SUCT ISOL VLV, and 1-FCV-069-0012, RWCU SYS RETURN ISOL VLV, are CLOSED.
- 1-FCV-001-0056, MAIN STEAM LINE DRAIN OUTBD ISOL VLV, is CLOSED.

1) RPS _____ has been de-energized.

2) Which of the following is an Operator IMMEDIATE ACTION for 2-AOI-99-1, Loss of Power to One RPS Bus?

A. 1) 1A
2) STOP all testing with potential RPS half-scrams or PCIS logic isolation signals.

B. 1) 1A
2) DISPATCH an Operator to the Aux Instrument Room to reset ATU GROSS FAILURES.

C. 1) 1B
2) DISPATCH an Operator to the Aux Instrument Room to reset ATU GROSS FAILURES.

D. 1) 1B
2) STOP all testing with potential RPS half-scrams or PCIS logic isolation signals.

Excerpt from 1-ARP-9-5B:

BFN Unit 1	Panel 9-5 1-XA-55-5B	1-ARP-9-5B Rev. 0025 Page 13 of 42

REACTOR PROT
120V PWR SYS
ABNORMAL

10

(Page 1 of 1)

Sensor/Trip Point:

Bus A	42T-A circuit breaker
Bus B	42T-B circuit breaker
Bus A	1A1 circuit protector
Bus A	1A2 circuit protector
Bus B	1B1 circuit protector
Bus B	1B2 circuit protector
Regulating Transformer TRP-1	1C1 circuit protector
Regulating Transformer TRP-1	1C2 circuit protector

Sensor Location: Bat Board 1, Panel 9A, Bus A and B
All circuit protectors Battery Board Room 1.

Probable Cause:
A. Circuit breaker open.
B. RPS MG set A or B abnormal.
C. One channel RPS on alternate power supply.
D. Circuit protector tripped on over voltage, under voltage or under frequency.
E. Sensor malfunction.
F. Cleared fuse for the control power to any circuit protector.

Automatic Action: De-energizing one RPS channel will actuate every scram-related alarm for the affected channel and actuate the PCIS inboard/outboard logic.

Operator Action:
A. **REFER TO** 1-AOI-99-1.
B. **REFER TO** Tech spec Sect 3.3.8.2.

References: 1-45E641-3, -4 1-45E620-6-2

Excerpts from 1-AOI-99-1:

BFN Unit 1	Loss of Power to One RPS Bus	1-AOI-99-1 Rev. 0023 Page 5 of 17

3.0 AUTOMATIC ACTIONS

NOTES

An overview of the automatic actions is provided here. A detailed list of the actions is provided in 1-OI-99, Illustration 1 which lists the actions that occur when RPS buses are de-energized on a transfer of power supply.

A. An RPS trip logic A(B) half scram occurs.

B. One-half of the PCIS Group 1 trip logic is de-energized.

C. Inboard (outboard) isolation of PCIS Group 2, shutdown cooling mode of RHR.

D. Inboard and outboard isolation of PCIS Group 3, RWCU on loss of RPS A and outboard isolation on loss of RPS B.

E. Inboard and outboard isolation of PCIS Group 6, primary containment vent and purge, reactor building ventilation.

F. Group 8, TIP.

G. Control Bay Emergency Pressurization System A & B start.

H. Standby Gas Treatment System starts.

Also Supports Distractors A2, C2:

BFN Unit 1	Loss of Power to One RPS Bus	1-AOI-99-1 Rev. 0023 Page 6 of 17

4.0 OPERATOR ACTIONS

4.1 Immediate Action

[1] **STOP** all testing that could cause RPS half scrams or PCIS Logic isolation signals.

4.2 Subsequent Actions

NOTES

1) If power can **NOT** be restored promptly to the de-energized RPS bus, plant operation may continue until repairs can be made provided all plant operational limits can be met. With Reactor Building ventilation isolated, the main steam tunnel area temperature can rise to the PCIS Group 1 isolation trip level in less than 10 minutes.
2) Loss of RPS will isolate 1-RM-90-256, Drywell Air Monitor, and TS LCO 3.4.5 Condition B should be entered.

[1] **ENSURE** automatic actions have occurred.

[2] **IF** desired to place RPS Bus A on Alternate Supply, **THEN**

PERFORM Step 4.2[7]. (Otherwise N/A)

[3] **IF** desired to place RPS Bus B on Alternate Supply, **THEN**

PERFORM Step 4.2[8]. (Otherwise N/A)

[4] **IF** desired to Start up and Load RPS MG Set 1A, **THEN**

PERFORM Step 4.2[9]. (Otherwise N/A)

[5] **IF** desired to Startup and Load RPS MG Set 1B, **THEN**

PERFORM Step 4.2[10]. (Otherwise N/A)

[6] **IF** Alternate RPS Supply **AND** RPS MG Set unavailable, **THEN**

PERFORM the following

[6.1] **ATTEMPT** to determine the cause of loss of RPS Circuit Protector by using the indicating lights inside the RPS cabinets.(Otherwise N/A)

[6.2] **CONTACT** Electrical Maintenance to correct the cause.

Excerpts from 1-OI-69:

BFN Unit 1	Reactor Water Cleanup System	1-OI-69 Rev. 0087 Page 121 of 155

8.11 Temporary Shut Down of RWCU System for RPS Bus Power Transfer

NOTES

1) This Section is performed in conjunction with 1-OI-99 while transferring RPS power supplies.
2) RWCU Pump seal purge is required during pump shut down unless maintenance activities require pump isolation. With seal purge in service, a flow path is established with the performance of Step 8.11[9].

[1] **REVIEW** Precautions and Limitations. REFER TO Section 3.0.

[2] **NOTIFY** Chemistry that RWCU is to be secured and **PERFORM** the following:

- **IMPLEMENT** sampling as required in 1-SI-4.6.B.1-4.
- **CHECK** the durability monitor and **EVALUATE** the need to remove it from service.

[3] **IF** Reactor Water Cleanup System return to service is to be delayed, **THEN**

NOTIFY Reactor Engineering for Heat Balance adjustments (Otherwise **N/A**).

[4] **REMOVE** the RWCU filter demineralizers from service.
REFER TO Section 6.1

[5] **STOP** RWCU PUMP 1A, using 1-HS-69-4A-A.

[6] **STOP** RWCU PUMP 1B, using 1-HS-69-4B-A.

[7] **IF** RPS Bus 1A power is to be transferred, **THEN:**

CLOSE the following valves: (Otherwise **N/A**)

- RWCU INBD SUCT ISOLATION VALVE, 1-FCV-069-0001 using 1-HS-69-1.
- RWCU OUTBD SUCT ISOL VLV, 1-FCV-069-0002 using 1-HS-69-2.
- RWCU SYS RETURN ISOL VLV, 1-FCV-069-0012 using 1-HS-69-12A.

Supports Distractors C1, D1:

BFN Unit 1	Reactor Water Cleanup System	1-OI-69 Rev. 0087 Page 122 of 155

8.11 Temporary Shut Down of RWCU System for RPS Bus Power Transfer (continued)

[8] **IF** RPS Bus 1B power is to be transferred, **THEN:**

CLOSE the following valves: (Otherwise **N/A**)

- RWCU OUTBD SUCT ISOL VLV, 1-FCV-069-0002 using 1-HS-69-2A.
- RWCU SYS RETURN ISOL VLV, 1-FCV-069-0012 using 1-HS-69-12A.

[9] **IF** seal purge is required, **THEN**

PERFORM ONE of the following to maintain a flow path:

[9.1] **THROTTLE** RWCU RETURN ISOLATION VALVE, using 1-HS-69-12A.

OR

[9.2] **ESTABLISH** Blowdown flow path by performing the following:

[9.2.1] **OPEN** RWCU BLOWDOWN TO MAIN CNDR, using 1-HS-69-16A.

[9.2.2] **POSITION** 1-HS-69-15A, DEFEAT/OPERATE (FOR 1-HC-69-15) to the **OPERATE** position.

[9.2.3] **THROTTLE OPEN** the RWCU BLOWDOWN PRESS CONTROLLER, using 1-HC-69-15.

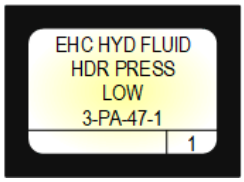
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295005 (APE 5) Main Turbine Generator Trip / 3	Tier #	1	------
G2.1.19 (10CFR 55.41.7)	Group #	1	------
Ability to use available indications to evaluate system or component status	K/A #	295005G2.1.19	
	Importance Rating	3.9	------

Proposed Question: **# 42**

Unit 3 is operating at 100% RTP when the following conditions occur:

- 3A Electro-Hydraulic Control (EHC) Pump is running
- An electrical fault trips 480V Unit Board 3B
- EHC HYDRAULIC FLUID HEADER PRESSURE LOW, (3-9-7B, Window 1) alarms
- 3-PI-47-7, EHC HEADER PRESSURE, currently indicates 1200 psig



Given the conditions above, 3B EHC Pump **(1)** received an automatic start signal and a Main Turbine trip **(2)** occurred.

A. (1) has
(2) has

B. (1) has
(2) has NOT

C. (1) has NOT
(2) has

D. (1) has NOT
(2) has NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses the EHC Pump power supplies since numerous components at BFN are not powered by conventional alignment means. Second part is incorrect but plausible if the candidate confuses the given EHC Header Pressure indication along with the given alarm to conclude that a Main Turbine trip signal has been reached.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See attached)* In accordance with the given Alarm Response Procedure for 3-9-7B, Window 1, alarms when EHC Header Pressure lowers to 1300 psig, the standby EHC Pump gets an automatic start signal. However, given that 480V Unit Board 3B has tripped, 3B EHC Pump has not automatically started since it is de-energized. For second part, in accordance with 3-OI-47, Turbine-Generator System, a Main Turbine trip will occur when EHC Fluid Actuator Supply (FAS) Pressure is less than 1100 psig which currently not the case. Therefore, the Unit 3 Main Turbine has not tripped.

RO Justification: Tests the candidate's ability to use available indications to evaluate system or component status as it relates to a Main Turbine trip and AC Electrical Distribution. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 3-OI-47, Rev. 135 (Attach if not previously provided)
3-ARP-9-7B, Rev. 31
OPL171.230, Rev. 5U1

Proposed references to be provided to applicants during examination: **EHC HYDRAULIC FLUID HEADER PRESSURE LOW, (3-9-7B, Window 1)**

Learning Objective: OPL171.230, Obj. 13 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.230-05 002 #3436
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

3436. OPL171.230-05 002

Given the following conditions:

- Unit 3 is operating at 75% power due to elevated river temperature
- An electrical fault results in the de-energization of 3B 480V Unit Board
- Subsequent to this, the following occurs:
  - Turbine Building AUO reports a significant EHC leak in the Turbine Building
  - EHC HYD FLUID HDR PRESS LOW (3-ARP-9-7B Window 1) alarm is received.
  - EHC HEADER PRESSURE 3-PI-47-7 currently indicates 1200 psig

Which ONE of the following completes the statements below?

3B EHC Pump is currently ___(1)___.

A Main Turbine Trip has ___(2)___.

A. (1) Running
(2) Occurred

B. (1) Running
(2) NOT occurred

C. (1) NOT Running
(2) Occurred

D. (1) NOT Running
(2) NOT occurred

Excerpt from 3-ARP-9-7B:

BFN Unit 3	Panel 9-7 3-XA-55-7B	3-ARP-9-7B Rev. 0031 Page 4 of 49

EHC HYD FLUID
HDR PRESS
LOW
3-PA-47-1

1

(Page 1 of 1)

Sensor/Trip Point:
3-PS-47-1C (PS-30) 1300 psig

Sensor Location: EHC pump unit
El 586'
Turbine Bldg

Probable Cause:
A. Inservice pump trip.
B. Inservice filter plugged.
C. Hyd fluid piping break.
D. Sensor malfunction.

Automatic Action:
A. Standby pump starts.
B. Red light above EHC PUMP 3B(3A) TEST pushbutton 3-HS-47-5A(4A) for selected pump extinguishes at 1300 psig lowering, illuminates at 1500 psig rising.

Operator Action:
A. **ENSURE** Standby EHC PUMP 3B(3A), 3-HS-47-2A(1A) running.
B. **CHECK** EHC HEADER PRESSURE, pressure indicator, 3-PI-47-7 between 1550 and 1650 psig.
C. **DISPATCH** personnel to inspect EHC pump unit.

NOTE
On EHC Hydraulic System failure, accumulator and check valve arrangement will provide approximately one minute bypass valve operation.

D. **IF** EHC Hydraulic system fails, **THEN** **ENSURE** turbine trips at or below 1100 psig.

References: 3-45E620-10 0-45E602-6 3-45E602-3

Excerpt from 3-OI-47:

BFN Unit 3	Turbine-Generator System	3-OI-47 Rev. 0135 Page 266 of 284

Attachment 4
(Page 1 of 6)

Turbine Trip Logic

The following is a list of main turbine trips:

A.	Generator trips	**REFER TO** page 3 of this Attachment for various generator trips
B.	Electrical overspeed	108% (1944 rpm)
C.	Electrical backup(Protech) overspeed	109% (1962 rpm)
D.	All speed sensors failed	Four passive speed probes supply two separate speed inputs to each micronet processor (kernal) for a total of eight speed inputs. When seven out of the eight speed inputs have completely failed or are out of range, a turbine trip will occur.
E.	Turbine failed to accelerate	Occurs when control valves reach 25% open and the turbine has not rolled off the turning gear.
F.	MSOP discharge pressure low	≤ 105 psig if turbine speed is > 1300 rpm
G.	Bearing Oil pressure low	≤ 8 psig
H.	Thrust Bearing Wear	± 35 mils
I.	EHC ETS Fluid low pressure	< 850 psig
J.	EHC FAS oil pressure low	< 1100 psig
K.	Moisture Separator high level	Level in drain tank at approx El. 594'9"
L.	Stator Cooling high temperature	≥ 85°C (185°F) 60 second time delay
M.	Stator Cooling low flow	≤ 508 gpm if ≥ 7726 amps 60 second time delay
N.	Power Load Unbalance (PLU)	> 40% mismatch between reheat pressure (LP inlet pressure representing turbine power) and generator current (representing generator load)
O.	High Reactor water level	+ 55 inches (Level 8)
P.	High Vibration	10 mils sensed by combinations of adjacent bearings. **REFER TO** page 5 of this Attachment for the specific combinations.

Excerpt from OPL171.230 Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
devices in the trip circuit	
2. **Design**: The HPU is a standard skid mounted unit which includes the fluid reservoir, system pumps, coolers, and filtering system	
3. **Components**	
a. **Fluid Reservoir**	
1) Total volume of 800 gallons, normally holds 500-550 gallons of hydraulic fluid	
2) Normal level is approximately 12" - 14" below the top of the tank	
a) Fluid level can be read locally and is referenced to "0" at Low level alarm	Obj. B.7(OF-2) Procedure Adherence
b) Abnormal levels actuate annunciators in the control room	What consequence is low fluid level?
(1) High - +16" above 0" (2" below top of tank)	
(2) Low - 0" (18" below top of tank)	
(3) +2" and +8", GOI-300-1 Low and High level limit	
3) Tank has internal baffles to allow time for any air entrained in the returned fluid to escape before the fluid reaches the pump suction	
4) The reservoir is vented to atmosphere through an air dryer/breather which uses activated alumina to help keep moisture out of the tank. 23-HT is a temperature detector which alarms in the control room if fluid temperature reaches 65°F decreasing or 150°F increasing	Obj. B.7 (OF-2)
5) If fluid temperature drops below 85°F, a thermostatically controlled set of heaters and fans automatically come on to blow hot air around the tank. There are no submersible heaters in the tank	Obj.B.3 (OF-1)
6) An alarm in the control room occurs if the heater control switches are placed in the off position	
a) HS-47-3B in off	
b) HS-47-3C in safe stop position	
b. **EHC Fluid Pumps**	
1) Each unit is equipped with two 100%, variable stroke, positive displacement pumps	Obj. B.9

OPL171.230 , Electro-Hydraulic Control (EHC) System: Hydraulics, Rev# 5U1

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
2) The pumps are 480 AC volt motor driven powered from the respective 480V Unit Board	
3) Each pump has a wire mesh - strainer on the suction side and a pair of full-flow combined 20 micron and 10 micron filters on the discharge side.	
a) The combined full-flow filters contain a disposable primary element (cartridge)	
b) The discharge full-flow filters have non-adjustable pressure switches which alarm in the control room when filter differential pressure exceeds 100 psid for 30 seconds	
c) The discharge filters have an auto bypass feature to maintain header flow if the filter becomes clogged	
d) The suction strainer and secondary elements are cleanable	
e) The suction strainers have pointer indications that the FILTER IS CLEAN OR FILTER NEEDS CLEANING OR BYPASSING	
4) An auxiliary pump is provided to support the main EHC pumps	
a) The auxiliary pump is driven by a chain and sprocket drive from the main pump shaft	
b) The auxiliary pump draws fluid directly from the hydraulic reservoir and supplies the main pump internals with fluid for lubrication and cooling.	
5) A pressure compensator on each pump maintains a preset pressure throughout the delivery range. That is, if system flow demand changes the pump automatically adjusts its stroke to just meet flow demand while maintaining system pressure	
6) The pumps are adjusted to deliver 1600 psig operating fluid pressure. Relief valves in the discharge lines are set to relieve pump discharge back to the tank should fluid header pressure exceed 2000 psig	
7) The pumps are operated with one in service and one on standby, controlled from the main control room	
a) At 1300 psig decreasing pressure on the running pump (PS-47-1A or PS-47-2A) the	Attention to detail: right unit/train/ component

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

Equipment Control

G2.2.39 (10CFR 55.41.7)

Knowledge of less than or equal to 1 hour technical specification action statements (This K/A does not include action statements of 1 hour or less that follow the expiration of a completion time for a technical specification condition for which an action statement has already been entered.)

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.39	

Importance Rating	3.9	

Proposed Question: **# 43**

In accordance with **TECHNICAL SPECIFICATIONS (TECH SPECS)**, if Reactor Steam Dome Pressure reaches 1200 psig, the REQUIRED ACTION to restore Pressure must be completed within _____.

A. 15 minutes

B. 30 minutes

C. 1 hour

D. 2 hours

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* Correct in that in accordance with Tech Spec 3.4.10, Reactor Steam Dome Pressure shall be ≤1050 psig. The given Reactor Steam Dome Pressure requires this pressure to be restored within the limit of ≤1050 psig within 15 minutes.

B INCORRECT: Incorrect but plausible in that in accordance with Tech Spec 3.4.9, RCS Pressure and Temperature Limits, RCS pressure, RCS temperature must be maintained within the limits of Figures 3.4.9.1 and 3.4.9.2 (Pressure vs. Temperature Limit Curves) at all times. If a violation of the Pressure vs. Temperature Limits occurs according to these graphs, parameters (Pressure and/or Temperature) must be restored within limits within 30 minutes. There are numerous Technical Specification requirements that have 1 hour or less Action Statements, and Unit Operators are responsible for this information. Therefore, it is easy to confuse these requirements.

C INCORRECT: There are numerous Technical Specification requirements that have 1 hour or less Action Statements, and Unit Operators are responsible for this information. Therefore, it is easy to confuse these requirements.

D INCORRECT: Incorrect but plausible in that in accordance with the Safety Limits (Tech Spec 2.0), action must be taken to restore Reactor Pressure below the Safety Limit within 2 hours. The Reactor Pressure Safety Limit is 1325 psig – Reactor Pressure has not exceeded this limit in the question.

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's knowledge of less than or equal to 1 hour Technical Specification Requirements. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U2 Tech Spec 3.4.10, Amend. 254 (Attach if not previously provided)
U2 Tech Spec 2.0, Amend. 217
U2 Tech Spec 3.4.9, Amend. 253

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.087 Obj. 9 (As available)

Question Source: Bank #
Modified Bank # BFN NRC 2104 #64 (Note changes or attach parent)
New

Question History: Last NRC Exam 2021

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

In accordance with Reactor Pressure Safety Limits, which **ONE** of the following completes the statements below?

The Reactor Steam Dome Pressure Safety Limit is **(1)**.

If violated, the REQUIRED ACTION is to restore Reactor Steam Dome Pressure Safety Limit compliance and insert **ALL** insertable Control Rods within **(2)**.

A. (1) 1325 psig
(2) 2 hours

B. (1) 1325 psig
(2) 10 hours

C. (1) 1375 psig
(2) 2 hours

D. (1) 1375 psig
(2) 10 hours

Proposed Answer: **A**

Excerpt from Tech Spec 3.4.10:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1050 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

Excerpt from Tech Spec 2.0:

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.06 for two recirculation loop operation or ≥ 1.08 for single loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Excerpt from Tech Spec 3.4.9:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1	Restore parameter(s) to within limits.	30 minutes
	AND		
	A.2	Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

(continued)

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
259002 (SF2 RWLCS) Reactor Water Level Control	Tier #	2	------
K1.16 (10CFR 55.41.7)	Group #	1	------
Knowledge of the physical connections and/or cause and effect relationships between the Reactor Water Level Control System and the following systems:	K/A #	259002K1.16	
• ECCS	Importance Rating	3.5	------

Proposed Question: **# 44**

Unit 3 is operating at 100% RTP when the following conditions occur:

- Reactor Pressure is at normal operating pressure
- Reactor Water Level is at normal operating level and slowly rising
- Drywell Pressure is 1.4 psig and steady
- Operator just reported that HPCI has initiated and is injecting
- **NO** Operator Actions have been taken

Given the conditions above, HPCI is injecting into ___**(1)**___ Feedwater System line and RFWCS will be in ___**(2)**___ Element Control.

A. (1) 'A'
(2) Single

B. (1) 'A'
(2) Three

C. (1) 'B'
(2) Single

D. (1) 'B'
(2) Three

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) Drawings and Lesson Plans illustrate that the HPCI injection flow path is from the CSTs to the HPCI Pump then discharges to the 'A' Feedwater System line. For second part, at normal operating conditions, RFWCS is in Three Element Control if all the permissives are met. HPCI is inadvertently injecting as given since there is not an initiation signal present. Since HPCI injects into the 'A' Feedwater line, this will be indicated on 3-FI-3-78A, FW Line 'A' Flow , lowering and 3-FI-3-78B, FW Line B Flow, rising to the point that both will be automatically bypassed. This mismatch results in two bad Feedwater Line flow signals no longer meeting the Three Element Control criteria, therefore RFWCS automatically swaps to Single Element Control.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that when operating at normal conditions, RFWCS is in Three Element Control if all the permissives are met.

C INCORRECT: First part is incorrect but plausible in that RCIC discharges to the 'B' Feedwater System line. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the physical connections and/or cause and effect relationship between the Reactor Water Level Control System and ECCS System. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-OI-3, Rev. 115 (Attach if not previously provided)
3-AOI-3-1, Rev. 17
OPL171.040, Rev. 25U1
OPL171.042, Rev. 32
1-47E610-3-1, Rev. 33

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171. 012, Obj. 3 (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

New	**X**
Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpt from OPL171.042 Lesson Plan:

The design of the ECCS Systems and the Reactor Pressure Vessel (RPV) must provide for adequate core cooling for all breaks, including the worst-case scenario. Specific criteria are stipulated in Code of Federal Regulations 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors."

II. Presentation

Obj. ILT 1-OF5
Obj. LOR 1
Obj. NLO 1

A. General Description

The High Pressure Coolant Injection System (HPCI) consists of a steam turbine-driven system driving a constant-flow pump assembly to inject either Condensate Storage Tank (CST) water or Suppression Pool (Torus) water into the Reactor under emergency conditions at the rate of at least 5000 gpm over an 1174 -150 psig Reactor pressure range. Flow controller setpoint is 5300 gpm.

1. System Design Basis
   a) To provide adequate core cooling for all small break sizes (up to approx. 6 in) which **do not** result in rapid depressurization of the Reactor vessel
   b) To function independent of off-site power sources and diesel generators
2. Components

Obj. NLOR 1-OF5
Obj. NLO 2

   a) Turbine
   b) Main and Booster Pumps
   c) Turbine Auxiliaries
3. Flow path
   a) One 100% System
   b) Steam Path

Obj. ILT 1
Obj. LOR 1
Obj. NLO 10

      (1) From B Main Steam Line upstream of the flow restrictor
      (2) Through isolation valves
      (3) Through stop valve and control valves
      (4) Exhaust through check valve to suppression pool
   c) Water Path

Figure 1

      (1) Normal condensate path from CST to the HPCI pump, to the A Feedwater line and into the Reactor vessel
      (2) Alternate suction path from suppression pool
      (3) Automatic swap over to suppression pool on high suppression pool level +5.25" or low CST level Elev < 552'6"

Approx. 7000 gallons left in CST piping when auto swap occurs on low CST Level.

Note: There is a 7-second delay for suction swap on high suppression pool level.

Excerpt from OPL171.040 Lesson Plan:

OPL171.040, Reactor Isolation Cooling (RCIC), Rev.# 32

require only DC power from the station batteries, therefore providing a high degree of assurance that RCIC will operate when required.

Appendix C is a RCIC Operating Experience Simulator Demonstration.

The demonstration is based on Browns Ferry and other BWR plant performance problems. These plant performance problems are derived from the following operating experience:

- PER 00-011480: Discharge Piping Overpressurization
- NRC IN 2000-01: BWR Operational Issues During Rx Scram & Transient
- GE SIL No. 623: Peak Pump Discharge Pressure during SRs
- OE10570: Turbine Trips Due to Check Valve

OE11133: Pump Trip on Lo Suct Press Due To Unfilled Discharge Piping

II. Presentation

A. General Description

1. The purpose of the RCIC System is to provide a source of high pressure coolant makeup to the Reactor vessel in case of a loss of feedwater flow. The system is used to maintain the Reactor water level and for Reactor pressure control under MSIV isolation conditions and loss of normal feedwater.

2. Safety Design Basis

RCIC operates automatically to maintain sufficient coolant in the vessel so that the fuel will not overheat in the event of Reactor isolation and loss of feedwater flow. The system is a consequence limiting system rather than an ECCS system. — Obj. ILT/NLO 1

B. The RCIC System Consists of :

1. Turbine-driven pump located in basement of Reactor Building (Elev. 519) — Obj. NLOR 1, Obj. NLO 2, IL-1 & IL-2
2. Turbine is driven by steam from Main Steam Line "C" and exhausts to the suppression pool.
3. Pump is normally lined up to take suction from the Condensate Storage Tank (CST), but can take suction from suppression pool (only done manually). — Obj. ILT 2.
4. Pump discharges to Reactor via feedwater line B — Obj. NLO 3

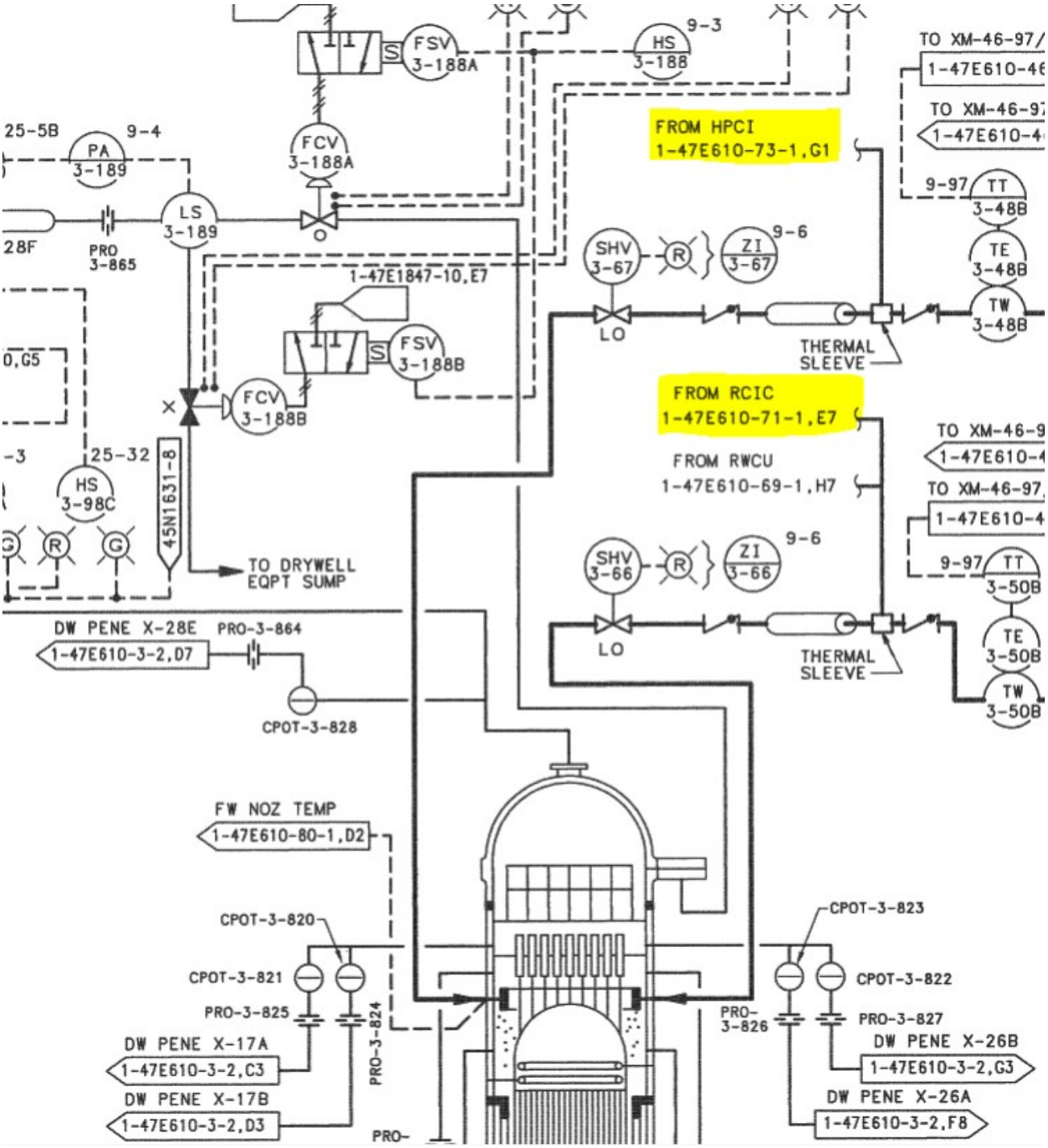
a. Turbine

(1) 100% capacity

(2) Delivers full pump design flow at Reactor pressures of 150 to 1120 psig — Obj. ILT 1

(3) 500 Hp at 1200 psig to 80 Hp at 225 psig

Excerpt from Drawing 1-47E610-3-1, Mechanical Control Diagram Reactor Feedwater System: (Same as Unit 3)



Excerpt from 3-AOI-3-1:

BFN Unit 3	Loss Of Reactor Feedwater or Reactor Water Level High/Low	3-AOI-3-1 Rev. 0017 Page 5 of 15

2.0 SYMPTOMS (continued)

I. Possible mismatch between Feedwater Line Flows (FI-3-78A,78B) during HPCI and/or RCIC inadvertent injections.

J. Possible Reactor power rise due to improved moderation for rising reactor water level and/or Feedwater flow.

3.0 AUTOMATIC ACTIONS

A. The following trip at greater than or equal to +55" (normal range level instruments 3-3-208A, 208B, 208C and 208D):

1. Main Turbine (this will cause a Reactor scram above 26% power)
2. RFPT

B. The following trip at greater than or equal to +51"(normal range level instruments 3-3-208A, 208B, 208C and 208D):

1. RCIC Turbine (if steam supply valve is open)
2. HPCI Turbine

C. Recirc Pumps receive a 75% speed run back signal from the following:

1. Reactor water level at 27" (normal range) and discharge flow of a RFP is less than 16% (rated flow).
2. Reactor Scram.

Excerpt from 3-OI-3:

Attachment 4
(Page 2 of 3)

Single/Three Element Control

3.0 DESCRIPTION

SINGLE ELEMENT An automatic operating mode where one feedback input is used, Narrow Range Reactor water level. In SINGLE ELEMENT control, the operator adjusted Reactor water level setpoint is compared to the average of the valid Narrow Range level signals. SINGLE ELEMENT is used with automatic operation of RFW Startup Level Control PDS and Reactor Water Level Control PDS (with at least one individual RFPT Speed Control PDS in AUTO).

THREE ELEMENT An automatic operating mode using three feedback inputs: Narrow Range water level, Total Steam Flow, and Total Feedwater Flow. In THREE ELEMENT control, the operator adjusted Reactor level setpoint is compared to the average of the valid Narrow Range Reactor water level signals. A resultant level error signal is generated. This level error signal is summed with a Total Steam Flow signal and sent to the Steam Flow - Feed Flow Mismatch control block. The control block takes demand signal and Total Feedwater Flow signal and produces a Steam Flow/Feed Flow signal and is processed through the single element control logic. The demand signal is then sent to RFPT Control logics, where it is combined with automatic flow balance (flow biasing) inputs.

The RFW Control System will **NOT** allow THREE ELEMENT operation until all of the following permissives are met:

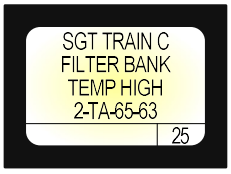
- Reactor Water Level Control (PDS), 3-LIC-46-5, in AUTO and at least one RFPT Speed Control (PDS) in AUTO.
- A valid Total Steam Flow signal exists (Four good steam line flows, or at least two good steam line flow signals and a valid Turbine First Stage pressure signal).
- A valid Total Feedwater line flow signal exists (Two good Feedwater line flow signals, or one good Feedwater line flow and three valid individual RFP flow signals).
- Total Steam Flow >16%.
- At least one valid Narrow Range Level signal.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
261000 (SF9 SGTS) Standby Gas Treatment	Tier #	2	------
K5.01 (10CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Standby Gas Treatment System:	K/A #	262001K5.01	
• Heat removal mechanisms	Importance Rating	3.0	------

Proposed Question: **# 45**

An event has occurred on Unit 2, requiring the use of the Standby Gas Treatment (SGT) System, with the following conditions:



- SGT TRAIN C FILTER BANK TEMP HIGH (2-9-3B, Window 25) alarms
- The Outside AUO reports that SGT 'C' charcoal filter temperature is 150 °F

Given the conditions above, the SGT decay heat removal mode **(1)** initiated.

In accordance with Tech Specs, while in decay heat removal mode, SGT is **(2)**.

A. (1) is automatically
(2) OPERABLE

B. (1) is automatically
(2) INOPERABLE

C. (1) must be manually
(2) OPERABLE

D. (1) must be manually
(2) INOPERABLE

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Standby Gas Trains automatically start when required by plant conditions. However, in accordance with 0-OI-65, Standby Gas Treatment System when filter bank temperature reaches 150 ºF, decay heat removal must be manually initiated. Second part is incorrect but plausible in that decay heat removal is a normal system operation, which normally would not be considered as an inoperability concern.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT**:** First part is correct (*See D*). Second part is incorrect but plausible (*See A*).

D **CORRECT**: *(See attached)* In accordance with 2-ARP-9-3B, Window 25 and 0-OI-65, Staandby Gas Treatment System, the operators must take action to initiate decay heat removal on Standby Gas Trains. For second part, in accordance with 0-OI-65, an open decay heat removal damper renders a Standby Gas Train inoperable for Secondary Containment purposes.

RO Level Justification: Tests the candidate's knowledge of the Standby Gas Treatment System design features and the operational implications of decay heat removal. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-65, Rev.56 (Attach if not previously provided)
2-ARP-9-3B, Rev. 40
OPL171.018, Rev. 12U1

Proposed references to be provided to applicants during examination: **SGT TRAIN C FILTER BANK TEMP HIGH (2-9-3B, Window 25)**

Learning Objective: OPL171.018, Obj 6e (As available)

Question Source: Bank # (Note changes or attach parent)
Modified Bank # BFN 2204 NRC #18
New

Question History: Last NRC Exam ILT 2204

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Copy of Bank Question:

Examination Outline Cross-reference:

261000 (SF9 SGTS) Standby Gas Treatment

K4.02 (10CFR 55.41.7)

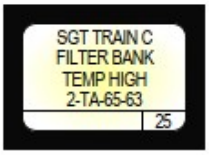
Knowledge of Standby Gas Treatment System design features and/or interlock that provide for the following:

- Charcoal bed decay heat removal

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	261000K4.02	
Importance Rating	3.0	------

Proposed Question: **# 18**

An event has occurred on Unit 2, requiring the use of Standby Gas Treatment (SGT) System, with the following conditions:



- SGT TRAIN C FILTER BANK TEMP HIGH (2-9-3B, Window 25) alarms
- The Outside AUO reports that SGT 'C' charcoal filter temperature is 150 ºF

Given the conditions above, the SGT decay heat removal mode **(1)** initiated and the carbon bed filters **(2)** designed to remove iodine.

A. (1) is automatically
(2) are

B. (1) is automatically
(2) are NOT

C. (1) must be manually
(2) are

D. (1) must be manually
(2) are NOT

Proposed Answer: **C**

Excerpt from 2-ARP-9-3B:

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0040 Page 29 of 39

SGT TRAIN C FILTER BANK TEMP HIGH
2-TA-65-63
25

(Page 1 of 1)

Sensor/Trip Point:
TE-65-63B TS-65-63 150°F

Sensor Location: Standby Gas Treatment Building
Charcoal Filter Bank C

Probable Cause:
A. Decay heat.
B. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **IF** high temperature is due to iodine absorption, **THEN**
REFER TO 0-OI-65, Section 8.0, **AND**
OPERATE SGT system in Decay Heat Removal mode.

B. **DISPATCH** personnel to SGT Bldg to reset temperature switch, TS-65-63A and TS-65-63B, locally (located in JB2 on north side of SGT train).

C. **NOTIFY** Unit SRO and Unit 1 and 3 operators.

D. **REFER TO** Tech Spec Section 3.6.4.3.

References:

45E620-3	0-47E610-65-1	45E614-16
45E771-2	Technical Specifications 3.6.4.3	

Excerpt from 0-OI-65:

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0056 Page 8 of 44

3.0 PRECAUTIONS AND LIMITATIONS

A. Upon a secondary containment isolation, the SGT System is designed to maintain a negative 1/4-inch of H_20 vacuum in Secondary Containment with an inleakage flow of 12,000 cfm.

B. [NRC/C] All three trains will remain in operation during an accident to satisfy single failure criteria and to minimize the potential release of radioactivity from the Reactor Building into the Control Building air supply intake ducts. [NRC NCO 88 0193 004]

C. [NER/C] Steps should be taken to minimize dust loading and to prevent paint vapors, petroleum fumes, welding smoke, and other airborne contaminants from reaching the HEPA filters and charcoal adsorbers. Normal ventilation should be in operation for a minimum of two (2) hours after painting, fire, smoke, or chemical release has terminated prior to operating SGT System. [CAQR SQP890064]

D. If the SGT System is run within 16 hours of the completion of painting in the areas specified in MAI-5.3 or MAI-5.7, Control of Volatile Organic Compounds section, a determination is to be made using those procedures as to whether additional actions are required to verify SGT System operability. Exceeding MAI-5.7 limits requires performing 0-SR-3.6.4.3.2(A)(B)(C) to verify SGT can perform its intended function.

E. When all SGT Trains are secured and any evolution has the potential to discharge radioactive effluents through the main stack, one Unit 2 and one Unit 3 Stack Dilution Fan should remain in operation. This requirement provides clean air flow through the dilution cross-tie to SGT ducts. This prevents the potential back flow of radioactive effluents through the SGT duct work.

F. The alignment of SBGT trains to perform the PURGING function cannot be used when the average reactor coolant temperature is above 212°F since a postulated LOCA could impact the ability for the SBGT trains to perform their safety function. If the primary containment purge system is inoperable and the average reactor coolant temperature is less than or equal to 212°F, the standby gas treatment system venting path will provide the required filtration. The standby gas treatment system is **NOT** the normal means for PURGING operations since the vent path from containment is a much more restrictive flowpath (slower) than the purge system.

G. In the event that the train charcoal filter temperature rises to 150°F due to iodine adsorption following a LOCA, decay heat removal mode of operation should be initiated when the train is no longer in service.

H. An open decay heat removal damper in a particular train renders that train inoperable for Secondary Containment purposes.

Excerpt from OPL171.018:

OPL171.018, Standby Gas Treatment (SGT) System, Rev.12U1

c) Moisture drains by gravity to SGT sump and is then pumped to Radwaste	LOR/NLOR Objective 4a NLO Objective 2
4. Relative Humidity Heater	
a) The relative humidity heater reduces relative humidity to <70%.	
b) 40kW heaters for relative humidity control. SGT A and B powered from A and B 480V DG Aux Boards respectively. SGT C from the 480V SGT Board.	ILT Objective 6b LOR/NLOR Objective 4b NLO Objective 2
c) 15kW charcoal bed heaters formerly maintained a 125ºF charcoal bed temperature when SGT was out of service. Heater control switches were spring-return-to-neutral and required resetting after SGT operation. Due to the Technical Specification requirement of 10 hours monthly operation with the relative humidity heaters in service, the charcoal bed heaters are no longer needed.	
d) The relative humidity heater is energized automatically on startup by the fan breaker closure and is de-energized on shutdown by fan breaker opening.	
e) The heater will also trip if ambient temperature reaches 180ºF.	
f) If 480V Load Shed logic is initiated, the Train A and B relative humidity heaters will automatically trip. They will restart after 40 seconds. Train C is not affected by the 480V load shed logic.	
g) Relative humidity heater control switches (12, 34, 60) in ON or OFF cause annunciation.	
5. Roughing Filter (Prefilter)	
Used to remove large particles (dust, dirt, lint) and to protect HEPA filter.	ILT Objective 6c LOR/NLOR Objective 4c NLO Objective 2
6. HEPA filter	
Removes 99.9% of 0.3 micron particles	
7. Charcoal Bed (Adsorber Type)	ILT Objective 6d LOR/NLOR Objective 4d NLO Objective 2 ILT Objective 6e LOR/NLOR Objective 4e NLO Objective 2
a) Designed to remove at least 95% of iodine in the form of methyl iodine (CH3I) and 99.9% of elemental iodine upon entering conditions of 70% relative humidity at 190ºF.	
b) Made up of individual rectangular canisters of activated charcoal.	
c) Vertical air flow	
8. HEPA afterfilter	
a) Identical to first HEPA filter	ILT Objective 6f LOR/NLOR Objective 4f

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
262001 (SF6 AC) AC Electrical Distribution	Tier #	2	------
K3.03 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the AC Electrical Distribution will have on the following systems or system parameters:	K/A #	262001K3.03	
• DC Electrical Distribution	Importance Rating	3.9	------

Proposed Question: **# 46**

Unit 3 was operating at 100% Rated Thermal Power (RTP) when the following occurs:

- Loss of Offsite power
- Drywell Pressure is 20 psig
- Reactor Pressure is 350 psig and lowering
- 3A, 3B, 3D EDGs start and tie to their respective 4KV Shutdown Boards
- 3C EDG is tagged out of service

Given the conditions above, 250V Battery Charger(s) **(1)** has(have) under gone load shed and 480V Common Board **(2)** is the alternate 480V AC supply to the affected Battery Charger(s).

A. (1) 3
(2) 1

B. (1) 3
(2) 3

C. (1) 3 **AND** 4
(2) 1

D. (1) 3 **AND** 4
(2) 3

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See Attached)* Unit 3 load shed occurs by division. For load shed to occur, there must be a Common Accident (CASA) signal present with that division's EDG supplying its 4KV Shutdown Board as the only source of power. In a normal power alignment, that's the 3A and 3C 4KV Shutdown Boards. 3C board does not have an EDG tied to it. For second part, 480V Common Board 1 has a single breaker that supplies a common bus that will supply 250V Battery Chargers 1-4, one at a time.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible since 480V Common Board 3 is a Unit 3 board and cannot be aligned to any battery charger.

C INCORRECT: First part is incorrect but plausible since the stem requires the candidate to understand the current electrical system alignments. Additionally, Unit 1 and Unit 2 would have experienced load shed on all divisions of both units. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's knowledge and ability to determine the implications of a Load Shed condition with respect to the Battery Chargers and which power supplies supply alternate AC power to the chargers. This question is rated as C/A due to the candidate being given multiple plant parameters and having to apply power supplies to an integrated series of events.

Technical Reference(s): OPL171.037, Rev.18U1 (Attach if not previously provided)
1/2-AOI-57-1D, Rev. 8
3-AOI-57-1D, Rev.13
0-OI-57/ATT-3F, Rev. 189

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.037 Obj. 7 (As available)

Question Source:

Bank #	
Modified Bank #	(Note changes or attach parent)
New	**X**

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpt from 3-AOI-57-1D:

BFN Unit 3	480V Load Shed	3-AOI-57-1D Rev. 0013 Page 4 of 22

1.0 PURPOSE

The 480V Load Shedding Logic System removes selected loads from 480V boards which are powered from the 4kV Shutdown Boards. Unit 3 Load Shed Logic is divisionalized. If a Unit 3 accident signal is present, **(122" Reactor Water Level or 2.45 psig High Drywell Pressure with 450 psig Low Reactor Pressure) and the appropriate conditions listed below:

DIVISION 1	DIVISION 2
3EA D/G or the U-1/2 crosstie breaker is the source of power to 3EA 4KV S/D BD (DGVA-A) and 3A 480V S/D BD normal feeder Bkr is closed	3EC D/G or the U-1/2 crosstie breaker is the source of power to 3EC 4KV S/D BD (DGVA-C) and 3B 480V S/D BD normal feeder Bkr is closed
OR	OR
3EB D/G or the U-1/2 crosstie breaker is the source of power to 3EB 4KV S/D BD (DGVA-B) and 3A 480V S/D BD alternate feeder Bkr is closed	3EB D/G or the U-1/2 crosstie breaker is the source of power to 3EB 4KV S/D BD (DGVA-B) and 3B 480V S/D BD alternate feeder Bkr is closed
THEN	THEN
Load Shed will occur for the loads on the following boards:	Load Shed will occur for the loads on the following boards:
3A 480V S/D Bd 3A 480V RMOV Bd 3EA 480V D/G AUX Bd	3B 480V S/D Bd 3EB 480V D/G AUX Bd 3B 480V RMOV Bd 3C 480V RMOV Bd *

* Normal Feeder Breaker for 480V RMOV Bd 3C is enabled for manual closure after 40 seconds and will auto CLOSE in 11 minutes

** -122 low-low-low level(Level 1)

Excerpt from 1/2-AOI-57-1D:

BFN Unit 1 & 2	**480V Load Shed**	**1/2-AOI-57-1D Rev. 0008 Page 4 of 26**

1.0 PURPOSE

The 480V Load Shedding Logic System removes selected loads from 480V boards which are powered from the 4kV Shutdown Boards. The load shedding is initiated by an accident signal from the units Core Spray System logic on low-low-low reactor water level (-122") (LEVEL 1) **OR** high drywell pressure (2.45 psig) with low reactor pressure (450 psig) with any Diesel Generator tied to its Shutdown Board or the Unit 3 crosstie breaker closed (DGVA) as the only source of power.

Excerpts from 0-OI-57B/Att-3F:

BFN Unit 0	Attachment 3F Unit 3 480V Shutdown Boards and Reactor MOV Boards Electrical Lineup Checklist	0-OI-57B/ATT-3F Rev. 0189 Page 6 of 32

3.0 ATTACHMENT DATA (continued)

Performed On: _____

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
	480V Shutdown Bd 3A Control Bay - Div I - El 621', R-20R (3-45E749 - 5)			
5C	3-BKR-068-0003A1 VFD 3A COOLANT PUMP 3A1		3	ALIGNED BY 3-OI-68
5D	3-BKR-068-0003B1 VFD 3B COOLANT PUMP 3B1		3	ALIGNED BY 3-OI-68
6A	3-BKR-047-0011 TURNING GEAR OIL PUMP		3	ALIGNED BY 3-OI-47B
6B	3-BKR-070-0005 CLOSED COOLING WATER PUMP 3A		3	ALIGNED BY 3-OI-70
6C	3-BKR-069-0004A RWCU RECIRCULATION PUMP 3A		3	ALIGNED BY 3-OI-69
6D	3-BKR-248-0003A/6D 250V BATTERY CHARGER 3		3	ALIGNED BY 0-OI-57D
7A	3-BKR 231-0003A/7A EMERGENCY FEEDER FOR 480V RMOV BD 3E	(2) CONNECTED AND OPEN	3	____ ____
7B	3-BKR-063-0006A 3A SLC PUMP MOTOR BREAKER		3	ALIGNED BY 3-OI-63
7C	3-BKR-078-0010 FUEL POOL COOLING PUMP 3A		3	ALIGNED BY 3-OI-78
7D	3-BKR-252-0003A/7D UNIT PREFERRED MG SET 3	CLOSED	3	____ ____
8C	3-BKR-231-0003A/8C EMERGENCY FEEDER FROM 4KV S/D BD 3EB	(2) CONNECTED AND OPEN	3	____ ____

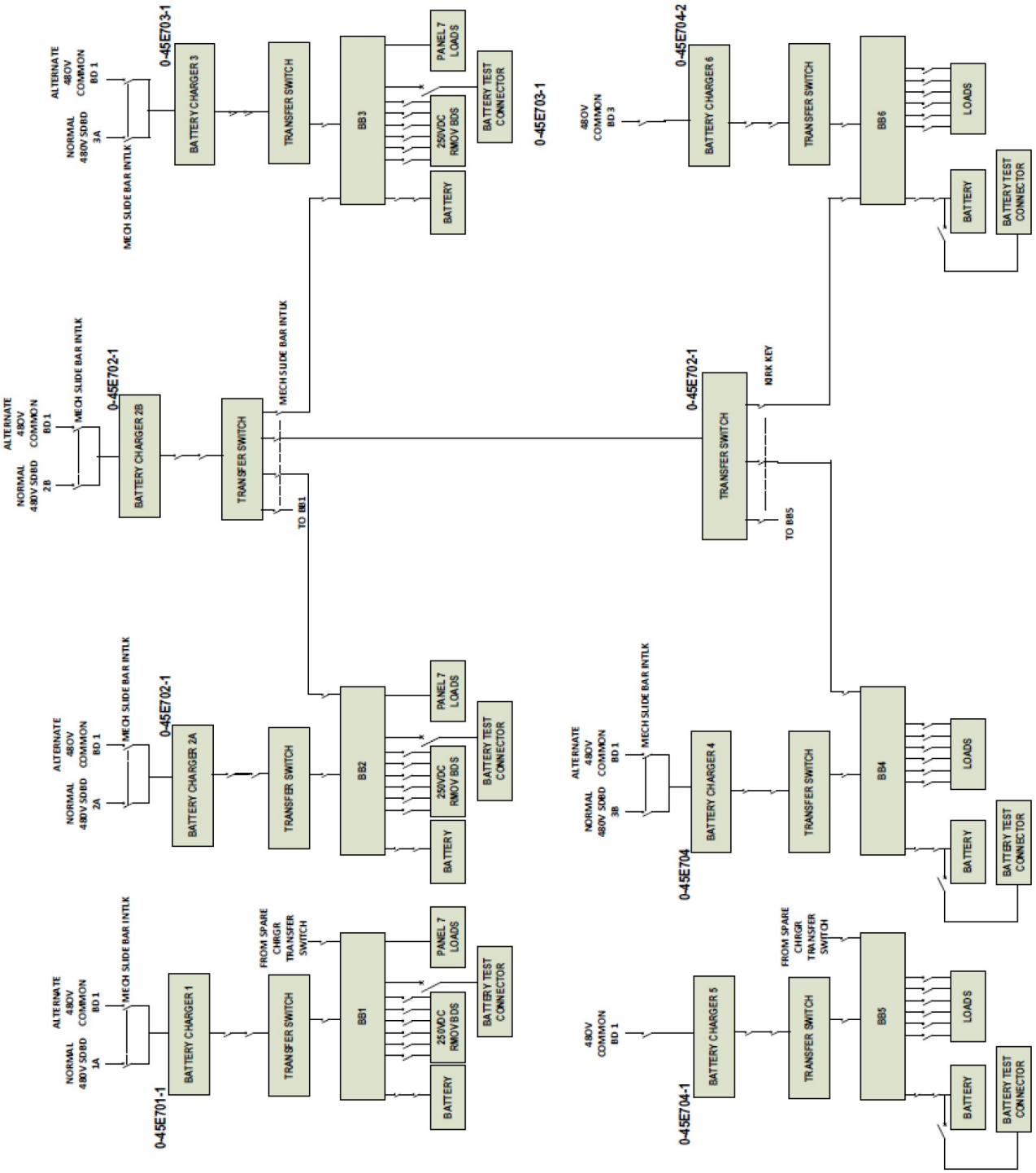
(2) Use of this breaker is restricted by NFPA 805 requirements. REFER TO drawing 3-45E749-5 for restrictions.

3.0 ATTACHMENT DATA (continued)

Performed On: _____

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
	480V Shutdown Bd 3B Control Bay - Div II - El 621', R-21R (3-45E749 - 6)			
3D	3-BKR-231-0003B/3D SPARE	OFF	3	____ ____
5A	3-BKR-231-003B/5A NOR FDR TO 480V DSL AUX BD 3EB ALTERNATE FDR TO 480V DSL AUX BD 3EA	CLOSED	3	____ ____
5B	3-BKR-231-0003B/5B INSTRUMENT AND CONTROL BUS B TRANSFORMER TIC3B	CLOSED	3	____ ____
5C	3-BKR-068-0003A2 VFD 3A COOLANT PUMP 3A2		3	ALIGNED BY 3-OI-68
5D	3-BKR-068-0003B2 VFD 3B COOLANT PUMP 3B2		3	ALIGNED BY 3-OI-68
6A	3-BKR-231-0003B/6A SPARE	OFF	3	____ ____
6B	3-BKR-070-0008 CLOSED COOLING WATER PUMP 3B		3	ALIGNED BY 3-OI-70
6C	3-BKR-069-0004B RWCU RECIRCULATION PUMP 3B		3	ALIGNED BY 3-OI-69
6D	3-BKR-248-0003B/6D 250V BATTERY CHARGER 4		3	ALIGNED BY 0-OI-57D

Excerpt from OPL171.037 Lesson Plan:



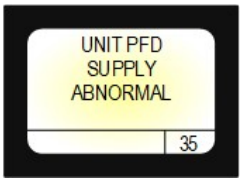
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC) **A2.01** (10CFR 55.41.5)	Tier #	2	------
Ability to (a) predict the impacts of the following on the Uninterruptable Power Supply (AC/DC) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations • Abnormal voltage	Group #	1	------
	K/A #	262002A2.01	
	Importance Rating	3.2	------

Proposed Question: **# 47**

Unit 3 is operating at 100% RTP when the following conditions occur:

- UNIT PREFERRED SUPPLY ABNORMAL (3-9-8B, Window 35) alarms
- Control Bay AUO reports that Unit 3 Unit Preferred MMG set is **NOT** running



Given the conditions above, which **ONE** of the following completes the statements below?

The **NORMAL** DC power source to the Unit 3 Unit Preferred MMG set, Battery Board **(1)** has been lost.

If Control Rod movement is required, a manual Reactor SCRAM **(2)** required in accordance with 3-AOI-57-4, Loss of Unit Preferred.

A. (1) 5
(2) is

B. (1) 5
(2) is NOT

C. (1) 6
(2) is

D. (1) 6
(2) is NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 250VDC Battery Board 5 is the alternate DC power source to the Unit 3 Unit Preferred MMG Set. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that both the BFN AC and DC Electrical Distribution Systems are complex and easily and often confused. If the candidate fails to recognize that the given loss of the Unit 3 Unit Preferred MMG set results in the loss of RPIS, then each respective Control Rod indication and movement still could be available.

C **CORRECT:** *(See attached)* In accordance with UNIT PREFERRED SUPPLY ABNORMAL (3-9-8B, Window 35), a loss of normal DC power source is a probable cause requiring the Operator to reference 3-AOI-57-4, Loss of Unit Preferred and 0-OI-57C, 208V/120V AC Electrical System. The MMG is normally driven by the AC motor, powered from 480V Shutdown Board 3A. If the AC source is lost, the AC motor is automatically disconnected and the DC motor starts which is powered from 250VDC Battery Board 6. Any transfers to alternate sources for the MMG are all manual. Given Unit 3 Unit Preferred MMG Set is not running, the normal source DC power, 250VDC Battery Board 6 has been lost in accordance 0-OI-57C. For second part, in accordance with 3-AOI-57-4, Loss of Unit Preferred, Rod Position Indicating System (RPIS) is lost requiring the use of 3-AOI-85-4, Loss of RPIS as well. Both 3-AOI-57-4 and 3-AOI-85-4 state that if Control Rod movement is required with a total loss of RPIS, then a manual Reactor SCRAM is required.

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's ability to predict the impacts of abnormal voltage on the Uninterruptable Power Supply (AC/DC) and use procedures to correct, control, or mitigate the consequences of those abnormal conditions. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must look at the conditions and put several pieces of mental data together to come up with a solution as it relates to the complex AC/DC power source capabilities of MMG operations supporting the 120VAC System.

Technical Reference(s): 0-OI-57C, Rev. 136 (Attach if not previously provided)
3-AOI-57-4, Rev. 42
3-AOI-85-4, Rev. 16
3-ARP-9-3B, Rev. 24
OPL171.102, Rev. 11U2

Proposed references to be provided to applicants during examination: **UNIT PREFERRED SUPPLY ABNORMAL, (3-9-8B, Window 35)**

Learning Objective: OPL171.102 Obj. 2b,h (As available)

Form 4.2-1	**Written Examination Question Worksheet**		
Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.102-02 023 #2759	(Note changes or attach parent)
Question History:	New		
	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

2759. OPL171.102-02 023

The Control Bay AUO reports that the Unit 1 Unit Preferred System Inverter, 1-INV-252-001, output has failed to zero.

Which ONE of the following completes both statements below?
(Consider each statement separately)

The Unit Preferred Inverter's DC Power Source is Battery Board __(1)__.

In accordance with 1-ARP 9-8B Window 35, UNIT PFD SUPPLY ABNORMAL, Battery Board 1, Cabinet 11 will automatically be powered from the __(2)__

A. (1) Four
(2) Unit 2 Unit Preferred MMG set

B. (1) Four
(2) Unit Preferred Regulating XFMR1

C. (1) Five
(2) Unit 2 Unit Preferred MMG set

D. (1) Five
(2) Unit Preferred Regulating XFMR1

Excerpt from 3-ARP-9-8B:

BFN Unit 3	Panel 9-8 3-XA-55-8B	3-ARP-9-8B Rev. 0024 Page 38 of 38

UNIT PFD SUPPLY ABNORMAL

35

(Page 1 of 1)

Sensor/Trip Point:

Relay SE -	loss of normal DC power source.
Relay TS -	DC transfer switch in emergency position.
Relay 2MS -	DC motor starts.
Relay 3k -	AC generator overvoltage, under frequency, or loss of output voltage.
Relay 13K -	difference between battery voltage and open circuit DC motor voltage greater than -7V DC or greater than +12V DC.

Sensor Location: 250V DC Battery Bd 3
El 593'

Probable Cause:

A. Loss of normal DC power source, UNIT PFD DC NORM/EMERG SELECTOR, 3-HS-252-02 in EMERG.
B. DC power transfer.
C. DC motor starts.
D. AC generator overvoltage.
E. AC generator under frequency.
F. AC generator loss of output voltage.
G. Relay failure.
H. Sensor malfunction.
I. Voltage differential between battery and motor is greater than -7V DC or greater than +12V DC.
J. Burned out light bulb on voltage indicating switch which activates Relay 13K.

Automatic Action: None

Operator Action:

A. **IF** 120V AC Unit Preferred is lost, **THEN** **REFER TO** 3-AOI-57-4.
B. After determining situation, **REFER TO** appropriate portion of 0-OI-57C.
C. **IF** voltage differential between battery and motor is greater than -7V DC or greater than +12V DC, **THEN** **PLACE** both local and remote OFF-AUTO-MANUAL select switch in MANUAL to prevent possible damage to DC drive motor.
D. **IF** burned out light bulb is cause of alarm, **THEN** **INITIATE** WO to change out the light bulb due to it being hard wired.

References: 0-45E641-2 3-45E620-11 3-3300D15A4585

Excerpts from 3-AOI-57-4:

BFN Unit 3	Loss of Unit Preferred	3-AOI-57-4 Rev. 0042 Page 3 of 31

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for Loss of Unit Preferred (Battery Board 3 Panel 11, Control Room Panel 3-9-9 Cabinet 5 Non-preferred and Unit Preferred to Panel 3-9-9 Cabinet 6).

NOTE

A loss of the Unit Preferred power source results in the following boards being de-energized:

- Battery Board 3 Panel 11
- Panel 3-9-9 Cabinet 5 Non-preferred
- Panel 3-9-9 Cabinet 6 Unit Preferred (if panel does **NOT** auto transfer to alternate)

2.0 SYMPTOMS

A. UNIT PFD SUPPLY ABNORMAL (3-XA-55-8B, Window 35) is in alarm.

B. Loss of units computer and RWM panel (if on normal supply).

C. Loss of indication to CRD rod control indicator lights Panel 3-9-5 (ie: rod out permissive, timer switch malfunction - select block, rod withdraw, insert and settle, notch override and stabilizing valve indicator lights.

D. The following recorders fail downscale (Panels 3-9-6, 3-9-7, 3-9-8):

1. Turbine Generator Vibration(3-XR-47-15),
2. Turb Gen ECC/Speed/Valve Posn (3-XR-47-16),
3. Turb Exp and Temp Mn XFMR Winding Temp (3-TR-47-20/3-XR-47-20),
4. RFPT 3A, 3B and 3C Vibration & Eccentricity recorder (3-XR-3-177).

E. PANEL 9-9 CABINET 1,2,3 & 6 CONTROL POWER TRANSFER (3-XA-55-7A, Window 15), is in alarm on transfer of Unit Preferred Cabinet 6.

F. Panel 3-9-9, Cabinet 6 UNIT PREFERRED 120V AC NORMAL and ALT SUPPLY lights, 3-XI-57-600A and 3-XI-57-600B extinguished.

G. Loss of RPIS. **REFER TO** 3-AOI-85-4.

2.0 SYMPTOMS (continued)

H. The following RFW Control System annunciators in alarm on Panel 3-9-6:

1. RFPT GOVERNOR POWER FAILURE OR GOVERNOR ABNORMAL, 3-XA-46-8, 9 & 10 (3-XA-55-6C window 12)

I. Loss of power to Cabinet 6 will cause a loss of flow signal to CNDS FLOW CONTROL SHORT CYCLE, 3-FIC-2-29 when in auto. This will cause 3-FCV-2-29A/B to open resulting in rising condensate flow. This can adversely affect Condensate & Feedwater system NPSH and Reactor water level.

J. The following EHC Control System Annunciators alarm on Panel 3-9-7 due to a loss of power to either Unit Preferred Panel 3-9-9 Cabinet 6 OR Non-preferred Panel 3-9-9 Cabinet 5. A loss of power to both cabinets will NOT result in an alarm.

1. EHC POWER ABNORMAL 3-EA-57-163 (3-XA-55-7B Window 5)

2. EHC/TSI SYSTEM TROUBLE 3-XA-57-165 (3-XA-55-7B Window 6)

K. EHC Control System PLU 1 (power load unbalance) can bypass with a sustained loss of power to Panel 9-9 Cabinet 5. An uninterruptible power supply will keep the PLU energized for approximately 15 minutes after normal power is lost.

L. EHC Control System HMI on Panel 3-9-31 may become blank if power is lost to Panel 9-9 Cabinet 6. An uninterruptible power supply will keep this component energized for approximately 15 minutes after normal power is lost. RECIRC FLOW SYSTEM TROUBLE ALARM (3-XA-55-4A, Window 23).

M. Loss of power to CRD select modules.

N. PNL 9-21 SYS LEAK DETECTION POWER FAILURE (3-XA-55-3D, Window 31) on loss of power to Panel 3-9-21 Steam Leak Detection Panel.

3.0 AUTOMATIC ACTIONS

A. Panel 3-9-9 Cabinet 6 Unit Preferred automatically transfers to the alternate power source.

B. 3-FCV-85-11(A/B) closes on loss of 3-9-9 cabinet 6 and opens if only 3-9-9 cabinet 5 is de-energized.

C. 3-FCV-1-58, UPSTREAM MSL DRAIN TO CONDENSER and BYPASS VLV UPSTREAM MSL DRAIN TO CONDENSER, 3-FCV-1-185 auto open.

D. Loss of one out of two auctioneered power sources for EHC Control System (This includes the PLU) (ICS is the other power source).

E. Loss of power to Turbine Supervisory Instrumentation Panel 3-9-46.

F. Loss of one out of two auctioneered power sources for RFPT 3C Woodward Governor and Final Driver (ICS is the other power source).

G. Loss of one out of two auctioneered power sources for Recirc Flow Control System (ICS is the other power source).

4.0 OPERATOR ACTIONS

NOTE

If necessary, place keeping marks may be made directly in the Control Room copy of this instruction. Management Services should be contacted for a replacement copy when time permits.

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **ENSURE** the following:

- No Control Rod movement as indicated by stable Reactor Power.
- RFW Control System is maintaining Reactor Water Level.
- **CHECK** Recirc Flow Control System maintaining Recirc pump speeds.
- EHC Control System maintaining Reactor Pressure and Turbine Control parameters.

[2] **IF** ANY EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). (Otherwise, NA)

CAUTION

While RPIS and the process computer are inoperable, control rod movement may only be performed by manual reactor scram. **REFER TO** 3-AOI-85-4

[3] **IF** control rod movement is required while RPIS and the process computer are inoperable, **THEN**

INSERT a MANUAL SCRAM. **REFER TO** 3-AOI-100-1. (Otherwise, NA)

Attachment 3
(Page 3 of 4)

Panel 3-9-9 Cabinet 6 Unit Preferred Loads

I. BKR 609, 48V DC SUPPLY FAILURE ALARM CKT

Panel 3-9-9, Cabinet 1, 48V DC Supply Failure Alarm CKT

J. BKR 610, PANEL 3-9-40 CMPTR PERIPHERAL TYPERS

1. Alarm Typer
2. Period Log Typer #2
3. OD Log Typer
4. Period Log Typer #1

K. BKR 611, PANEL 3-9-8 H2 DUMP VLV CONTROL

Hydrogen Dump Valve Control CKT

L. BKR 612, PANEL 3-9-27 ROD POSITION INFO SYS

Rod Position Information System

M. BKR 613, PANEL 3-9-31 EHC SOV

N. BKR 614, MN STM LN DRN VLV INTERLOCK & CONT

Main Steam Line Drain Valve Interlocks, control ckt, and indication for 3-FCV-1-168 through 171, 57, 58 and 185.

O. BKR 615, VFD 3A CONT PWR, RLYS, & ICS INPUT

P. BKR 616, VFD 3B ALT CONT PWR & RLYS

Q. BKR 617, PANEL 3-9-46 TSI

Excerpt from 3-AOI-85-4:

BFN Unit 3	Loss of RPIS	3-AOI-85-4 Rev. 0016 Page 6 of 16

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **STOP** all control rod movement.

4.2 Subsequent Actions

NOTE
Reference TRM 3.3.5, RPIS Indicated Channel Operability, for applicable 7 or 30 day LCO relating to an inoperable RPIS indication.

[1] **CHECK** with Operations Superintendent and Reactor Engineer for actions to be taken in a timely manner.

[2] **CHECK** with Technical Support to help determine the extent of loss of RPIS.

[3] **IF** control rod was in motion when RPIS failed and position of that control rod can **NOT** be determined, **THEN**

DECLARE that Control Rod Inoperable. **REFER TO** Tech. Spec. 3.1.3.

[4] **IF** control rod movement is required with a total loss of RPIS, **THEN**

MANUALLY SCRAM reactor.

[5] **REFER TO** EPIP-1.

[6] **CHECK ON** Breaker 612, PANEL 3-9-27 ROD POSITION INFO SYS FEED FROM UNIT PREFERRED 120VAC, at Panel 3-9-9 Cabinet 6.

Excerpts from 0-OI-57C: Also Supports Distractors A1, B1

5.5 Unit Preferred MMG Set Startup and Synchronization (continued)

[4] **ENSURE CLOSED** the UNIT PREFERRED MG SET 2(3) breaker, Compartment 7D on 480V Shutdown Board 2(3)A.

TABLE 1

Unit No.	Battery Board	Breaker No.	Description
2	4	313	0-BKR-280-0004/313 NORMAL SUPPLY FOR U2 PFD AC SYS MMG SET
	2	714	0-BKR-280-0002/714 U-2 PREFERRED AC SYSTEM SYNCHRONIZING BREAKERS
	6	303	0-BKR-280-0006/303 ALTERNATE SUPPLY FOR U2 PFD AC SYS MMG SET
3	5	303	0-BKR-280-0005/303 ALT SUPPLY U3 PFD AC SYSTEM MMG SET
	3	714	0-BKR-280-0003/714 U-3 PREFERRED AC SYSTEM SYNCHRONIZING BREAKERS
	6	313	0-BKR-280-0006/313 NOR SUPPLY U3 PFD AC SYSTEM MMG SET

[5] **ENSURE CLOSED** the following 250V DC Breakers for the desired unit preferred MMG set. (**SEE** Table 1 above for location and compartment number):

A. UNIT 2(3) PFD AC SYS MMG SET DC MOTOR ALTERNATE FDR

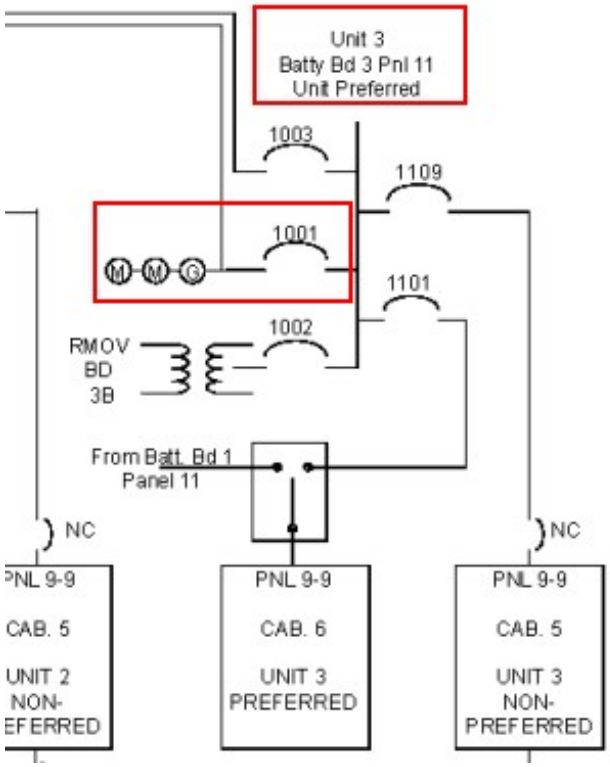
B. UNIT 2(3) PFD AC SYSTEM SYNCHRONIZING BREAKERS

C. UNIT 2(3) NORMAL SUPPLY FOR U2(3) PFD AC SYS MMG SET

NOTE

Steps 5.5[6] is performed from Battery Board Room 2, UNIT PFD Sync. Control Panel.

[6] **PLACE** the UNIT 2(3) MMG LOCAL-REMOTE CONTROL, 2(3)-XS-252-02A in LOCAL.


Unit 3
Batty Bd 3 Pnl 11
Unit Preferred
1003
1109
1001
M M G
1101
1002
RMOV
BD
3B
From Batt. Bd 1
Panel 11
NC
NC
PNL 9-9
CAB. 5
UNIT 2
NON-
EFERRED
PNL 9-9
CAB. 6
UNIT 3
PREFERRED
PNL 9-9
CAB. 5
UNIT 3
NON-
PREFERRED

Excerpts from OPL171.102:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
powered from the Standby Auxiliary Power System with an alternate supply from the 250VDC Power System. Each unit has a separate UPS. A Synchronizing and Control Panel for all three units is located on Panel 11 of 250V Battery Board 2.	NLO Obj. 2.h.i
(2) Each unit's battery board, panel 11 is the distribution point for UPS power. There are 3 sources of power to the battery board panel 11.	Note: The normal and alternate supply breakers are on panel 10 of the Battery Board.
(a) The MMG (Unit 2 & 3 only): Normal UPS bus power supply is a motor-motor-generator (MMG). The MMG Set consists of a 480VAC motor, a 250VDC motor, and a 240/120VAC three-wire single-phase generator, all close-coupled on a single shaft. There is an inertial flywheel between the generator and the motors, to override power supply fluctuation or momentary interruption, and to prevent a loss of generator output on an AC to DC motor transfer.	ILT/NLO/NLOR Obj. 2.a NLO/NLOR Obj. 2.e ILT Obj. 2.f
(b) The INVERTER Unit (Unit 1): Unit 1 is powered by an uninterruptible power supply (inverter unit).Normal supply is 480VAC to the rectifier/ inverter unit itself where it is converted to DC volts then back to a 'smooth' 120/ 240VAC signal fed to Batt. Bd 1 Panel 10/11. There is a backup 250VDC backup power supply fed to the inverter unit for a bumpless transfer in case of loss of AC power. Additionally there is a regulated AC alternate power to the inverter static switch for a continuation of power in case of inverter failure.	IL-2 NLO Obj. 2.i NLOR Obj. 2.h
(c) Unit Preferred Transformer: The alternate power source is the unit preferred transformer. This transformer receives power from the 480V portion of the standby AC power system. Unit 1 transformer is from the 480 V RMOV Bd 1A Unit 2 & 3 transformers are powered from 480V RMOV Board 2B & 3B.	Note: The UPS transformer is also the alternate RPS power supply for U2.

Lesson Plan Content

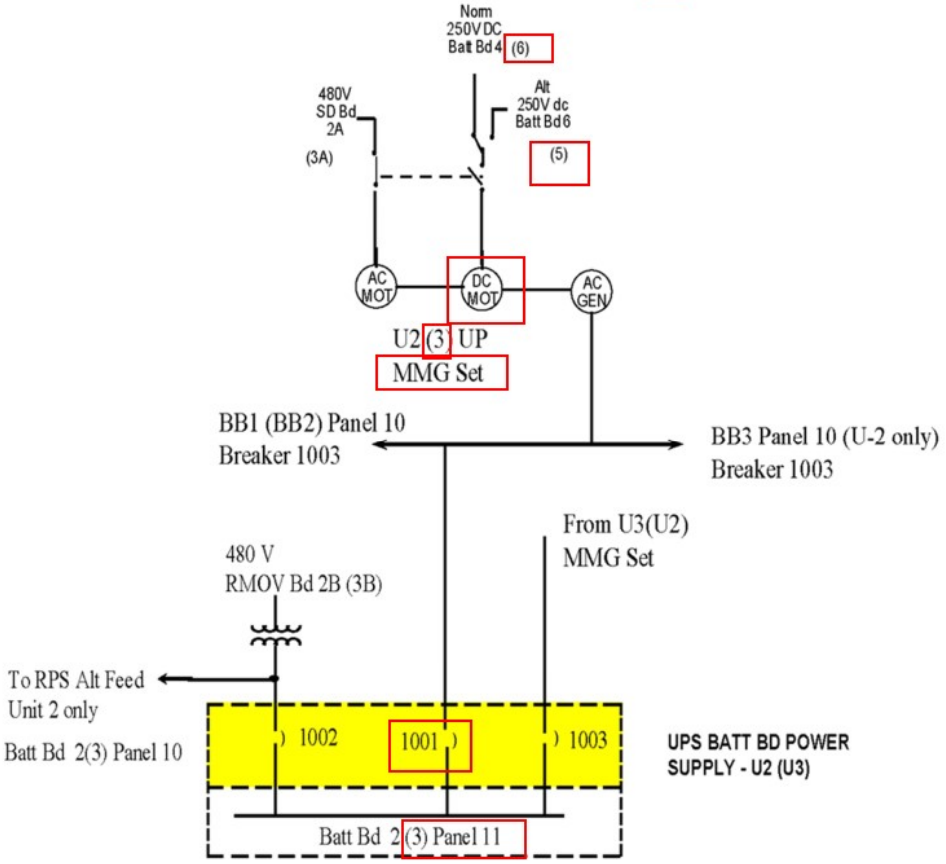
Outline of Instruction	Instructor Notes and Methods (Optional)
Transfers to this source are done manually at battery board 2 panel 11. (d) Another Unit's MMG set: The second alternate is from another unit's MMG set output. Unit 2 MMG is the second alternate for either Unit 1 or Unit 3; Unit 3 is the second alternate for Unit 2. Transfers to this source are done manually at Battery Board 2 panel 11.	
b) MMG Sets (Unit 2&3) (1) The MMG is normally driven by the AC motor, powered from 480V Shutdown Board A. Should this supply fail, the AC motor is automatically disconnected and the DC motor starts, powered from 250V Battery Board. The DC motor has an alternate power supply from another 250V Battery Board. Transfer to the alternate DC source is manual. Under-frequency on the generator output will trip the DC motor. Transfer of the MMG set back to the AC motor is manual.	IL-11 ILT Obj. 2.b, 2.h LOR Obj. 3.e NLO Obj. 2.c, 2.d, 2.j NLOR Obj. 2.c, 2.d, 2.i
(2) The 1001 and 1003 breakers from an MMG set will trip on overvoltage or under-frequency at the output of the MMG. Also Unit 2 MMG breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.	
(3) When an under-frequency or overvoltage condition exists at the Generator Output the following occurs:	ILT Obj. 2.h LOR Obj. 3.e NLO Obj. 2.j NLOR Obj. 2.i
(a) BB panel 10 breakers from the MMG set trip: U2 1001 (U2) 1003 (U1&3) U3 1001 (U3) 1003 (U2)	
(b) Excitation is lost and the MMG Set continues to run. (The Hold to build up voltage switch must be depressed to restore voltage.)	
(4) The starting sequence for the MMG is as follows:	
(a) Start the AC motor first. This is a larger motor and has enough power to compensate for the flywheel load.	HU Tools Procedure Use & Adherence

Also Supports Distractors A1,B1

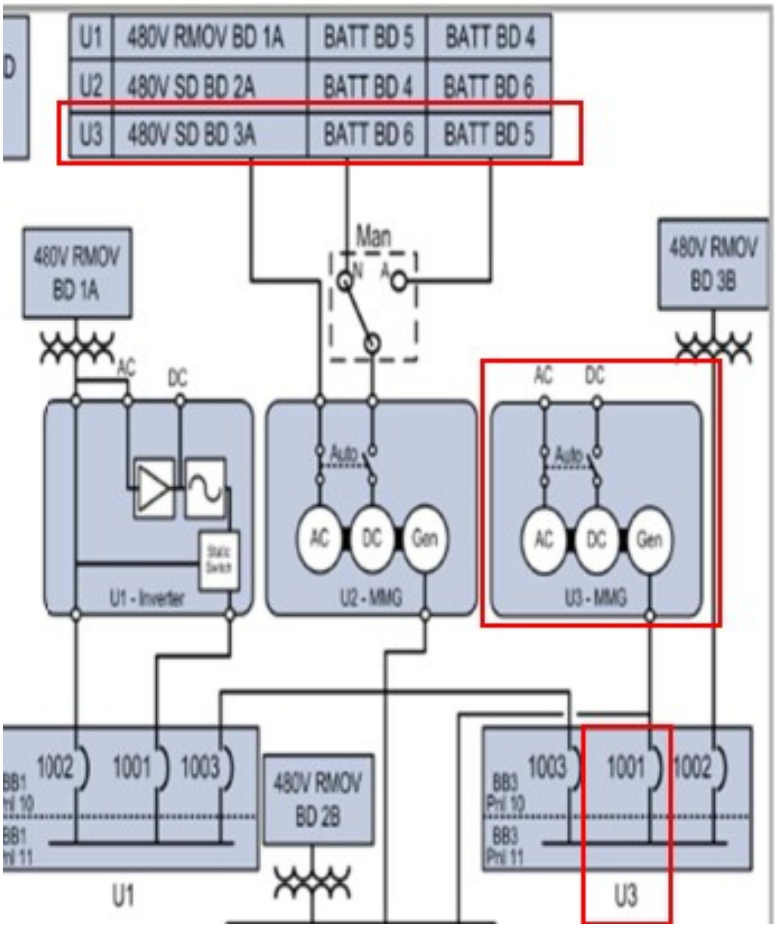
OPL171.102 , 120V AC Power Supplies and Distribution Systems, Rev# 11U2

Appendix A: Illustrations

IL-11: Power Supply to UPS Batt Bd (U-2/3)



Also Supports Distractors A1,B1



Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
263000 (SF6 DC) DC Electrical Distribution **K1.01** (10CFR 55.41.7)	Tier #	2	------
Knowledge of the physical connections and/or cause and effect relationships between the DC Electrical Distribution and the following systems: • AC electrical distribution	Group #	1	------
	K/A #	263000K1.01	
	Importance Rating	4.0	------

Proposed Question: **# 48**

Unit 1 is operating at 100% RTP when an event occurs resulting in 480V Load Shed.

In accordance with 1/2-AOI-57-1D, 480V Load Shed, if the load shed logic **CANNOT** be reset, Battery Charger 1 may be returned to service by placing the charger select switch in **(1)**.

If Battery Charger 1 will not reset, Battery Charger 2B **(2)** be aligned to Battery Board 1.

A. **(1)** EMERG
(2) can

B. **(1)** EMERG
(2) can NOT

C. **(1)** OFF and then back ON
(2) can

D. **(1)** OFF and then back ON
(2) can NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 1/2-AOI-57-1D, 480V Load Shed, when Battery Charger 1 operation is required, the battery charger control switch must be placed in EMERG to bypass the Load Shed of the charger if the Load Shed cannot be reset. For second part, 2B is the Spare Charger and can be aligned to any Unit or Station Battery Board.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible as the candidate could misapply the battery charger lineup due to the complexity of the DC System at BFN. There are 7 battery chargers (two of which carry a Unit 2 UID – 2A and 2B), any of which could be a spare or only be aligned to one unit's battery.

C INCORRECT: First part is incorrect but plausible in that many breaker schemes at BFN require taking the control switch to OFF to reset the anti-pump coil prior to restarting a component. Second part is correct (*See A*).

D INCORRECT: First part is incorrect (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the operational implications of AC electrical distribution Load Shed has on the DC Distribution System. This question is rated as memory due to strictly recalling facts related to Load Shed and its effect on the DC Distribution System.

Technical Reference(s): 1/2-AOI-57D, Rev.8 (Attach if not previously provided)
0-OI-57D, Rev.196
OPL171.037, Rev.18U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.037, Obj. 7,8 (As available)

Question Source:

Bank #	NRC ILT 1909 #2
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2019

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 2**

An event occurs on Unit 3 that results in 480V Load Shed. In accordance with 3-AOI-57-1D, 480V Load Shed, if the load shed logic cannot be reset, Battery Charger 3 may be returned to service by placing the charger select switch in **(1)**.

If Battery Charger 3 will not reset, Battery Charger **(2)** may be used as a spare.

A. **(1)** EMERG
(2) 2A

B. **(1)** EMERG
(2) 2B

C. **(1)** OFF and then back ON
(2) 2A

D. **(1)** OFF and then back ON
(2) 2B

Proposed Answer: **B**

Excerpts from 1 / 2-AOI-57-1D:

BFN Unit 1 & 2	480V Load Shed	1/2-AOI-57-1D Rev. 0008 Page 9 of 26

3.0 AUTOMATIC ACTIONS (continued)

12. 480V RMOV Board 2C:
    a. Drywell Blowers 2A-5 and 2B-5
    b. Recirc Drive 2A and 2B cooling fans and lights
    c. Steam Vault Exhaust Booster Fan

B. The following loads are "Locked Out" until the accident signal has been reset and the Core Spray logic is RESET or the DGVA resets:

1. Unit 1 and 2 Turning Gear Oil Pumps
2. RWCU pumps and Holding pumps
3. Unit 1 and 2 Recirc drive cooling water pumps A1, A2, B1, B2, and Unit 2 Normal Aux Power VFD cooling fans and lights
4. Drywell Blowers A3, A4, A5, B3, B4 and B5 on the accident unit
5. D/G Starting Air Compressors
6. Battery Chargers 1, 2A, and 2B
7. SHUTDOWN BOARD ROOM Exhaust Fans 1A and 2A

C. The following loads will automatically restart after a 40 second time delay:

1. A RBCCW Pump, if its control switch is in the Normal After Start position
2. If RBCCW pump A fails to start RBCCW Pump B will automatically start after a 3 second time delay.(with the control switch in Normal After Start)
3. Standby Gas Treatment Fans A and B and their associated humidity control heater

D. The following loads will automatically restart after a 40 to 90 second time delay

1. Drywell Blowers auto restart on the non-accident unit with staggered start times

4.2 Subsequent Actions (continued)

NOTE

When operation of the 1, 2A or 2B Battery Chargers is required, the control switch is placed in EMERG to bypass load shed of the charger if the load shed logic can **NOT** be reset.

[14] **PERFORM** the following for the 250V Unit Batteries:

- **MONITOR** Batt Bd amps and VOLTS on Panel 1(2)-9-8.
- **PLACE** the battery charger back in service within 30 minutes after loss of the charger to the battery.

[15] **IF** load shed logic can **NOT** be reset and operation of the battery charger is required, **THEN** (Otherwise N/A)

[15.1] **PLACE** Battery Charger EMER/OFF/ON Select Switch in EMERG.

[15.2] **WHEN** load shed logic is reset, **THEN**

PLACE Battery Charger EMERG/OFF/ON Select Switch in ON.

Excerpt from 0-OI-57D:

3.0 PRECAUTIONS AND LIMITATIONS (continued)

F. 250V Unit Battery Charger 1, 2A, 2B and 3 Emergency ON select switch bypasses battery charger emergency load shed contacts. Placing the select switch in Emergency ON reestablishes charger operations with an accident signal present and Diesel Generator voltage available. Battery Charger 4 supply breaker, 480V Shutdown Board 3B, Compt 6D, receives a trip signal from the load shed logic and the breaker must be manually re-closed after a 40 second time delay to restore the charger to service. The annunciation circuit for the 250V Unit Battery Charger 3 does NOT work when the EMER/OFF/ON Select Switch is in the EMER Position.

G. The Autostart mode will be enabled for Main Bank Battery 3, 4, 5 and 6 and Shutdown Board Battery SB-A, SB-C, SB-D, 3EB and SB-Spare Chargers. When AC input power is applied to the charger the HMI display will show "System type SDC" for 10 seconds and then changes to "Standby". If the DC output breaker is closed (ON) the charger will start automatically 60 seconds later. The HMI display will indicate "Float charge Load" and the 250V Battery is being charged. If the Main Bank or Shutdown Board Battery Charger is manually started, using the HMI softkey "ON" pushbutton, within the 60 seconds then autostart will be ignored.

H. [II/C] Neutron monitoring battery chargers are NOT stand alone power supplies and shall only be operated while connected to the neutron monitoring batteries. [BFPER 940862]

I. Within 30 minutes after the loss of the normal charger to a 250V Unit Battery another charger shall be placed in service to that battery and load reduced so that the battery is NOT discharging.

J. [NRC/C] Upon return to service of 24V DC Neutron Monitoring Battery A or B, Instrument Maintenance must perform functional tests on SRMs and IRMs that are powered from the affected battery board (In that the IRMs and SRMs are normally inoperable after entering RUN mode due to lack of testing, these tests are N/A for the IRMs and the SRMs if the Unit is in RUN Mode and the IRMs and SRMs are inoperable). Prior to calling the IRMs and SRMs operable, the tests have to be performed. [NRC IE Inspect Follow-up Item 86-40]

K. To return equipment to service following a failure or trip, the shutdown section of this instruction should be performed on the equipment failed. The initial conditions may NOT be applicable in this case.

L. [NRC/C] The transfer of 250VDC control power to a 4kV Shutdown Board with a diesel generator operating may cause an inadvertent start of a RHRSW pump. [LER 88021/25]

8.0 INFREQUENT OPERATIONS

8.1 Placing the 250V BATTERY CHARGER 2B, 0-CHGA-248-0002B in Service to Battery Board 1(2,3,4,5,6)

[1] **REVIEW** all Precautions and Limitations in Section 3.0.

NOTE

A 250V DC battery charger should NOT be considered operable if its safety related supply is NOT available. If normal power (safety related supply) is available but the charger is on its alternate supply it is still considered operable.

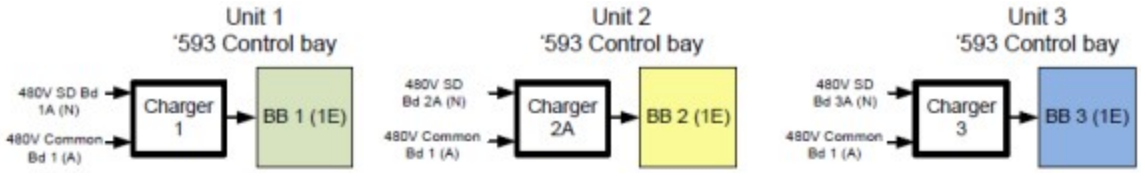
[2] **ENSURE** the following initial conditions are satisfied:

- Normal Source: 480V SHUTDOWN BD 2B, COMP 6D, 2-BKR-248-0002B/6D is CLOSED.
- Alternate Source (Charger Service Bus): 480V Common Board 1, COMP 3A, 0-BKR-215-0001/03A is CLOSED.
- 250V Battery Charger 2B, 0-CHGA-248-0002B is **NOT** already in service to any Battery Board (1, 2, 3, 4, 5, and 6).
- The 250V Battery Charger (1, 2A, 3, 4, 5, 6) has been removed from service in accordance with Section 7.2.

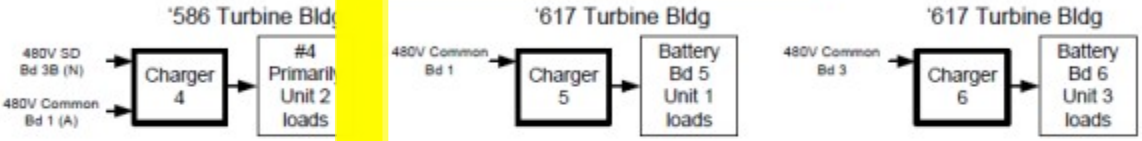
Excerpt from OPL171.037 Lesson Plan:

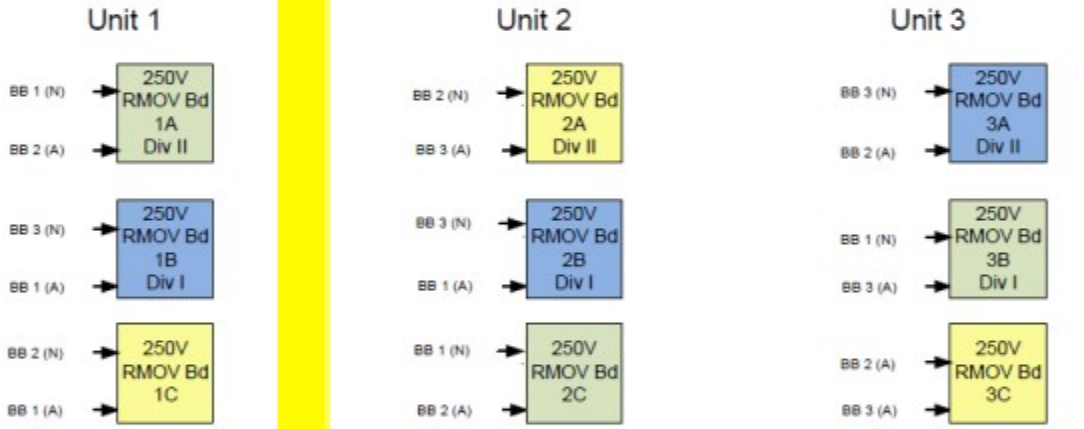
OPL171.037, DC Systems, Rev: 18U1

I. 250V Unit and Station DC System	
A. Description 1. Consists of six batteries, seven battery chargers, and the associated circuitry, switches, distribution panels, indicators, and alarms. The DC System provides motive power for DC motor operated valves, pumps, and control/logic power. 2. The three "Unit" batteries and associated chargers (1, 2, and 3) supply the engineered safeguards Class 1E safety related loads. There is one "Unit" battery and one charger per Unit. 3. The three "Station" batteries and associated chargers (4, 5, and 6) supply only loads that are not essential to the safe shutdown and cool down of the nuclear system. There is one "station" battery and charger per unit for Class Non-1E systems. Batteries 5 and 6 were added to supply non-Class 1E loads that were initially supplied from Unit Batteries. This reduced the loading on the Unit batteries to improve the ability to withstand a 4-hour Station Blackout (SBO) event during multi-unit operations. Most Class Non-1E loads were transferred from the unit batteries to the plant/station batteries 4, 5, and 6 4. There is one spare charger (2B) that may be aligned to any Unit or Station battery board.	Information in this column is for use at instructors discretion Key Diagram DC Systems 731E700 Obj 1 a,b Fig-1,2,6 Obj 1a,b Instructor will use various questioning techniques throughout the lecture.
B. Location 1. Unit Batteries (1, 2, and 3) are located in their respective Battery rooms on each Unit on 1C (Elev. 593) in the Control Bay. The battery boards and chargers for the Unit Batteries are located in Battery Board Rooms adjacent to the associated batteries. 2. Battery (4) Room is located on Unit 2 in the Turbine Building on Elev. 586. Access is from U3. Battery Boards 4 and its charger is located in Battery Board Room adjacent to the associated battery. 3. Battery (5 & 6) Rooms are located on the Turbine Floor, Elev. 617. Battery Boards 5 & 6 and their associated chargers are located adjacent to the battery rooms, but are in the open space of the turbine floor 4. Spare charger 2B is located in the Unit 2 Battery Board room.	Obj 1a,b,c BB1-6 45E701 series 45E702 series 45E703 series 45E704 series Obj 1a,b,c Obj 1a,b,c Obj 1b









250V RMOV Board 2A Loads

HPCI
Pump Discharge Vlv 73-34
Steam Supply Isolation Vlv 73-3
Pump Suction from CST Vlv 73-40
Min Flow Vlv 73-30
Pump Discharge Vlv 73-44
Test Bypass to CST Vlv 73-35
Pump Suction from Suppression Chamber 73-26
Pump Suction from Suppression Chamber 73-27
Steam Supply to Turbine 73-16
Gland Seal Cond Blower
Gland Seal Cond Pump
Turbine Aux Oil Pump
System Logic Div II, Pnl 9-39

Other
ECCS Div II ATU inverter
ECCS Div II ATUs, Panel 9-82
Core Spray System Logic, Div II, Panel 9-33
RHR System Logic, Div II, Panel 9-33
RHR Shutdown Cooling Suction Outboard Vlv 74-47
SRVs 1-30, 1-22, 1-4, 1-41
Hardened Suppression Chamber Vent Vlv 64-222

Unit 1 and 3 are similar

250V RMOV Board 2B Loads

HPCI
Redundant Shut-Off to CST Vlv 73-36
System Logic, Div I, Panel 9-32

RCIC
Steam Supply Isolation Vlv 71-3
Min Flow Vlv 71-34
System Logic Div I, Panel 25-31

ADS
System Logic Bus A, Div I, Panel 9-30
ADS Div I, Bus B, Panel 9-30

Other
RWCU Isolation Vlv 69-2
Backup Scram Vlv, Panel 9-17
Core Spray System Logic, Div I, Panel 9-32
RHR System Logic, Div I, Panel 9-32
ECCS Div I ATU, Panel 9-81
SRV 1-179
RHR Flush Discharge Vlv 74-108
Main Steam Line Warming Isolation Vlv 1-56
SRVs 1-31, 1-22 (alt), 1-19, 1-18, 1-42, 1-179
Hardened Suppression Chamber Vent Vlv 64-221

Unit 1 and 3 are similar

250V RMOV Board 2C Loads

RCIC
Turbine Trip and Throttle Vlv 71-9
Pump Discharge Vlv 71-37
Pump Injection Vlv 71-39
Turbine Steam Supply Vlv 71-8
CST Suction Vlv 71-19
Test Bypass Vlv 71-38
Pressure Suppression Chamber Suction Vlv 71-18
Turbine Trip Solenoid Circuit
Pressure Suppression Chamber Suction Vlv 71-17
Gland Seal Vacuum Tank Condensate Pump
Gland Seal Vacuum Pump
Lube Oil Cooling Water Vlv 71-25

Other
SRV 1-180, 1-34, 1-23, 1-5, 1-30(alt)

Unit 1 and 3 are similar

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG **A2.11** (10CFR 55.41.5) Ability to (a) predict the impacts of the following on the Emergency Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: • Failure of emergency generator to start/load	Tier #	2	------
	Group #	1	------
	K/A #	264000A2.11	
	Importance Rating	4.6	------

Proposed Question: **# 49**

Unit 1 was operating at 100% RTP when an event results in the following plant conditions:

- Reactor Water Level lowering to (-) 130 inches
- Drywell Pressure is 2.2 psig
- Reactor Pressure is 500 psig

Given the above conditions, **(1)** EDGs have received an automatic start signal.

In accordance with 0-OI-82, Standby Diesel Generator System, if 'A' EDG fails to automatically start, it can be started from **(2)**.

A. **(1) ALL**
(2) Panel 0-9-23 **ONLY**

B. **(1) ALL**
(2) either Panel 0-9-23 **OR** 'A' 4KV Shutdown Board

C. **(1) ONLY** A, B, C and D
(2) Panel 0-9-23 **ONLY**

D. **(1) ONLY** A, B, C and D
(2) either Panel 0-9-23 **OR** 'A' 4KV Shutdown Board

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that while the Standby Diesel Generators (EGDs) can be started at Panel 0-9-23, in accordance with 0-OI-82, Standby Diesel Generator System, in the event EDGs fail to automatically fast start, a manual fast start is allowed from their respective 4KV Shutdown Board.

B **CORRECT**: (*See attached*) In accordance with 0-OI-82, all eight EDGs will start on any Pre-Accident Signal (Reactor Vessel Low Low Low Water Level or High Drywell Pressure). Reactor Water Level is given in the stem of the question as (-) 130 inches, which is below the Reactor Water Level Low Low Low setpoint of (-) 122 inches – therefore, all eight EDGs will receive an automatic start signal. For second part, in accordance with 0-OI-82, Standby Diesel Generator System, in the event EDGs fail to automatically fast start, a manual fast start is allowed from their respective 4KV Shutdown Board.

D INCORRECT: First part is incorrect but plausible in that the question stem gives a low Reactor Water Level on Unit 1, so it is reasonable that only the Unit 1 and 2 EDGs would start for a Pre-Accident signal on Unit 1. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the effect of an Emergency Diesel Generator failing to start on a valid automatic initiation signal on plant operations. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

Technical Reference(s): 0-OI-82, Rev.180 (Attach if not previously provided)
OPL171.038, Rev. 27U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.038 Obj. 4, 7 (As available)

Question Source:

Bank #	
Modified Bank #	(Note changes or attach parent)
New	**X**

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Excerpts from 0-OI-82:

BFN Unit 0	Standby Diesel Generator System	0-OI-82 Rev. 0180 Page 14 of 226

3.0 PRECAUTIONS AND LIMITATIONS (continued)

P. Personnel working in the D/G rooms should remain aware that the possibility exists of CO_2 discharge into the room. Upon CO_2 initiation, an alarm will sound. Personnel then have 20 seconds to evacuate the area before CO_2 is dispensed. For detection purposes, a wintergreen odor is injected into CO_2 discharge.

Q. Environmental calculations assume DG battery ambient temperatures are within 40°F to 110°F.

R. When the D/G is the only feed to the shutdown board and in single unit operations, starting an RHR Pump with other 4kV motor loads running on the associated board may result in D/G overload.

S. After operation of 4160V breakers, the charging spring is required to be verified to have recharged by verifying locally the breaker closing spring target indicates charged and the amber breaker spring charged light is on to ensure future breaker operation.

T. Diesel Generators will automatically start, as follows:

1. Degraded voltage, undervoltage, OR sustained voltage unbalance on 4-kV Shutdown Board A, B, C, or D will start its associated Diesel Generator.

2. A Pre-Accident Signal (Reactor Vessel Low Low Low water level OR High Drywell pressure) on Unit 1, Unit 2 or Unit 3 will start all eight Diesel Generators.

U. Under normal conditions, any of the following will auto trip the Diesel Generator output breaker:

1. Differential overcurrent

2. Timed overcurrent

3. Reverse power

4. Loss of field

5. Overspeed

6. Common Accident Signal (Low Low Low Reactor water level OR Low Reactor pressure in conjunction with High Drywell pressure on Unit 1, 2 or Unit 3.)

5.3 Manual Fast Start at Panel 9-23

[1] **ENSURE** the following initial conditions:

A. All Precautions and Limitations in Section 3.0 have been reviewed.

B. The Diesel Generator to be started is in Standby Readiness. **REFER TO** Section 4.0.

NOTE

To minimize wear on the engines, the Diesel Generators should normally be slow started using Section 5.6. However, it is not always practical or desired to perform a slow start since performance of a slow start renders the Diesel Generator inoperable. If a slow start is impractical, the Unit SRO may authorize a manual fast start at his/her discretion to allow for flexibility in supporting tests which require operation of the Diesel Generator(s).

[2] **ENSURE** a manual fast start is allowed by the following criteria:

A. A Diesel Generator failed to fast start automatically when required **OR**

B. An operability check is required by Tech Spec 3.8.1 and 3.8.2 **OR**

C. Performance of a slow start is impractical due to the need to maintain minimum operable equipment required by Tech Specs **OR**

D. The Unit SRO has authorized a manual fast start to support testing.

Excerpts from OPL171.038:

OPL171.038 Diesel Generator and Auxiliary Power System Rev 27 U1

Outline of Instruction	Instructor Notes and Methods (optional)
(1) Synchronizing equipment	
(2) Indicators for voltage, current, KW, KVAR are provided for each diesel. (Panel 9-23 KVAR meter indicates on-scale for lagging (outgoing) KVAR only.)	NLOR OBJ. 9 NLO OBJ. 9
(3) Voltage and current loads on each bus, board and breaker are provided	
(4) All Control Room equipment and controls are duplicated for individual diesels, buses or boards at the respective 4kV shutdown boards. Controls & Indications may vary slightly in nomenclature and physical layout	
The NORM / EMERG transfer switch (43) associated with the output breaker, when placed to EMERG transfers control of the Diesel Generator to the 4KV Shutdown Board	NLO OBJ. 11 NLOR OBJ. 11
(5) Central Information Panel (units1/2)	
(a) Located in Diesel Generator Building	
(b) Presents diesel loading information	
(c) Communications center to monitor load, dispatch supervision when diesels are on backup control	
4. Accident Signal Logic	
a) Two separate start signals are produced from Core Spray initiation logic, which then seal-in until reset at PNL 9-3	
(1) **PAS** (Pre Accident Signal from any unit will start all 8 diesel generators) Drywell pressure above 2.45 psig OR RPV water level below -122 inches.(Level 1)	OI-75
(2) **CAS** (Common Accident Signal from any unit will trip DG output breakers if closed) Drywell pressure above 2.45 psig AND Reactor pressure below 450 psig OR RPV water level below -122 inches.(Level 1)	
(3) **CAS** will trip the diesel generator output breaker if it is closed at the time the CAS signal developed. In this case, re-closure of the breaker is inhibited for three seconds to allow the closing spring to recharge	
(4) **CAS** will remove ALL generator protective trips into the 86G lockout relay except differential overcurrent (87). (overspeed will also trip DG breaker with CAS present)	
b) Unit Priority Re-Trip	
(1) Following a Common Accident Signal (which trips the diesel breakers), a second diesel breaker trip on a "unit priority" basis is provided to ensure that the diesel supplied 4kV SD	ILT OBJ. 6 LOR OBJ. 4

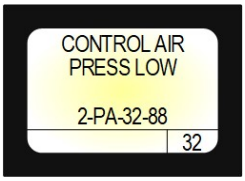
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
300000 (SF8 IA) Instrument Air **K3.12** (10CFR 55.41.7) Knowledge of the effect that a loss or malfunction of the Instrument Air System will have on the following systems or system parameters: • Feedwater system	Tier #	2	------
	Group #	1	------
	K/A #	300000K3.12	
	Importance Rating	3.5	------

Proposed Question: **# 50**

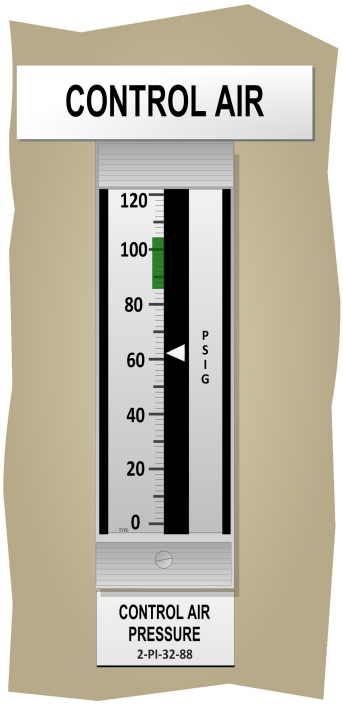
Unit 2 is operating at 100% RTP when the following conditions occur:

- CONTROL AIR PRESSURE LOW (2-9-20B, Window 32) alarms
- Control Air Pressure is as indicated and lowering



Given the conditions above, 2-FCV-3-20(13)(6), RFP 2A(B)(C) MINIMUM FLOW VALVES, **(1)** if the trend continues.

By design, RFP 2A(B)(C) MIN FLOW VALVES, normally automatically open when RFP flow is less than **(2)**.



A. (1) OPEN
(2) 2000 gpm

B. (1) OPEN
(2) 3000 gpm

C. (1) CLOSED
(2) 2000 gpm

D. (1) CLOSED
(2) 3000 gpm

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with the given Alarm Response Procedure for CONTROL AIR PRESSURE LOW (2-9-20B, Window 32), when Control Air Pressure lowers to 70 psig, the alarm comes in and the Operator is required to reference 2-AOI-32-2, Loss of Control Air. 2-AOI-32-2 list expected system responses for the Feedwater during a loss of Control Air resulting in 2-FCV-3-20(13)(6), RFP 2A(B)(C) MINIMUM FLOW VALVES, failing open and can be isolated locally at RFP rooms. Given that Control Air Pressure is as indicated (just above 60 psig) and lowering, the candidate must take into account that the 0-FCV-33-1, SERVICE AIR TO CONTROL AIR SUPPLY VALVE, should have opened at 85 psig Control Air Pressure to augment Control Air. Additionally, 0-FCV-33-1 fails closed on a loss of Control Air Pressure at 30 psig and lowering in order to still supply Service Air loads.
For second part, in accordance with 2-OI-3, Reactor Feedwater System, if RFP Turbines are running, RFP 2A(B)(C) MINIMUM FLOW VALVES, automatically open when RFP flow is less than 2000 gpm.

B INCORRECT: First part is correct (*See B*). Second part in incorrect but plausible in that if RFP Turbines are running, RFP 2A(B)(C) MINIMUM FLOW VALVES, automatically close when RFP flow is greater than 3000 gpm.

C INCORRECT: First part is incorrect but plausible in that various system valves fail as is, open, or closed when a loss of air or power occurs often leading to confusion for candidates. 2-LCV-3-53, STARTUP BYPASS VALVE, is air to open, spring to close and fails AS IS on a loss of Control Air and on a loss of power/control signal, fails closed.

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the effect that a loss or malfunction of the Instrument Air System will have on the Feedwater System and parameters. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine the operational effect based on alarms and readings impacting Control Air, Service Air, and Feedwater Systems.

Technical Reference(s): 2-OI-3, Rev. 163 (Attach if not previously provided)
2-ARP-9-20B, Rev. 29
2-AOI-32-2, Rev. 39

Proposed references to be provided to applicants during examination: **CONTROL AIR PRESSURE LOW (2-9-20B, Window 32)**
Panel 2-9-20, 2-PI-32-88, Control Air Pressure

Learning Objective: OPL171.054, Obj. 8 (As available)

Form 4.2-1 **Written Examination Question Worksheet**

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.012-01 002 #596	(Note changes or attach parent)
Question History:	New		
	Last NRC Exam		

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

10 CFR Part 55 Content:
- 55.41 **X**
- 55.43

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

596. OPL171.012-01 002

Given the following:

- Unit 2 is performing a reactor startup following a refueling outage
- 'G' Control Air Compressor is tagged for repairs
- Reactor pressure is currently approximately 145 psig
- Reactor water level is being controlled using 2-LCV-3-53
- The following events then occur
    - An electrical fault results in a loss of 480V Common Board 1
    - The remaining two control air compressors ('A' and 'D') trip on fault
    - Control air pressure has lowered to approximately 62 psig

Which ONE of the following completes the statements below?

As a result of this transient, 2-LCV-3-53 is expected to ___(1)___.

As a result of this transient, the reactor feedwater pump minimum flow valves (2-FCV-3-20, 2-FCV-3-13, and 2-FCV-3-6) are expected to ___(1)___.

A. (1) fail **AS-IS** and can be manually isolated or bypassed locally
(2) fail **OPEN** and can be isolated locally

B. (1) fail **AS-IS** and can be manually isolated or bypassed locally
(2) fail **CLOSED** and can be opened locally

C. (1) fail **OPEN** but can be manually isolated or bypassed locally
(2) fail **OPEN** and can be isolated locally

D. (1) fail **OPEN** but can be manually isolated or bypassed locally
(2) fail **CLOSED** and can be opened locally

Excerpt from 2-ARP-9-20B:

BFN Unit 2	2-XA-55-20B	2-ARP-9-20B Rev. 0029 Page 39 of 42

CONTROL AIR PRESS LOW
2-PA-32-88

32

(Page 1 of 1)

Sensor/Trip Point:

Relay 20B 2-PS-32-88 70 psig

Sensor Location: Relay 20B PS-32-24, Panel 9-20 Panel 25-275

Probable Cause:
A. Leak(s).
B. Valve malfunction or incorrect alignment.
C. Control air dryer malfunction.
D. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** for indications of control air pressure lowering.
- On Panel 2-9-20, CONTROL AIR PRESSURE, 2-PI-32-88.
- ICS Display (AIR COMPRESSOR G).
- ANN: SCRAM PILOT AIR HEADER PRESS LOW (2-XA-55-5B, Window 28)

B. **DISPATCH** PERSONNEL to investigate. (Coordinate with Units 1 and 3.)

C. **REFER TO** 0-AOI-32-1 and 2-AOI-32-2.

References: 0-45E620-13 0-45E769-5 2-45E769-13
2-47E610-32-3

Excerpts from 2-AOI-32-2:

BFN Unit 2	Loss of Control Air	2-AOI-32-2 Rev. 0039 Page 6 of 25

4.0 OPERATOR ACTIONS

NOTE

[NER/C] Attachment 1 provides expected system responses, critical components that do not fail in intended positions should be placed in the required positions. [INPO SOER 88-001]

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **IF** a RFP Minimum Flow Valve failed open and flow is required from the condensate/feedwater system to reactor vessel or to prevent pump overload, **THEN**

ISOLATE the associated RFP minimum flow lines in the appropriate RFPT Room as follows: (N/A any RFP valves not affected).

- RFP 2A MIN FLOW SHUTOFF, 2-SHV-003-0508
- RFP 2B MIN FLOW SHUTOFF, 2-SHV-003-0517
- RFP 2C MIN FLOW SHUTOFF, 2-SHV-003-0526

[2] **IF** CNDS BSTR PUMPS DISCH BYPASS TO COND B, 2-FCV-2-29A

and

CNDS BSTR PUMPS DISCH BYPASS TO COND C, 2-FCV-2-29B fail CLOSED, **THEN** (Otherwise N/A)

- **ENSURE** a flow path for condensate system

  OR

- **STOP** the condensate pumps/booster pumps using 2-OI-2.

[3] **IF** any outboard MSIVs fails closed, **THEN**

PLACE associated hand-switch on Panel 2-9-3 to CLOSE position. (Otherwise N/A)

Attachment 1
(Page 2 of 6)

Expected System Responses

3.0 FEEDWATER

A. RFP 2A(B)(C) MIN FLOW VLVS 2-FCV-3-20(13)(6) fail open and can be isolated locally at RFP rooms.

B. STARTUP BYPASS VALVE, 2-LCV-3-53, fails as-is and can be manually isolated or bypassed locally.

4.0 HEATER DRAINS AND VENTS

A. FEEDWATER HEATER 2A(B)(C)-1 DRAIN TO HTR 2A(B)(C)-2, 2-LCV-6-1(19)(37) fail OPEN.

B. FEEDWATER HEATER 2A(B)(C)-2 DRAIN TO HTR 2A(B)(C)-3, 2-LCV-6-4A(22A)(40A) fail CLOSED.

C. FEEDWATER HEATER 2A(B)(C)-2 COND DUMP, 2-LCV-6-4B(22B) (40B) fail OPEN.

D. FEEDWATER HEATER 2A(B)(C)-3 DRAIN TO HTR 2A(B)(C)-4, 2-LCV-6-7(25)(43) fail OPEN.

E. FEEDWATER HEATER 2A(B)(C)-4 DRAIN TO HTR 2A(B)(C)-5, 2-LCV-6-11A(29A)(47A) fail CLOSED.

F. FEEDWATER HEATER 2A(B)(C)-4 COND DUMP, 2-LCV-6-11B(29B)(47B) fail OPEN.

G. FEEDWATER HEATER 2A(B)(C)-5 DRAIN TO CNDR, 2-LCV-6-14A(32A)(50A) fail CLOSED.

H. FEEDWATER HEATER 2A(B)(C)-5 COND DUMP, 2-FCV-6-14B(32B)(50B) fail OPEN. These LCVs can be isolated by closing 2-FCV-6-97(98)(99) from Panel 2-9-7.

I. Moisture Separator level control reservoir high level dump drain LCVs to condenser fail open.

J. Moisture Separator level control reservoir isolation valves(FCV-6-62A, 62B, 73A, 73B, 84A, 84B) to A2, B2, and C2 high pressure heater shell side fail closed.

K. Condensate Injection valve (2-FCV-002-0144) to Moisture Separator Drain System fails closed.

Attachment 1
(Page 3 of 6)

Expected System Responses

5.0 RCW

A. All RCW temperature control valves fail open except for 2-TCV-24-80B and 2-TCV-24-85B on 2A and 2B RBCCW heat exchangers and 2-TCV-024-0075B on the Main Turbine Oil Coolers (4" line) which fail CLOSED.

6.0 RAW SERVICE WATER

A. RSW Head Tank outlet valve FCV-25-32 fails CLOSED. A high pressure fire pump should be started immediately to maintain fire system pressure.

7.0 HIGH PRESSURE FIRE PROTECTION

A. Fire header pressure control valve 0-PCV-26-4 located at the Intake will fail CLOSED. Relief valve will maintain fire header pressure < 175 psig.

8.0 SERVICE AIR

A. Service air supply to control air 0-FCV-33-1 opens at 85 psig Control Air press but will fail CLOSED on loss of control air at 30 psig.

9.0 CONTROL AIR

A. Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

B. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

Excerpts from 2-OI-3: Also supports Distractors B2, D2, C1, D1

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Y. RFP 2A(2B)(2C) MIN FLOW VALVE, 2-FCV-3-20(13)(6) operation is as follows:

1. When handswitch 2-HS-3-20(13)(6) is in AUTO:

   a. Valve will open if RFPT is running and RFP flow is less than 2000 gpm (~1 x 10^6 lbm/Hr).

   b. Valve will close if RFPT is running and RFP flow is greater 3000 gpm (~than 1.5x10^6 lbm/Hr).

2. When handswitch 2-HS--3-20(13)(6) is taken to CLOSE, valve will travel closed after 150 second time delay.

3. When handswitch 2-HS--3-20(13)(6) is in OPEN, valve will open.

4. With 2-HS-3-20(13)(6) in AUTO, minimum flow valve will close when associated RFPT is tripped.

Z. Startup bypass valve, 2-LCV-3-53, is air to open, spring to close. The valve fails as is on loss of air (65 psig). On a loss of power or loss of control signal, 2-LCV-3-53 fails closed.

AA. RFP 2A(2B)(2C) SUCTION VALVE, 2-FCV-2-83(95)(108), will not open if bearing oil pressure is low (less than 4 psig).

BB. RFP Suction Valves may require manual operation to open when a Condensate Booster Pump is in service due to elevated differential pressure across the valve.

CC. Total Feedwater flow provides inputs to Feedwater Level Control System, Rod Worth Minimizer, Recirc Flow Control System, Feedwater Integrator, and Feedwater flow recorder.

DD. RX WATER LEVEL NARROW RANGE instruments, 2-LI-3-208A(D) on Panel 2-9-5 and 2-LI-3-208B(C) on Panel 2-9-3 should be monitored closely during reactor level transients. These instruments provide inputs to Main Turbine, Reactor Feedpump Turbine, HPCI, and RCIC high water level trip logic.

EE. [TSAR] Operation of Reactor Feed Pump Turbines with inoperable high level trip instrumentation may result in equipment damage or personnel hazard. [Item D-91]

FF. It is acceptable to Shut Down Vapor Extractor for up to 24 hours since this should not cause any detrimental effect to the system. If the Extractor needs to be out longer than 24 hours, an evaluation of the effects on the system should be done by System Engineering. [SEOPR 96-2/3 047-003]

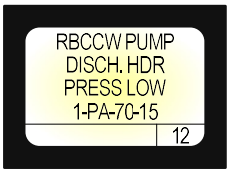
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
400000 (SF8 CCW) Component Cooling Water **K6.11** (10CFR 55.41.7) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Component Cooling Water System: • Recirculation system	Tier #	2	------
	Group #	1	------
	K/A #	400000K6.11	
	Importance Rating	2.9	------

Proposed Question: **# 51**

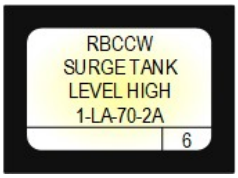
Unit 1 is operating at 100% RTP with the following plant conditions:

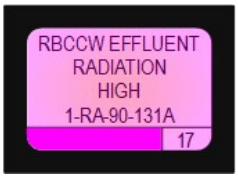
- RBCCW PUMP DISCHARGE HEADER PRESSURE LOW (1-9-4C, Window 12) in alarm
- 1-FCV-70-48, RBCCW SECTIONALIZING VALVE, is closed



Subsequently, events occur resulting in the following conditions:

- RBCCW SURGE TANK LEVEL HIGH (1-9-4C, Window 6) in alarm
- RBCCW EFFLUENT RADIATION HIGH (1-9-3A, Window 17) in alarm





The conditions above indicate leakage into the RBCCW System from _____.

A. RWCU Pump seal coolers

B. Reactor Recirc Pump seal coolers

C. Fuel Pool Cooling Heat Exchangers

D. RWCU Non-Regenerative heat exchangers

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that Reactor Water Cleanup (RWCU) is cooled by Reactor Building Closed Cooling Water (RBCCW) but is a non-essential loop load, and is isolated when 1-FCV-70-48, RBCCW SECTIONALIZING VALVE, is closed. Therefore, RWCU Pump seal coolers cannot be the source of the leak, since 1-FCV-70-48 is closed.

B **CORRECT**: (*See attached*) Reactor Recirc Pump seal coolers is an essential loop load, and is not isolated when 1-FCV-70-48 is closed. Therefore, given the answer choice, Reactor Recirc Pump seal coolers are the likely cause of the leakage into RBCCW.

C INCORRECT: Incorrect but plausible in that Fuel Pool Cooling is cooled by RBCCW, but is a non-essential loop load and is isolated when 1-FCV-70-48, RBCCW SECTIONALIZING VALVE, is closed. Therefore, Fuel Pool Cooling Heat Exchangers cannot be the source of the leak, since 1-FCV-70-48 is closed.

D INCORRECT: Incorrect but plausible in that the RWCU Non-Regenerative Heat Exchangers are cooled by RBCCW, but are a non-essential loop load and is isolated when 1-FCV-70-48, RBCCW SECTIONALIZING VALVE, is closed. Therefore, the RWCU Non-Regenerative Heat Exchangers cannot be the source of the leak, since 1-FCV-70-48 is closed.

RO Level Justification: Tests the candidate's ability to recognize and diagnose the effect of a Recirculation System leak into Component Cooling Water. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-OI-70, Rev. 62 (Attach if not previously provided)
1-ARP-9-4C, Rev. 31
1-ARP-9-3A, Rev. 55
OPL171.047, Rev. 14

Proposed references to be provided to applicants during examination: **RBCCW PUMP DISCHARGE HEADER PRESSURE LOW (1-9-4C, Window 12)**
RBCCW SURGE TANK LEVEL HIGH (1-9-4C, Window 6)
RBCCW EFFLUENT RADIATION HIGH (1-9-3A, Window 17)

Learning Objective: OPL171.047, Obj. 2, 3 (As available)

Question Source:

Bank #	ILT 1102 NRC #25
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam 2011

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 25**

Unit 1 is at 100% Reactor Power with the following system line ups:

- Reactor Building Closed Cooling Water (RBCCW) Pumps 1A **AND** 1B are in service
- Reactor Water Cleanup (RWCU) Pumps 1A **AND** 1B are in service
- Fuel Pool Cooling and Cleanup (FPCC) Pump 1A is in service

Unit 1 Reactor **Scrams AND** the following alarms / indications are received:

- 480 V Shutdown Board 1A is locked out
- RBCCW SURGE TANK LEVEL HIGH, (1-9-4C, Window 6)
- RBCCW EFFLUENT RADIATION HIGH, (1-9-3A, Window 17)
- RX BLDG,TURB BLDG, RF ZONE EXH RADIATION HIGH, (1-9-3A, Window 4)

Which ONE of the following is a potential cause of the alarms?

Leakage into RBCCW from ____.

A. Reactor Recirc Pump seal coolers

B. Fuel Pool Cooling Heat Exchangers

C. Reactor Water Cleanup Pump Seal Coolers

D. Reactor Water Cleanup Non-Regenerative Heat Exchangers

Proposed Answer: **A**

Excerpt from 1-OI-70:

BFN Unit 1	Reactor Building Closed Cooling Water System	1-OI-70 Rev. 0062 Page 7 of 86

3.0 PRECAUTIONS AND LIMITATIONS

A. The Spare RBCCW Pump and RBCCW Heat Exchanger are common to all three (3) units. When it becomes necessary for the spare RBCCW Pump and RBCCW Heat Exchanger to be placed in service, operators on all three units shall communicate and be fully aware of RBCCW conditions. Common equipment will be aligned in accordance with 2-OI-70.

B. When removing a RBCCW heat exchanger from service, RBCCW flow should be stopped prior to stopping RCW or EECW cooling water flow.

C. Any water removed from RBCCW is considered potentially contaminated.

D. On low RBCCW pump discharge header pressure (57 psig), nonessential equipment isolation valve 1-FCV-070-0048 automatically closes and must be reopened manually using 1-HS-70-48. This interlock is bypassed when RBCCW SECTIONALIZING VLV XFER SW, 1-XS-070-0048 on 480V RMOV Board 1B, compartment 5A, is placed in EMERGENCY.

E. When placing an RBCCW heat exchanger in service, prior to opening an RBCCW heat exchanger inlet valve, placing the RBCCW SECTIONALIZING VLV XFER SW at the 480V reactor MOV board 1B, compartment 5A to EMERG will temporarily bypass the header low pressure auto closure for 1-FCV-070-0048.

F. When the transfer switches for RBCCW pump breakers are in the EMERGENCY position, Units 1 and 2 RBCCW pump load shedding logic is defeated.

G. [NRC/C] When the RCW, EECW, or RBCCW supplied to any RBCCW heat exchanger is put into service or taken out of service, Chemistry shall be notified so any required sampling can be initiated or stopped as determined by the status of RCW, EECW, and RBCCW. [NRC LER 259/88010]

H. [CAQR/C] When the RBCCW system is drained for more than 30 days, the Technical Support Supervisor shall be notified in order that a specific lay-up configuration is established by the System Engineer to prevent system degradation. All planned draining of RBCCW should be coordinated with the Chemistry Supervisor and the Radwaste Coordinator. In addition, Chemistry should be notified of any leaks discovered which allows RBCCW to discharge to floor drains. [CAQR BFP900249]

Excerpt from OPL171.047 Lesson Plan:

OPL171.047, Reactor Building Closed Cooling Water System, Rev. 14

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
2. RBCCW Heat Loads	
a Essential loop loads	Obj. ILT 2, NLO/NLOR 2
• Drywell Blowers(10)	
• Reactor recirculation pump motor coolers (2)	
• Reactor recirculation pump seal coolers (2)	
• Drywell equipment drain sump heat exchanger (1)	
b Non-essential loop loads	Obj. ILT 3
• Reactor Building equipment drain sump heat exchanger (1)	Obj. NLO/NLOR 3
• Reactor water cleanup pump heat exchanger (2)	
• Non-regenerative heat exchangers (2)	
• Fuel pool cooling heat exchangers (2)	
• Reactor recirculation pump sample discharge cooler (1)	
3. RBCCW Heat Exchangers	
a These provide the means for heat removal from RBCCW by RCW with Emergency Equipment Cooling Water (EECW) as a backup.	
b They are counter-flow type, 50% capacity each.	
• RBCCW flow makes one pass through the shell side.	
• RCW makes one pass through the tube side.	
• RBCCW flow is in the opposite direction to RCW flow.	
c The spare RBCCW heat exchanger has manual isolation valves which allow it to be lined up to Unit 1, 2, or 3.	

Excerpts from 1-ARP-4C:

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0031 Page 19 of 43

RBCCW PUMP DISCH HDR PRESS LOW

1-PA-70-15

12

(Page 1 of 1)

Sensor/Trip Point:

1-PS-70-15	50 psig (57 psig on local pump discharge pressure gauge)

Sensor Location: 1-LPNL-925-0008
Elevation 593
Column R-2 R-Line

Probable Cause:
A. Failed pump.
B. Broken coupling.

Automatic Action: Closes RBCCW SECTIONALIZING VLV, 1-FCV-70-48.

Operator Action:
A. **ENSURE** RBCCW SECTIONALIZING VLV, 1-FCV-70-48 closed.
B. **CHECK** RBCCW pumps A and B in service.
C. **CHECK** RBCCW Surge Tank Level Low alarm is RESET.
D. **DISPATCH** personnel to check:
- RBCCW surge tank level locally.
- RBCCW pumps for proper operation.

E. **REFER TO** 1-AOI-70-1 for RBCCW System failure and 1-OI-70 for starting spare pump.

References: 1-45E779-6 1-47E610-70-1
FSAR Sections 10.6.4 and 13.6.2

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0031 Page 12 of 43

RBCCW
SURGE TANK
LEVEL HIGH

1-LA-70-2A

6

(Page 1 of 2)

Sensor/Trip Point:

1-LS-070-0002A 4 Inches Above Center Line of Tank

Sensor Location: RBCCW surge tank on the fourth floor in the M-G set room.

Probable Cause:
A. Makeup valve 1-FCV-70-1 open.
B. Bypass valve 1-2-1369 leaking.
C. Leak into the system.

Automatic Action: None

Operator Action:
A. **ENSURE** make-up valve 1-FCV-70-1 closed, using RBCCW SYS SURGE TANK FILL VALVE, 1-HS-70-1, on Panel 1-9-4.
B. **CHECK** RBCCW PUMP SUCTION HDR TEMP, 1-TIS-70-3, indicates water temperature is 100°F or less, on Panel 1-9-4.
C. **DISPATCH** personnel to verify high level, ensure bypass valve, 1-2-1369, is closed and observe sight glass level.
D. **OPEN** surge tank drain valve, 1-70-609
   1. **CLOSE** surge tank drain valve, 1-70-609 when desired level is obtained.

E. **IF** level continues to rise, **THEN**

   REQUEST Chemistry to pull and analyze a sample for total gamma activity and attempt to qualify source of leak.
F. **CHECK** activity reading on 1-RM-90-131D (1-RR-90-134 Ch 3).

Continued on Next Page

RBCCW SURGE TANK LEVEL HIGH 1-LA-70-2A, Window 6
(Page 2 of 2)

Operator Action: (Continued)

NOTE

[NER/C] Reactor Recirculation Pump seal cooler leakage may be indicated by a rise in 1-RM-90-131 (Panel 1-9-10) activity (1-RR-90-134 Ch 3 on Panel 1-9-2) or 1-TE-68-54 or 67 temperature (Panel 1-9-21) or lowering of any Recirc pump seal pressure.

G. **IF** it is suspected that the Reactor Recirculation Pump seal cooler is leaking, **THEN**
PERFORM the following:
1. **DETERMINE** which Reactor Recirculation loop is leaking
2. **ISOLATE** affected loop, at the discretion of the Unit SRO (Refer to 1-OI-68 Section 7.1 or 7.2 as applicable).

CAUTION

COOLDOWN may be required prior to re-start of isolated Reactor Recirculation Pump to prevent hanger or shock suppressors from exceeding their maximum travel range [GE SIL 251, 430, and 517]

3. **WHEN** primary system pressure is below 125 psig and at the discretion of the Unit SRO, **THEN**
ISOLATE, as necessary, the RBCCW System to preclude damage to the RBCCW PIPING.[IEN 89-054, GE SIL-459]

H. **START** selective valving to determine in-leakage source, if present.

References: 1-45E620-4 1-47E610-70-1 1-47E822-1
FSAR Section 10.6.4 and 13.6.2

Excerpts from 1-ARP-3A:

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0055 Page 29 of 60

RBCCW EFFLUENT
RADIATION
HIGH
1-RA-90-131A

17

(Page 1 of 2)

Sensor/Trip Point:

	HI	HI-HI
1-RM-90-131D	(Note 1) Alarm from recorder	(Note 1) Alarm from drawer

(Note 1) Contact chemistry for current setpoints per 0-TI-45.

Sensor Location: 1-RE-090-0131A (off-line) RBCCW HX, Rx Bldg, EL 593', R-R2

Probable Cause: Hx tube leak into RBCCW system.

Automatic Action: None

Operator Action:

A. **DETERMINE** cause of alarm by observing the following:
   1. RBCCW AND RCW EFFLUENT RADIATION recorder, 1-RR-90-134 Channel 3, on Panel 1-9-2
   2. RBCCW EFFLUENT OFFLINE RAD MON, 1-RM-90-131D, Panel 1-9-10.

B. **NOTIFY** Chemistry to sample RBCCW for total gamma activity to verify condition.

C. **DETERMINE** if source of leak is RWCU Non-Regenerative Heat Exchanger, Fuel Pool Cooling, or Reactor Water Sample, Recirc Pump A or B Seal Water Heat Exchanger(s).

D. [NER/C] **CHECK** the following for indication of Recirc Pump Seal Heat Exchanger leak:
   - LOWERING in Reactor Recirculation Pump 1A(1B) N0. 1 or 2 SEAL, 1-PI-68-64A or 1-PI-68-63A (1-PI-68-76A or 1-PI-68-75A) on Panel 1-9-4.
   - Temperature rise on CLG WTR FROM SEAL CLG 1-TE-68-54, on RECIRC PMP MTR 1A WINDING AND BRG TEMP temperature recorder, 1-TR-68-58, on Panel 1-9-21.
   - Temperature rise on CLG WTR FROM SEAL CLG 1-TE-68-67, on RECIRC PMP MTR 1B WINDING AND BRG TEMP temperature recorder, 1-TR-68-84, on Panel 1-9-21.

Continued on Next Page

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0055 Page 30 of 60

RBCCW EFFLUENT RADIATION HIGH 1-RA-90-131A, Window 17
(Page 2 of 2)

Operator Action: (Continued)

E. **IF** it is determined the source of leakage is from Reactor Recirc Pump A(B), **THEN**, at the discretion of the Unit SRO, perform the following:
   1. **ISOLATE** Reactor Recirculation Loop A(B) per 1-OI-68 as applicable.

NOTE

Cooldown is required to prevent hangers or shock suppressors from exceeding their maximum travel range.

   2. **WHEN** primary system pressure is less than 125 psig, **THEN,** at the discretion of the Unit SRO
      a. **ISOLATE** RBCCW System to preclude damage to RBCCW piping. [IEN 89-054, GE SIL-459]

F. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References:

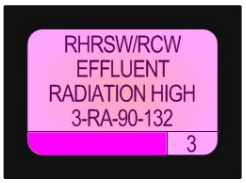
1-47E610-90-3	1-45E620-3	0-47W600-178
729E814 Series	0-TI-45	GE 1-729E814-3

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
510000 (SF4 SWS) Service Water (Normal and Emergency)	Tier #	2	------
K5.02 (10CFR 55.41.4)	Group #	1	------
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Service Water System:	K/A #	510000K5.02	
♦ Radiation alarm response	Importance Rating	3.0	------

Proposed Question: **# 52**

Unit 3 is placing RHR Loop II in Shutdown Cooling in service to support a Refueling outage when the following occurs:



- RHRSW/RCW EFFLUENT RADIATION HIGH (3-9-3A, Window 3) alarms

Once the high radiation condition clears, 3-RM-90-133D(134D), RHR HEAT EXCHANGER A/C(B/D) SERVICE WATER DISCHARGE OFFLINE RADIATION MONITOR, must be reset on Panel **(1)** **BEFORE** the associated annunciator window can be manually reset.

In accordance with 0-OI-23, RHRSW System, to ensure the Off-line Radiation monitor sample pumps will auto start, the **MINIMUM** flow rate from any in-service RHRSW heat exchanger must exceed **(2)**.

A. (1) 9-3
(2) 600 gpm

B. (1) 9-3
(2) 1700 gpm

C. (1) 9-10
(2) 600 gpm

D. (1) 9-10
(2) 1700 gpm

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the majority of operational controls, indications, and alarms for the RHRSW System are located on Panel 3-9-3 in the Main Control Room. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that in accordance with 0-OI-23, RHRSW System, to prevent damage to RHRSW pumps, minimum flow is 1700 gpm per pump in service.

C **CORRECT**: *(See attached)* In accordance with the given Alarm Response Procedure for 3-9-3A, Window 3, alarms for 3-RM-90-132, 3-RM-90-133, 3-RM-90-134 latch-in and must reset on Panel 3-9-10 in the Main Control Room using reset pushbutton on the Radiation Monitor in alarm before the window on Panel 3-9-3 can be reset. For second part, in accordance with 0-OI-23, RHRSW System, during RHRSW low flows, such as Shutdown Cooling split flows, the initial flow rate from any in-service RHRSW heat exchanger must exceed 600 gpm. This will ensure operation of the off-line Radiation Monitors which receive their start signal from a Time Delay Pickup relay energized by the RHRSW heat exchanger's discharge flow element.

D INCORRECT**:** First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's knowledge of the operational implications of radiation alarm response as it applies to the Residual Heat Removal Service Water System (RHRSW). This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to the RHRSW System.

Technical Reference(s): 0-OI-23, Rev.109 (Attach if not previously provided)

3-ARP-9-3A, Rev. 60

OPL171.046, Rev. 20

Proposed references to be provided to applicants during examination: **RHRSW/RCW EFFLUENT RADIATION HIGH, (3-9-3A, Window 3**

Learning Objective: OPL171.046, Obj. 7c (As available)

Question Source: Bank #

Modified Bank # ILT EXAM BANK OPL171.046-07 004 #1899 (Note changes or attach parent)

Question History: New

Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

Comments:

Copy of Bank Question:

1899. OPL171.046-07 004

Unit 3 is placing Loop II of Suppression Chamber Sprays in service for testing. The Unit Operator is placing the D RHRSW Heat Exchanger in operation.

Which ONE of the following completes the statements below?

In accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW) System, Off-line Radiation monitors receive their start signal from a relay energized by the RHRSW Heat Exchanger's __(1)__ flow element.

The flow rate from any in-service RHRSW Heat exchanger must exceed a **MINIMUM** of __(2)__ to ensure the Off-Line Radiation monitor sample pump will auto start.

A. (1) discharge
(2) 1700 gpm

B. (1) inlet
(2) 600 gpm

C✓ (1) discharge
(2) 600 gpm

D. (1) inlet
(2) 1700 gpm

Excerpts from 3-ARP-9-3A:

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 8 of 59

RHRSW/RCW EFFLUENT RADIATION HIGH
3-RA-90-132
3

(Page 1 of 3)

Sensor/Trip Point:

	HI	HI-HI
3-RM-90-132D (RCW)	(NOTE 1)	(NOTE 1)
3-RM-90-133D (RHRSW A&C)	(NOTE 1)	(NOTE 1)
3-RM-90-134D (RHRSW B&D)	(NOTE 1)	(NOTE 1)

(1) 0-TI-45 provides HI and HI-HI actual alarm setpoint values for each affected radiation monitor channel ratemeter. Alarm setpoints are conservative and serve to alert UO of approach to ODCM radionuclide discharge limits.

0-TI-45 also provides ODCM calculated setpoint limits as HI and HI-HI maximum allowable setpoint limits which are to be used to monitor ODCM compliance using affected ratemeter readings.

Sensor Location:
3-RE-090-0132A Panel 25-336, El 565', Col R-20 U-LINE
3-RE-090-0133A Panel 25-337, El 565', Col R-19 U-LINE
3-RE-090-0134A Panel 25-338, El 565', Col R-19 U-LINE

Probable Cause:
A. High radionuclide levels in RCW effluent due to possible leaks in RBCCW HX.
B. High radionuclide levels in RHRSW due to possible tube leaks in RHR HX.
C. Sensor malfunction.

Automatic Action:
None

NOTE
Alarms for 3-RM-90-132 / 3-RM-90-133 / 3-RM-90-134 latch-in and must be reset on Panel 3-9-10 using reset pushbutton on the radiation monitor in alarm before the window can be reset

Operator Action:
A. **DETERMINE** cause of alarm by observing following:
1. RHRSW, RBCCW AND RCW EFFLUENT RADIATION recorder, 3-RR-90-134 Channels 1, 2 & 4 on Panel 3-9-2.
2. RCW EFFLUENT OFFLINE RAD MON, 3-RM-90-132D on Panel 3-9-10.
3. RHR HX A/C SW DISCH OFFLINE RAD MON 3-RM-90-133D.
4. RHR HX B/D SW DISCH OFFLINE RAD MON 3-RM-90-134D.

B. **IF** 3-RM-90-132 / 3-RM-90-133 / 3-RM-90-134 is latch-in, **THEN RESET** using pushbutton on the radiation monitor in alarm.

Continued on Next Page

RHRSW/RCW EFFLUENT RADIATION HIGH 3-RA-90-132, Window 3
(Page 2 of 3)

Operator Action: (Continued)

C. **IF** 3-RM-90-133 or 3-RM-90-134 is causing alarm, **THEN PERFORM** the following:
   1. **NOTIFY** Unit SRO.
   2. **IMMEDIATELY SUSPEND** liquid effluent discharge by removing affected RHR heat exchanger from service. **REFER TO** 3-OI-74.
      - **CLOSE** affected RHR HX RHRSW discharge valve 3-FCV-23-34(40)(46)(52).
      - **CLOSE** affected RHR HX RHRSW inlet valve 3-SHV-023-0031(0037)(0043)(0049).
      - **PLACE** Caution Order on affected RHR HX RHRSW inlet SHV and outlet FCV.

D. **IF** 3-RM-90-132 is causing alarm and is reading at or above 1700 cpm, **THEN**
   PERFORM the following:
   1. **NOTIFY** Unit SRO.
   2. **IMMEDIATELY SUSPEND** liquid effluent discharge by removing affected RBCCW heat exchanger from service.
   3. **REFER TO** 3-OI-70.

E. **NOTIFY** RAD PRO.

F. **REQUEST** Chemistry perform radiochemical analysis to determine source.

G. **MONITOR** release rate for ODCM compliance using 0-TI-45 HI maximum allowable setpoint value for affected channel.

H. **IF** ODCM limits are exceeded, **THEN**
   REFER TO EPIP-1.

I. **MONITOR** other radiation monitor channels providing input to this annunciator frequently as these channels will be masked from alarming while this alarm is sealed-in.

J. **IF** alarm is due to sensor malfunction, **THEN**
   REFER TO 0-OI-55.

Continued on Next Page

Excerpt from 0-OI-23:

BFN Unit 0	Residual Heat Removal Service Water System	0-OI-23 Rev. 0109 Page 12 of 135

3.4 Initiation/Isolation/Trips

A. Off-line radiation monitors receive their start signal from a TDPU relay energized by the RHRSW heat exchanger's discharge flow element. During RHRSW low flows, such as shutdown cooling split flows, the initial flow rate from any in-service RHRSW heat exchanger must exceed 600 gpm. This will ensure operation of the off-line Radiation Monitor. After reaching 600 gpm, the flow can be split or lowered, while maintaining minimum flow requirements, to establish the desired cooldown rate or maintain consistent shutdown cooling temperatures.

B. An initiation signal from either Division I or Division II of any Units' Common Accident signal logic will result in the trip of RHRSW Pumps A2 and C2. This trip signal is sealed in for 60 seconds.

3.5 EECW System

A. RHRSW Pumps A3, B3, C3 and D3 are assigned to the EECW System. 0-OI-67 should be referred to for their operation.

B. RHRSW Pumps A1, B1, C1 and D1 may supply either the RHRSW or EECW systems, when used for EECW. REFER TO 0-OI-67 for their operation.

C. The start logic for RHRSW pump B1 has been modified by replacing the limit switch function associated with 0-SHV-067-0089 , RHRSW PMP B1 CROSSTIE TO EECW with a mode switch located on 4KV SD BD 3EC compt. 8. The mode switch will be placed in the EECW mode when 0-SHV-067-0089, RHRSW PMP B1 CROSSTIE TO EECW, is opened which aligns the RHRSW swing pump to the EECW header.

D. The start logic for RHRSW pump A1 has been modified by replacing the limit switch function associated with 0-SHV-067-0088 , RHRSW PMP A1 CROSSTIE TO EECW with a mode switch located on 4KV SD BD A compt. 10. The mode switch will be placed in the EECW mode when 0-SHV-067-0088, RHRSW PMP A1 CROSSTIE TO EECW, is opened which aligns the RHRSW swing pump to the EECW header.

E. The start logic for RHRSW Pump C1 has been modified by replacing the limit switch function associated with 0-SHV-067-0049 , RHRSW PMP C1 CROSSTIE TO EECW with a mode switch located on 4KV SD BD B compt. 10. The mode switch will be placed in the EECW mode when 0-SHV-067-0049, RHRSW PMP C1 CROSSTIE TO EECW, is opened which aligns the RHRSW swing pump to the EECW header.

Excerpt from OPL171.046:

G. System Interrelationships with RHRSW

1. EECW uses the # 3 RHRSW pumps to supply cooling water for the components cooled by EECW. The # 1 RHRSW pumps can be manually aligned to EECW if a # 3 pump fails and will automatically start on EECW automatic start signals when aligned to EECW use. However, the auto-start signals for the # 1 RHRSW pumps are not the same as the associated # 3 RHRSW pumps. They auto-start from different diesel starts or different spray pump starts. A #1 pump valved in for EECW must be running to maintain operability.

   EECW can also provide minimum flow protection during periods of RHRSW flow <1700 gpm.

   Obj. 7 ILT
   Obj. 6 LOR
   If a #1 RHRSW pump is used to supply EECW, the pump should be placed in service per 0-01-67.

   Crosstie of the RHRSW pump #2 to EECW requires all 3 pumps in the room to be declared inoperable along with associated RHR HXs.

2. RHRSW supplies flow to the RHR heat exchangers to cool RHR water in the suppression pool cooling, containment spray modes of operations, and to cool reactor water during the shutdown cooling mode of operation. RHRSW also supplies Standby Coolant flow to the RHR loops if the RHR pumps are not able to inject to the reactor.

   If RHRSW is lost while in shutdown cooling this would require entry into AOI-74-1

   Obj. 8 NLOR
   Obj. 11 NLO

3. The Process Radiation Monitoring system checks the common discharge piping of each RHRSW header for excessive radiation before it is discharged offsite. Alarms in the control room alert the Unit Operator if leakage from the RHR to the RHRSW system is occurring.

4. Power for each of the RHRSW pumps is provided by either a Unit 1/2 4KV shutdown board or a Unit 3 4KV shutdown board. Specific power supplies were listed previously in the lesson plan.

5. The Raw Service Water system maintains the RHRSW system pressurized when the pumps are not running (prevent water hammer upon pump start). The connections located on the Unit 2 supply headers are placed such that all three unit's headers remain pressurized.

6. Demineralized water is supplied to the heat exchanger for the purpose of flushing the heat exchangers and for refilling them after maintenance.

   Obj. 7.f ILT
   Obj. 6.f LOR

7. Chemistry control of RHRSW is provided by the Calgon Corp. The system connections to RHRSW are in the pump rooms.

   Chemistry should be notified prior to starting or stopping a RHRSW pump.

8. CCW pump pits provide water through the sluice gates to the RHRSW pits which provide suction for the RHRSW pumps.

   Obj. 7.h ILT
   Obj. 6.h LOR

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
201001 (SF1 CRDH) CRD Hydraulic	Tier #	2	------
291003, K1.04 (10CFR 55.41.5 / 7)	Group #	2	------
Controllers and Positioners	K/A #	291001K1.04	
• Function and operation of pressure and temperature controllers, including pressure and temperature control valves	Importance Rating	3.3	------

Proposed Question: **# 53**

Unit 2 is operating at 50% RTP with the following plant conditions:

- 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, is in automatic
- The OATC places 2-FCV-85-23A, CRD DRIVE WATER PRESSURE CONTROL VALVE, to CLOSE for one (1) second

As a result of the OATC's actions, CRD Drive Water Header Differential Pressure will **(1)**.

Once system parameters stabilize following the actions, CRD Cooling Water Flow will be **(2)**.

A. (1) rise
(2) lower

B. (1) rise
(2) the same

C. (1) lower
(2) lower

D. (1) lower
(2) the same

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that closing the Drive Water Pressure Control Valve could cause a restriction in the flow of any part of the CRD system. When 2-FCV-85-23A, CRD DRIVE WATER PRESSURE CONTROL VALVE, is closed, pressure rises and flow initially lowers. FCV-85-11A will automatically operate to return CRD Cooling Water Header Flow to normal.

B **CORRECT**: (*See attached*) Drive Water Header D/P is controlled by the position of 2-FCV-85-23A. When 2-FCV-85-23A is placed in CLOSE, Drive Water Header DP rises. For second part, when 2-FCV-85-23A, CRD DRIVE WATER PRESSURE CONTROL VALVE, is closed, pressure rises and flow lowers. FCV-85-11A will automatically operate to return CRD Cooling Water Header Flow to normal.

C INCORRECT: First part is incorrect in that operators routinely operate 2-FCV-85-23A, CRD DRIVE WATER PRESSURE CONTROL VALVE in the wrong direction when adjusting CRD Drive Water Header Differential Pressure, so there is confusion surrounding the operation of 2-FCV-85-23. Drive Water Header Differential Pressure rises when 2-FCV-85-23Ais closed. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's ability to monitor the automatic response of CRD System Flow Control following operation of a manual valve. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-85, Rev. 156 (Attach if not previously provided)
OPL171.005, Rev. 23

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.xxx Obj. xx (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK OPL171.005-13 001 #266 (Note changes or attach parent)
New	
Question History: Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

266. OPL171.005-13 001

The CRD system is in operation with the reactor operating at 50% power. The CRD System Flow Controller FIC-85-11 is in automatic.

During a pause in power ascension, the Unit Operator places the Drive Water Pressure Control Valve (85-23) control switch to the OPEN position for two seconds.

Which ONE of the following describes the new steady state values of drive water pressure and cooling water flow?

A. Drive water pressure will rise.
Cooling water flow will lower.

B. Drive water pressure will lower.
Cooling water flow will lower.

C. Drive water pressure will rise.
Cooling water flow will remain the same.

D. Drive water pressure will lower.
Cooling water flow will remain the same.

Excerpt from 2-OI-85:

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0156 Page 151 of 271

8.15 CRDM Low Drive Water DP Flush

[1] **REVIEW** all Precautions and Limitations in Section 3.6 and 3.7.

NOTE

The following steps will flush the CRDM without the possibility of moving a control rod.

[2] **LOWER** the CRD DRIVE WTR HDR DP, 2-PDI-85-17A to less than 60 psid, using CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A.

[3] **PLACE** CRD POWER, 2-HS-85-46, in ON.

[4] **SELECT** the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40.

[5] **PLACE** CRD CONTROL SWITCH, 2-HS-85-48, in ROD IN and **HOLD** to flush the control rod.

[5.1] **IF** control rod moves, **THEN**

RELEASE 2-HS-85-48 to STOP rod motion. (Otherwise N/A)

[5.1.1] **NOTIFY** Unit SRO and Reactor Engineer for further instructions.

[6] **CHECK** CRD DRIVE WTR HDR FLOW, 2-FI-85-15A has stabilized. (Rod should **NOT** insert)

[7] **RELEASE** CRD CONTROL SWITCH, 2-HS-85-48.

[8] **PLACE AND HOLD** CRD NOTCH OVERRIDE, 2-HS-85-47 in NOTCH OVERRRIDE.

[9] **PLACE AND HOLD** CRD CONTROL SWITCH, 2-HS-85-48 in ROD OUT NOTCH.

[9.1] **IF** control rod moves, **THEN**

RELEASE 2-HS-85-47 & 2-HS-85-48 to STOP rod motion. (Otherwise N/A)

[9.1.1] **NOTIFY** Unit SRO and Reactor Engineer for further instructions.

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0156 Page 152 of 271

8.15 CRDM Low Drive Water DP Flush (continued)

[10] **CHECK** CRD DRIVE WTR HDR FLOW, 2-FI-85-15A has stabilized. (Rod should **NOT** withdraw)

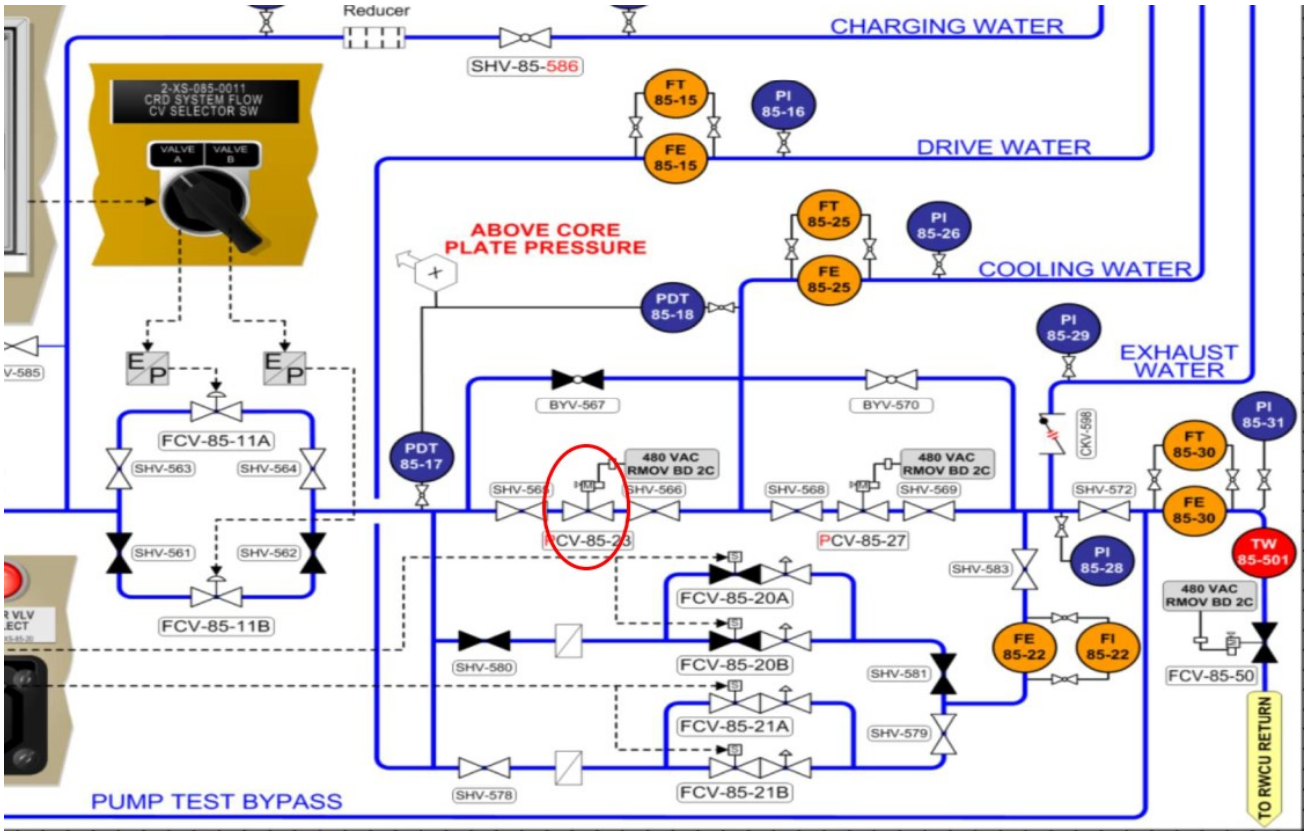
[11] **RELEASE** both CRD NOTCH OVERRIDE, 2-HS-85-47 and CRD CONTROL SWITCH, 2-HS-85-48.

[12] **PLACE** CRD POWER, 2-HS-85-46, in OFF.

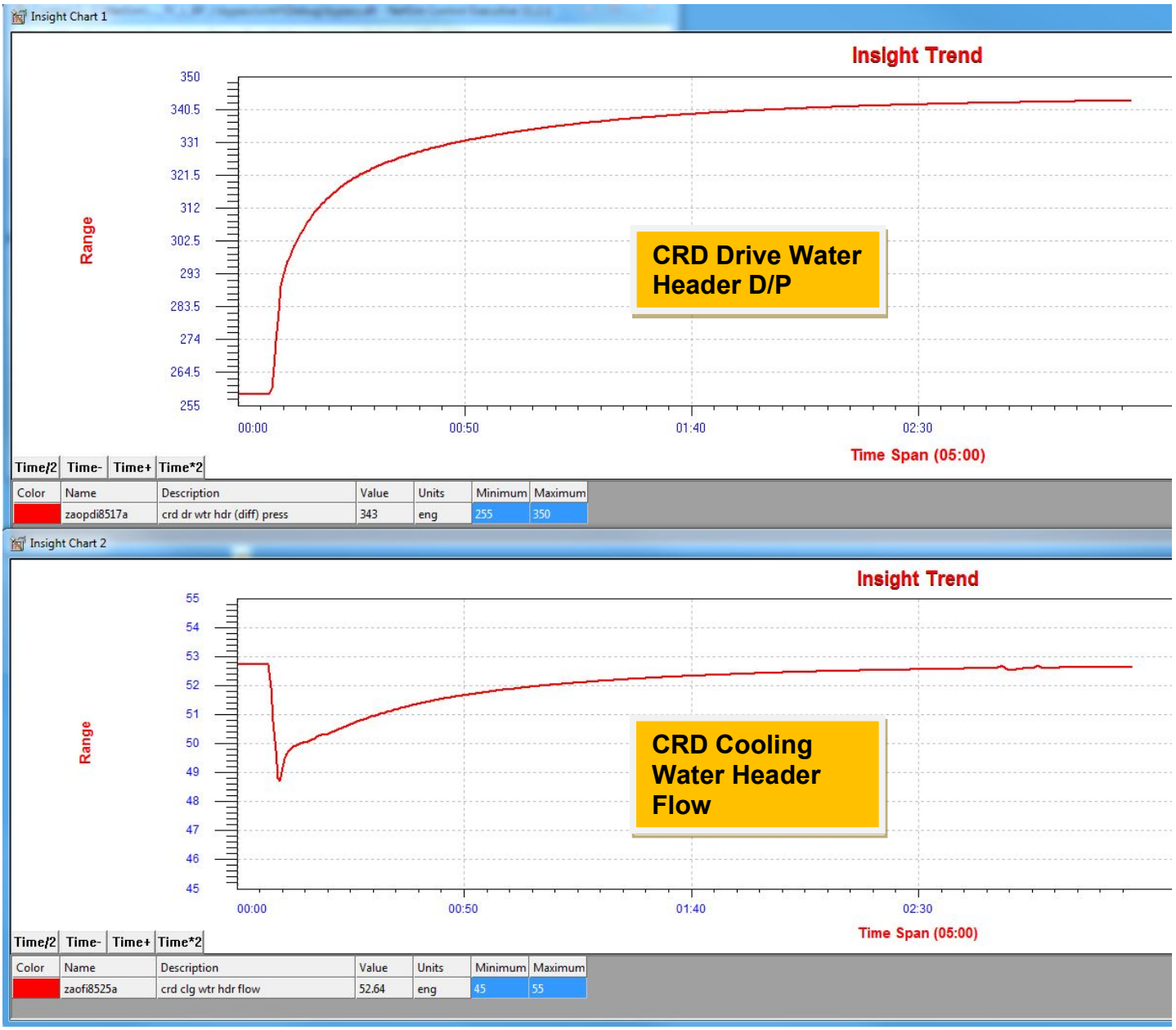
[13] **REPEAT** steps 8.15[3] through 8.15[12] for all desired control rods requiring low drive water DP flush.

[14] **RAISE** the CRD DRIVE WTR HDR DP, 2-PDI-85-17A to between 250 psid and 270 psid, using CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A.

Excerpts from OPL171.005 Lesson Plan:



Simulator trending:



Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
201006 (SF7 RWMS) Rod Worth Minimizer **A1.04** (10CFR 55.41.7)	Tier #	2	------
Ability to predict and/or monitor changes in parameters associated with operation of the Rod Worth Minimizer System, including: • Rod Withdrawal Blocks	Group #	2	------
	K/A #	201006A1.04	
	Importance Rating	3.6	------

Proposed Question: **# 54**

During a Reactor Startup, Rod Worth Minimizer Rod Blocks are **ENFORCED** until Reactor Power reaches the **(1)** to mitigate the consequences of **(2)**.

NOTE: LPAP – Low Power Alarm Point
LPSP – Low Power Set Point

A. (1) LPAP
(2) a Control Rod Drop Accident

B. (1) LPAP
(2) Control Rod Withdrawal error

C. (1) LPSP
(2) a Control Rod Drop Accident

D. (1) LPSP
(2) Control Rod Withdrawal error

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that accordance with 2-OI-85, Control Rod Drive System, the Low Power Alarm Point (LPAP) is above which the Rod Worth Minimizer System (RWM) no longer will provide alarms to the operator. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that while the RWM allows up to three (3) withdrawal blocks, in accordance with Technical Specification Bases, the purpose of the system is to mitigate the consequences of a Control Rod Drop accident.

C **CORRECT**: *(See attached)* In accordance with 2-OI-85, the Rod Worth Minimizer enforces Rod Blocks until Total Steam and Feed Flow are above 22%. For second part, in accordance with Technical Specification Bases 3.1.6, Rod Pattern Control, the purpose of the Rod Worth minimizer is to mitigate the consequences of a Control Rod Drop Accident.

D INCORRECT: First part is correct (*See C*). Second part is incorrect and plausible (*See B*).

RO Level Justification: The question tests the candidate's ability to predict the operation of the Rod Worth Minimizer System based on plant conditions. This question is rated as Memory due to the requirement to strictly recall facts in relation to Rod Worth Minimizer operation.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): 2-OI-85, Rev. 156 (Attach if not previously provided)
U2 Tech Spec Bases 3.1.6, Rev. 61
OPL171.024, Rev.16

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.024 Obj.11 As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.024-07 009 #939
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Copy of Bank Question:

939. OPL171.024-07 009

Which ONE of the following completes the statement below?

On **Unit 3**, for RWM to enforce, Total Feedwater Flow or Total Steam Flow must be less than ___(1)___.

During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of ___(2)___.

A. (1) 22%
(2) a control rod drop accident (CRDA)

B. (1) 22%
(2) control rod withdrawal error

C. (1) 27%
(2) a control rod drop accident (CRDA)

D. (1) 27%
(2) control rod withdrawal error

Excerpt from 2-OI-85:

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0156 Page 25 of 271

3.8 Rod Worth Minimizer (RWM)

A. The RWM System Rod Test/Touch screen function allows any one rod to be selected and moved to any position only if all other control rods are fully inserted. To get out of the rod test, the pushbutton needs to be depressed again (otherwise any single rod in any group can be selected and withdrawn).

B. [NER/C] When the RWM is bypassed, a second licensed operator, or other qualified member of the technical staff, is required to verify the Control Rod Sequence is followed. [INPO SOER-84-002]

C. 2-SR-3.3.2.1.7 is used to document independent verification of the RWM whenever the reactor is in startup or run, below 10% power.

D. [NER/C] Activities that can directly affect core reactivity are of a critical nature. Strict procedural compliance and conservative actions are required to be followed. [INPO SOER-84-002]

E. For RWM to enforce, Total Feedwater Flow or Total Steam Flow is required to be less than 22%. To take RWM out of service automatically, Low Power Set Point (LPSP), Total Steam Flow AND Total Feedwater Flow is required to be greater than 22 %.

The Low Power Alarm Point (LPAP) for the RWM is 27%, as sensed by Total Steam Flow. When the RWM is operating in the transition zone, between the LPSP (22%) and the LPAP (27%), no rod blocks are applied as a result of insert or withdraw errors, but the RWM will continue to provide alarm indications and error displays.

F. The monitoring functions of the RWM are automatically bypassed at power levels above the LPAP.

G. All the RWM blocks are applied in the event of a system hardware or software failure, when power is below the LPAP. At any Rx power, when a loss of ICS 2A occurs, a select block occurs due to the loss of power and cannot be bypassed using the RWM Bypass key.

H. An insert error occurs if:

1. A rod in the currently latched group is inserted past the insert limit for this group.

2. A rod in a group lower than the one that is presently latched is inserted past the withdraw limit for the lower group.

Excerpt from Tech Spec Bases 3.1.6:

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 10 and 11.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 10 and 11. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy

(continued)

Excerpt from OPL171.024:

control rod must be withdrawn a single notch at a time within the indicated range of high notch worth.

E. ** **Rod Worth Minimizer Purpose and Terms**

1. The RWM system design is based on Banked Position Withdrawal (BPWS) system design requirements.

SOER 84-2 Recommendation 7d

2. The RWM, in conjunction with the control rod velocity limiter, limits the amount of fuel damage that could occur during a control rod drop accident.

OBJ. ILT/LOR 3
OBJ. NLOR/NLO 1
Figure-3

T.S. Basis 3.1.6.1

   a) The RWM acts to enforce of the programmed control rod patterns and generates a rod block if significant deviation from the programmed sequence is detected.

Obj. ILT/LOR 7.a
Obj.NLOR/NLO 2.a

   b) The programmed sequence is designed to limit the maximum worth of any single control rod.

Obj. ILT/LOR 11
Why are control rod sequence restraints not required when reactor power is above the LPSP? No possible control rod configuration that results in rod worth that would exceed 280 cal/gm fuel damage limit during a CRDA.
The steam/feed flow values increase with EPU. DCN 67325

3. **Low Power Set Point**
   a) The low power set point (LPSP) is the power level below which the RWM enforces adherence to the programmed sequence of control rod withdrawals or insertions.
   b) When the reactor power level is above the LPSP, the RWM program does not impose any sequence constraints.
   c) The LPSP is determined by a steam flow/feedwater flow measurement and is set by adjustment of these sensors
   d) The LPSP varies depending on the unit and is found in the Precautions and Limitations of the unit's OI-85
   e) Either steam or feedwater flow can place the system in service, but both are needed to remove the system from service.

Obj. ILT/LOR11

4. **Low Power Alarm Point**
   a) The low power alarm point (LPAP) is the power level above which RWM does not enforce the rod sequence

Obj. ILT/LOR 7.b
Obj. NLOR/NLO 2.b

   b) The low power alarm point setting is based ONLY on reactor steam flow.
   c) the LPAP varies depending on the unit and is found in the Precautions and Limitations of the unit's OI-85. It is always set at some point higher than the LPSP setting.
   d) The AUTO light on the RWM operator's panel is lit whenever the reactor is operating above the LPAP setpoints indicating that the RWM functions are automatically bypassed. All RWM control rod blocks are automatically bypassed upon reaching power levels above the LPAP setpoint.

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
202002 (SF1 RSCTL) Recirculation Flow Control **A2.12**(10CFR 55.41.5) Ability to predict the impacts of the following on the Recirculation Flow Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: • APRM/LPRM	Tier #	2	------
	Group #	2	------
	K/A #	202002A2.12	
	Importance Rating	3.8	------

Proposed Question: **# 55**

Unit 2 is operating at 100% RTP when the following conditions occur:

- RECIRC DRIVE 2B TRIPPED, (2-9-4B, Window 7), is in alarm
- Core Flow indicates 50%
- **NO** Operator actions have been taken



Given the conditions above, the APRM Flow Biased SCRAM setpoint will be **(1)** Simulated Thermal Power.

In accordance with 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease, single loop operation in the MELLLA+ Operating Region **(2)** allowed.

A. (1) 97.9%
(2) is

B. (1) 97.9%
(2) is **NOT**

C. (1) 92.5%
(2) is

D. (1) 92.5%
(2) is **NOT**

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that in accordance with Technical Specifications, the Reactor can be operated indefinitely in single loop. However, in accordance with 2-OI-68, Reactor Recirculation System, single loop operation in the MELLA+ operating domain is prohibited.

B **CORRECT**: *(See attached)* Given that no operator actions have been taken, and in accordance with 2-OI-92B, Average Power Range Monitoring, the flow-biased Reactor SCRAM setpoint is set by the equation (0.61W + 67.4%), where W = Core Flow. Therefore, the SCRAM setpoint in this instance is (0.61(50) + 67.4) = 97.9%. For second part, in accordance with 2-OI-68-1, 2-ARP-9-4B, and 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease, single loop operation is prohibited in the MELLA+ operating domain.

C INCORRECT: First part is incorrect in that in accordance with 2-OI-92B, the Control Rod Block setpoint is set by the equation (0.61W + 62.0%), where W = Core Flow. Therefore, the Control Rod Block Second setpoint in this instance is (0.61(50 + 62.0) = 92.5%. Second part is incorrect but plausible *(See A).*

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's ability to predict Reactor SCRAM setpoint change based on single loop operations and the procedural limitations of single loop operation. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-68, Rev. 168 (Attach if not previously provided)
2-AOI-68-1, Rev. 39
2-OI-92B, Rev. 44
2-ARP-9-4B, Rev. 51
U2 Tech Spec Bases, Amend. 258
OPL171.148, Rev. 18U1

Proposed references to be provided to applicants during examination: **RECIRC DRIVE 2B TRIPPED, (2-9-4B, Window 7)**

Learning Objective: OPL171.007, Obj 24 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.074-02 074 #2427
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

2427. OPL171.074-02 074

Unit 3 was at 100% Reactor Power when Reactor **Recirc Pump 3A tripped**. Total Core Flow indication lowered to 50%.

Which ONE of the following completes the statements below?

Immediately following the trip, APRM Flow Biased Scram set point will be__**(1)**__ Simulated Thermal Power. (**Assume NO Operators actions**)

Single recirculation loop operation ___(2)___ in the MELLLA+ operating domain.

A. **(1)** 97.9%
(2) is allowed

B. **(1)** 86.5%
(2) is allowed

C. **(1)** 86.5%
(2) is prohibited

D. **(1)** 97.9%
(2) is prohibited

Excerpt from 2-ARP-9-4B:

BFN Unit 2	Panel 2-XA-55-4B	2-ARP-9-4B Rev. 0051 Page 12 of 50

RECIRC DRIVE 2B TRIPPED
2-XA-96-47
7

(Page 1 of 1)

Sensor/Trip Point:

External Digital Output 01g (ICS Group display GD @VFDBDF)
Fatal Fault (DRIVE TRIPPED)
Coast Stop Active
Coolant High Temp
Emergency Stop
Excessive drive losses
Input Protection Fault
Input One Cycle
Output Ground Fault
Latch Relay Status
For a complete list **REFER TO** ICS Group dis GD @VFDBDF

Sensor Location: Elevation 639' in Recirc Drive Room

Probable Cause:
A. Drive Fault
B. Ground fault
C. Loss of cooling
D. Emergency stop push button depressed
E. Sensor Malfunction

Automatic Action: Trip/Shutdown of 2B Recirc Drive.

Operator Action:
A. **CONFIRM** Recirc Drive 2B shutdown.
B. **REFER TO** 2-AOI-68-1, RECIRC PUMP TRIP/CORE FLOW DECREASE.
C. **REFER TO** ICS Group display GD @VFDBDF (Drive Tripped) to determine cause of alarm.
D. **IF** in the MELLLA+ operating domain, **THEN** (Otherwise N/A)
IMMEDIATELY take actions to INSERT control rods to exit the MELLLA+ operating domain. **REFER TO** 0-TI-464, Reactivity Control Plan, 2-OI-85 and Tech Spec 3.4.1.
E. **CORRECT** cause of trip.
F. **INITIATE** CR if equipment failures are indicated.

References: 2-45E620-5, 2-731E320RF-21, 2-47E610-68-2, GE 731E320RE, FSAR 13.6.2

Excerpts from 2-OI-92B:

BFN Unit 2	Average Power Range Monitoring	2-OI-92B Rev. 0044 Page 21 of 30

Attachment 1
(Page 1 of 6)

APRM Trip Outputs

APRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
APRM Downscale	≤5%	1. Rod Block if REACTOR MODE SWITCH in RUN.
APRM Inop	1. APRM Chassis Mode not in OPERATE (keylock to INOP). 2. Loss of Input Power to APRM. 3. Self Test detected Critical Fault in the APRM instrument. 4. Firmware Watchdog timer has timed out	1. One Channel detected, no alarm or RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).
APRM Inop Condition	1. < 20 LPRMs in OPERATE, or < 3 per level.	1. <20 LPRMs total or <3 per level results in a Rod Block and a trouble alarm on the display panel. This does not yield an automatic APRM trip, but does, however, make the associated APRM INOP.
APRM High	1. DLO ≥ (0.61W + 62.0%) SLO ≥ (0.55(W-dw) + 58.5%) [W = Total Recirc Drive Flow in % rated]. 2. Neutron Flux Clamp Rod Block ≥ 113% 3. ≥ 8% APRM Flux.	1. Rod Block if REACTOR MODE SWITCH in RUN. 2. Rod Block if REACTOR MODE SWITCH in RUN. 3. Rod Block in all REACTOR MODE SWITCH positions except RUN.

BFN Unit 2	Average Power Range Monitoring	2-OI-92B Rev. 0044 Page 22 of 30

Attachment 1
(Page 2 of 6)

APRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
APRM High High	1. DLO ≥ (0.61W + 67.4%) SLO ≥(0.55(W-dw) + 64.5%) [W = Total Recirc Drive Flow in % rated]. 2. ≥ 119% APRM Flux. 3. ≥ 12% APRM Flux.	1. Scram. 2. Scram. 3. Scram in all REACTOR MODE SWITCH positions except RUN.
Recirc Flow Compare Recirc Flow Upscale	1. ≥ 5% mismatch between APRM Channels. 2. 107% Flow monitor upscale.	1. Flow compare inverse video alarm. 2. Rod Block.
OPRM Inop	Less than 8 responsive OPRM cells and the Rx Mode Switch is in RUN	Annunciation Only
OPRM ALARM	Any one of two algorithms, period or CDA exceeds its pre-trip alarm setpoint for an operable OPRM cell.	Rod Block
OPRM Trip	Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.	1. One Channel detected, no RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).
All OPRM setpoints are bypassed when the Reactor Mode Switch is not in RUN or the Reactor is not operating in the Power/Flow region where instabilities can occur (≥ 23% Power & <75% Recirc Drive Flow).		

Excerpt from 2-AOI-68-1:

4.2 Subsequent Actions (continued)

NOTES

1) Step 4.2[2] through Step 4.2[10.3] apply to any core flow lowering event.
2) Power To Flow Map is maintained in 0-TI-248"Station Reactor Engineer" and on ICS.
3) If a cell bypasses while a recirc pump is running, a drop of ≈200 rpm will occur. The drive will remain at the speed at which it stopped. Operator action is required to raise speed following a cell bypass.
4) Single recirculation loop operation (SLO) is prohibited in the MELLA+ operating domain. **REFERENCE** Tech Spec 3.4.1.

[2] **IF** a single Recirc Pump tripped, **THEN**

CLOSE tripped Recirc Pump 2A(2B) discharge valve 2-FCV-068-0003(0079).

[2.1] **IF** Recirc Pump was tripped due to dual seal failure, **THEN** (Otherwise **MARK** N/A)

[2.1.1] **ENSURE** TRIPPED, RECIRC DRIVE 2A(2B) NORMAL FEEDER, 2-HS-57-17(14).

[2.1.2] **ENSURE** TRIPPED, RECIRC DRIVE 2A(2B) ALTERNATE FEEDER, 2-HS-57-15(12).

[2.1.3] **CLOSE** tripped recirc pump suction valve using, RECIRC PUMP 2A(2B) SUCTION VALVE, 2-HS-68-1(77).

CAUTION

The temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other. If this limit cannot be maintained, a plant cool down should be initiated. Failure to maintain this limit and not cool down could result in hangers and/or shock suppressers exceeding their maximum travel range. [GE SIL 251, 430 and 517]

[2.1.4] **IF** it is evident that 75°F between the dome AND the idle Recirc loop CANNOT be maintained, **THEN**

COMMENCE plant shut down and cool down.
REFERENCE 2-GOI-100-12A.

Excerpt from 2-OI-68:

BFN Unit 2	Reactor Recirculation System	2-OI-68 Rev. 0168 Page 13 of 211

3.3 Tech Specs

A. With both recirculation pumps out of service, refer to Tech Spec 3.4.1.

B. [NRC/C] Core Thermal-Hydraulic Stability.

1. The reactor SHALL be verified outside Regions I, II & III.

2. When OPRMS are INOP, Refer to 2-SR-3.3.1.1.I. [NCO 940245010]

C. [NRC/C] Single Loop Operation: Per Technical Specifications, the reactor can be operated indefinitely with one Recirc loop out of service, provided the requirements of T.S. 3.4.1 are implemented within 24 hours of entering single loop operations.

D. Single recirculation loop operation (SLO) is prohibited in the MELLLA+ operating domain. **REFER TO** Tech Spec 3.4.1.

E. TS BASES SR 3.5.1.5, If Recirc Pump 2A(2B) Discharge Valve, 2-FCV-068-0003(0079), is declared inoperable while the valve is open, the associated LPCI Subsystem must be declared INOPERABLE.

F. TS SR 3.4.2.1, 2-SR-3.4.2.1, Jet Pump Mismatch and Operability, is required to be performed 24 hours after reaching greater than 23% RTP and/or within 4 hours after returning an idle recirc pump to service.

Excerpt from Tech Spec Bases 3.4.1:

BASES (continued)

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR Limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, 'MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power-High Setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the assumptions of References 7 and 8.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

(continued)

Excerpts from OPL171.148 Lesson Plan:

e. The average of the LPRM input signals yields an instantaneous average neutron flux value.

ILT Objective 13b
ILT Objective 13c
NLO Objective 7b
NLO Objective 7c

1) Reactor Scram High Flux setpoints vary depending upon Reactor Mode Switch position.

a) Set at 119% with Reactor Mode Switch in RUN.

Control rod block - 113%

b) Set at 12% with Reactor Mode Switch in Startup/Hot Standby.

Control rod block - 8%

f. This average flux value is filtered to produce a Simulated Thermal Power (STP) signal.

1) STP takes the neutron flux input and filters the signal through a 6 second filter.

a) This approximates the thermal power being transferred into the reactor coolant.

ILT Objective 12

2) The STP signal is used by the APRM for flow biased rod blocks and scram set points.

neutron flux scram signal

3) Flow Biased Scram and Rod Block generation.

a) The APRM calculates a flow-biased setpoint by comparing reactor power and reactor recirculation flow.

b) The flow biased scram signal is not a fixed value, but is dynamically calculated based on reactor recirculation flow.

A dynamic flow based scram ensures a quicker scram during a power excursion compared to a high fixed scram setpoint.

c) With only one recirculation pump in service, a 10% bias is added to the flow biased calculation (added conservatism).

d) Flow biased setpoint for reactor scram is:
0.61w + 67.4% (DLO)

0.55w + 64.5 - 0.55Δw (SLO)

DLO = Dual Loop Operation
SLO = Single Loop Operation

w = recirc flow as calculated by the APRM instrument.

e) Flow biased setpoint for Control Rod Block is:
0.61w + 62.0% (DLO)

0.55w + 58.5% - 0.55Δw (SLO)

Δw = 10% for SLO. This conservative bias accounts for loop back flow. Δw is automatically entered when APRMs are placed in SLO mode.

f) Examples:

(1) Given that Reactor Recirculation Flow indicates 40% flow, calculate the setpoint for the alarm and rod block. (Assume both reactor recirculation pumps are running.)

(a) 0.61(40) + 67.4 = scram setpoint = 91.8% STP

The change in scram and control rod block setpoints are manually inputted into the APRMs by I&C Maintenance by performance of 1(2)(3)-SR-3.4.1(SLO) and 1(2)(3)-SR-3.4.1(DLO)

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18U1

(b) 0.61(40) + 62.0 = rod block setpoint = 86.4% STP

(2) How is this different if in single loop operation?

(a) 0.55(40) + 64.5 - 0.55(10)= 81% STP

(b) 0.55(40) + 58.5 - 0.55(10)= 75% STP

Delta-W shows on the APRM drawer as 0 vice 10% due to the MELLA+ upgrade. The reason for this change is because in single loop operations, you cannot be in MELLA+. Therefore, the slope is different for single loop operation than for two-loop operations, and the entire algorithm is different. A delta-W of 10% is still used when manually calculating flow-biased setpoint in single-loop operations

g) The STP is also used by the RBM for reference APRM power level.

4) An APRM downscale condition (at ≤ 5%) also produces a rod block when the reactor mode switch is in RUN.

5) Recirculation Flow Monitor

a) Each flow monitor channel consists of two flow inputs used to calculate a Total Recirculation Flow, one from Recirculation Loop A and one from Recirculation Loop B.

b) Each APRM receives the inputs from two (4 to 20 mA) differential pressure (ΔP) transmitters used to measure the recirculation loop flows.

c) Each RBM receives the four Total Recirculation Flow values from the APRM channels to determine the status of the flow compare alarm.

d) The Recirculation Flow Monitor Function provides the following alarm functions for each Total Recirculation Flow level:

(1) Flow Upscale Alarm (generated by the APRM)

(2) Flow Compare Alarm (generated by the RBM, display alarm only)

ILT Objective 13a
NLO Objective 7a

ILT Objective 10
NLO Objective 6

e) The table below shows the relationship between the flow monitors and each APRM channel.

Displays on Input Status display

	Loop A	Loop B
APRM 1	FT-68-5A	FT-68-81A
APRM 2	FT-68-5B	FT-68-81B
APRM 3	FT-68-5C	FT-68-81C
APRM 4	FT-68-5D	FT-68-81D

Inputs to flow upscale alarm APRM flow alarm is when flow is

Form 4.3-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

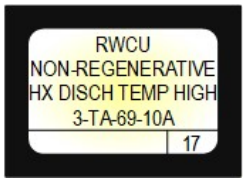
204000 (SF2 RWCU) Reactor Water Cleanup
A4.04 (10CFR 55.41.7)
Ability to manually operate and/or monitor in the control room:

- Heat exchanger temperature

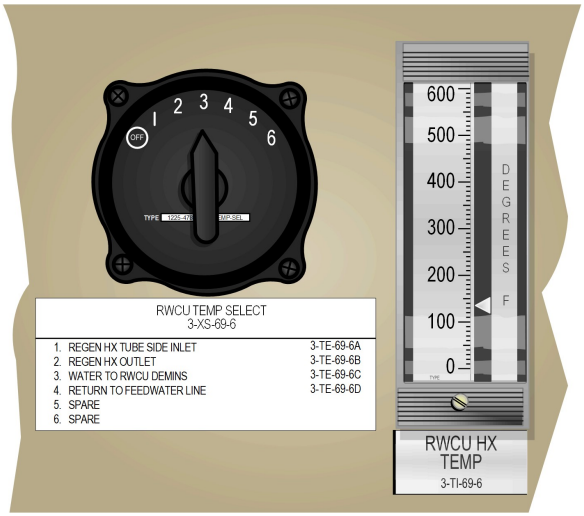
Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	204000A4.04	
Importance Rating	3.1	------

Proposed Question: **# 56**

Unit 3 is operating at 100% RTP, when the following conditions exist:



- RWCU NON-REGENERATIVE HEAT EXCHANGER DISCHARGE TEMPERATURE HIGH (3-9-4B, Window 17)
- RWCU NON-REGENERATIVE HEAT EXCHANGER DISCHARGE TEMPERATURE is 135 °F



Given the conditions above, the Reactor Water Cleanup (RWCU) System **(1)** receive an isolation signal.

RWCU Pumps receive a direct trip signal when **(2)** is **NOT** full OPEN.

Note: RWCU INBOARD SUCTION ISOLATION VALVE, 3-FCV-69-1

RWCU RETURN ISOLATION VALVE, 3-FCV-69-12

A. (1) will
(2) 3-FCV-69-1

B. (1) will
(2) 3-FCV-69-12

C. (1) will **NOT**
(2) 3-FCV-69-1

D. (1) will **NOT**
(2) 3-FCV-69-12

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the given alarm is to alert operators to a High Temperature in the Reactor Water Cleanup (RWCU) System. In accordance with 3-9-4B, Window 17, the alarm setpoint is 130 ºF. In accordance with 3-OI-69, Reactor Water Cleanup System, the isolation setpoint for NRHX Discharge Temperature for Primary Containment Isolation System (PCIS) Group 3 – RWCU is 140 ºF. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that the RWCU Pumps will trip only when 3-FCV-69-12, RWCU RETURN ISOLATION VALVE, is fully closed. If 3-FCV-69-1, RWCU INBOARD SUCTION ISOLATION, or 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION are not fully open the RWCU Pumps will trip.

C **CORRECT:** (*See attached)* In accordance with 3-OI-69, the given NRHX Discharge Temperature has not reached the isolation setpoint of 140 ºF. Second part is correct in that in accordance with 3-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation, RWCU Pumps will trip when 3-FCV-69-1, RWCU INBOARD SUCTION ISOLATION, or 3-FCV-69-2, RWCU OUTBOARD SUCTION ISOLATION are not fully open.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's ability to monitor Reactor Water Cleanup Heat Exchanger Temperature and determine system response to the given temperature. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-ARP-4B, Rev.54 (Attach if not previously provided)
3-AOI-64-2A, Rev.12
3-OI-69, Rev.115

Proposed references to be provided to applicants during examination: **RWCU NON-REGENERATIVE HEAT EXCHANGER DISCHARGE TEMPERATURE HIGH (3-9-4B, Window 17)**
3-XS-69-6, RWCU TEMP SELECT SWITCH
3-TI-69-6, RWCU HX TEMP INDICATOR

Learning Objective: OPL171.013, Obj. 4a (As available)
OPL171.013, Obj. 5g

Form 4.3-1 **Written Examination Question Worksheet**

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.013-03 004 #657	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:
- 55.41 **X**
- 55.43

Copy of Bank Question:

657. OPL171.013-03 004

Unit 3 is operating at 100% Reactor Power when the following alarms AND indications are received:

- A Partial Loss of Reactor Building Closed Loop Cooling Water occurs
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH, (3-9-4B, Window 17) is in alarm
- RWCU Non- Regenerative Heat Exchanger Discharge Temperature is 140º F

Which ONE of the following correctly completes the statement below?

3-FCV-69-1 - RWCU INBD SUCT ISOLATION VALVE
3-FCV-69-2 - RWCU OUTBD SUCT ISOLATION VALVE
3-FCV-69-12 - RWCU RETURN ISOLATION VALVE

The Reactor Water Cleanup (RWCU) Pumps will receive a **DIRECT** trip input from _____.

A. the non-Regenerative Heat Exchanger high temperature signal

B. either the 3-FCV-69-1, 3-FCV-69-2 or 3-FCV-69-12 isolation valve position

C. the non-Regenerative Heat Exchanger high temperature signal **ONLY**

D. a low flow condition after 30 seconds **ONLY**

Excerpt from 3-ARP-9-4B:

BFN Unit 3	Panel 9-4 3-XA-55-4B	3-ARP-9-4B Rev. 0054 Page 23 of 48

RWCU
NON-REGENERATIVE
HX DISCH TEMP HIGH
3-TA-69-10A

17

(Page 1 of 1)

Sensor/Trip Point:
3-TS-069-0010A 130°F

Sensor Location:
Elevation 593'
Rx Building
Column T-16 M-LINE
NRHX outlet

Probable Cause:
A. High RWCU flow.
B. RBCCW TCV malfunction.
C. RWCU Heat Exchangers RBCCW Flow Controller malfunction.
D. Failed sensor.

Automatic Action:
None

Operator Action:
A. **CHECK** RWCU NRHX discharge temperature, 3-XS-69-6, on Panel 3-9-4.

B. **CHECK** RBCCW system temperature indication normal, Panel 3-9-4.

C. **IF** temperature continues to rise, **THEN**
PERFORM the following, otherwise, **MARK** steps N/A:
- **REDUCE** system flow or reject flow as necessary to control temperature..
- **REFER TO** 3-OI-69.

D. **DISPATCH** personnel to CHECK the following:
- RWCU Heat Exchangers RBCCW Flow Controller (normally in auto with setpoint at approx. 110°F), located on 3-LPNL-925-0002 Rx Bldg 593'.
- 3-TCV-70-49 operating properly (RBCCW to NRHX), located in RWCU HX room.

E. **IF** due to loss of RBCCW, **THEN**
REFER TO 3-AOI-70-1.

References:
3-45N620-5 3-47E810-1 3-47E822-1
3-47E610-69-1 FSAR Section 13.6.2

Excerpt from 3-AOI-64-2A :

BFN Unit 3	Group 3 Reactor Water Cleanup Isolation	3-AOI-64-2a Rev. 0012 Page 4 of 8

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 3 Reactor Water Cleanup Isolation.

2.0 SYMPTOMS

NOTE

Reactor Water Cleanup System Isolation is initiated by any one of the following signals:

- Reactor Vessel Water Level Low (PCIS Group 3 isolation)
- SLC Injection Initiation
- RWCU Isolation Logic for Area Temperatures (PCIS Group 3 isolation)
- RWCU Non-Regenerative HX Discharge Temperature High

2.1 Any One or More of the Following Annunciator in Alarm:

- RWCU ISOL LOGIC CHANNEL A(B) TEMP HIGH (3-XA-55-5B, Window 32 and/or 33)
- RX VESSEL WTR LEVEL LOW HALF SCRAM (3-XA-55-4A, Window 2)
- RWCU LEAK DETECTION TEMP HIGH (3-XA-55-3D, Window 17)
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (3-XA-55-4B, Window 17)
- SLC INJECTION FLOW TO REACTOR (3-XA-55-5B, Window 14)
- SCRAM, PRIMARY CNTMT PRESS HIGH (3-XA-55-1, Window 24)

3.0 AUTOMATIC ACTIONS

- RWCU INBD SUCTION ISOLATION VALVE, 3-FCV-69-1 **CLOSES**
- RWCU OUTBD SUCTION ISOLATION VALVE, 3-FCV-69-2 **CLOSES**
- RWCU RETURN ISOLATION VALVE, 3-FCV-69-12 **CLOSES**
- Reactor Water Cleanup Recirc Pumps 3A and 3B **TRIP**.

Excerpt from 3-OI-69:

BFN Unit 3	Reactor Water Cleanup System	3-OI-69 Rev. 0115 Page 15 of 182

3.7 RWCU Pump Trip Signals

A. Low flow 56 gpm (30 second time delay if control switch in NORMAL after start).

B. 3A Inboard Bearing high temperature 180°F (30 sec. time delay).
3B Inboard Bearing high temperature 180°F (30 sec. time delay).

C. RWCU INBD SUCT ISOLATION VALVE, 3-FCV-69-1 not full open.

D. RWCU OUTBD SUCT ISOLATION VALVE, 3-FCV-69-2 not full open.

E. RWCU RETURN ISOLATION VALVE, 3-FCV-69-12 fully closed.

F. 480V Shutdown Board Undervoltage (5 second TD) or Overcurrent.

3.8 RWCU Isolation Signals

A. Reactor water level low (LEVEL 3).

B. Non-regenerative heat exchanger outlet high temperature 140°F.

C. RWCU Pump Room 3A high temperature 148°F.

D. RWCU Pump Room 3B high temperature 148°F.

E. Main Steam Tunnel/RWCU Piping high temperature 197°F.

F. RWCU System Pipe Trench 131°F.

G. RWCU Heat Exchanger Room Pipe Chase Area high temperature 166°F.

H. RWCU Heat Exchanger Room high temperature 139°F.

I. Standby Liquid Control system initiation.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295006 (APE 6) SCRAM / 1	Tier #	1	------
AK1.01 (10CFR55.41.5)	Group #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to SCRAM:	K/A #	295006AK1.01	
• Decay heat generation and removal	Importance Rating	4.0	------

Proposed Question: **# 57**

Unit 1 is operating at 100% Rated Thermal Power (RTP) when the following events occur:

- An automatic Reactor SCRAM
- Reactor Water Level reaches a **MINIMUM** of (-) 110 inches, and stabilizes at (+) 32 inches
- Reactor Pressure reaches a **MINIMUM** of 840 psig and is rising
- **NO** Operator Actions have been taken

Given the conditions above, which **ONE** of the following lists the decay heat removal method(s) which is/are **IMMEDIATELY** available?

A. MSRVs **ONLY**.

B. MSRVs, Reactor Water Cleanup, **AND** Main Steam Line Drain Valves **ONLY**.

C. MSRVs, Main Steam Line Drain Valves, **AND** Main Turbine Bypass Valves **ONLY**.

D. MSRVs, Reactor Water Cleanup, Main Steam Line Drain Valves, **AND** Main Turbine Bypass Valves.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** In accordance with 1-OI-1, Main Steam System, whenever Reactor Pressure is reaches 852 psig and the Reactor MODE Switch is in RUN position, the MSIVs will close due to a PCIS Group 1 Isolation signal. As far as the other Main Steam System components, once the MSIVs close, the downstream Main Steam Line Drain Valves and Main Turbine Bypass Valves are no longer available. The MSRVs are located upstream of the MSIVs by design for safety purposes. Therefore, the MSRVs are still immediately available for decay heat removal.

B INCORRECT: Incorrect but plausible in that once Reactor Water Level lowers to (+) 2 inches, RWCU is isolated due to a PCIS Group 3 Isolation signal. Main Steam Line Drain Valves close due to a PCI Group 1 Isolation signal with Reactor Pressure reaches 852 psig *(See A).*

C INCORRECT: Incorrect but plausible the MSRVs are the only available decay heat removal method available *(See A).*

D INCORRECT: Incorrect but plausible the MSRVs are the only available decay heat removal method available *(See A and B).*

RO Level Justification: Tests the candidate's knowledge of the operational implications from a Reactor SCRAM occurring and the lowering of Reactor Water Level to determine available decay heat removal methods. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine available decay heat removal methods.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 1-OI-1, Rev. 17 (Attach if not previously provided)
1-OI-69, Rev. 87
OPL171.009, Rev. 17U1
OPL171.017, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.009 Obj. 11 (As available)

Question Source:

Bank #	BFN 1006 AUDIT#5
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

ILT 10-06 Audit Written Exam

5. A Unit 1 Reactor startup is in progress with Reactor Power at 18%. A failure causes **ALL** Turbine Control Valves to fail OPEN, resulting in an automatic Scram.

Plant parameters are as follows:

- Reactor Water Level reached a **MINIMUM** of (-) 110 inches, and stabilized at (+) 32 inches
- Reactor Pressure reached a **MINIMUM** of 840 psig and is rising

With **NO** operator recovery action, which ONE of the following lists the decay heat removal method(s) which is/are **IMMEDIATELY** available?

A. MSRVs **ONLY**.

B. MSRVs, Reactor Water Cleanup, **AND** Main Steam Line Drain Valves **ONLY**.

C. MSRVs, Main Steam Line Drain Valves, **AND** Main Turbine Bypass Valves **ONLY**.

D. MSRVs, Reactor Water Cleanup, Main Steam Line Drain Valves, **AND** Main Turbine Bypass Valves.

Excerpts from 1-OI-1:

BFN Unit 1	Main Steam System	1-OI-1 Rev. 0017 Page 9 of 67

3.2 Main Steam Isolation Valves (MSIV)

3.2.1 MSIV Closure

A. The MSIVs should be fast closed when the reactor is shutdown and no steam flow, unless required to be slow closed by surveillance, test instruction, or an abnormal condition. [BFNPER 164499]

B. When an MSIV is closed at power, the potential exists for an isolation of the Hydrogen Water Chemistry System to occur. This is due to the possibility of a hydrogen bubble becoming entrained in the main steam line drains and subsequently being released when the main steam line drains reposition in response to a MSIV closure. This scenario can result in a small Off Gas System hydrogen spike of sufficient strength to cause an automatic isolation of the Hydrogen Water Chemistry System.

C. Closure of all MSIVs could cause turbine shaft damage if main condenser vacuum is maintained and seal steam supply is not established from the auxiliary boiler.

3.2.2 MSIV Isolation

A. Main steam tunnel temperature should not be allowed to exceed 189°F to prevent MSIV isolation.

B. Whenever reactor pressure is reduced to 852 psig and the reactor mode switch is in RUN position, the MSIVs will close.

C. The MSIVs will close if 250 Vdc and 120 Vac power to the MSIV control logic is de-energized.

D. Reactor power should be less than or equal to 75% prior to closing an MSIV > 15% during closure testing. This should prevent a steam line high flow MSIV closure.

E. All MSIV Handswitches need to be placed in the Close Position to allow the PCIS group one trip logic to be reset, and prevent rapid re-opening upon reset of any one not in close. Placing all INBOARD MSIV Handswitches in the Close Position allows the PCIS group one A1/B1 Division I trip logic to be reset and placing all OUTBOARD MSIV Handswitches in the Close Position allows the PCIS group one A2/B2 Division II trip logic to be reset. (PER 658229-011)

3.2.2 MSIV Isolation (continued)

F. The PCIS group one trip parameters do not exceed trip setpoints.

1. Reactor water level above -122 in.
2. MSL flow less than 135%.
3. MSL tunnel temperature less than 189°F.
4. MSL pressure greater than 852 psig if in Mode 1.

3.2.3 RPS and Limit Switches

A. Whenever more than two main steam lines are isolated (>10% closed) and the reactor mode switch is in RUN position, a reactor scram will occur.

B. The LS-3 and LS-4 limit switches for the Unit 1 MSIVs actuate when MSIVs are 90% open to initiate a reactor scram via the RPS. During MSIV testing a half-scram initiation signal will be received before the green position indicator light illuminates.

C. LS-5 limit switches cause the MSIV green position (closed) indicator light to illuminate when MSIVs are 85% or less open.

D. Attachment 1 references the RPS 1A fuses relays and Logic.

E. Attachment 2 references the RPS 1B fuses relays and Logic.

Excerpt from OPL171.009 Lesson Plan:

(7) Signals which cause automatic closure of MSIVs are:

- (a) Low-low-low reactor water level (Level 1) -122"
- (b) Steam tunnel high temperature, 189°F
- (c) High steam flow in any main steam line ≥ 135% .45 sec TD.
- (d) Main Steam Line low pressure (852 psig) with Mode SW in "RUN"
- (e) An automatic isolation causes:
  - All MSIVs to close
  - Drain valves 1-55 and 1-56 to shut.

Obj. ILT 10
Obj. NLO 6, OF-1

Obj. ILT 11
Obj. NLO 7
Briefly discuss other systems affected

(8) Reasons for isolation signals:

- (a) -122" Rx Water Level (Level 1)
  - i. Low enough to prevent spurious initiation
  - ii. High enough to initiate isolation and ECCS so that:
  - iii. No melting of the fuel cladding occurs.
  - iv. Post accident cooling may be accomplished.
  - v. The guidelines of *10CFR50.67* are not exceeded.
- (b) 189°F Steam Tunnel High Temp
  - i. Detect small (15 gpm) steam leaks in the steam tunnel.
  - ii. Provide backup to high steam flow isolation on large breaks outside containment.
  - iii. Prevent exceeding *10CFR50.67* guidelines
- (c) 135% Steam Flow In conjunction with the flow restrictors and main steam isolation valve closures, limits the mass inventory loss such that:
  - i. Fuel is not uncovered.
  - ii. *10CFR50.67* guidelines are not exceeded.

-(OF-5 Solid understanding of plant design)
-Discuss isolation signals and bases for why each isolation is required
-Isolation on Low RPV level may be bypassed with EOI App8A. DCN 72702 installs 4 keylocks on 9-4 Switches in the MCR to Replace the jumpers

Excerpt from 1-OI-69:

BFN Unit 1	Reactor Water Cleanup System	1-OI-69 Rev. 0087 Page 16 of 155

3.7 RWCU Pump Trip Signals

A. The following signals will cause an automatic trip of an RWCU Pump:

1. Low flow 56 gpm (30 second time delay if the control switch is in NORMAL after start)
2. 1B Inboard Bearing high temperature 180°F (30 sec. time delay).
   1A pump Cooling water high temperature 140°F (30 sec. time delay).
3. RWCU INBD SUCT ISOLATION VLV, 1-FCV-069-0001 not full open.
4. RWCU OUTBD SUCT ISOL VLV, 1-FCV-069-0002 not full open.
5. RWCU SYS RETURN ISOL VLV, 1-FCV-069-0012 fully closed.
6. 480V Shutdown Board Undervoltage (5 second TD) or Overcurrent.

3.8 RWCU Isolation Signals

B. The following signals will cause an automatic isolation of the RWCU system:

1. Reactor water level low (Level 3).
2. Non-regenerative heat exchanger outlet high temperature 140°F.
3. RWCU Pump Room 1A area high temperature 148°F.
4. RWCU Pump Room 1B area high temperature 148°F.
5. Main Steam Tunnel/RWCU piping high temperature 197°F.
6. RWCU pipe trench area high temperature 131°F.
7. RWCU Heat Exchanger Room pipe chase area high temperature 166°F.
8. RWCU Heat Exchanger Room high temperature 139°F.
9. Standby Liquid Control system initiation.

Excerpts from OPL171.017 Lesson Plan:

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 22

Table 1
Primary Containment Isolation Valves
Numbers in parentheses are keyed to notes list at lend of Table 1.)

Isolation Signals	Group 1	Valve type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)	Re-Open Manipulation
Level 1; Rx low-low-low level -122"(5) OR MSL Hi Flow 135% .45 sec. TD OR MSL Space Hi Temp 189°F (O.I.) 200°F (T.S.) OR MSL Low Pressure 852 psig in run mode (825 psig TS)	Main steam line Isolation valves (FCV 1-14, -26, -37, 51)	AO Globe	Inside	Air & AC DC	Air & spring (1)	(8)
	Main steam line Isolation valves (FCV 1-15, 27, 38, and 52)	AO Globe	Outside	Air & AC DC	Air & spring (1)	(8)
	*Reactor water sample line Isolation valves (FCV 43-13) (2)	AO Valve	Inside	Air & AC "A" I&C	Spring	(9)
	* Reactor water sample line Isolation valve (FCV 43-14) (2)	AO Valve	Outside	Air & AC "B" I&C	Spring	(9)
	Main steam line drain valves (FCV 1-55)	MO Gate	Inside	AC	AC	(9)
	Main steam line drain valves (FCV 1-56)	MO Gate	Outside	DC	DC	(9)
	Mechanical Vacuum Pump trip & suction valve closure 38, and 52)					
*The only signal which isolates the Reactor Water Sample Line Isolation valves (FCV-43-13 and FCV-43-14) is Rx low-low-low level (Level 1, -122")						

ILT- 2a,l
LOR- 2a,l

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 22

Isolation Signals	Group 3	Valve type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)	Re-Open Manipulation
Level 3; Rx low level: +2" OR Rx water cleanup space Hi temp (U1/U3- 131- 197°F, U2- 131- 185°F) OR SLC Pump Hand Switch **NOT** in STOP OR RWCU Non-Regenerative HX Discharge Temperature High. (Not PCIS)	Reactor water cleanup supply and return valves (FCV 69-1)	MO Gate	Inside	AC	AC	(9)
	Reactor water cleanup supply and return valves (FCV 69-2)	MO Gate	Outside	DC	DC	(9)
	Reactor water cleanup supply and return valves (FCV 69-12)	MO Gate	Outside	AC	AC	(9)

ILT- 2b,l
LOR- 2b,l

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215002 (SF7 RBMS) Rod Block Monitor	Tier #	2	------
A2.03 (10CFR 55.41.5)	Group #	2	------
Ability to (a) predict the impacts of the following on the Rod Block Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:	K/A #	215002A2.03	
• Loss of associated reference APRM channel	Importance Rating	3.3	------

Proposed Question: **# 58**

A Reactor Startup is in progress on Unit 3, when the following condition occurs:

- The reference signal from APRM 2 to the Rod Block Monitor (RBM) fails downscale

APRM 2 is the **PRIMARY** reference channel for RBM Channel **(1)**.

Given the conditions above and in accordance with 3-OI-92C, Rod Block Monitor, the RBM **(2)** automatically transfer to the **ALTERNATE** APRM.

A. (1) A
(2) will

B. (1) A
(2) will NOT

C. (1) B
(2) will

D. (1) B
(2) will NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that there are four APRMs and two Rod Block Monitors (RBM). Any APRM could be selected as the reference channel for any RBM. Additionally, both RBMs receive flow inputs from all four APRM Channels and inputs from various LPRMs to add to the difficulty. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that the APRMs and RBM can be bypassed and manipulated manually at Panel 3-9-5 and Panel 3-9-14.

C **CORRECT**: (*See attached*) First part is correct in that in accordance with 3-OI-92C, Rod Block Monitor, APRM 2 is the primary reference channel for RBM B. Second part is correct in that in accordance with 3-OI-92C, if the primary reference channel APRM becomes unavailable for a RBM, the alternate is automatically selected.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B).*

RO Level Justification: Tests the candidate's ability to predict the impact of a loss of a reference APRM Channel on the RBM System. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to primary and alternate APRM Reference Channels for the RBM System.

Technical Reference(s): 3-OI-92C, Rev.20 (Attach if not previously provided)
OPL171.148, Rev. 18U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.148 Obj. 27 (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

New	**X**
Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpts from 2-OI-92C:

BFN Unit 3	Rod Block Monitor	3-OI-92C Rev. 0020 Page 7 of 18

3.0 PRECAUTIONS AND LIMITATIONS (continued)

3.0 PRECAUTIONS AND LIMITATIONS

A. If both RBM channels are inoperable, at least one Rod Block Monitor channel shall be placed in the tripped condition within one hour. (Ref. TS Section 3.3.2.1 and Table 3.3.2.1-1 and TRM Section 3.3.4.)

B. The minimum number of LPRM inputs required for each RBM channel to avoid an inoperative trip is greater than or equal to 50% of the assigned LPRM inputs. This is adjusted automatically by the rod selection matrix when a new rod is selected.

C. Both RBM channels are bypassed when APRM reference power is less than or equal to 25% and the RBM channels are **NOT** inoperable.

D. Both RBM channels are bypassed when a peripheral (edge) control rod is selected.

E. If a rod withdrawal block is initiated by the RBM, it applies only to the control rod selected and should **NOT** inhibit movement of other rods.

F. Illustration 1 lists the trip signals and associated actions for the Rod Block Monitoring system.

G. Each RBM channel can select from a designated hierarchy of normal and alternate APRM channels to use as their reference. If an Alternate was selected by the RBM and the Primary becomes available, it will automatically be selected as the reference APRM channel.

H. RBM reference signals from APRM channels:

RBM Channel	Reference APRM Channel
A	CHANNEL 1 (PRIMARY)
A	CHANNEL 3 (FIRST ALTERNATE)
A	CHANNEL 4 (SECOND ALTERNATE)
B	CHANNEL 2 (PRIMARY)
B	CHANNEL 4 (FIRST ALTERNATE)
B	CHANNEL 3 (SECOND ALTERNATE)

I. [QA/C] NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with RBMs bypassed unless bypassing is specifically allowed within approved procedures. [ISE-NPS-92-R01]

3.0 PRECAUTIONS AND LIMITATIONS (continued)

J. The new RBM system uses Simulated Thermal Power (STP) for imposing its rod blocks. STP is APRM flux power with a 6 second time constant.

K. The RBM trip points have been changed from flow biased to power biased.

L. Calibration of the RBM Process computer requires the inputs from both Reactor Recirculation loops to be interrupted, thereby removing the flow bias signal from the APRM and the RBM causing a reactor scram. Therefore, this calibration can only be performed during an outage.

M. REFER TO TS Table 3.3.2.1-1 for RBM operability requirements.

N. RBM "A" and RBM "B" interface panels are powered from RPS "A" and "B" respectively.

O. Loss of either RPS "A" or RPS "B" will render the associated RBM INOPERABLE, generate a Rod Block and an Inverse Video trouble alarm.

P. RBM "A" and "B" both receive flow inputs from all four APRM Channels.

Q. The Flow compare function is performed in the RBM function only. They compare Total Recirculation flow differences among the four APRM channels (> 5% mismatch yields an Inverse Video trouble alarm).

R. Each RBM Channel receives inputs from all four "C" level LPRM detectors, 2 from "B" level LPRM detectors, and 2 from "D" level LPRM detectors. Reference Illustration 3, RBM/LPRM LOCATIONS.

S. The RBM determines the LPRM operability on each rod selection, and uses the input only if the LPRM is indicating ≥ 3%.

T. With regards to the Recirculation flow monitor, APRMs generate the upscale function, and the RBM generates the flow compare function.

U. CRITICAL FAULTS cause RBM INOP TRIPS and Rod Blocks. NON-CRITICAL FAULTS cause Inverse Video trouble alarm only.

V. RBM provides protection for Control Rod Withdrawal Events at higher power.

Excerpts from OPL171.148 Lesson Plan:

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18U1

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
f. Provides input into two of the four Operator Display Assemblies (ODAs) on Main Control Room Panel 9-5.	
2. General Description	
a. RBM consists of two redundant channels for monitoring of reactor power in the immediate vicinity of a control rod selected for movement.	ILT Objective 22d
1) Labeled as RBM 'A' and RBM 'B'.	
b. RBM is active above 25% Simulated Thermal Power (STP) as determined by the reference APRM for the associated RBM Channel.	LOR Objective 6b
1) 'A' RBM receives Simulated Thermal Power (STP) input from APRM #1 with alternate APRM being APRM #3 and second alternate channel being APRM #4.	
a) Alternate APRM is automatically selected when associated primary APRM is bypassed.	ILT Objective 22b
2) 'B' RBM receives Simulated Thermal Power (STP) input from APRM #2 with alternate APRM being APRM #4 and second alternate channel being APRM #3	
a) Alternate APRM is automatically selected when associated primary APRM is bypassed.	
3) Both RBM channels are bypassed when APRM reference power is less than 25% STP.	
4) Both RBM channels are also bypassed when a peripheral (edge) control rod is selected.	
a) RBM must have an internal control rod selected to be active.	
c. If a control rod withdrawal block is initiated by the RBM due to exceeding specified values, it applies only to the control rod selected.	
1) Movement of other control rods should not be inhibited.	
2) If the RBM is inoperable, none of the selected control rods will be able to be withdrawn.	
3) One channel of the RBM ('A' or 'B') can be manually bypassed using a joystick on Panel 9-5.	
a) Only one channel of the RBM can be bypassed at a time.	
3. Basic Operation	

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18U1

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
e. Receives logic power from its associated RBM instrument and from an internal power supply, powered from its associated RPS 120VAC bus.	
1) Upon a loss of power from its associated RPS power supply, the associated interface module will lose power, resulting in a critical fault and a control rod withdrawal block being generated.	
6. Determining selected LPRM channels	ILT Objective 22a ILT Objective 23 LOR Objective 6a LOR Objective 1b
a. The LPRMs are selected based on the control rod selected.	
1) There are 16 LPRMs (four strings, 4 LPRMs per string) in the vicinity of each interior control rod.	
2) The 'A' RBM channel selects two 'B' level and two 'D' level LPRMs, as well as all four 'C' level LPRMs.	
3) The 'B' RBM channel selects the remaining two 'B' level and two 'D' level LPRMs, as well as all four 'C' level LPRMs.	
4) The RBM does not use any input from the 'A' level LPRMs.	
5) The following tables show the LPRM selection for the RBM:	

RBM Channel 'A'				
Level	A	B	C	D
Upper Right		X	X	
Lower Right			X	X
Upper Left			X	X
Lower Left		X	X	
RBM Channel 'B'				
Level	A	B	C	D
Upper Right			X	X
Lower Right		X	X	
Upper Left		X	X	
Lower Left			X	X

ILT Objective 23

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18U1

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
b. LPRM values are not used by the RBM if they are from LPRMs that are bypassed in the APRM or auto-bypassed by the RBM.	ILT Objective 25
c. LPRMs that are downscale (<3%) are also bypassed by the RBM.	
1) The LPRM status display will show "AUTO/BYPASS" for the affected LPRM if the associated LPRM is manually or automatically bypassed.	ILT Objective 24
7. RBM Instrumentation	
a. RBM Instrumentation is shown in Appendix D	
8. RBM Operator Display Assembly (ODA)	
a. Two RBM ODAs are located on Panel 9-5.	
1) Each ODA receives input signals from each of the two RBM channels.	
b. The RBM ODA provides the operator on Panel 9-5 with the following indications:	
1) Local Power Range Monitor (LPRM) detector signals.	
2) Reference APRM	
3) Simulated Thermal Power	
4) The RBM ODA is only capable of displaying the status of RBM instruments.	
c. Abnormal conditions are displayed in inverse video	LOR Objective 6d
1) Flow compare	
2) Rod Block	
3) Bypass Auto	
4) Bypass Manual	
5) INPO-CAL	
6) INOP-SET.	
9. Signal Processing	Null Initiate and Gain change will each rod selection
a. The Functional Computer	
1) Delays processing of the nulling sequence (RBM Gain calculation) after receipt of the null initiate signal from the RMCS.	

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295016 (APE 16) Control Room Abandonment / 7	Tier #	1	------
G2.1.31 (10CFR 55.41.10)	Group #	1	------
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup	K/A #	295016G2.1.31	
	Importance Rating	4.6	------

Proposed Question: **# 59**

Unit 2 is operating at 100% RTP when the Nuclear Unit Senior Operator (NUSO) directs the Control Room to be abandoned due to toxic gas.

(1) Emergency Diesel Generators (EDGs) will start when 2-HS-82-1, DIESEL GENERATORS START, on Panel **(2)** is placed in the START position.

A. (1) **ALL** eight
(2) 2-9-8

B. (1) **ALL** eight
(2) 2-25-32

C. (1) **ONLY** Unit 1 and 2
(2) 2-9-8

D. (1) **ONLY** Unit 1 and 2
(2) 2-25-32

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that all of the EDGs that support Unit 1 and 2 start when either 1-HS-82-1 or 2-HS-82-1 is placed to START and 2-AOI-100-2, Control Room Abandonment requires that EDGs are started. Unit 3 also has the 3-HS-1, but once taken to START, it will only start Unit 3's EDGs. Additionally, much of the AC Electrical Distribution System can be connected across the Unit 1, 2 and 3 to provide alternate power. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that given the Control Room has to be abandoned, 2-AOI-100-2 requires that the crew proceed to the 2-25-32, Backup Control Panel.

C **CORRECT**: (*See attached*) In accordance with 0-OI-82, Standby Diesel Generator System, placing either 1-HS-82-1 or 2-HS-82-1, DIESEL GENERATORS START, will result in the simultaneous start of all Unit 1 and 2 EDGs. For second part, 1-HS-82-1 or 2-HS-82-1, DIESEL GENERATORS START, are located on Panel 1(2)-9-8 respectively.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's ability to locate Control Room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup as it relates to Control Room Abandonment. This question is rated as Memory due to the requirement to strictly recall procedural facts as it relates Control Room Abandonment and Main Control Room switches, controls and indications.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-OI-82, Rev. 180 (Attach if not previously provided)
3-OI-82, Rev. 157
2-AOI-100-2, Rev. 61

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.038 Obj. 7 (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK OPL171.038-12 001 #1396 (Note changes or attach parent)
New	

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

1396. OPL171.038-12 001

Unit 1 and 2 Control Room evacuation was conducted. A Unit 2 Operator placed handswitch 2-HS-82-1, Diesel Generators Start, on Panel 2-9-8 to START. The Unit 1 operators took **NO** actions to start diesel generators.

In accordance with 1/2- AOI-100-2, Control Room Abandonment the required appropriate actions and attachments have been completed.

One hour later, a Loss of ALL Offsite Power occurs.

Which ONE of the following completes the statements below?

When the Unit 2 Operator placed handswitch 2-HS-82-1 on panel 2-9-8 to START, Diesel Generators __(1)__ started.

When Offsite power was lost, the Unit 1 and 2 Diesel Generators __(2)__ their respective shutdown boards.

A. (1) B and D ONLY
(2) automatically energized

B. (1) B and D ONLY
(2) have to be manually synched in order to energize

C✓ (1) A, B, C and D
(2) automatically energized

D. (1) A, B, C and D
(2) have to be manually synched in order to energize

Excerpt from 0-OI-82:

5.2 Manual Fast Start at Panel 9-8

NOTES

1) Unit 1 Panel 9-8 and Unit 2 Panel 9-8 both have a DIESEL GENERATORS START switch, 1-HS-82-1. Placing either of these switches in the START position will result in the simultaneous start of all Unit 1 and 2 Diesel Generators.
2) Any Unit 1/2 Diesel Generator start will result in an auto start of the EECW pumps powered by U1/2 4KV Shutdown Boards (normally B3 and D3, A1 and C1 if aligned) after associated time delay (14 second delay if diesel supplying board or 28 second delay if normal voltage available).
3) Diesel generator frequency indication is not available unless the associated synchroscope switch on Panel 9-23 is placed in the ON position. When observing generator frequency, the synchroscope switch should only be placed in the ON position long enough to obtain a reading and then placed back in the OFF position

CAUTION

This section should only be performed during an emergency condition which requires the simultaneous start of all Unit 1 and 2 Diesel Generators.

[1] **START** all available Diesel Generators by **PLACING either** DIESEL GENERATORS START switch in START.

Unit	Handswitch Name	Handswitch No.	Panel
1	DIESEL GENERATORS START	1-HS-82-1	1-9-8
2	DIESEL GENERATORS START	2-HS-82-1	2-9-8

[2] **CHECK** all operable Diesel Generators start and stabilize between 4250 Volts and 4400 Volts and between 59 HERTZ and 61 HERTZ on Panel 9-23.

[3] **MONITOR and RECORD** Diesel Generator operating parameters as conditions allow using Attachment 2.

Excerpt from 3-OI-82:

BFN Unit 3	Standby Diesel Generator System	3-OI-82 Rev. 0157 Page 47 of 223

5.2 Manual Fast Start at Panel 9-8

NOTES

1) Placing the DIESEL GENERATORS START switch, 3-HS-82-1, in the START position will result in the simultaneous start of all Unit 3 Diesel Generators.
2) Any Unit 3 Diesel Generator start will result in an auto start of the EECW pumps powered by U3 4KV Shutdown Boards (normally A3 and C3, B1 and D1 if aligned) after associated time delay (14 second delay if diesel supplying board or 28 second delay if normal voltage available).
3) Diesel Generator frequency indication is not available unless the associated synchroscope switch on Panel 9-23 is placed in the ON position. When observing generator frequency, the synchroscope switch should only be placed in the ON position long enough to obtain a reading and then placed back in the OFF position.

CAUTION

This section should only be performed during an emergency condition which requires the simultaneous start of all Unit 3 Diesel Generators.

[1] **START** all available Diesel Generators by **PLACING** the DIESEL GENERATORS START switch, 3-HS-82-1, in START.

[2] **CHECK** all operable Diesel Generators start and stabilize between 4250 VOLTs and 4400 VOLTs and between 59 HERTZ and 61 HERTZ on Panel 9-23.

[3] **MONITOR and RECORD** Diesel Generator operating parameters in accordance with Attachment 2.

Excerpts from 2-AOI-100-2:

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0061 Page 6 of 95

4.0 OPERATOR ACTIONS

4.1 Immediate Action

NOTES

1) The immediate action to "DEPRESS REACTOR SCRAM A and B pushbuttons" is required to be completed prior to evacuating the control room.
2) Steps should be performed in order, however, Steps 4.1[7], 4.1[10], 4.1[11], and 4.1[12] may be performed at anytime while performing the immediate actions.

[1] **IF** core flow is above 60%, **THEN:**(Otherwise N/A)

LOWER core flow to between 50-60%.

[2] **DEPRESS** REACTOR SCRAM A and B pushbuttons.

[3] **PLACE** REACTOR MODE SWITCH in SHUTDOWN.

NOTE

If rods fail to insert or scram solenoids fail to deenergize in Steps 4.1[4] and 4.1[5], then Step 4.2[1] will pull RPS Scram Solenoid Fuses.

[4] **CHECK** ALL control rods fully inserted.

[5] **CHECK** all eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights extinguished.

[6] **TRIP** Reactor Recirc Pumps.

[7] **ISOLATE** RWCU.

[8] **ENSURE** Main Turbine tripped.

[9] **TRIP** Reactor Feed Pumps as necessary to prevent tripping on high water level.

[10] **START** Emergency Diesel Generators.

[11] **ENSURE** each EECW header has at least one pump in service.

4.1 Immediate Action (continued)

[12] **ANNOUNCE** to all plant personnel:

- Units 1/2 Main Control Room is being evacuated.
- All Operations personnel report to your assigned backup control stations.

[13] **OBTAIN** hand-held radios from Unit 2 Control Room.

[14] **PROCEED TO** Backup Control Panel 2-25-32.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
256000 (SF2 CDS) Condensate	Tier #	2	------
K1.30 (10CFR 55.41.3)	Group #	2	------
Knowledge of the physical connections and/or cause and effect relationships between the Condensate System and the following systems:	K/A #	256000K1.30	
• Oxygen injection system	Importance Rating	2.7	------

Proposed Question: **# 60**

Oxygen is injected in the Condensate System at the suction of the **(1)** Pumps in order to **(2)**.

A. (1) Condensate
(2) control pH

B. (1) Condensate
(2) minimize carbon steel piping corrosion

C. (1) Condensate Booster Pumps
(2) control pH

D. (1) Condensate Booster Pumps
(2) minimize carbon steel piping corrosion

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that BWR Chemistry must be maintained by controlling pH and oxygen content within limits to control corrosion. Oxygen is injected to the Condensate System to control corrosion, but not to control pH.

B **CORRECT:** (*See attached*) In accordance with 2-OI-2, Condensate System, oxygen is injected to the suction of the Condensate Pumps. Second part is correct in that oxygen is injected into the suction of the Condensate Pumps in order to minimize carbon steel piping corrosion in the Condensate System.

C INCORRECT: First part is incorrect but plausible in that Hydrogen is injected into the suction of the Condensate Booster Pumps. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

RO Level Justification: Tests the candidate's knowledge of the reason for oxygen injection into the Condensate System and the injection point. This question is rated as Memory due to the requirement to strictly recall facts in relation to Condensate oxygen injection.

Technical Reference(s): 2-OI-2, Rev. 114 (Attach if not previously provided)
OPL171.001, Rev. 19U1
OPL171.011, Rev. 24

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.001, Obj.12 (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

New	**X**
Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Excerpt from 2-OI-2:

BFN Unit 2	Condensate System	2-OI-2 Rev. 0114 Page 9 of 136

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. Prior to starting or stopping of Condensate Pumps, Chemistry should be notified for realignment of oxygen injection to Condensate Pumps.

B. Reactor power should be verified less than or equal to 95% prior to starting or stopping a Condensate Booster Pump or Reactor Feedwater Pump. Pumps must be removed from high pressure to low.

If testing in accordance with 0-TI-704, (Condensate Pump, Condensate Booster Pump and Feedwater Pump Testing), then adjust Reactor power as required to support 0-TI-704 prior to stopping a Condensate Booster Pump.

C. There is NOT adequate flow/NPSH to maintain 100% power when one Condensate Pump is out of service. To maintain adequate NPSH available to the Condensate Booster Pumps, Condensate flow is limited to (14.1 M lbm/hr).

D. To remove a Condensate pump along with any other pump, then additional restrictions apply(REFER TO drawing 2-47E800-3):

1. Pumps must be removed from high pressure to low
2. Condensate flow should be reduced to 76% (12.0 M lbm/hr) prior to removing the Condensate Pump.
3. Condensate flow can be restored to 87.5% (14.1 M lbm/hr).
4. The Condensate/Feedwater System is susceptible to perturbations from other evolutions. Any evolutions that have the potential to affect Condensate/Feedwater System flow/NPSH should be considered High Risk.

E. Condensate System should be in operation during Reactor Water Cleanup System operation in the blowdown mode.

F. To prevent motor damage, the following start limitations should be observed for the Condensate and Condensate Booster Pumps:

1. Two starts per hour from ambient temperature.
2. One start per hour from rated temperature.
3. When motor has run for a minimum of 15 min it is considered to have reached rated temp and can be restarted immediately if tripped. If motor subsequently retrips, must wait one hour to attempt restart.

Excerpts from OPL171.011:

OPL171.011, CONDENSATE SYSTEM, Rev. 24

Lesson Plan Content	Instructor Notes and Methods (Optional)
I. General Introduction	
Notes and Methods	
A. Introduce self and/or guest(s)	
1. Take Attendance	
2. Handout trainee feedback forms	
3. Introduce Topic / Goal	
4. Learning Objectives	
5. Description of how class will be conducted	
6. Evaluation Method	
7. What's In It For Me?	
II. Presentation	
A. Purpose	
The purpose of the condensate system is to supply the reactor feedwater pumps with heated demineralized water at a rate and pressure sufficient to provide adequate NPSH.	Obj. ILT 1 Obj. NLO 1
B. Condensate System Flow Path	Figure-1 Obj. ILT 2 Obj. NLOR 1 Obj. NLO 2
1. Main condenser hotwell provides NPSH for Condensate pumps.	
2. Makeup to hotwell is provided by the Condensate Storage and Supply system. See OI-2 for Alternate Hotwell Makeup.	
3. The Hydrogen Water Chemistry system injects **oxygen** into the condensate system at the suction of the condensate pumps.	
4. Tap off the Condensate pump discharge common line provides for turbine exhaust hood sprays.	
5. Condensate next flows through the tube side of the following parallel components:	Review Operator Fundamentals
a. Both SJAE condensers	
b. The steam packing exhauster (SPE) condenser	
c. The Off-gas condenser	
d. The steam packing exhauster motor operated bypass valve (FCV-2-190) for flow control through the parallel components.	
6. Condensate then passes into the Condensate filter/demineralizers. Ten demineralizers per unit with 28% capacity Bypass valve (FCV-2-130).	OPL171.011A

OPL171.011, CONDENSATE SYSTEM, Rev. 24

Lesson Plan Content	Instructor Notes and Methods (Optional)
f. Local controls for Motor operated valves are in close proximity to the valves.	
F. System Interrelationships	Figure-1 Obj. ILT 8 Obj. LOR 5 Obj. NLOR 6 Obj. NLO 8
1. Condensate supplies cooling water for:	
a. Steam packing exhauster condenser.	
b. Steam jet air ejector condenser.	
c. Off-gas condenser.	
d. Heaters and drain coolers.	
e. Turbine exhaust hood spray.	
f. RFP injection sealing water.	CST 1,2,3: 375,000 gal 12,000 gals/inch CST 4 & 5: 500,000 gal 12,500 gals/inch
2. Condensate is the backup supply to Gland Seal Water System.	
3. Condensate reject and makeup are via the CS & S system.	
4. Radioactive Waste Disposal System processes wastes from filter/demineralizers.	
5. The Hydrogen Water Chemistry system injects **oxygen** into the condensate system at the suction of the condensate pumps. The purpose of oxygen injection is to raise the oxygen concentration in the condensate system. Because oxygen is reduced in the steam, condensate oxygen is reduced. At oxygen levels < 15 ppb, carbon steel corrosion rates are accelerated due to stripping of the oxide layer.	
6. The Hydrogen Water Chemistry system injects **hydrogen** into the condensate system at the suction of the condensate booster pumps. The purpose of this is to mitigate Intergranular Stress Corrosion Cracking (IGSCC) in the reactor recirc. piping and reactor internals. The injected hydrogen causes a reduction in dissolved oxygen which lowers the radiolytic production of hydrogen and oxygen in the vessel core region. The result is a less oxidizing environment, thus, lower corrosion potentials.	
7. Condensate and CBP motors are powered from their associated 4kV Unit boards. The higher horsepower motors on Unit 1 and 2 can cause overloading on Unit 1 and 2 4kV Unit boards under certain conditions.	0-OI-57A See notes on drawings 1/2-45E721 DCN 51053 Unit Difference
8. Raw Cooling Water system provides cooling water for the condensate pump motors, CBP lube oil coolers, CBP motors and the condensate and CBP area coolers.	

Excerpt from OPL171.001:

OPL171.001, BWR Chemistry, Rev.# 19U1

C. Types of Corrosion

1. Uniform (General) Corrosion

a) Oxidation of the metal occurs over its entire surface, resulting in a slightly thinner wall thickness.

b) Easiest type to keep track of by measuring the thickness of the pipe periodically

c) This type of corrosion is controlled by maintaining the pH of the water and the dissolved oxygen content within limits.

d) System piping is designed such that minor wall loss due to general thinning caused by uniform corrosion is not a system failure concern (piping failure).

(1) The major concern is corrosion product input from the Condensate and Feedwater system, causing more activated corrosion products and a rise in radiation levels throughout the plant.

(2) Condensate Demineralizers help to minimize the buildup of corrosion products in the core by removing these corrosion products from the Main Condenser and Condensate Piping.

(3) Maintaining a thin adherent oxide layer on the interior of the piping aids in metal protection.

(4) At very low dissolved oxygen concentrations in the Condensate and Feedwater Systems, the adherent oxide layer is stripped from the interior of the piping, accelerating corrosion.

(5) Oxygen is injected into the Condensate System via the Condensate Pump suction piping to minimize the chance for this loss of oxide layer to occur.



Figure-1
ILT Objective 4a

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

271000 (SF9 OG) Offgas

K3.02 (10CFR 55.41.7)

Knowledge of the effect that a loss or malfunction of the Offgas System will have on the following systems or system parameters:

- Offsite release rate

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	271000K3.02	
Importance Rating	3.7	------

Proposed Question: **# 61**

Unit 2 is operating at 100% RTP when 2-RM-90-**265A**, OFFGAS POST TREAT RADIATION MONITOR, fails downscale.

Given the conditions above, which **ONE** of the following completes the statements below?

NOTE: CONSIDER EACH STATEMENT BELOW SEPARATELY.

If 2-RM-90-**266A**, OFFGAS POST TREAT RADIATION MONITOR, subsequently failed downscale, 2-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE, would **(1)**.

If 2-RM-90-**266A**, OFFGAS POST TREAT RADIATION MONITOR, subsequently reached its High-High-High setpoint, 2-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE, would **(2)**.

A. (1) remain open
(2) remain open

B. (1) remain open
(2) automatically close after 5 seconds

C. (1) automatically close after 5 seconds
(2) remain open

D. (1) automatically close after 5 seconds
(2) automatically close after 5 seconds

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the trip logic signals from numerous systems are often and easily confused among candidates. Second part is incorrect but plausible in that in accordance with the ARP for OFFGAS POST TREATMENT RADIATION MONITOR HIGH-HIGH, (2-9-4C, WINDOW 34), 2-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE, does not automatically close leading the candidate to believe it would remain open.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D).*

C INCORRECT: First part is correct (*See D)*. Second part is incorrect but plausible. (*See A*).

D **CORRECT:** (*See attached*) In accordance with the ARP OFFGAS POST TREATMENT RADIATION MONITOR DOWNSCALE, (2-9-4C, WINDOW 32), downscale in both sensors 2-RM-90-265A (266A), 2-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE, automatically closes after a 5 second time delay (See 2-45E614-2). For second part, in accordance with the Alarm Response Procedure (ARP) for OFFGAS POST TREATMENT RADIATION MONITOR HI-HI-HI/INOP, (2-9-4C, WINDOW 35), 2-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE, automatically closes after a 5 second time delay (See 2-45E614-2). This occurs with the combination of HI-HI-HI in 2-RM-90-266A and the given downscale in 2-RM-90-265A.

RO Level Justification: Tests the candidate's knowledge of the effect that a malfunction or loss of the Offgas System will have on Offsite release rate as it relates to the operation of 2-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE. This question is rated as C/A due to the requirement to correctly assemble given parameter from abnormal plant conditions to determine the correct system combination requirements to predict the correct outcome for trip logic signals.

Technical Reference(s): 2-ARP-9-4C, Rev. 38 (Attach if not previously provided)
OPL171.030, Rev. 21
2-45E614-2, Rev. 25

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.030, Obj. 10 (As available)

Question Source:

Bank #	BFN NRC 1102 #61
Modified Bank #	(Note changes or attach parent)
New	

Question History:

Last NRC Exam	2011

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 61**

Unit 2 Offgas Post Treat Radiation Monitor, 2-RM-90-265A, has failed downscale.

Which ONE of the following completes the statements below?

If Offgas Post Treat Radiation Monitor, 2-RM-90-266A, reaches the High-High-High setpoint, Off-Gas System Isolation Valve, 2-FCV-66-28, __**(1)**__ close.

If Offgas Post Treat Radiation Monitor, 2-RM-90-266A, fails downscale, Off-Gas System Isolation Valve, 2-FCV-66-28, __**(2)**__ close.

A. **(1)** will
(2) will

B. **(1)** will **NOT**
(2) will

C. **(1)** will
(2) will **NOT**

D. **(1)** will **NOT**
(2) will **NOT**

Proposed Answer: **A**

Excerpts from 2-ARP-9-4C:

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0038 Page 40 of 44

OG POST TRTMT
RAD MONITOR
DOWNSCALE
2-RA-90-265D

32

(Page 1 of 1)

Sensor/Trip Point:
2-RM-90-265A ≤ 1 cps
2-RM-90-266A

Sensor Location: 2-RE-90-265 Panel 2-25-94, Off-Gas Building
2-RE-90-266 Elevation 538.5

Probable Cause:
A. Circuit failure.
B. SI in progress.

Automatic Action: Downscale in both sensors of a monitor will cause OFFGAS SYSTEM ISOLATION VALVE, 2-FCV-66-28, to close.

Operator Action:
A. **CHECK** alarm condition on the following:
- OFFGAS RADIATION recorder, 2-RR-90-266 on Panel 2-9-2.
- OG POST-TREATMENT CHAN A RAD MON RTMR radiation monitor, 2-RM-90-266A on Panel 2-9-10.
- OG POST-TREATMENT CHAN B RAD MON RTMR radiation monitor, 2-RM-90-265A on Panel 2-9-10.

B. **MONITOR** indication and **CHECK** radiation levels below release limits on the following radiation recorders and associated radiation monitors:
- OFFGAS RADIATION, 2-RR-90-266 on Panel 2-9-2.
- STACK GAS/CONT RM RADIATION, 0-RR-90-147 on Panel 1-9-2.

C. **REFER TO** ODCM 1.1.2.

References: 2-45E620-4 2-47E610-90-1 GE 2-729E814 series
FSAR Sections 1.6.4.4.6, 7.12.2.2, 7.12.2.3, 7.12.3.3, 9.5.4, and 13.6.2
ODCM 1.1.2

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0038 Page 42 of 44

OG POST TRTMT
RADIATION
HIGH-HIGH
2-RA-90-265B

34

(Page 1 of 2)

Sensor/Trip Point:

2-RM-90-265A	7.73E+4 CPS
2-RM-90-266A	7.73E+4 CPS

Sensor Location: 2-RE-90-265 Panel 2-25-94 Off-Gas Building
2-RE-90-266 Elevation 538.5

Probable Cause:
A. Off-Gas flow change.
B. Adsorber lineup change.
C. Resin trap failure (RWCU or Condensate demins).
D. Fuel damage.

Automatic Action: None

Operator Action:

A. **CHECK MONITOR** high activity on the following:
- OFFGAS RADIATION recorder, 2-RR-90-266 on Panel 2-9-2.
- OG POST-TREATMENT CHAN A RAD MON RTMR radiation monitor, 2-RM-90-266A on Panel 2-9-10.
- OG POST-TREATMENT CHAN B RAD MON RTMR radiation monitor, 2-RM-90-265A on Panel 2-9-10.

B. **ENSURE** Charcoal Adsorbers in service.

C. **NOTIFY** Unit 1 and 3 operators of conditions and that verification of proper operation of Unit 1 and 3 Off-Gas system is required.

D. **CHECK** STACK GAS/CONT RM RADIATION RECORDER, 0-RR-90-147 on Panel 1-9-2.

E. **NOTIFY** Radiation Protection.

F. **REQUEST** Chemistry perform radiochemical analysis to determine source.

G. **REFER TO** 0-SI-4.8.b.1.a.1 and 0-SR-3.4.6.1-a for Technical Specification compliance and to determine if power level reduction is required.

H. **IF** directed by Shift Manager or Unit Supervisor/SRO, **THEN** **REDUCE** reactor power to maintain off-gas radiation within ODCM limits.

Continued on Next Page

Excerpt from OPL171.030:

ODCM 1.2.2 are required to be checked. EPIP-1 is referenced if release limits have been exceeded.

b) CARBON BED VAULT RAD MONITOR DOWNSCALE

Located on Panel 9-4C window 25. Alarms when RM-90-280 senses ≤1 mR/HR.

25. Offgas Post Treatment Radiation Monitors

a) OG POST TREATMENT RADIATION HI HI HI / INOP

Located on Panel 9-4C window 35. Alarms when RM-90-265A or 266A senses 620,000 cps. This is indicative of fuel damage or resin trap failure.

FCV-66-28 OFFGAS SYSTEM ISOLATION VALVE will automatically close after a 5 second time delay. Entry into AOI-66-2 is required.

b) OG POST TREATMENT RADIATION HI HI

Located on panel 9-4C window 34. Alarms when RM-90-265A or 266A senses 310,000 cps. This is also indicative of fuel damage or resin trap failure. 0-SI-4.8.b.1.a.1 and 0-SR-3.4.6.1-a are to be performed to determine if limits have been exceeded.

c) OG POST TREATMENT RADIATION HI

Located on Panel 9-4C window 33. Alarms when RM-90-265A or 266A senses 62,000 cps. This is also indicative of fuel damage or resin trap failure. 0-SI-4.8.b.1.a.1 and 0-SR-3.4.6.1-a are to be performed to determine if limits have been exceeded.

FCV-66-113B ADSORBER BYPASS VALVE will automatically close and FCV 66-113A will receive an open signal to place the Charcoal Adsorbers in service.

d) OG POST TREATMENT RAD MONITOR DOWNSCALE

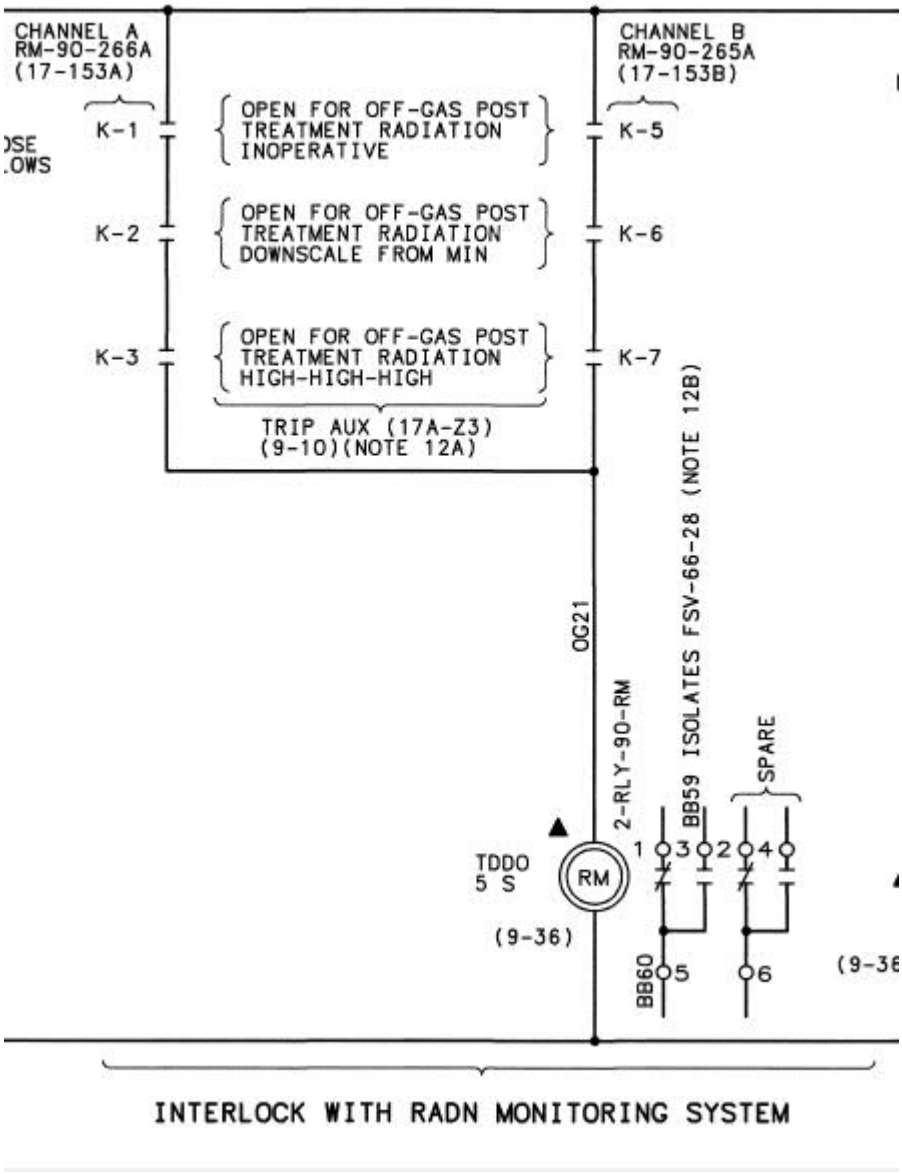
Located on Panel 9-4C window 32. Alarms when RM-90-265A or 266A senses less than 1 cps.

e) OG POST TREATMENT SAMPLE LINE PRESS ABN.

Located on Panel 9-4C window 28. Alarms when PS-90-262 senses low flow at the rad monitor.

26. Offgas Isolation Valve Closed

Excerpt from 2-45E614-2:



Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
286000 (SF8 FPS) Fire Protection	Tier #	2	------
A3.06 (10CFR 55.41.7)	Group #	2	------
Ability to monitor automatic operation of the Fire Protection System, including:	K/A #	286000A3.06	
• Fire dampers	Importance Rating	2.8	------

Proposed Question: **# 62**

Unit 1 is operating at 100% RTP when a fire is reported in 'A' EDG room.

Given the condition above, upon the dispensing of Carbon Dioxide (CO_2), 'A' EDG room supply dampers **(1)** automatically close.

When CO_2 is released, the Main Control Room (MCR) receives alarm(s) on Panel(s) **(2)**.

A. (1) will
(2) 1-9-8 **ONLY**

B. (1) will
(2) 1-9-8 **AND** 1-9-22

C. (1) will NOT
(2) 1-9-8 **ONLY**

D. (1) will NOT
(2) 1-9-8 **AND** 1-9-22

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that Panel 1-9-8 is the only alarm in the Main Control Room with an Alarm Response Procedure associated with CO_2 dispensing which the Operators will respond to.

B **CORRECT:** (*See attached*) In accordance with 0-OI-39, CO_2 System, upon the manual or automatic initiation of the CO_2 System, after at least a 20 second time delay, discharge pressure switches will provide a means to cause the affected exhaust fans to deenergize and dampers (both inlet and outlet) to close to help isolate the fire. After the time delay, CO_2 will dispense into the affected area and local alarms will activate. This is illustrated on the attached drawings 0-45E644-3 and 0-45E771-3. For second part, in accordance with 1-ARP-9-8A, Window 24, the Main Control Room's indication that CO_2 has dispensed is indicated from Panels 1-9-8 and 1-9-22, Fire Detection and Alarm System. 1-9-8 Window 24 alarming informs the MCR crew that a potential CO_2 release has occurred. Additional detailed information regarding the CO_2 release and Fire Protection status is available on 1-9-22, specifically 0-XI-026-0151, Fire Protection Display.

C INCORRECT: First part is incorrect but plausible is that in accordance with 0-OI-39, the CO_2 System is required to be manually initiated for each Unit's respective Auxiliary Instrument Rooms and the Computer Rooms. It is plausible for the candidate to think that with this manual initiation of the CO_2 System, the affected area dampers will not automatically close. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is correct *(See C*). Second part is correct (*See B*).

RO Level Justification: Tests candidate's ability to monitor automatic operation of the Fire Protection System as it relates to fire dampers and the CO_2 System. This question is rated as Memory due to the fact that it requires the strict recall of facts.

Technical Reference(s): 0-OI-39, rev. 38 (Attach if not previously provided)
1-ARP-9-8A, rev. 37
0-45E644-3, Rev. 14
0-45E771-3, Rev. 55

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.049, Obj. 3 (As available)

Question Source: Bank # ____ (Note changes or attach parent)
Modified Bank # ____
New **X**

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpts from 0-OI-39: Also supports Distractors C1, D1

BFN Unit 0	CO_2 System	0-OI-39 Rev. 0038 Page 8 of 29

3.0 PRECAUTIONS AND LIMITATIONS

A. [NRC/C] This Operating Instruction is used for three units. Valves, electrical boards, switches, and instruments have a specific unit prefix designated unless common to more than one unit, in which case "0" prefix is used. [RPT 82-13]

B. Minimum required CO_2 tank level is 5 units on the associated level indicator. (Unit 1 & 2, 5 = 8.5 tons; Unit 3, 5 = 3 tons)

C. CO_2 Fire Protection is required to be manually initiated for the following areas:

1. Auxiliary Instrument Rooms.

2. Computer Rooms.

D. Upon CO_2 initiation, an alarm will sound. Personnel then have 60 seconds (20 seconds for the Lube Oil Purification Room) to evacuate the area before CO_2 is dispensed. Evacuation times are posted at the entrance of each room. For detection purposes a wintergreen odor is injected into the CO_2 discharge.

E. Oxygen Deficiency Monitors 0-O2A-39-24 for Unit 1/2 CO_2 Storage Tank Room, and 3-O2A-39-24 for Unit 3 CO_2 Storage Tank Room, provide local audible and visual indication of low oxygen levels ($\leq$19.5%).

F. When work such as welding, cutting, or burning occurs in an area protected by CO_2, permission is required to be obtained from Unit SRO and Fire Protection Supervisor, or his designee, to place local CO_2 cutout switch in OFF.

G. When the refrigeration unit is SHUT DOWN for an extended period, approximately 6 to 20 pounds of CO_2 per hour will be lost through the respective bleeder valve located on the pilot supply line on the Unit 1 and 2 Cardox Unit and on the 3 way relief selector valve on the Unit 3 Cardox Unit.

H. If CO_2 fire protection is lost to any diesel generator building area, a fire watch is required to be established immediately in accordance with the Fire Protection Report/Fire Protection Requirement Manual and is required to be continued until CO_2 fire protection is restored.

I. Prior to operating the CO_2 System from the pilot control valve station, make sure the area that CO_2 will discharge into is evacuated, as manual operation of the pilot control valve will bypass the alarm and 60 second time delay (20 second time delay for Lube Oil Purification Room).

J. Manual operation of the CO_2 System is performed only under the direction of the Unit SRO.

K. CO_2 discharge into the Auxiliary Instrument Rooms may have an adverse effect on the instrumentation.

BFN Unit 0	CO_2 System	0-OI-39 Rev. 0038 Page 11 of 29

5.1 Automatic CO_2 Initiation (continued)

NOTE

The following actions occur upon automatic CO_2 system fire detection:

- Pushbutton station light for hazard area extinguishes.
- Alarm warning bell sounds.
- After at least 20 seconds CO_2 is discharged, discharge pressure provides a means to cause doors and dampers to close and isolate the fire.
- After a predetermined discharge period the CO_2 supply valves to the affected area close.
- Pushbutton station light for hazard area illuminates.

[5] **IF** it is determined the fire is **NOT** extinguished, **THEN**

ADD additional CO_2 using the manual pushbutton station by performing the following:

[5.1] **PULL** the hinged cover of the pushbutton station (this safely breaks the glass covering).

[5.2] **DEPRESS** the initiation pushbutton to actuate CO_2.

[6] **ENSURE** the following actions occur:

A. Pushbutton station light for hazard area extinguishes.

B. CO_2 is discharged for a predetermined time, and then the CO_2 supply valves will close.

C. Pushbutton station light for hazard area illuminates.

[7] **WHEN** the fire has been extinguished, **THEN**

PERFORM an inspection to determine extent of damage (fused detectors or damaged wiring).

[8] **WHEN** the inspection is complete, **THEN**

GO TO Section 8.6 of this instruction to return the CO_2 system to standby readiness.

5.2 Manual CO_2 Initiation (continued)

WARNING

Compartment entry is required to be made in accordance with the TVA Safety Manual, Procedure 801, Confined Space Entry.

[8] **ENSURE** CO_2 initiation is required and discharge area is evacuated.

[9] **PULL** the hinged cover of the pushbutton station (this safely breaks the glass covering).

[10] **DEPRESS** the initiation pushbutton to actuate CO_2.

[11] **ENSURE** the following actions occur:

A. Pushbutton station light for hazard area extinguishes.

B. Alarm warning bell sounds.

C. After at least 20 seconds CO_2 is discharged (discharge pressure provides a means to cause doors and dampers to close and isolate the fire).

D. After a predetermined discharge period the CO_2 supply valves to the affected area close.

E. Pushbutton station light for hazard area illuminates.

[12] **WHEN** the fire has been extinguished, **THEN**

PERFORM an inspection to determine extent of damage (fused detectors or damaged wiring)

[13] **WHEN** the inspection is complete, **THEN**

GO TO Section 8.6 of this instruction to return the CO_2 system to standby readiness.

6.0 SYSTEM OPERATIONS

None

Excerpt from 0-45E644-3:

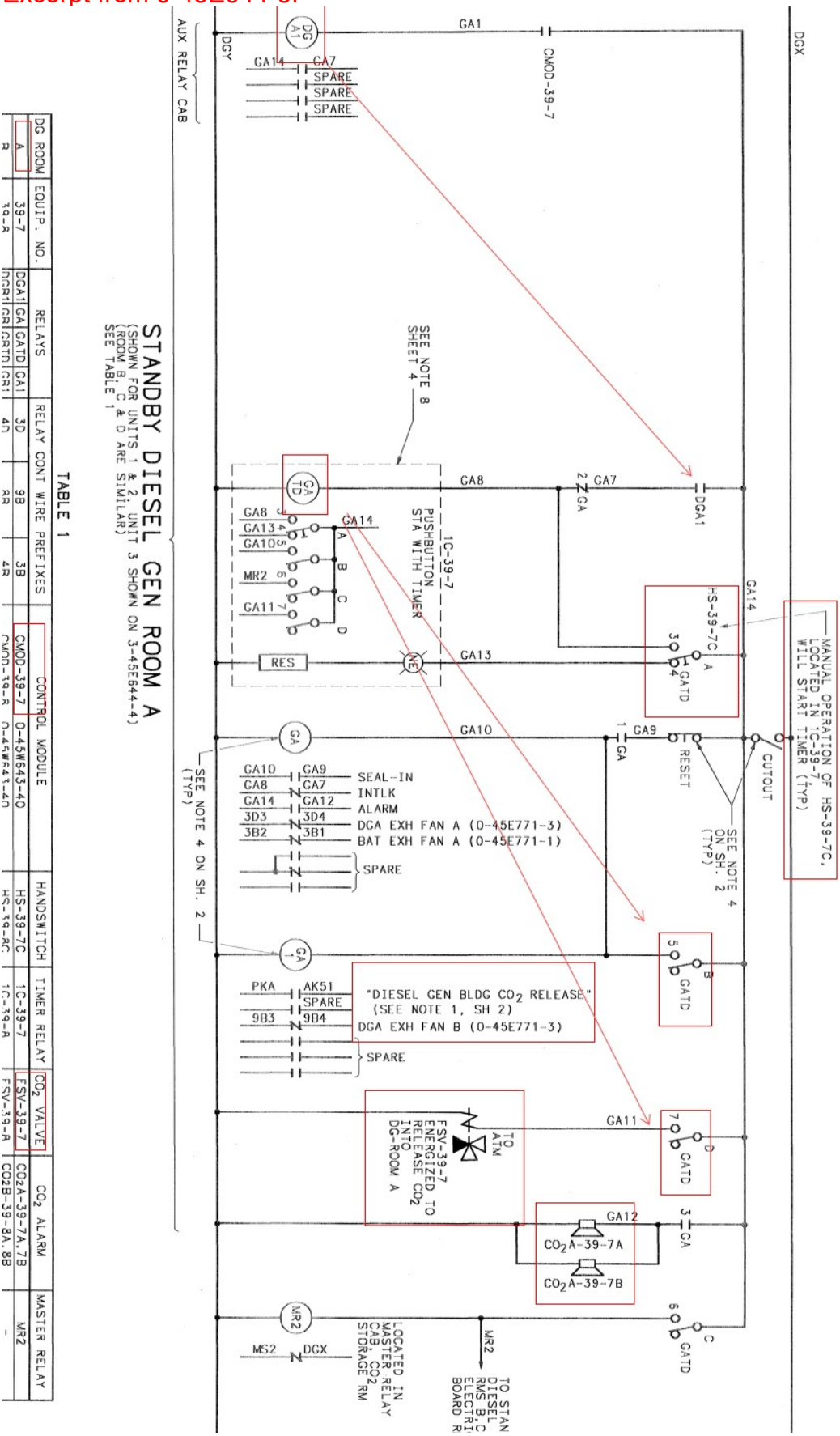
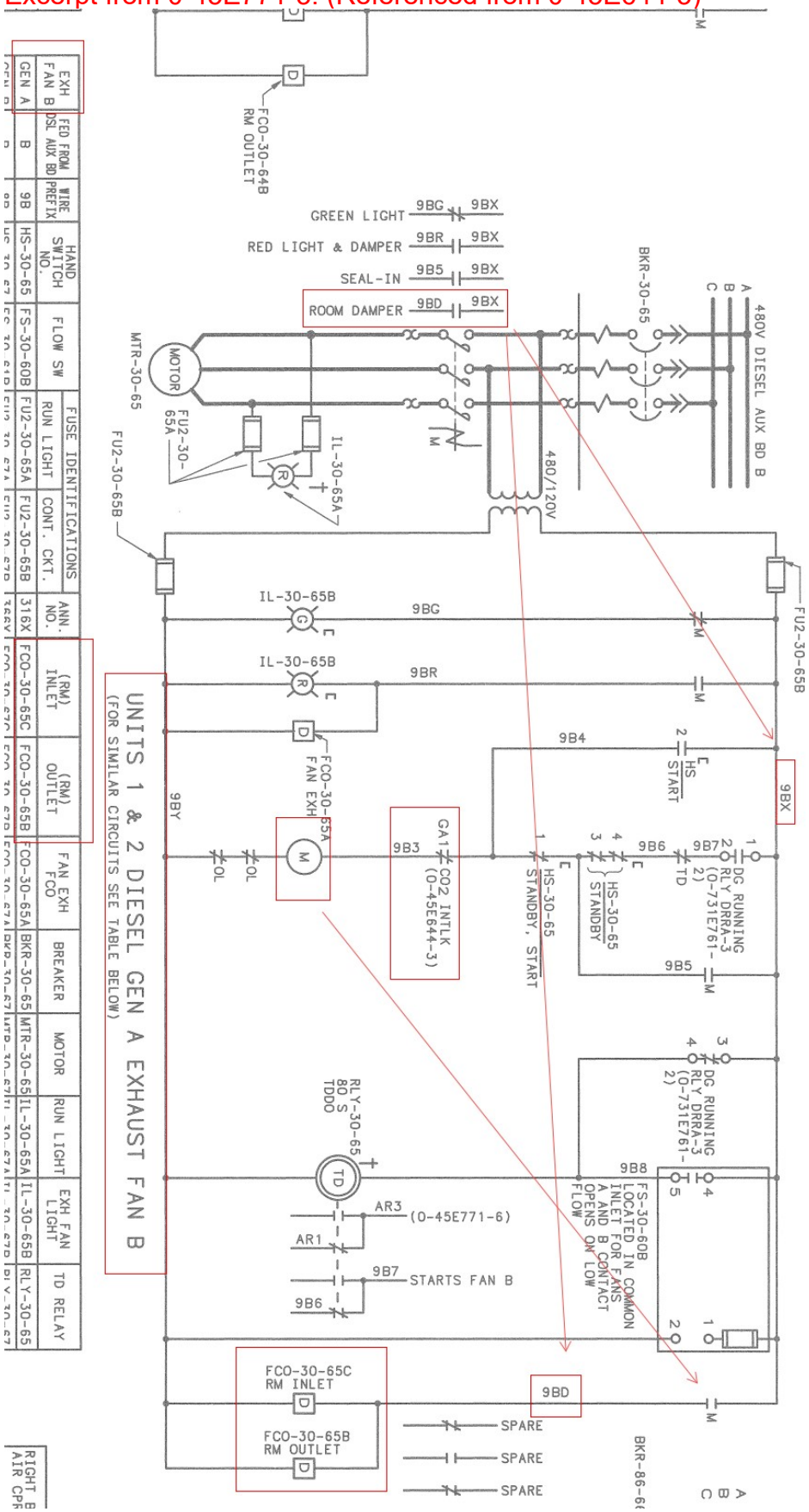


TABLE 1

DG ROOM	EQUIP. NO.	RELAYS			RELAY CONT WIRE PREFIXES	CONTROL MODULE	HANDSWITCH	TIMER RELAY	CO_2 VALVE	CO_2 ALARM	MASTER RELAY
A	39-7	DGA1	GA	GATD	GA1	3D	9B	3B	CMOD-39-7	0-45W643-40	HS-39-7C	1C-39-7	FSV-39-7	CO_2A-39-7A,7B	MR2
B	39-8	DGB1	GB	GBTD	GB1	4D	8B	4B	CMOD-39-8	0-45W643-40	HS-39-8C	1C-39-8	FSV-39-8	CO_2B-39-8A,8B	-

Excerpt from 0-45E771-3: (Referenced from 0-45E644-3)



Excerpt from 1-ARP-9-8A: Also supports Distractors A2, C2

BFN Unit 1	Panel 1-9-8 1-XA-55-8A	1-ARP-9-8A Rev. 0037 Page 33 of 44

CO_2 RELEASED
OR ABNORMAL
1-XA-57-153

24

(Page 1 of 1)

Sensor/Trip Point:

0-PIS-39-3A/B	324 psig rising or 276 psig lowering
Affected area thermostats	160°-190° by 2 out of 3 local "or pairs"

Sensor Location:

Pressure Switches:
- CO_2 Storage Unit
- Diesel Generator Building

Thermostats:
- Diesel Generator Building
- Lube Oil Purification Room
- Electrical Board Room
- Fuel Oil Transfer Rooms A or B
- Permanent Records Storage
- Oil Dispensing Room

Probable Cause:

A. Area initiation due to fire
B. Loss of CO_2 storage refrigeration
C. Filling CO_2 Tank
D. Sensor malfunction

Automatic Action:

Affected area header solenoid valve opens.

Operator Action:

A. **CHECK** CO_2 released by observing area fire alarms on Panels 1-9-8 and 1-9-22.
B. **DISPATCH** personnel to local alarm panel or CO_2 Storage Unit to investigate cause of alarm.
C. **IF** the alarm is due to an abnormal situation at the CO_2 Storage Unit, **THEN REFER TO** 0-OI-39, CO_2 System.

References:

0-45E644-2, 3 0-47E610-39-1

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
510001 (SF8 CWS*) Circulating Water	Tier #	2	------
K6.04 (10CFR 55.41.7)	Group #	2	------
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Circulating Water System:	K/A #	510001K6.04	
• Valve malfunctions	Importance Rating	3.0	------

Proposed Question: **# 63**

Unit 2 is operating at 75% RTP and conducting Condenser waterbox maintenance, with the following conditions:

- 2A, 2B, and 2C CCW Pumps are running
- When the Operator takes 2-FCV-27-13, CCW PUMP 2A DISCHARGE ISOLATION VALVE, to CLOSE, it fails to close and indicates mid position
- The AUO is asked to investigate locally and reports back that 2-FCV-27-13 indicates 50% closed

NOTE: CONSIDER EACH STATEMENT BELOW SEPARATELY.

Given the conditions above, 2A CCW **(1)** automatically tripped.

If restoring from maintenance with the 2A CCW Pump off and 2-FCV-27-13 closed, then 2-FCV-27-13 will **(2)** when 2A CCW Pump is started from the Main Control Room.

A. (1) has
(2) automatically open

B. (1) has
(2) require manual opening

C. (1) has NOT
(2) automatically open

D. (1) has NOT
(2) require manual opening

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that there are several systems in which pumps trip when their discharge valve begins to travel off of its open seat. In accordance with 2-OI-27, Condenser Circulating Water System, CCW Pumps will not trip until the discharge valve is ≥95% closed. Additionally, since the other two CCW Pumps are running, 2A CCW Pump will not trip with this valve configuration. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that CCW Discharge Valves will not fully close automatically, and must be manually closed once movement stops after switch operation from the Control Room.

C **CORRECT**: (*See attached)* In accordance with 2-OI-27, CCW Pumps will not trip unless their discharge valve is ≥95% closed. In the stem of the question, it is given that the discharge valve is 50% open, which would not cause the CCW Pump to trip on valve position. Second part is correct in that in accordance with 2-OI-27, a CCW Pump Discharge Valve will automatically open when its associated CCW Pump is started.

D INCORRECT**:** First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the effect of valve malfunctions on the CCW system, and the interlocks between the pumps and the valves. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-27, Rev. 102 (Attach if not previously provided)
OPL171.050, Rev. 24

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.050 Obj. 7 (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

New	**X**
Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Excerpt from 2-OI-27:

BFN Unit 2	Condenser Circulating Water System	2-OI-27 Rev. 0102 Page 8 of 126

3.0 PRECAUTIONS AND LIMITATIONS

A. A Condenser Circulating Water Pump will NOT start unless its discharge valve is closed and the remaining CCW pumps are either running or have their discharge valves closed.

B. Condenser Circulating Water Pumps will automatically trip if any of the following conditions occur:

1. 4160V undervoltage.
2. CCW Pump Discharge Valve motor controller closing coil is energized and neither of the two remaining CCW Pumps is running.
3. CCW Pump Discharge Valve motor controller closing coil is energized and the valve is ≥ 95% closed.
4. CCW Pump motor overcurrent.
5. CCW Pump motor high differential phase current (83 or 50 device).

C. Condenser Circulating Water Pump Discharge Valves:

1. Automatically open when its CCW Pump is started.
2. Automatically close if the associated CCW Pump is stopped or tripped unless the associated CCW Pump was the last one running.

D. Radwaste Discharge Valve, 2-FCV-77-61, automatically closes or is prevented from opening if less than two Unit 2 CCW Pumps are running or Gate 1A is less than full open.

E. During initial startup of the system, the time all three CCW Pump Discharge Valves are fully closed should be minimized. The CCW System is the suction supply for the Raw Cooling Water System.

F. If time permits the Unit 2 condenser Amertap balls should be collected before starting any Unit 2 CCW Pump.

G. Chemistry needs to be notified anytime a CCW pump is removed from service since CCW flow provides dilution water for raw water treatment chemical injection prior to discharge to the river.

Excerpts from OPL171.050 Lesson Plan:

Outline of Instruction	Instructor Notes and Methods (Optional)
Emergency Stop button on the Remote Controls ONLY cut off the signal from the Remote Controls to the reciever and DO NOT kill power to the Hoist Power Rail.	
5) The Trolleys utilize hydraulic fluid for Talon actuation. Great attention must be given to the hoses and fittings of the Talon Rakes prior to and during operation in order to prevent hydraulic fluid from entering the Waters of the U.S.	
6) Security Personnel MUST be on-site at the Control Panel in order to Enable the system. Security owns the Enable Handswitch enclosure lock and must be present any time that the system is operated as the Trolleys/Talons breech the Protected Area.	
e. There are no system interlocks or Control Room alarms associated with this system.	
f. Refer to 3-OI-27A for operational procedure.	
g.	
Interlocks and System Interrelationships	Obj. ILT 7 Obj. LOR 7 Obj. NLO 6 Obj. NLOR 6
D. CCW Pumps and associated discharge valves Numerous interlocks exist when both starting and stopping CCW pumps, relative from one pump to another, and to associated discharge valve positions. These interlocks are as follows:	
1. A CCW pump can be started only if its discharge valve is closed and the other two pumps are either running or their discharge valves are closed.	FIGURE-10 and FIGURE-11
a. Once a CCW pump is started, its associated discharge valve opens automatically.	
b. When manually stopping a CCW pump with other CCW pumps running, the pump will continue to run while its discharge valve travels closed.	
c. Once the discharge valve reaches 95% closed, the associated CCW pump will stop.	
1) This arrangement prevents reverse rotation of the pump when being stopped.	
2) If only one CCW pump is running and is manually stopped, the pump will stop immediately.	
3) Additionally, the pump discharge valve will remain open to provide a suction path for the Raw Service Water (RSW) and Raw Cooling Water (RCW) Systems.	
4) If a pump trips due to electrical fault with other pumps running, the associated discharge valve will automatically close.	

OPL171.050, Condenser Circulating Water (CCW) System, Rev. 24
Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
5) If a pump discharge valve for a running CCW pump is given a CLOSE signal, the valve will travel closed.	
6) If other pumps are running, as soon as the valve reaches 95% closed, the associated pump will trip.	
a) If no other pumps are running, the associated pump will trip immediately.	
b) Raw Service Water (RSW), High Pressure Fire Protection (HPFP)	
2. RSW normally supplies motor cooling water for the CCW pumps, and bearing lubrication water for the CCW pumps and Cooling Tower Lift pumps.	Obj. NLO 3.b Obj. NLOR 3.b
a. If RSW pressure drops to < 45 psig and any unit screen wash pump is not running, the spare screen wash pump will automatically start to supply CCW pump cooling and lubrication. 0-PCV-27-79 crosstie valve opens at 45 psig to crosstie screen wash to RSW.	Obj. ILT 8 Obj. LOR 8 **DCN 70515 currently prohibits Spare Screen Wash Auto Function**
b. At 20-psi RSW supply pressure, 7 gpm is supplied to the CCW cooling coils. Each column bearing requires an additional 8 gpm for lubrication, as well as 8 gpm for the stuffing box.	
c. Pressure indicating switches (PIS-25-4, 6, 8) located in the RSW supply lines will provide annunciation should pressure indicate insufficient (<27 gpm) or excessive (>40 gpm) RSW flow to the CCW pump.	
d. Screen Wash Pumps and Traveling Screens	FIGURE-12
3. When in AUTO, a screen wash pump will automatically start when differential pressure across the screen exceeds 7-inches H2O.	Questioning attitude (Unit 2 & 3 Only, U-1= 5"H_2O DP) **(Note: This mode is not used at BFN)**
a. As the screen wash pump pressure increases to 70 psig, the traveling screen motor would then automatically start.	Obj. NLO 8 Obj. NLOR 8
1) The system can be aligned such that the spare screen wash pump will start automatically rather than the normal screen wash pump	FIGURE-13
2) This is accomplished by placing the control switch for the normal screen wash pump (either the control switch on Panel 9-20 OR the control switch on the intake control panel) in the PULL OUT position.	
a) The spare screen wash pump would then automatically start if both:	Obj. ILT 8 Obj. LOR 8

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

Conduct of Operations

G2.1.7 (10CFR 55.41.5)

Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.1.7	
Importance Rating	4.4	------

Proposed Question: **# 64**

Unit 3 is at 100% RTP when the following event occurs:

- An EHC failure results in four (4) Turbine Bypass Valves failing open
- Generator output begins to lower
- Reactor Pressure is lowering

Given the conditions above, in accordance with 3-AOI-47-2, Turbine EHC Control System Malfunctions, the Operator is **IMMEDIATELY** required to insert a manual SCRAM if Reactor Pressure lowers to or below **___(1)___** and then close the **___(2)___**.

A. (1) 800 psig
(2) MSIVs

B. (1) 800 psig
(2) Bypass Valves using the jack controls

C. (1) 900 psig
(2) MSIVs

D. (1) 900 psig
(2) Bypass Valves using the jack controls

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the Primary Containment Isolation System (PCIS) will automatically close the Main Steam Isolation Valves (MSIVs) and the Reactor will SCRAM if Reactor Pressure lowers to 852 psig with the MODE SWTCH in RUN (PCIS Group 1 Isolation). Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that the Turbine Bypass Valve Jack push button is the way operators would manually control the Bypass Valves, either opening or closing. However, there is no guarantee that the Bypass Valves will respond during an EHC System Failure, and in accordance the proper action is the close the MSIVs.

C **CORRECT**: (*See attached*) In accordance with 3-AOI-47-2, Turbine EHC Control System Malfunctions, the Immediate Action for an EHC System failure is to insert a manual Reactor SCRAM if Reactor Pressure lowers below 900 psig. Second part is correct in that in accordance with 3-AOI-47-2, the immediate action if Reactor Pressure lowers below 900 psig is to close the Main Steam Isolation Valves (MSIVs).

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the required Immediate Operator Actions pertaining to abnormal plant conditions and the correct action(s) to take in response. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-AOI-47-2, Rev. 9 (Attach if not previously provided)
3-OI-1-1, Rev. 48
OPL171.017, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.074, Obj. 2 (As available)

Question Source:

Bank #	ILT EXAM BANK OPL171.074-02 073 #2482
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

2482. OPL171.074-02 073

Unit 2 is at 100% power. The BOP reports that an EHC malfunction has resulted in the following:

- 3 Bypass Valves failing full open
- Generator output lowering
- Reactor Pressure is 915 psig and lowering rapidly

The NUSO directs entry into 2-AOI-47-2, Turbine EHC Control System Malfunctions.

If Reactor Pressure lowers to or below ___(1)___ then ___(2)___.

A. 1) 852 psig
   2) manually scram ONLY

B. 1) 852 psig
   2) manually scram and close MSIVs

C✓ 1) 900 psig
   2) manually scram and close MSIVs

D. 1) 900 psig
   2) manually scram ONLY

Excerpt from 3-AOI-47-2:

BFN Unit 3	Turbine EHC Control System Malfunctions	3-AOI-47-2 Rev. 0009 Page 7 of 11

4.0 OPERATOR ACTIONS

NOTE

If necessary, place keeping marks may be made directly in the Control Room copy of this instruction. Management Services should be contacted for a replacement copy when time permits.

4.1 Immediate Actions

[1] **IF** Reactor Pressure lowers to or below 900 psig, **THEN**

MANUALLY SCRAM the Reactor and **CLOSE** the MSIVs.
[PER 03-006187-000]

4.2 Subsequent Actions

[1] **IF** ANY EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s).

[2] **ENSURE** Automatic Actions have occurred.

[3] **IF** a Group 1 isolation has occurred, **THEN**

PLACE EHC PUMP 3A and 3B, 3-HS-47-1A and 3-HS-47-2A, to PULL TO LOCK, (Otherwise N/A).

[4] **WHEN** the turbine bypass valves close, **THEN**

RESET the Group 1 PCIS isolation and **OPEN** MSIVs, as desired. **REFER TO** 3-OI-1.

[5] **USE** EHC WORKSTATION computer to aide in diagnosing the problem.

[6] **IF** Turbine Control Valve is oscillating AND the problem is believed to be with the electrical connection, **THEN**

PERFORM the following:

[6.1] **REDUCE** Reactor power to less than 84% IAW 3-GOI-100-12, Power Maneuvering.

[6.2] **ENSURE** offending Turbine Control Valve has been positively identified.

Excerpt from 3-OI-1:

BFN Unit 3	Main Steam System	3-OI-1 Rev. 0048 Page 9 of 71

3.2 Main Steam Isolation Valves (MSIV)

3.2.1 MSIV Closure

A. The MSIVs should be fast closed when the reactor is shutdown and no steam flow, unless required to be slow closed by surveillance, test instruction, or an abnormal condition. [BFNPER 164499]

B. When an MSIV is closed at power, the potential exists for an isolation of the Hydrogen Water Chemistry System to occur. This is due to the possibility of a hydrogen bubble becoming entrained in the main steam line drains and subsequently being released when the main steam line drains reposition in response to a MSIV closure. This scenario can result in a small Off Gas System hydrogen spike of sufficient strength to cause an automatic isolation of the Hydrogen Water Chemistry System.

C. Closure of all MSIVs could cause turbine shaft damage if main condenser vacuum is maintained and seal steam supply is not established from the auxiliary boiler.

3.2.2 MSIV Isolation

A. Main steam tunnel temperature should not be allowed to exceed 189°F to prevent MSIV isolation.

B. Whenever reactor pressure is reduced to 852 psig and the reactor mode switch is in RUN position, the MSIVs will close.

C. The MSIVs will close if 250 Vdc and 120 Vac power to the MSIV control logic is de-energized.

D. Reactor power should be less than or equal to 75% prior to closing an MSIV greater than 15 percent during closure testing. This should prevent a high steam line flow MSIV closure and subsequent reactor scram.

E. All MSIV Handswitches need to be placed in the Close Position to allow the PCIS group one trip logic to be reset, and prevent rapid re-opening upon reset of any one not in close. Placing all INBOARD MSIV Handswitches in the Close Position allows the PCIS group one A1/B1 Division I trip logic to be reset and placing all OUTBOARD MSIV Handswitches in the Close Position allows the PCIS group one A2/B2 Division II trip logic to be reset. (PER 658229-013)

Excerpt from OPL171.017 Lesson Plan:

6. What is important to note is that no single manipulation will re-open a PCIS valve. It takes two deliberate actions (resetting the Isolation signal and opening the valve). A complete listing of PCIS Valves is in the FSAR Table 5.2.2

   NOTE: Details on manipulations required to reset the various groups/Isolation valves are contained in Table 1.

F. Valve Groups

1. The Isolation valves at BFN are categorized into one of seven "groups" (1,2,3,4,5,6,& 8), based generally upon the type of system(s) Isolated and associated Isolation signals.

   NLO / NLOR- 2

   NOTE: Group 7 was removed from PCIS long ago. These are HPCI/RCIC drain valves which close on system initiation.

2. Details regarding valves in each group are provided in Table 1. A basic description of each group is as follows:

   a) Group 1

      (1) This group includes the Main Steam Isolation Valves (MSIVs), main steam line drains, and reactor water sample line Isolation valves.

      (2) The signals which will initiate a Group 1 Isolation are as follows:
      (a) RPV low-low-low level (-122" or Level 1)
      (b) *MSL High Flow .45 sec TD (135%)
      (c) *MSL Area High Temperature (189°F)
      (d) *MSL Low Pressure (852 psig with the Mode switch in RUN)
      *(MSIVs and MSL Drains only)

   ILT- 2a
   LOR- 2a
   730E927-7,8
   730E927-10
   730E927-15

   NLO / NLOR-2

   b) Group 2
      (1) This group includes Residual Heat Removal (RHR), drywell floor/equipment drain sump, and PSC Head Tank Pump suction valves.
      (2) The signals which will initiate a Group 2 Isolation are: (2-730E927-13)
      (a) RPV low level (+2"or Level 3)
      (b) Drywell High Pressure (+2.45 psig)
      (c) Reactor High pressure (100 psig) (SDC) only.

   ILT- 2g
   LOR- 2g
   730E927-7,8
   730E927-13

   NLO / NLOR- 2

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8	Tier #	1	------
AK2.03 (10CFR 55.41.7)	Group #	1	------
Knowledge of the relationship between Partial or Complete Loss of Instrument Air and the following systems or components:	K/A #	295019AK2.03	
• Feedwater system	Importance Rating	3.7	------

Proposed Question: **# 65**

In accordance with 2-AOI-32-2, Loss of Control Air, when Control Air Pressure drops to less than or equal to **(1)**, 2-LCV-3-53, Reactor Feedwater Start-up Bypass Valve, fails **(2)**.

A. (1) 65 psig
(2) as is

B. (1) 65 psig
(2) closed

C. (1) 75 psig
(2) as is

D. (1) 75 psig
(2) closed

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with 2-AOI-2, Loss of Control Air, when Control Air Pressure drops to less than or equal to 65 psig, 2-LCV-3-53, Reactor Feedwater Start-up Bypass will be directly impacted. For second part, when Control Air Pressure drops to less than or equal to 65 psig, 2-LCV-3-53 fails as is and can be manually isolated or bypassed locally.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that if open, 0-PCV-33-1, Service Air Crosstie, closes when Control Air Pressure drops to less than or equal to 30 psig.

C INCORRECT: First part is incorrect but plausible in that 2-PCV-85-654, Unit 2 CAD/Control Air Flow Select, automatically aligns nitrogen from Containment Atmospheric Dilution (CAD) Tank A to FSV-64-20(21), Suppression Chamber Vacuum Relief Valves. Second part is correct *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C).* Second part is incorrect but plausible *(See B).*

RO Level Justification: Tests the candidate's knowledge of the relationship between partial or complete Loss of Instrument Air as it relates to the Feedwater System. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to Control Air and Feedwater System.

Form 4.2-1 **Written Examination Question Worksheet**

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents. (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 2-AOI-32-2, Rev. 39 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.026 Obj. 7d (As available)

Question Source:

	Bank #	ILT EXAM BANK OPL171.012-02 074 #606
	Modified Bank #	
	New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

606. OPL171.012-02 074

When Control Air pressure drops to __(1)__ psig, then 2-LCV-3-53, Feedwater Startup Bypass Valve, fails ___(2)___.

A. 1) < 65 psig
   2) as is

B. 1) < 65 psig
   2) closed

C. 1) < 75 psig
   2) as is

D. 1) < 75 psig
   2) closed

Excerpts from 2-AOI-32-2:

Attachment 1
(Page 2 of 6)

Expected System Responses

3.0 FEEDWATER

A. RFP 2A(B)(C) MIN FLOW VLVS 2-FCV-3-20(13)(6) fail open and can be isolated locally at RFP rooms.

B. STARTUP BYPASS VALVE, 2-LCV-3-53, fails as-is and can be manually isolated or bypassed locally.

4.0 HEATER DRAINS AND VENTS

A. FEEDWATER HEATER 2A(B)(C)-1 DRAIN TO HTR 2A(B)(C)-2, 2-LCV-6-1(19)(37) fail OPEN.

B. FEEDWATER HEATER 2A(B)(C)-2 DRAIN TO HTR 2A(B)(C)-3, 2-LCV-6-4A(22A)(40A) fail CLOSED.

C. FEEDWATER HEATER 2A(B)(C)-2 COND DUMP, 2-LCV-6-4B(22B) (40B) fail OPEN.

D. FEEDWATER HEATER 2A(B)(C)-3 DRAIN TO HTR 2A(B)(C)-4, 2-LCV-6-7(25)(43) fail OPEN.

E. FEEDWATER HEATER 2A(B)(C)-4 DRAIN TO HTR 2A(B)(C)-5, 2-LCV-6-11A(29A)(47A) fail CLOSED.

F. FEEDWATER HEATER 2A(B)(C)-4 COND DUMP, 2-LCV-6-11B(29B)(47B) fail OPEN.

G. FEEDWATER HEATER 2A(B)(C)-5 DRAIN TO CNDR, 2-LCV-6-14A(32A)(50A) fail CLOSED.

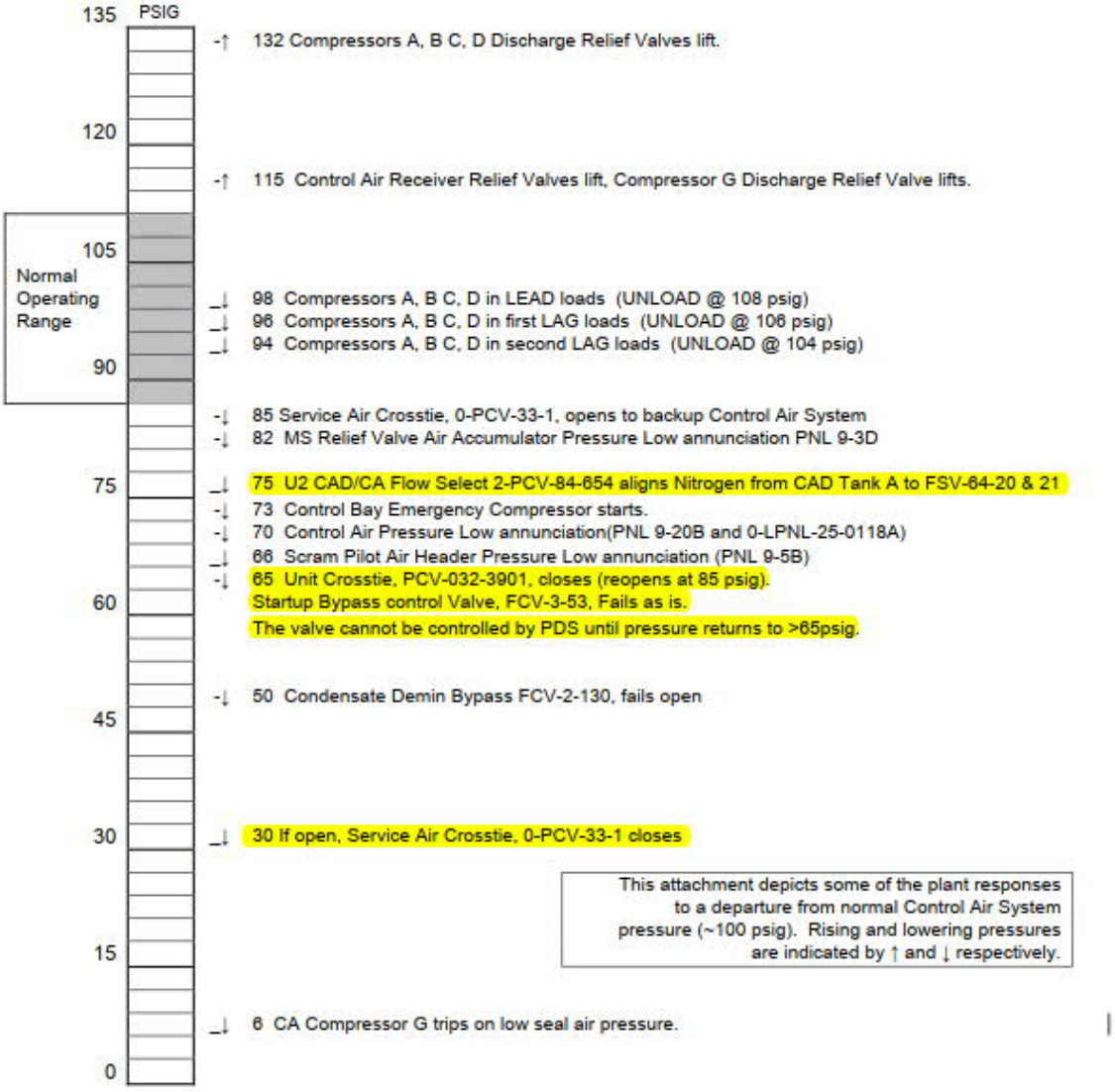
H. FEEDWATER HEATER 2A(B)(C)-5 COND DUMP, 2-FCV-6-14B(32B)(50B) fail OPEN. These LCVs can be isolated by closing 2-FCV-6-97(98)(99) from Panel 2-9-7.

I. Moisture Separator level control reservoir high level dump drain LCVs to condenser fail open.

J. Moisture Separator level control reservoir isolation valves(FCV-6-62A, 62B, 73A, 73B, 84A, 84B) to A2, B2, and C2 high pressure heater shell side fail closed.

K. Condensate Injection valve (2-FCV-002-0144) to Moisture Separator Drain System fails closed.

Attachment 3
(Page 1 of 1)
Control Air System Pressure Spectrum



Supports Distractors C1, D1:

2.0 SYMPTOMS (continued)

O. REACTOR WATER LEVEL ABNORMAL 2-LA-3-53 annunciator is in alarm (2-XA-55-5A, Window 8).

P. REACTOR PRESS HIGH 2-PA-3-53 annunciator is in alarm (2-XA-55-5A, Window 1).

Q. REACTOR CHANNEL A(B) AUTO SCRAM annunciator in alarm if any scram setpoint is exceeded (2-XA-55-5B, Window 1(2)).

R. MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW 2-PA-32-31 annunciator in alarm (2-XA-55-3D, Window 18).

3.0 AUTOMATIC ACTIONS

A. Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve.

B. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will close to separate Units 1 and 2 when Control Air Header pressure reaches 65 psig and lowering at the valve.

C. 2-PCV-84-0654, CAD/CA FLOW SEL, will select nitrogen from CAD tank A to supply 2-FSV-64-20, 2-FSV-64-21, 2-FSV-64-221, and 2-FSV-64-222 at ≤75 psig.

D. 2-PCV-84-0033, will select nitrogen from CAD tank A to supply 2-FSV-84-19, 2-FSV-64-29, and 2-FSV-64-32.

E. 2-PCV-84-0034, will select nitrogen from CAD tank B to supply 2-FSV-84-20, 2-FSV-64-31, and 2-FSV-64-34.

Supports Distractors B2, D2:

BFN Unit 2	Loss of Control Air	2-AOI-32-2 Rev. 0039 Page 20 of 25

Attachment 1
(Page 3 of 6)

Expected System Responses

5.0 RCW

A. All RCW temperature control valves fail open except for 2-TCV-24-80B and 2-TCV-24-85B on 2A and 2B RBCCW heat exchangers and 2-TCV-024-0075B on the Main Turbine Oil Coolers (4" line) which fail CLOSED.

6.0 RAW SERVICE WATER

A. RSW Head Tank outlet valve FCV-25-32 fails CLOSED. A high pressure fire pump should be started immediately to maintain fire system pressure.

7.0 HIGH PRESSURE FIRE PROTECTION

A. Fire header pressure control valve 0-PCV-26-4 located at the Intake will fail CLOSED. Relief valve will maintain fire header pressure < 175 psig.

8.0 SERVICE AIR

A. Service air supply to control air 0-FCV-33-1 opens at 85 psig Control Air press but will fail CLOSED on loss of control air at 30 psig.

9.0 CONTROL AIR

A. Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

B. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will close when Control Air Header pressure reaches 65 psig and lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295018 (APE 18) Partial or Complete Loss of CCW / 8

AA1.05 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of Component Cooling Water:

- Reactor recirculation system

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295018AA1.05	
Importance Rating	3.6	------

Proposed Question: **# 66**

Reactor Recirculation System Variable Frequency Drive (VFD) heat exchangers are cooled by **(1)**. The standby VFD Cooling Water Pump **(2)** automatically start upon a VFD low cooling water flow condition.

NOTE: RCW – Raw Cooling Water

RBCCW - Reactor Building Closed Cooling Water

A. (1) RCW
(2) will

B. (1) RCW
(2) will NOT

C. (1) RBCCW
(2) will

D. (1) RBCCW
(2) will NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 2-OI-68, Reactor Recirculation System, the Recirc VFD heat exchangers are cooled by RCW. For second part, in accordance with the Alarm Response Procedure for RECIRC DRIVE 2B COOLANT FLOW LOW, 2-ARP-9-4B, (Window 12), the standby VFD Cooling Water Pump will auto start when flow lowers to 72 gpm.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that RECIRC PUMP B COOLING WATER FLOW LOW, 2-ARP-9-4B, (Window 34), alarms when RBCCW flow to the Recirc Pump lowers to 22 gpm, however no automatic actions occur as a result.

C INCORRECT: First part is incorrect but plausible in that RBCCW does provide cooling flow to the Recirc Pump's seals and motor oil. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests candidate's ability to operate and/or monitor the Reactor Recirc System as it relates to a partial or complete loss of CCW. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to Recirc System component cooling operations.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-OI-68, Rev. 168 (Attach if not previously provided)
2-ARP-9-4B, Rev. 51
OPL171.007, Rev. 31U4
OPL171.007A, Rev. 13U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.007, Obj. 19k (As available)

Question Source:

Bank #	
Modified Bank #	(Note changes or attach parent)
New	**X**

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Excerpt from 2-OI-68:

BFN Unit 2	Reactor Recirculation System	2-OI-68 Rev. 0168 Page 154 of 211

8.14 Bypassing VFD Cooling Water TCV

NOTE

Bypassing the cooling water system TCV provides maximum cooling to the recirc drive. The operator should remain aware that this action may also decrease cooling efficiency to other plant components cooled by RCW. To control the temperature of the recirc drive cooling water, the RCW supply to the heat exchanger will need to be controlled to ensure that the recirc drive cooling water inlet temperature does not fall below 25°C.

[1] **IF** desired to bypass the TCVs on Recirc Drive 2A, **THEN** (Otherwise N/A):

[1.1] **OPEN** VFD 2A TEMP CONTROL VLV BYPASS VLV, 2-BYV-068-0649A.

[1.2] **CLOSE** VFD 2A TCV-68-600AA INLET BYPASS SHUTOFF VLV, 2-SHV-068-0645A

[2] **IF** desired to bypass the TCVs on Recirc Drive 2B, **THEN** (Otherwise N/A):

[2.1] **OPEN** VFD 2B TEMP CONTROL VLV BYPASS VLV, 2-BYV-068-0649B.

[2.2] **CLOSE** VFD 2B TCV-68-600BA INLET BYPASS SHUTOFF VLV, 2-SHV-068-0645B

Excerpts from 2-ARP-9-4B:

BFN Unit 2	Panel 2-XA-55-4B	2-ARP-9-4B Rev. 0051 Page 17 of 50

RECIRC DRVIE 2B
COOLANT FLOW LOW
2-FA-68-76

12

(Page 1 of 1)

Sensor/Trip Point:

Thorton 770 72 gpm

Sensor Location:
Elevation 639'
Rx Building
Column R-11 T-LINE

Probable Cause:
A. Recirc Drive cooling water pump failed and standby pump failed to start.
B. Break in the RECIRC DRIVE cooling water system.
C. Sensor malfunction.

Automatic Action:
Standby Recirc Drive cooling water pump will auto start.

Operator Action:
A. **CONFIRM** RECIRC DRIVE cooling water pump running.
B. **DISPATCH** personnel to the RECIRC DRIVE to check the operation of the Recirc Drive cooling water system.

References:
GE 731E320 Series 2-45E620-5 2-731E320RF-21
47W600-1 FSAR 13.6.2

BFN Unit 2	Panel 2-XA-55-4B	2-ARP-9-4B Rev. 0051 Page 49 of 50

RECIRC PUMP B COOLING WATER FLOW LOW
2-FA-68-66
34

(Page 1 of 1)

Sensor/Trip Point:

2-FIS-68-66 22 gpm

Sensor Location: Recirculation Pump 2B area Drywell.

Probable Cause:
A. Partial or complete loss of RBCCW system.
B. High RBCCW flow through Fuel Pool Cooling heat exchangers

Automatic Action: None

Operator Action:

A. **CHECK** following on RBCCW system:
- RBCCW PRI CNTMT OUTLET VALVE handswitch, 2-HS-70-47A (2-FCV-70-47) OPEN.
- RBCCW PUMP 2A, 2-HS-70-5A in service.
- RBCCW PUMP 2B, 2-HS-70-8A in service.
- Surge tank level normal indicated by RBCCW SURGE TANK LEVEL LOW 2-LA-70-2B (window 13 on 2-XA-55-4C) NOT illuminated.

B. **REFER TO** 2-AOI-70-1.

NOTE
Using ICS to trend bearing temperatures may be helpful if time and resources allow.

C. **IF** the upper or lower seal cavity temperature on RECIRC PUMP MTR 2B WINDING AND BRG TEMP temperature recorder, 2-TR-68-71 on Panel 2-9-21 is ≥ 200°F, **THEN**
PERFORM the following, otherwise, **MARK** steps N/A:
1. **SHUT DOWN** Recirculation Pump 2B by **FIRMLY DEPRESSING** .2-HS-96-20
2. **REFER TO** 2-AOI-68-1, RECIRC PUMP TRIP/CORE FLOW DECREASE.

D. **IF** CRD or RBCCW system is lost, **THEN**
REFER TO 2-OI-68, Precautions and Limitations Sections to determine requirements for continued pump operation.

References: 2-45E620-5 2-47E610-68-1 FSAR 13.6.2

Excerpt from OPL171.007A Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
2. Raw Cooling Water	
a. Supplies cooling water for the closed loop internal VFD cooling water system. The VFD heat exchangers are supplied from a four inch header connected to the suction of the RCW Booster Pump. Flow to the heat exchangers is limited to approximately 160 gpm.	Obj: NLO / NLOR- 5
3. Heat Balance	
a. The recirc pump MW value inputs to the heat balance equation. On a loss of a recirc pump or loss of this indication the heat balance is adjusted accordingly.	
b. The APLHGR limit is also automatically adjusted for a lower value.	
4. ICS Computer	
a. The ICS Computer provides system status using the following screens:	
1) RECIRC	
2) VFDPMPA and VFDPMPB	
3) RECRCFC	Figure-8
4) VFDAAL and VFDBAL	
5. RPT Breakers	
a. The RPT breakers can be reset from the control room on panel 9-4. These reset switches are in parallel with the ones in the Auxiliary Equipment Room. Divisional separation can be seen on Panel 9-4. Division 1 is in the left area and Division II is in the right area.	Obj: ILT/LOR-4 Figure-9/10
6. Feed Water Level Control System	
a. The FWLC System provides the low feed flow initiation signals for the 75% Limiter and the 28% Limiter.	
b. The FWLC system also provides the steam flow feedback/ permissive signals for the Upper and Mid Power Manual Runback circuits.	

Excerpts from OPL171.007 Lesson Plan:

2) RHR Loop II discharges into Recirculation Loop 'A'.

c) If, for whatever reason, the Recirculation Loop discharge valve will not close, the associated RHR System (LPCI Mode) must be declared Inoperable.

1) Ensures that during a significant design basis LOCA (suction line break) that all the LPCI flow travels through the reactor, removing heat, before it is discharged through the break.

2. Average Power Range Monitor (APRM)/Rod Block Monitor (RBM) System. — ILT Objective 19b, LOR Objective 9c

a) The Recirculation Loop flow elements (68-5, 68-81) provide the flow signals to the APRM for flow biased setpoint determination.

b) The APRM flow signals are then sent to the Rod Block Monitor, where they are evaluated for differences in flow between APRM drawers (5% disparity will result in 'flow compare' alarm on panel drawer).

c) Failure of one of the two flow transmitters at high power will result in a reduction in the measured flow, with possible rod block or scram signals resulting.

d) Unit difference: A failure of the flow element in the U1 'B' Recirculation Loop forced a T-Mod (BFN-1-2020-068-003) that required loop flow to be measured using the dP across the 1B Recirculation Pump as opposed to the flow element. Because of this, the recirculation discharge and suction valves must be fully open in order to determine actual loop flow. Refer to 1-OI-68 for further precautions.

3. Reactor Building Closed Cooling Water (RBCCW) — ILT Objective 9, 19c; LOR Objective 9b; NLOR Objective 4b; NLO Objective 6b

a) RBCCW cools the recirculation pump seals and recirculation pump motor oil.

b) During a partial loss of RBCCW, Recirculation Pump motor temperatures are monitored closely.

c) During a complete loss of RBCCW, Recirculation Pumps are tripped (following core flow runback and reactor scram) to prevent overheating of the Recirculation Pump motors.

4. Control Rod Drive System — ILT Objective 19d; NLOR Objective 4c; NLO Objective 6c

a) CRD Hydraulic System provides seal purging water to the #1 seal cavity on both Recirculation Pumps.

b) If a loss of CRD flow occurs, the seal cavity is still cooled using RBCCW and no over temperature condition should occur.

5. Reactor Water Cleanup System — ILT Objective 19e; LOR Objective 9d; NLOR Objective 4d; NLO Objective 6d

a) Recirculation Loop 'A' provides the suction source for the RWCU System via the same line that provides a suction source for RHR Shutdown Cooling Mode.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
239002 (SF3 SRV) Safety Relief Valves	Tier #	2	------
A1.05 (10CFR 55.41.7)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operation of the Safety Relief Valves, including:	K/A #	239002A1.05	
• Reactor Water Level	Importance Rating	4.2	------

Proposed Question: **# 67**

Unit 2 is operating at 100% RTP when an event occurs resulting in the following conditions:

- A manual SCRAM was successfully inserted
- MSIVs are **CLOSED**
- MSRVs are being used for Reactor Pressure Control
- Reactor Pressure is 825 psig and lowering

Subsequently, 2-AOI-1-1, Relief Valve Stuck Open, is entered.

Given the conditions above and in accordance with 2-AOI-1-1, the Operator is allowed to cycle the control switch of the potentially stuck open SRV **(1)** from CLOSE to OPEN to CLOSE positions.

When the MSRV closes due to Operator action, **INDICATED** Reactor Water Level will **INITIALLY** **(2)**.

A. (1) **ONLY** one (1) time
(2) rise

B. (1) **ONLY** one (1) time
(2) lower

C. (1) **UP TO** three (3) times
(2) rise

D. (1) **UP TO** three (3) times
(2) lower

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that as a standard, the majority of Main Control Room switches are only allowed to be operated once if the desired response is not obtained in the subsequent actions (inhibit or ON/OFF switches, breakers). Second part is incorrect but plausible in that when the MSRV is opened, the lowering Reactor Pressure results in a rise in the specific volume of the steam vapor, raising the flow resistance inside the core shroud. Water in the annular region swells due to the rising resistance to flow.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See attached)* In accordance with 2-AOI-1-1, Relief Valve Stuck Open, the Immediate Actions state that while observing the indications for the affected SRV on the acoustic monitor; CYCLE the affected SRV control switch as required up to three (3) times in an attempt to close the SRV. For second part, when the MSRV closes due to Operator action, indicated Reactor Water Level will initially lower due to Reactor Pressure no longer lowering. This is directly related to shrink which occurs when the MSRV closes causing the steam vapor inside the Reactor Vessel to collapse that existed due to the initial lowering Reactor Pressure. The resulting collapse reduces the two phase resistance inside the shroud. Water now more easily flows from the annular region outside the shroud to the inside of the shroud resulting in lowering Reactor Water Level, also known as shrink.

RO Level Justification: Tests the candidate's ability to predict and/or monitor changes in parameters associated with operation of the Safety Relief Valves as it relates to Reactor Water Level. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-AOI-1-1, Rev. 30 (Attach if not previously provided)

GFE Thermodynamics-Fluid Statics and Dynamics, Rev. 0

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.009 Obj. 14b (As available)

Question Source:

Bank #		
Modified Bank #	BFN 1909 NRC #51	(Note changes or attach parent)
New		

Question History:

Last NRC Exam	2019

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

Form 4.2-1	**Written Examination Question Worksheet**

10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

Copy of Bank Question:

Proposed Question: **# 51**

Unit 2 was operating at 100% RTP when the following occurs:



- MAIN STEAM RELIEF VALVE OPEN, (2-9-3C, Window 25) alarms
- SRV 1-31 indicates full OPEN on the SRV TAILPIPE FLOW MONITOR
- 2-AOI-1-1, Relief Valve Stuck Open actions are in progress

Given the conditions above, which **ONE** of the following completes the statements below?

The given annunciator is **DIRECTLY** actuated by the tailpipe **___(1)___**.

In accordance with 2-AOI-1-1, the Operator is allowed to cycle the control switch for SRV 1-31 **___(2)___** from CLOSE to OPEN to CLOSE positions.

A. **(1)** acoustic monitor
(2) UP TO three (3) times

B. **(1)** acoustic monitor
(2) ONLY one (1) time

C. **(1)** discharge temperature
(2) UP TO three (3) times

D. **(1)** discharge temperature
(2) ONLY one (1) time

Proposed Answer: **A**

Excerpt from 2-AOI-1-1:

BFN Unit 2	Relief Valve Stuck Open	2-AOI-1-1 Rev. 0030 Page 4 of 28

4.0 OPERATOR ACTION

NOTE

Once a MSRV is operated, a time delay of 15 to 30 seconds can be expected before a response can be detected on 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE. ICS can be used to monitor the discharge tailpipe temperature, but the appropriate indications on 2-TR-1-1 must be confirmed.

4.1 Immediate Action

[1] **IDENTIFY** stuck open relief valve by

OBSERVING the following:

- SRV TAILPIPE FLOW MONITOR, 2-FMT-1-4, on Panel 2-9-3,

  OR

- MSRV DISCHARGE TAILPIPE TEMPERATURE recorder, 2-TR-1-1 on Panel 2-9-47. ☐

[2] **IF** relief valve transient occurred while operating above 90% power, **THEN**

REDUCE reactor power to ≤90% RTP with recirc flow. (Otherwise N/A) ☐

[3] **WHILE OBSERVING** the indications for the affected Relief valve on the Acoustic Monitor;

CYCLE the affected relief valve control switch as required up to three times:

- CLOSE to OPEN to CLOSE positions ☐

[4] **IF** all SRVs are CLOSED, **THEN**

CONTINUE at Step 4.2.4. (Otherwise N/A) ☐

Excerpts from GFE Thermodynamics-Fluid Statics and Dynamics:

4.3 Two Phase Flow

Until now we have assumed that all the fluid was a single-phase of fluid, whether liquid or vapor.

However, when the fluid undergoes a phase change, and a mixture of liquid and vapor exists, the resulting fluid friction is much higher than single phase.

The two-phase flow friction multiplier is defined as the ratio of the two-phase head loss divided by the head loss evaluated using single-phase saturated liquid properties.

Two-phase flow friction is greater than single-phase flow friction for the same pipe dimensions and mass flow rate. Two-phase flow friction losses are experimentally determined by measuring pressure drops across different piping elements. The two-phase flow losses are generally related to single-phase flow losses through the same elements. Although the relationships are complicated, there are ways to find a two-phase flow head loss:

$$R = \frac{H_{f\ two\ phase}}{H_{f\ sat\ liquid}}$$

Where:

R = two-phase flow friction multiplier

$H_{f\ two\text{-}phase}$ = *two-phase flow head loss due to friction (ft)*

$H_{f\ sat\ liquid}$ = *single-phase saturated liquid flow head loss due to friction (ft)*

Shrink occurs upon a pressurization event (i.e. SRV closure). When the SRV closes, the steam vapor inside the reactor vessel collapses. The resulting collapse due to pressurization

INSTRUCTOR NOTES

Display "Two Phase Flow"

©2015 IntelliQlik LLC

reduces the two phase resistance inside the shroud. Water now more easily flows from the annular region outside the shroud to the inside of the shroud resulting in lowering reactor water level "shrink". The opposite is true when an SRV is opened. The lowering pressure results in an increase in the specific volume of the steam vapor, increasing the flow resistance inside the shroud. Water in the annular region "swells" due to the increased resistance to flow.

NOTE:
Reactor level control during SRV actuation must take into account the effects of "shrink" and "swell".

In addition to the head loss due to fluid friction, two-phase flow involves several phenomena that are different from single-phase flow. One of the most important is that the average velocities of the two phases are not the same. The average velocity of the liquid phase is less than the average velocity of the vapor phase .

The slip ratio (SR) of a two-phase flow is a measure of the relative velocities of the liquid and vapor portions. It is defined as follows:

$$SR = \frac{V_{avg\ vapor}}{V_{avg\ liquid}}$$

Where:
SR = slip ratio
$V_{avg\ vapor}$ = *average velocity of the vapor phase (ft/sec)*
$V_{avg\ liquid}$ = *average velocity of the liquid phase (ft/sec)*

Knowledge 1.21
Define or explain two-phase flow

INSTRUCTOR NOTES

Display "Slip Ratio"

©2015 IntelliQlik LLC

Examination Outline Cross-reference:	Level	RO	SRO
293007 Thermodynamics – Heat Transfer (Condenser applications of heat transfer)	Tier #	4	------
K1.09 (10CFR 55.41.14)	Group #	------	------
Discuss operational implications of condensate depression.	K/A #	293007K1.09	
	Importance Rating	2.7	------

Proposed Question: **# 68**

At Browns Ferry, the purpose of Condensate Depression in the Main Condenser cycle is to _____.

A. maximize plant efficiency

B. maximize condenser vacuum

C. minimize Condensate Pump cavitation

D. minimize thermal gradients in the Condenser Hotwell

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that if the Condensate leaves the Condenser Hotwell at Saturation Temperature and Saturation Pressure, the most work has been removed from the steam entering the Condenser from the Main Turbine Exhaust and no additional heating the of Condensate or Feedwater is necessary prior to entering the Reactor. If any excess cooling of the Condensate occurs (Condensate Depression) then additional heat from the fission process must be used to heat the Feedwater prior to entering the Reactor. The heat from the fission process used to warm the Feedwater could not be used to convert the Feedwater to steam. Condensate Depression REDUCES plant efficiency; however, it is necessary to ensure adequate net positive suction head to prevent cavitation of the Condensate Pumps and is a known cause that reduces plant efficiency.

B INCORRECT: Incorrect but plausible in that Condenser Vacuum is formed when the steam exhausted from the Main Turbine is cooled and condensed. If all the steam is cooled from the Turbine Exhaust the reduction in volume created by converting steam vapor into liquid water creates the vacuum. Condenser vacuum is not improved by any additional subcooling of the Condensate.

C **CORRECT**: (*See attached*) Correct in that the temperature difference between the saturation temperature for the existing condenser vacuum and the temperature of the condensate, is termed condensate depression. It is how much the condensate is subcooled in the Main Condenser, and is necessary to prevent cavitation of the Condensate Pumps.

D INCORRECT: Incorrect but plausible in that, if anything, REDUCED subcooling would mean a narrower band of temperature in the condenser thereby reducing any thermal gradients.

RO Level Justification: Tests the candidate's knowledge of the operational importance of Condensate Depression. This question is rated as Memory due to the requirement to strictly recall facts in relation to the reason for Condensate Depression.

Technical Reference(s): BWR Thermodynamics, Chapter 8, Thermal Hydraulics (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: GFES Thermal Hydraulics, Chapter 8, Obj. 12 , (As available)

Question Source:

	Bank #	GFES Thermodynamics QID#B1084
	Modified Bank #	
	New	
Question History:	Last NRC Exam	

(Note changes or attach parent)

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

10 CFR Part 55 Content:
- 55.41 **X**
- 55.43

Comments:

Copy of Bank question:

TOPIC: 293007
KNOWLEDGE: K1.09 [2.5/2.7]
QID: B1084

The purpose of condensate depression in a nuclear power plant steam cycle is to...

A. maximize condenser vacuum.

B. maximize total plant efficiency.

C. minimize cavitation of the condensate pumps.

D. minimize thermal gradients in the condenser hotwell.

ANSWER: C.

Excerpts from BWR GFES, Thermodynamics, Heat Transfer Question Justification:

0 / 0 QID: B1084 Operational Implications of Condensate Depression

Explanation

The purpose of condensate depression in the turbine/condenser phase of the plant steam cycle is to...

Condensate depression is the temperature difference between the saturation temperature for the existing condenser vacuum and the temperature of the condensate. Increasing condensate subcooling means the condensate has been cooled below the saturation temperature. Some amount of condensate subcooling or condensate depression is required to cool the condensate liquid below the saturation temperature and saturation pressure to provide a margin between system pressure and required condensate pump NPSH to prevent cavitation and allows for proper operation of pumps.

A. maximize condenser vacuum.
Incorrect – condenser vacuum is vacuum is formed when the steam exhausted from the main turbine is cooled and condensed. If all the steam is cooled from the turbine exhaust the reduction in volume created by converting steam vapor into liquid water creates the vacuum. Condenser vacuum is not improved by any additional subcooling of the condensate.

B. maximize total plant efficiency.
Incorrect – if the condensate leaves the condenser hotwell at saturation temperature and saturation pressure, the most work has been removed from the steam entering the condenser from the main turbine exhaust and no additional heating the of condensate or feedwater is necessary prior to entering the boiler (BWR, steam generator PWR). If any excess cooling of the condensate occurs (condensate depression) then additional heat from the fission process must be used to heat the feedwater prior to entering the boiler (BWR, steam generator PWR). The heat from the fission process used to warm the feedwater could not be used to convert the feedwater to steam. Condensate depression REDUCES plant efficiency, however it is necessary to ensure adequate net positive suction head to prevent cavitation of the condensate pumps and is a know cause that reduces plant efficiency.

C. minimize cavitation of the condensate pumps.
CORRECT – see above.

D. minimize thermal gradients in the condenser hotwell.
Incorrect – this is a distracter, there is no direct relationship between subcooling and thermal gradients. If anything, REDUCED subcooling would mean a narrower band of temperature in the condenser thereby reducing any thermal gradients.

The correct answer is ANSWER: C. minimize cavitation of the condensate pumps.

Excerpt from BWR GFES, Thermodynamics, Chapter 7 Student Guide:

energy directly to the bulk fluid. The exact opposite happens in a large steam condenser. The steam gives up the latent heat of condensation (equal in magnitude but opposite in heat transfer direction to the latent heat of vaporization) to the tube surface and through the tube to the cooling water. Then the droplet formed drops off of the tube and falls via gravity toward the hotwell. As it does so it mixes with the hotter bulk fluid (steam) and also transfers heat directly from the steam to the cooler droplets. Back at the tube in the place of the droplet, more steam contacts the tube and the process of condensation repeats. In this manner, the stagnant layer is minimized and higher energy steam is constantly contacting the tubes directly. The heat transfer mechanism is therefore a combination of conduction and convection and is much more efficient than pure single phase conduction.

Note that before leaving the tube surface the droplets transfer some amount of sensible heat as well. This continued cooling is called subcooling; it is desirable to a certain degree. Subcooling a few degrees is necessary to prevent cavitation of the condensate pumps. The temperature difference between the saturation temperature for the existing condenser vacuum and the temperature of the condensate, is termed condensate depression. It is expressed as the number of degrees condensate depression, or the number of degrees subcooled.

Determine the condensate depression in a condenser operating at 1 psia with a condensate temperature of 95°F.

Example 7-8

Excessive condensate depression decreases the operating efficiency of the plant, since the subcooled condensate must be reheated in the reactor vessel, thus requiring more heat from the reactor core. Excessive condensate depression also allows for an increased absorption of air by the condensate that causes accelerated oxygen corrosion of plant materials.

© 2007 GENERAL PHYSICS CORPORATION REV 4

Excerpt from BWR GFES, Thermodynamics, Chapter 7 Instructor Guide:

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Objective 12 **Example 7-8 / TP 7-64** Determine condensate depression in condenser operating at 1 psia with condensate temperature of 95°F. From steam tables, 1 psia corresponds to saturation temperature of 101.74°F.	5) As it does so it mixes with the hotter bulk fluid (steam) and also transfers heat directly from the steam to the cooler droplets. 6) Back at the tube in the place of the droplet, more steam contacts the tube and the process of condensation repeats. 7) In this manner the stagnant layer is minimized and higher energy steam is constantly contacting the tubes directly. 8) The heat transfer mechanism is therefore a combination of conduction and convection and is much more efficient than pure single phase conduction. 5. Note that before leaving the tube surface the droplets transfer some amount of sensible heat as well. a. This continued cooling is called subcooling; it is desirable to certain degree b. Subcooling few degrees is necessary to prevent cavitation of condensate pumps c. The temperature difference between saturation temperature for existing condenser vacuum and temperature of condensate, is termed condensate depression d. It is expressed as number of degrees condensate depression, or number of degrees subcooled.

© 2007 GENERAL PHYSICS CORPORATION

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures / Plan	Tier #	3	------
G2.4.14 (10CFR 55.41.10)	Group #	------	------
Knowledge of general guidelines for emergency and abnormal operating procedures usage	K/A #	G2.4.14	
	Importance Rating	3.8	------

Proposed Question: **# 69**

An event has occurred on Unit 3 with the following plant conditions:

- The Reactor has been SCRAMMED
- 3-EOI-1 has been entered due to low Reactor Water Level

Subsequently, the following occurs:

- Drywell Cooling is lost due to an electrical failure
- Drywell Temperature is 165°F and steady

Given the conditions above, the crew will _____.

NOTE: 3-EOI-1 – RPV Control Modes 1-3
3-EOI-2 – Primary Containment Control

A. re-enter 3-EOI-1 **ONLY**

B. enter 3-EOI-2 **ONLY**

C. exit 3-EOI-1 **AND** enter 3-EOI-2

D. re-enter 3-EOI-1 **AND** enter 3-EOI-2

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that re-entering EOI-1 is required based on some Drywell parameters (Drywell Pressure above 2.45 psig), but Drywell Temperature is not an entry condition for EOI-1. Re-entering EOI-1 would be required before Drywell Temperature rises to 280 ºF, but the stem of the question indicates that Drywell Temperature is not rising and not challenging 280 ºF.

B **CORRECT**: (*See attached)* Drywell Temperature above 160°F requires entry into EOI-2.

C INCORRECT: Incorrect but plausible in that entry into EOI-2 is required based on Drywell Temperature above 160°F, but EOI-1 cannot be exited until the reason for entering EOI-1 has been resolved.

D INCORRECT: Entering 3-EOI-2 is correct, however re-entering 3-EOI-1 is incorrect but plausible in that re-entering EOI-1 is required based on some Drywell parameters (Drywell Pressure above 2.45 psig), but Drywell Temperature is not an entry condition for EOI-1. Re-entering EOI-1 would be required before Drywell Temperature rises to 280 ºF, but the stem of the question indicates that Drywell Temperature is not rising and not challenging 280 ºF.

RO Level Justification: Tests the candidate's knowledge of entering the EOIs based on given plant parameters. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-EOI-1, Rev. 16 (Attach if not previously provided)
3-EOI-2, Rev. 15

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.201, Obj. 4 (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK #2970 OPL171.201-04 024 (Note changes or attach parent)
New	

Question History:

Last NRC Exam	

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

2970. OPL171.201-04 024

You are executing EOI-1, "RPV Control," in response to a Reactor Scram and subsequent low water level.

- Drywell cooling has been lost due to an electrical malfunction to the fans
- After 10 minutes, Drywell Pressure is at 2.45 psig and slowly rising

Determine which **ONE** of the following is the correct action to take?

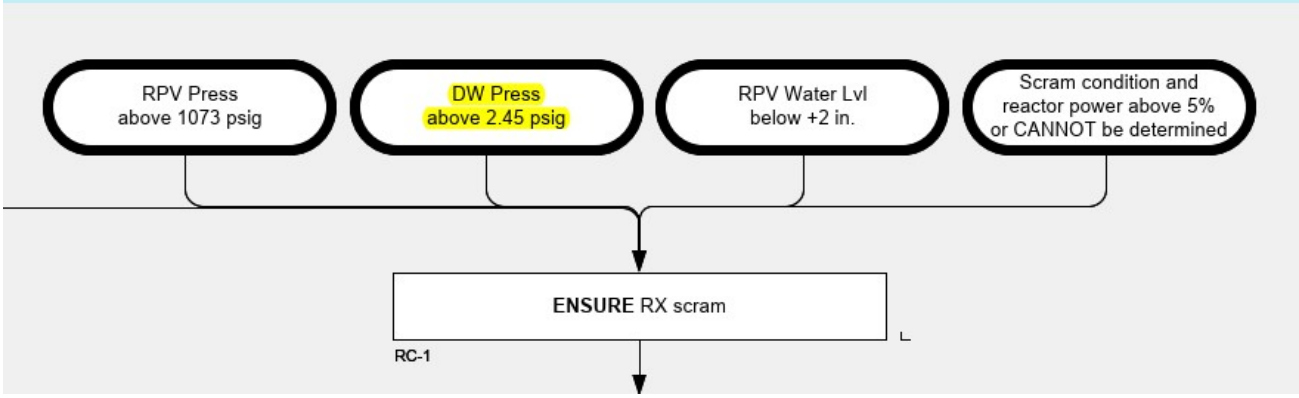
A. Re-enter EOI-1 only

B. Enter EOI-2 only

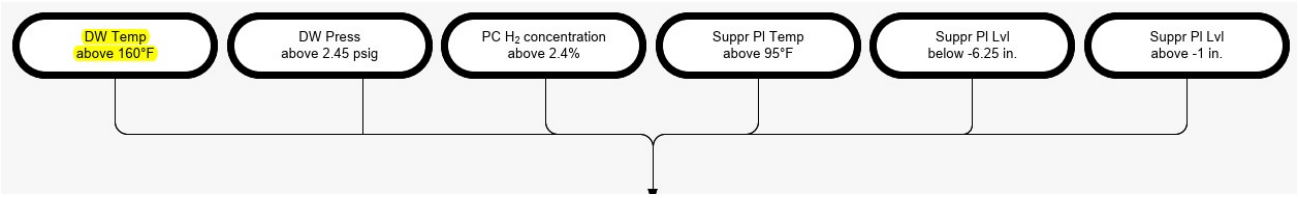
C. Exit EOI-1 AND enter EOI-2

D. ✔ Re-enter EOI-1 AND enter EOI-2

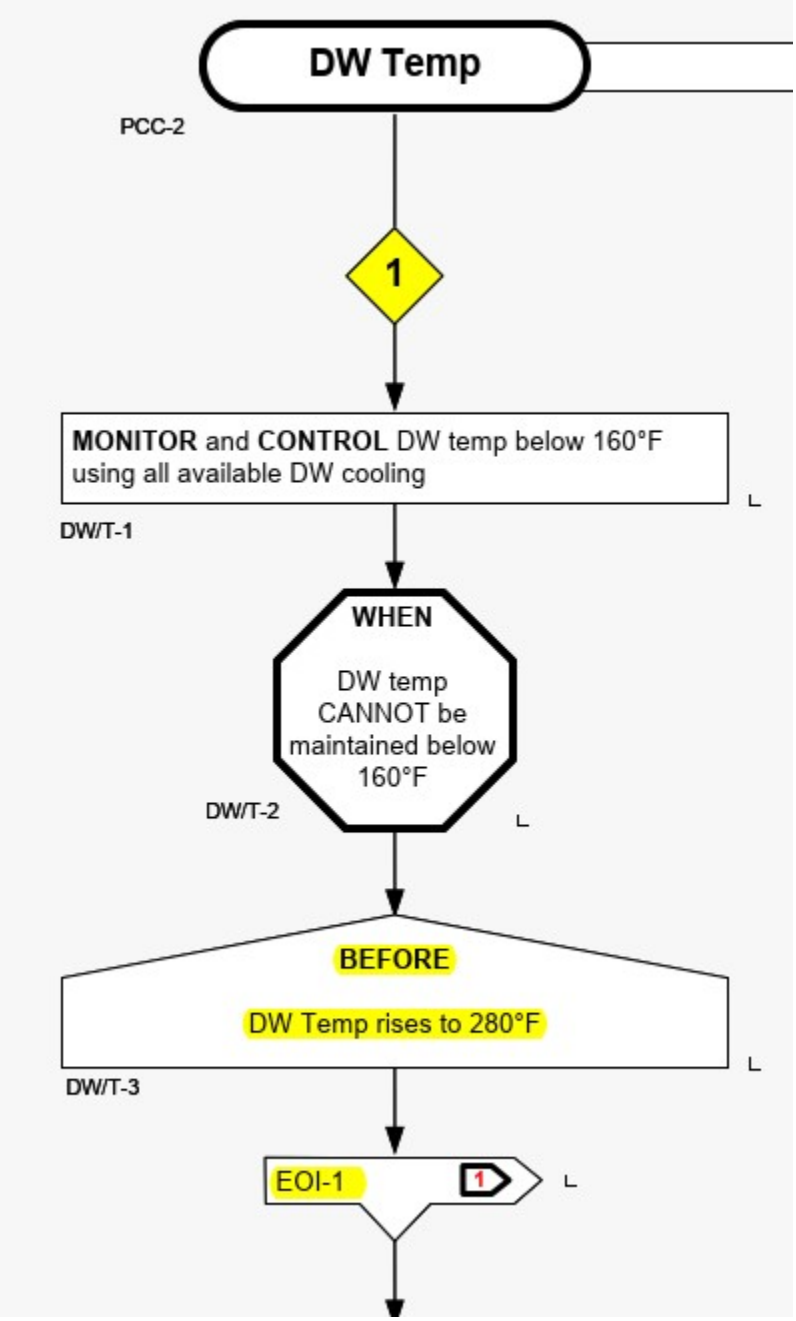
Excerpt from3-EOI-1:



Excerpt from 3-EOI-2:



Excerpt from 3-EOI-2:



Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

292003 Reactor Kinetics and Neutron Sources
K1.07 (10CFR 55.41.1)
Explain prompt critical, prompt jump, and prompt drop

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292003K1.07	
Importance Rating	3.3	------

Proposed Question: **# 70**

A Reactor Startup is in progress on Unit 1.

Which **ONE** of the following describes a condition in which the Unit 1 Reactor is prompt critical?

A. A very long Reactor Period makes Reactor control very sluggish and unresponsive.

B. Fissions are occurring so rapidly that the Effective Delayed Neutron Fraction approaches zero.

C. Any increase in Reactor Power requires a reactivity addition equal to the fraction of prompt neutrons in the core.

D. The net positive reactivity in the core is greater than or equal to the magnitude of the Effective Delayed Neutron Fraction.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that a very long Reactor Period implies a SLOW power change, rather than the extremely rapid power change that occurs when the Reactor is prompt critical.

B INCORRECT: Incorrect but plausible in that the Effective Delayed Neutron Fraction is a term to consider with respect to Reactor Period, but fission rate does not affect the Effective Delayed Neutron Fraction.

C INCORRECT: Incorrect but plausible in that the prompt neutron fraction is roughly 99.34%. The reactivity equivalent to the Effective Delayed Neutron Fraction is sufficient to make the reactor prompt critical.

D **CORRECT**: (See attached) Correct in that prompt criticality occurs when enough positive reactivity is added such that the positive reactivity is greater than or equal to the Effective Delayed Neutron Fraction. This causes the Reactor to be critical on prompt neutrons alone.

RO Level Justification: Tests the candidate's knowledge concerning the cause for prompt criticality. This question is rated as Memory due to the requirement to strictly recall facts in relation to Reactor prompt criticality.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): GFES Reactor Kinetics and Neutron Sources (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: BWR GFES Reactor Theory, CH.3, Obj.7 (As available)

Question Source:

Bank #	GFES 292003 QID B3351
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank question:

TOPIC: 292003
KNOWLEDGE: K1.07 [3.3/3.3]
QID: B3351 (P549)

Which one of the following describes a condition in which a reactor is prompt critical?

A. A very long reactor period makes reactor control very sluggish and unresponsive.

B. Fissions are occurring so rapidly that the effective delayed neutron fraction approaches zero.

C. Any increase in reactor power requires a reactivity addition equal to the fraction of prompt neutrons in the core.

D. The net positive reactivity in the core is greater than or equal to the magnitude of the effective delayed neutron fraction.

ANSWER: D.

Excerpt from BWR GFES Reactor Theory, Reactor Kinetics and Neutron Sources Question Justification:

1 / 0 QID: B3351 (P549) Prompt critical, prompt jump, and prompt drop
Explanation
Which one of the following describes a condition in which a reactor is prompt critical? A. A very long reactor period makes reactor control very sluggish and unresponsive. Incorrect – A very long reactor period implies a SLOW power change, rather than the extremely rapid power change that occurs when the reactor is prompt critical. B. The fission process is occurring so rapidly that the delayed neutron fraction approaches zero. Incorrect – Fissioning rate does not affect the delayed neutron fraction. C. Any increase in reactor power requires a reactivity addition equal to the fraction of prompt neutrons in the core. Incorrect – The prompt neutron fraction is roughly 99.34%. The reactivity equivalent to the effective ***delayed*** neutron fraction is sufficient to make the reactor prompt critical. D. The net positive reactivity in the core is greater than or equal to the magnitude of the average effective delayed neutron fraction. CORRECT - The condition of the reactor reaching criticality on prompt neutrons alone. It will occur when positive reactivity added is equal to, or greater than, the average effective delayed neutron fraction The correct answer is ANSWER D. The net positive reactivity in the core is greater than or equal to the magnitude of the average effective delayed neutron fraction.

© 2006-2008 GENERAL PHYSICS CORPORATION

Excerpt from BWR GFES Reactor Theory, Reactor Kinetics and Neutron Sources

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Solution: $\tau = \frac{\ell^*}{\rho} + \frac{(\bar{\beta}_{eff} - \rho)}{\rho\lambda_{eff}}$ Assume that this delayed neutron term does not exist. $\tau = \frac{\ell^*}{\rho}$ $\tau = \frac{10^{-4}\ \text{sec}}{0.001}$ **Objective 7** **Equation 3-39 / TP 3-79** $\tau = \frac{\ell^*}{\rho} + \frac{(\bar{\beta}_{eff} - \rho)}{\rho\lambda_{eff}}$ $\tau = \frac{\ell^*}{\rho}$	C. Prompt criticality occurs when reactivity added is greater than or equal to core average delayed neutron fraction causing reactor to go critical on prompt neutrons alone 1. The reactor would be a highly supercritical reactor in which effect of delayed neutrons in slowing down reactor power changes is lost and power is increasing very rapidly 2. In this condition, period equation is not valid because delayed term breaks down 3. (Actually, the delayed term would be negative which has no meaning for a positive reactivity addition.) The period equation now becomes: D. Many design considerations and procedure controls insure that reactor periods of this magnitude cannot be established in commercial reactor

© 2007 GENERAL PHYSICS CORPORATION
REV 4

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295023 (APE 23) Refueling Accidents / 8

AA1.03 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to Refueling Accidents:

- Fuel Handling Equipment

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295023AA1.03	
Importance Rating	3.1	------

Proposed Question: **# 71**

Unit 3 is in a Refueling Outage with the following conditions:

- 3-HS-99-5A-S1, REACTOR MODE SWITCH, is in REFUEL
- **ONE** Control Rod is out for testing
- The fuel grapple is full up **AND** unloaded
- The Refueling bridge Operator has moved the Refueling Bridge over the Reactor Core
- Gas bubbles have just been observed coming up from the Core

Note: Valve stroking is **NOT** in progress

The grapple **(1)** be lowered into the Reactor Core.

In accordance with 3-AOI-79-1, Fuel Damage During Refueling, the IMMEDIATE ACTION is to **(2)**.

A. (1) can
(2) evaluate the Radiation Levels

B. (1) can
(2) evacuate all non-essential personnel from Refueling Floor

C. (1) can NOT
(2) evaluate the Radiation Levels

D. (1) can NOT
(2) evacuate all non-essential personnel from Refueling Floor

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 0-GOI-100-3A contains the Rod, Bridge, and Hoist Blocks, Block Diagram which illustrates the numerous possible interlocks and conditions related to Refueling operations. Given that one Control Rod is withdrawn for testing, as long as that Rod is never deselected, the Refueling Bridge travel can continue and not be blocked. Second part is incorrect but plausible in that evaluate the Radiation Levels is a subsequent action from 3-AOI-79-1.

B INCORRECT: INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See attached)* In accordance with 0-GOI-100-3A, Refueling Operations (In-Vessel Operations), fuel grapple operation is blocked when all Control Rods are not full in and Refuel platform is near or over the Reactor Core as given in the stem of the question. For second part, in accordance 3-AOI-79-1, Fuel Damage During Refueling, given that gas bubbles have just been observed coming up from the Core, the IMMEDIATE ACTION is to evacuate all non-essential personnel from the Refuel Floor.

RO Level Justification: Tests the candidate's ability to operate and/or monitor Fuel Handling Equipment as it relates to Refueling Accidents. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-GOI-100-3A, Rev. 97 (Attach if not previously provided)
3-AOI-79-1, Rev. 11

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.060, Obj. 3a (As available)

Question Source:

Bank #		
Modified Bank #	ILT 1909 NRC #72	(Note changes or attach parent)
New		

Question History:

Last NRC Exam	2019

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41	**X**
55.43	

Comments:

Copy of Bank question:

Unit 1 is in a Refueling Outage with the following conditions:

- 1-HS-99-5A-S1, REACTOR MODE SWITCH is in REFUEL
- Fuel movements are in progress

Which **ONE** of the following completes the statements below?

Three direction movements are allowed **ONLY** in the **(1)**.

In the event gas bubbles are visible in the Spent Fuel Pool, the **IMMEDIATE ACTION** in accordance with 1-AOI-79-1, Fuel Damage During Refueling is to **(2)**.

A. **(1)** Reactor Vessel area
(2) evacuate the Refueling Floor

B. **(1)** Reactor Vessel area
(2) evaluate Radiation Levels

C. **(1)** Spent Fuel Pool
(2) evacuate the Refueling Floor

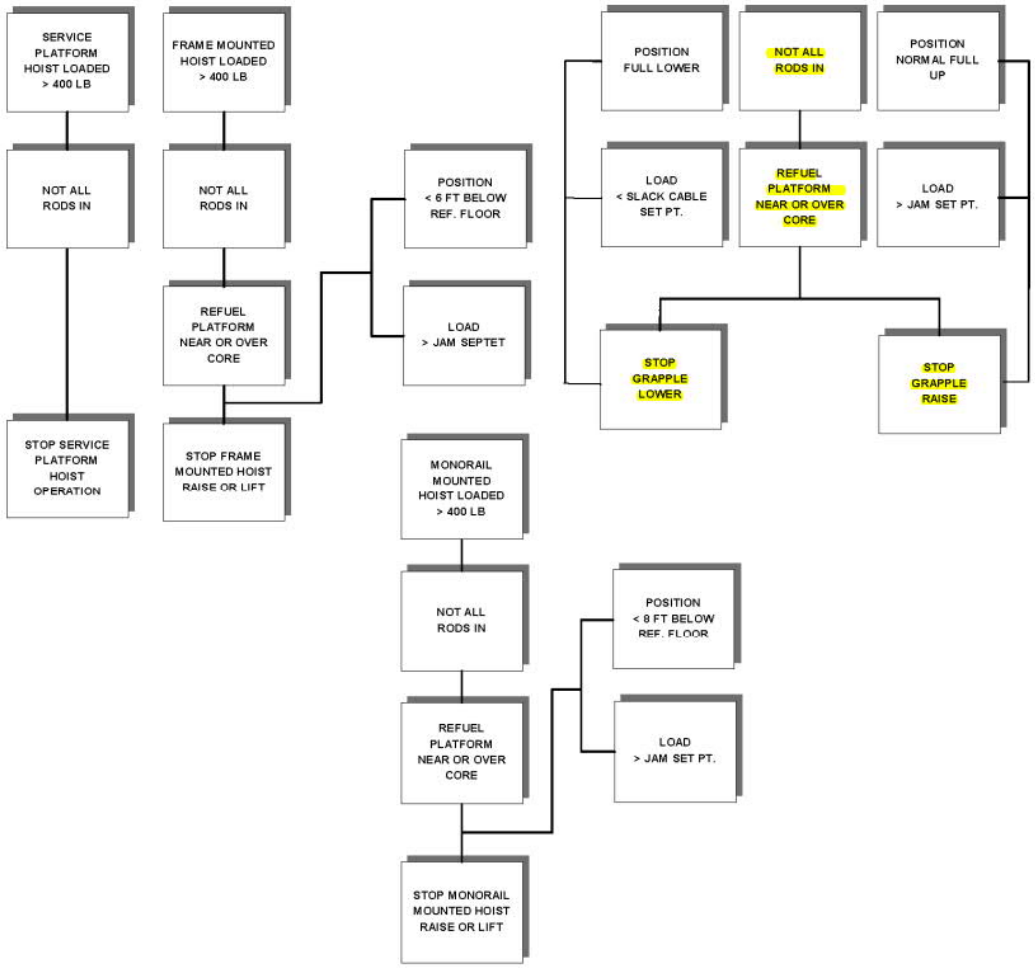
D. **(1)** Spent Fuel Pool
(2) evaluate Radiation Levels

Proposed Answer: **A**

Excerpts from 0-GOI-100-3A:

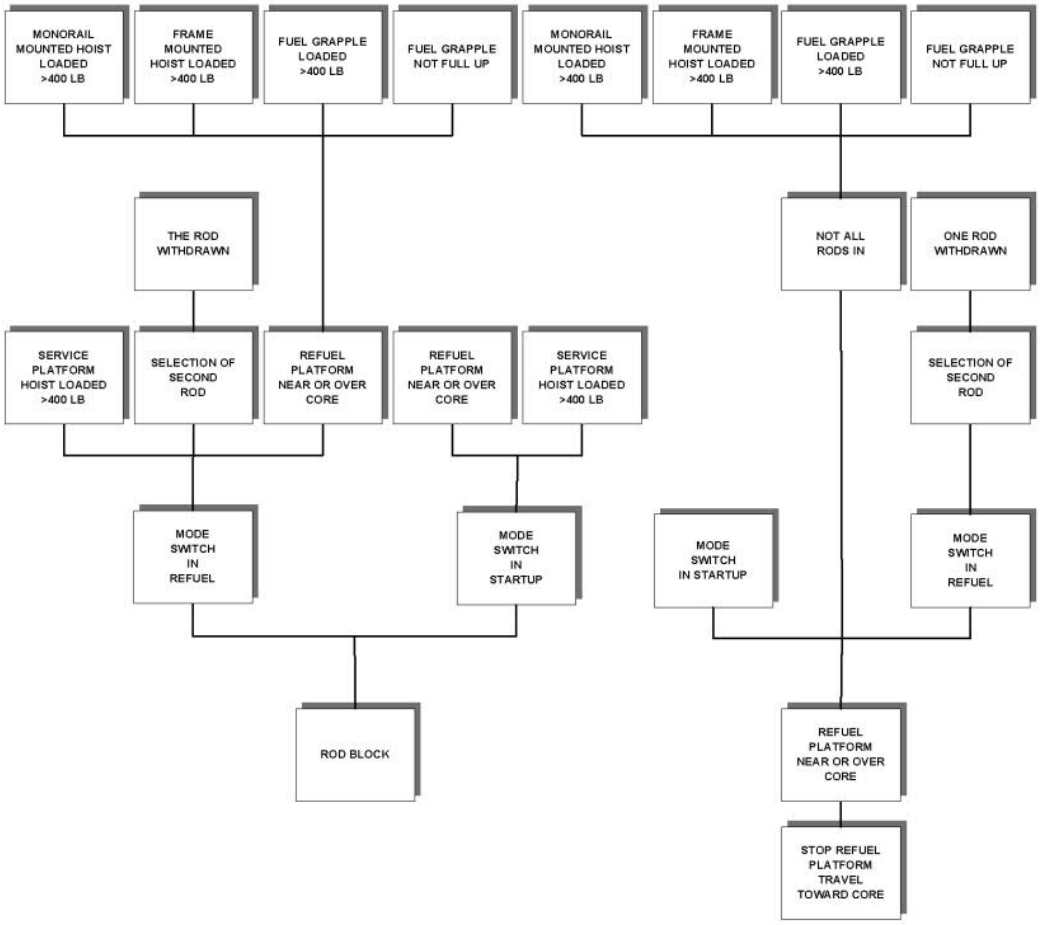
Attachment 30
(Page 1 of 2)

Rod, Bridge, and Hoist Blocks, Block Diagram



Attachment 30
(Page 2 of 2)

Rod, Bridge, and Hoist Blocks, Block Diagram



3.3 Refuel Bridge Operation

A. For safety and cleanliness reasons, fuel handling areas and equipment travel paths shall remain free of unnecessary equipment and control or power cables that cross travel paths shall be elevated or diverted.

B. Refueling equipment operators shall **NOT** rely solely on limit switches and stops for termination of refueling equipment travel. These switches and devices shall serve as backup protection. The primary means for stopping travel is operator action.

C. When operating the refuel bridge in any speed other than JOG, ensure that the grapple or devices being transported have adequate clearance above items stored in the SFSP and Reactor Cavity.

D. Bridge travel toward the core will be stopped if any of the following conditions are met (except when interlocks are jumpered out by instruction in this procedure):

1. Any platform hoist loaded or main grapple **NOT** full up and all rods **NOT** full in with the platform near or over the core.

2. Platform near or over the core with the Mode Switch in other than REFUEL.

3. One rod withdrawn and when withdrawn rod is initially deselected with the Mode Switch in REFUEL. (As long as the rod that is withdrawn is never deselected bridge travel may continue and not be blocked by this interlock.)

E. The Associated Hoist operation will be stopped if any of the following exist:

1. Main Grapple position at full lower (46 ft). Stops main hoist lower.

2. Main Grapple slack cable signal (< 50 lb tension on cable). Stops main hoist lower.

3. Associated Hoist loaded with all rods **NOT** full in with the platform near or over the core. Stops raise.

4. Associated Hoist overloaded (> 1000 lbs). Stops hoist raise.

5. All rods **NOT** full in with Platform near or over the core. Stops main hoist raise or lower.

6. Associated hoist at full up. Stops raise.

Excerpts from 3-AOI-79-1:

1.0 PURPOSE

This instruction provides the symptoms, automatic actions and operator actions for a fuel damage accident.

2.0 SYMPTOMS

A. Possible annunciators in alarm:

1. FUEL POOL FLOOR AREA RADIATION HIGH (3-XA-55-3A, Window 1)
2. AIR PARTICULATE MONITOR RADIATION HIGH (3-XA-55-3A, Window 2)
3. RX BLDG, TURB BLDG, RF ZONE EXH RADIATION HIGH (3-XA-55-3A, Window 4)
4. REACTOR ZONE EXHAUST RADIATION HIGH (3-XA-55-3A, Window 21)
5. RX BLDG AREA RADIATION HIGH (3-XA-55-3A, Window 22)
6. REFUELING ZONE EXHAUST RADIATION HIGH (3-XA-55-3A, Window 34).

B. Gas bubbles visible, in the Spent Fuel Storage Pool and/or Reactor Cavity, attributed to physical fuel damage.

C. Known dropped or physically damaged fuel bundle.

D. Portable CAM in alarm.

E. Radiation level on the Refuel Floor is greater than 25 mr/hr and cause is unknown.

Date _____

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **STOP** all fuel handling.

[2] **EVACUATE** all non-essential personnel from Refuel Floor.

4.2 Subsequent Actions

CAUTION

The release of iodine is of major concern. If gas bubbles are identified at any time, Iodine release should be assumed until RADCON determines otherwise.

[1] **ENSURE** secondary containment is intact. (**REFER TO** Tech Spec 3.6.4.1)

[2] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s).

[3] **CHECK** automatic actions.

[4] **NOTIFY** RADCON to perform the following:

- **EVALUATE** the radiation levels.
- **MAKE** recommendation for personnel access.
- **MONITOR** around the Reactor Building Equipment Hatch, at levels below the Refuel Floor, for possible spread of the release.

[5] **REFER TO** EPIP-1 for proper notification.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

292008 Reactor Theory – Reactor Operational Physics (Normal reactor shutdown)
K1.29 (10CFR 55.41.1)
Define decay heat

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292008K1.29	
Importance Rating	3.6	------

Proposed Question: **# 72**

Unit 1 was operating at 100% RTP for 10 months, when a Reactor SCRAM occurred.

IMMEDIATELY following the Reactor SCRAM, decay heat is approximately _____,

due to _____.

A. (1) 1%
(2) fission products

B. (1) 1%
(2) delayed neutrons

C. (1) 7%
(2) fission products

D. (1) 7%
(2) delayed neutrons

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Decay Heat drops to 1% approximately 1 hour after the Reactor SCRAM. The question asks for Decay Heat IMMEDIATELY following the Reactor SCRAM. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that while delayed neutrons are produced over a period of a few minutes following the SCRAM, they are not responsible for decay heat. Neutron flux is not responsible for decay heat, as Beta and Gamma decay of fission products supplies most of the energy for decay heat. Decay Heat is independent of neutron flux.

C **CORRECT**: *(See attached)* The decay of fission products produces approximately 7% of the energy from fission, and once the Reactor is SCRAMMED, after a few seconds only fission products are available to provide thermal heat. Second part is correct in that Beta and Gamma decay of fission products supplies most of the energy for decay heat.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's knowledge of Generic Fundamentals, specifically of how decay heat is produced in a Reactor following a Reactor SCRAM. This question is rated as Memory due to the requirement to strictly recall facts in relation to decay heat production.

Technical Reference(s): BWR Reactor Theory, Chapter 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: GFES Reactor Theory, Ch. 8, Obj. 30, 31. (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

Question History:

New	X
Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Excerpt from BWR GFES Reactor Theory, Chapter 8, Operational Physics:

DECAY HEAT

Approximately 200 MeV of energy is released per fission event. The majority of this energy is in the form of the kinetic energy of the fission fragments and appears instantaneously. Some time later, as the fission products decay, they release approximately 7% of the energy. Beta and gamma decay of the fission products supply most of the energy. This energy is *decay heat* and is independent of the neutron flux.

Once the reactor is shut down and the fission process stops, the fission fragments continue to decay, adding heat to the reactor core and vessel materials. The amount of decay heat added depends on the power history of the reactor and the amount of time since the reactor was shut down. The number of fissions that have occurred determines the number of fission fragments in the core. If the reactor operates at 25% power for a long time, the fission products are in equilibrium concentrations for that power level.

However, if the reactor operates at 100% power for a long time, equilibrium concentrations of the fission products for the higher power level are present within the core. When the reactor shuts down, the previous operating power level plays an important role in the amount of decay heat produced.

Following a scram, thermal power remains high for several seconds because of the stored heat within the fuel. The time lag between the power production from fission and heat being added to the coolant causes thermal output to remain the same for seven to nine seconds. Decay heat produced is at a level dependent on power history. Therefore, from a scram at 100% power, initially the thermal output of the reactor will be about 100% power, 7% supplied by decay heat. Thermal heat output decreases rapidly to the decay heat level. Eight to ten seconds after the scram, thermal output is due mainly to decay heat and drops to 7% of rated thermal output. After approximately one minute, thermal output is 3 to 5% of rated and drops to about 2% after about ten minutes. One hour after a scram, decay heat is about 1% rated thermal output. Figure 8-9 plots decay heat following a reactor shutdown.

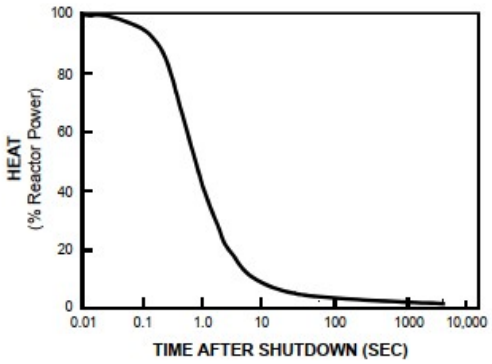


Figure 8-9 Decay Heat vs. Time after Shutdown

The most significant amount of heat generated by the core after shutdown is produced by:

a. Prompt neutrons

b. Delayed neutrons

c. Gamma and beta decay

d. Alpha and neutron emission

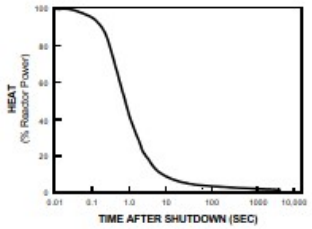
Example 8-14

© 2007 GENERAL PHYSICS CORPORATION

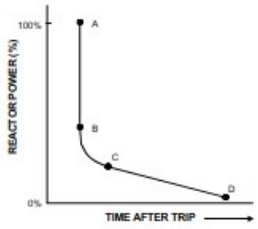
Excerpts from BWR GFES Reactor Theory, Chapter 8, Operational Physics, Instructor Guide:

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Objective 30	XI. DECAY HEAT A. Approximately 200 MeV of energy is released per fission event 1. The majority of this energy is in form of kinetic energy of fission fragments and appears instantaneously 2. Some time later, as fission products decay, they release approximately 7% of energy 3. Beta and gamma decay of fission products supply most of energy 4. This energy is *decay heat* and is independent of neutron flux B. Once reactor is shut down and fission process stops, fission fragments continue to decay, adding heat to reactor core and vessel materials 1. The amount of decay heat added depends on power history of reactor and amount of time since reactor was shut down 2. The number of fissions that have occurred determines number of fission fragments in core 3. If reactor operates at 25% power for long time, fission products are in equilibrium concentrations for that power level C. However, if reactor operates at 100% power for long time, equilibrium concentrations of fission products for higher power level are present within core 1. When reactor shuts down, previous operating power level plays important role in amount of decay heat produced
Objective 31	D. Following scram, thermal power remains high for several seconds because of stored heat within fuel

© 2007 GENERAL PHYSICS CORPORATION

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Figure 8-9 / TP 8-46  **Example 8-14 / TP 8-47** The most significant amount of heat generated by the core after shutdown is produced by: a. Prompt neutrons b. Delayed neutrons c. Gamma and beta decay d. Alpha and neutron emission Answer: c	1. The time lag between power production from fission and heat being added to coolant causes thermal output to remain same for seven to nine seconds 2. Decay heat produced is at level dependent on power history 3. Therefore, from scram at 100% power, initially thermal output of reactor will be about 100% power, 7% supplied by decay heat 4. Thermal heat output decreases rapidly to decay heat level 5. Eight to ten seconds after scram, thermal output is due mainly to decay heat and drops to 7% of rated thermal output 6. After approximately one minute, thermal output is 3 to 5% of rated and drops to about 2% after about ten minutes 7. One hour after scram, decay heat is about 1% rated thermal output 8. Figure 8-9 plots decay heat following reactor shutdown

© 2007 GENERAL PHYSICS CORPORATION

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Example 8-12 / TP 8-44 Which equation best describes the decrease in reactor power (neutron flux) from point A to point B in the figure below?  a. $T = (\bar{\beta}_{eff} - \rho)/\bar{\lambda}\rho$ b. $P = P_o e^{t/\tau}$ c. $P = P_o 10^{SUR(t)}$ d. $P = P_o \bar{\beta}_{eff}/(\bar{\beta}_{eff} - \rho)$ Answer: d	C. Shortly after scram, appearance of delayed neutrons from shorter- and longer-lived delayed neutron precursors dominates neutron population 1. These precursors, which were formed when reactor was at 100% power, decay over period of few minutes D. Once shorter-lived precursors decay, appearance of delayed neutrons from longest-lived precursors controls neutron population 1. The power falls at constant -80 second period until neutron population is low enough to see effect of source neutrons and to reach subcritical equilibrium

© 2007 GENERAL PHYSICS CORPORATION

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
Radiation Control	Tier #	3	------
G2.3.5 (10CFR 55.41.11)	Group #	------	------
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms or personnel monitoring equipment	K/A #	G2.3.5	
	Importance Rating	2.9	------

Proposed Question: **# 73**

The Wide Range Gaseous Effluent Monitor System (WRGERMS) measures effluent noble gas concentration in the **(1)** Exhaust and can be monitored on Unit **(2)** Main Control Room's Panel 9-10.

A. (1) Off-Gas Stack
(2) 2 **ONLY**

B. (1) Off-Gas Stack
(2) 1, 2, **OR** 3

C. (1) Turbine Building
(2) 2 **ONLY**

D. (1) Turbine Building
(2) 1, 2, **OR** 3

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached)* First part is correct in that in accordance with 2-OI-90, Radiation Monitoring System, the Wide Range Gaseous Effluent Monitoring System (WRGERMS) monitors Stack Release and Flow Rate. Second part is correct in that in accordance with 2-OI-90, the remote WRGERMS panel is located in the Unit 2 Control Room.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that there are numerous Radiation Monitors in all three Control Rooms.

C INCORRECT: First part is incorrect but plausible in that the Turbine Building Exhaust is monitored by Radiation Monitors 2-RM-90-249, Turbine Building Roof Exhaust Vent Continuous Air Monitor, and 2-RM-90-251, Turbine Building Roof Exhaust Vent Continuous Air Monitor. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

RO Level Justification: Tests the candidate's knowledge of the use of fixed radiation monitoring systems, including the location of the monitors. This question is rated as Memory due to the requirement to strictly recall facts in relation to which radiation monitor is installed to monitor Offgas Stack Radiation and the location of the monitor.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): 2-OI-90, Rev. 91 (Attach if not previously provided)
OPL171.033, Rev. 16

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.033, Obj. 2 (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK #1155 OPL171.033-02 001 (Note changes or attach parent)
New	

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**
55.43

Copy of Bank Question:

1155. OPL171.033-02 001

Which ONE of the following completes the statement below?

The Wide Range Gaseous Effluent Radiation Monitor System (WRGERMS) consists of __**(1)**__ ranges, **AND** has _____**(2)**_____ .

A. **(1)** TWO
(2) remote monitors in **ALL** three Units Control Rooms

B. **(1)** THREE
(2) remote monitors in **ALL** three Units Control Rooms

C. **(1)** TWO
(2) a remote monitor in Unit 2 Control Room

D. **(1)** THREE
(2) a remote monitor in Unit 2 Control Room

Excerpt from 2-OI-90:

BFN Unit 2	Radiation Monitoring System	2-OI-90 Rev. 0091 Page 55 of 79

Attachment 2
(Page 4 of 11)

Wide Range Gaseous Effluent Radiation Monitor Operation

Remote (U2 MCR) Display

The Remote Computer located in the U2 Control Room on Panel 2-9-10 updates the remote display which is used to convey WRGERM status to Control Room personnel. This display also provides means for calibrating and troubleshooting portions of the system. The remote display is composed of Stack Release Rate indication, Status indication, Stack Flow Rate and Detector channels virtual instruments, Annunciators, and online tool buttons.

The Stack flow and Release Rate, as well as Detector Channel indications and Virtual instruments are self-explanatory. The status indication functions in the same manner as the Host Computer. Specifically, the status indication changes from NORM RNG (green font) to MID RNG (yellow font) to HIGH RNG (red font) depending upon the Stack Release Rate or level of radioactive Noble Gases in the effluent stream. The choice of noble gas detector channel to use is based upon the status indication. If the status indicator shows that WRGERM is in MID RNG mode, mid range noble detector (0-RE-90-98A) readings are being used to calculate stack release rate. Annunciator windows change color whenever an alarm condition is present.

The online function buttons provide the following actions when selected:

A. The RESET button is used to reset the HI/LO arrows on each vertical and horizontal virtual instrument. The HI/LO arrows move with the process value cursor and provide indication of the highest and lowest values read by the instrument since the RESET function was last performed.

B. Selection of the ABOUT button shows the application About Box which contains release and other pertinent information about the software.

C. The TEST button provides a few troubleshooting functions. Specifically, the three WRGERM Annunciator alarms can be manually actuated. There is also a recorder control that allows 0-RR-90-306 to be manually adjusted in order to check its operation. A third feature allows the Host Computer to be queried for information and the Host Computer response recorded for troubleshooting purposes.

Excerpts from OPL171.033 Lesson Plan:

6. The ratemeter can be taken out of test and returned to the normal display by pressing the MODE key.

n. CONTROL ROOM HIGH RADIATION ISOLATION (55-6B-3) Alarms when either monitor (90-259A or B) reaches 221 cpm above background. This starts CREVs and isolates the fresh air paths to Control Bay Elevation 3C.

o. CONTROL ROOM ISOLATION MONITOR DNSCL / INOP (55-6B-10) Alarms if either monitor (90-259A or B) is at 10 cpm or less.

7. **Stack-Gas Radiation Monitoring System**

0-RM-90-306(Wide Range Gaseous Effluent [WRGERMS])

a. The objectives of this stack-gas radiation monitoring system are twofold:

1. Indicate and record release rates from the stack during normal operation and alarm when limits are reached.
2. Indicate and record release rates from the stack during accident conditions which could result in gross radiation release.

b. Wide Range Gaseous Effluent Radiation Monitor (0-RM-90-306)consists of the following:

1. 0-RE-90-093 Normal Range noble gas detector
2. 0-RE-90-98A Intermediate Range noble gas detector
3. 0-RE-90-98B High range noble gas detector

c. If a high radiation level is detected by the Normal Range noble gas detector the sample flowpath automatically swaps to the Intermediate/High Range flowpath. The Intermediate/High Range flowpath has an Intermediate Range detector (0-RE-90-098A) and a High Range noble gas detector (0-RE-098B) share a common sample chamber collector/filter. Transfer from the Intermediate to the High Range noble gas detector is done automatically on high radiation.

d. There are two normal range sample pumps and two mid/high range pumps with one of the normal range sample pumps is always running when the monitor is sampling.

Installed after the Three Mile Island accident

Controls and Indications are on **Unit 2 ONLY**

NLO/NLOR OBJ 8

Normal range particulate and Iodine filters are abandoned in place

The Host Computer will alternate operation of the normal range pumps on a periodic basis (e.g., 6 hour intervals)

OPL171.033 , Process Radiation Monitoring, Rev# 16

d. NUMAC Digital Display is provided on Panel 9-10. Each monitor also provides input to a recorder on panel 9-2. ICS Display "MISCRAD".

140 Refuel Zone
141 Refuel Zone
142 Reactor Zone
143 Reactor Zone

4. **Plant Ventilation Exhaust Radiation Monitoring System**

   a. Normal exhaust from turbine building, reactor building and refueling floor (1/2/3-RM 90-250), Normal exhaust from radwaste building (0-RM 90-252), and Turbine building roof vents exhaust (1/2/3-RM 90-249 and 1/2/3-RM-90-251) make up the plant ventilation exhaust radiation monitoring system.

ILT/LOR OBJ 3i
NLO/NLOR OBJ 3H

   b. Monitors the total noble gas activity of the ventilation exhaust utilizing a skid mounted Continuous Air Monitor (CAM). A sample is drawn from the ventilation exhaust flow by a vacuum pump and the sample stream passes through a flow element and a flow control valve, which maintain the desired sample flow rate.

   c. The sample is filtered for particulates and iodine which are monitored by Chemistry Dept. during weekly filter swap. The sample stream is then processed through a lead shielded chamber with a scintillation detector that detects gamma radiation from the noble gases.

   d. Each of the ventilation exhaust CAM's sends its output to a touch screen Monitor (0-MON-90-361) on the **Unit 1** panel 9-2

   e. High radiation in the exhaust flow will give an alarm on panel 9-3 of the respective unit.

      1. RM-90-250 Alert level alarm at 3297 cpm, High level alarm at 6594 cpm

Alarm Function Only

      2. RX BLDG, TURBINE BLDG, RF ZONE EXHAUST RAD MONITOR DNSCL. (55-3A-11) Alarms if RM-90-250 monitor senses $\leq$ 51 cpm for 10 minutes, or low flow for 5 seconds.

Alarm Function Only both input to this alarm

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

293008 Thermal Hydraulics – Recirculation System

K1.26 (10CFR 55.41.14)

Explain jet pump operating principle.

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	293008K1.26	
Importance Rating	3.1	------

Proposed Question: **# 74**

Unit 2 is operating at 100% RTP. Refer to the drawing below of a Recirculation Jet Pump.

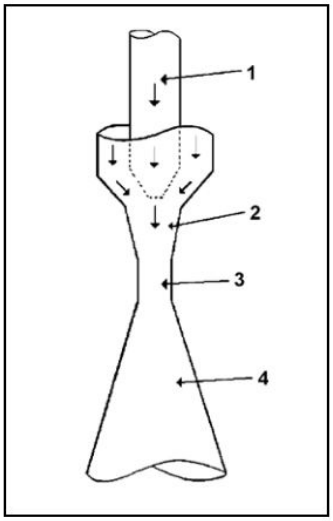
The **HIGHEST** water **VELOCITY** in the Jet Pump will exist at point _____.

A. 1

B. 2

C. 3

D. 4



Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** Incorrect but plausible in that in the Jet Pump, Point is where the Recirculation Pump Discharge Flow (Driving Flow) is directed to the Jet Pump at high velocity and high pressure; however, this is not the point of the highest water velocity.

B INCORRECT**:** Incorrect but plausible in that Point 2 is the point where the Driving Flow is accelerated (with a corresponding pressure drop) to entrain the surrounding fluid (Driven Flow) from the Downcomer Region to raise the amount of water being sent through the Jet Pump. However, this is not the point of the highest water velocity.

C **CORRECT:** Correct in that as the Driving and Driven Flow are forced through the Jet Pump, a momentum transfer is caused by mixing of steam and water in the Jet Pump at Point 3 (mixing section), causing pressure to rise slightly, but keeping velocity at its highest point in the Jet Pump.

D INCORRECT**:** Incorrect but plausible in that as water exits the Jet Pump, the area at Point 4 increases, causing Pressure to rise. However, at Point 4 velocity drops.

RO Level Justification: Tests the candidate's knowledge of Jet Pump operation, specifically how water velocity varies as it flows through the various portions of the Jet Pump. This question is rated as Memory due to the requirement to strictly recall facts in relation to how a Jet Pump operates.

Technical Reference(s): BWR Thermodynamics, Chapter 8, Thermal Hydraulics (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **Jet Pump Diagram**

Learning Objective: GFES Thermal Hydraulics, Chapter 8, Obj. 24 (As available)

Question Source:

Bank #	GFES Thermal Hydraulics QID B2791
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content:

55.41 **X**

55.43

Comments:

Copy of Bank Question:

TOPIC: 293008
KNOWLEDGE: K1.26 [2.9/3.1]
QID: B2791

Refer to the drawing of a core recirculation jet pump (see figure below).

During normal operation, the lowest pressure will exist at point _____; and the highest velocity will occur at point _____.

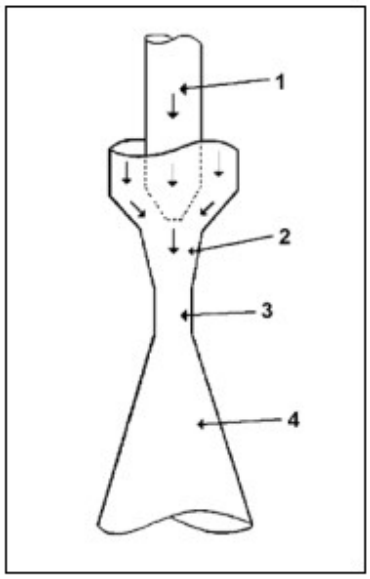
A. 3; 3

B. 3; 4

C. 4; 3

D. 4; 4

ANSWER: A.



Excerpt from BWR GFES, Thermodynamics, Thermal Hydraulics Question Justification:

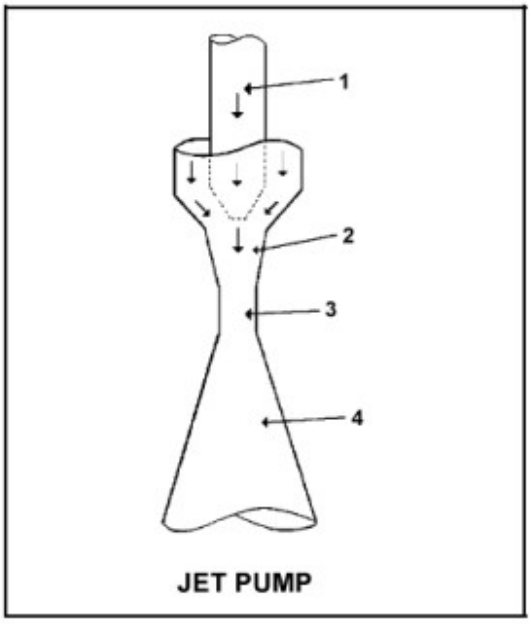
Refer to the drawing of a core recirculation jet pump (see figure below). The lowest pressure will exist at point _3_, and the highest velocity will occur at point _3_.

The options for lowest pressure are 3 or 4:

Since from point 3 to point 4 the area increases, an increase in area results in a decrease in velocity and an increase in pressure $(A \uparrow \bar{v} \downarrow P \uparrow)$.
Therefore of the options available point 4 has a higher pressure than point 3.

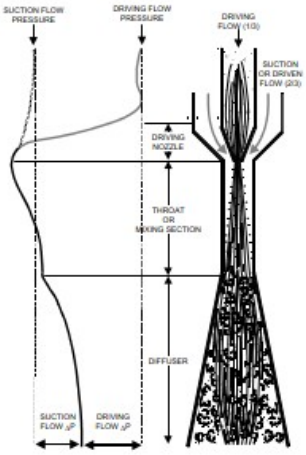
The options for highest velocity are 3 and 4:

Since from point 3 to point 4 the area increases, an increase in area results in a decrease in velocity and an increase in pressure $(A \uparrow \bar{v} \downarrow P \uparrow)$.
Therefore of the options available point 3 has a higher velocity than point 4.



JET PUMP

Excerpts from BWR GFES, Thermodynamics, Thermal Hydraulics Instructor Guide:

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Figure 8-18 / TP 8-42  	7. Accurate determination of individual jet pump and total core flow is important, because it affects core power distribution and is used in evaluation of Technical Specification Thermal Limits 8. As shown in Figure 8-18, the driving flow, developed by an external source, enters the inner nozzle at a high pressure and high velocity and is accelerated to a higher velocity jet flow at low pressure as a result of the constriction in area at the nozzle outlet $(A \downarrow \bar{v} \uparrow P \downarrow)$. a. The resulting low pressure at the outlet of jet nozzle draws the surrounding fluid (the driven flow) from the downcomer region into the low pressure throat or mixing section. b. The suction flow is accelerated through the converging suction inlet nozzle that surrounds the driving fluid nozzle. 9. The suction (driven) flow enters at a low pressure that is reduced further as the nozzle area decreases and the velocity increases. 10. The high velocity, low pressure driving flow mixes with and entrains the low velocity, low pressure suction (driven) flow in the mixer section. a. In the mixer section, a pressure rise occurs because the area of the driving flow expands in the constant area section and the velocity decreases as the stream expands and the momentum transfer caused by mixing. b. The rate of pressure rise decreases near the end of the mixer section because mixing essentially is completed.

© 2007 GENERAL PHYSICS CORPORATION

Excerpt from BWR GFES, Thermodynamics, Chapter 8, Thermal Hydraulics:

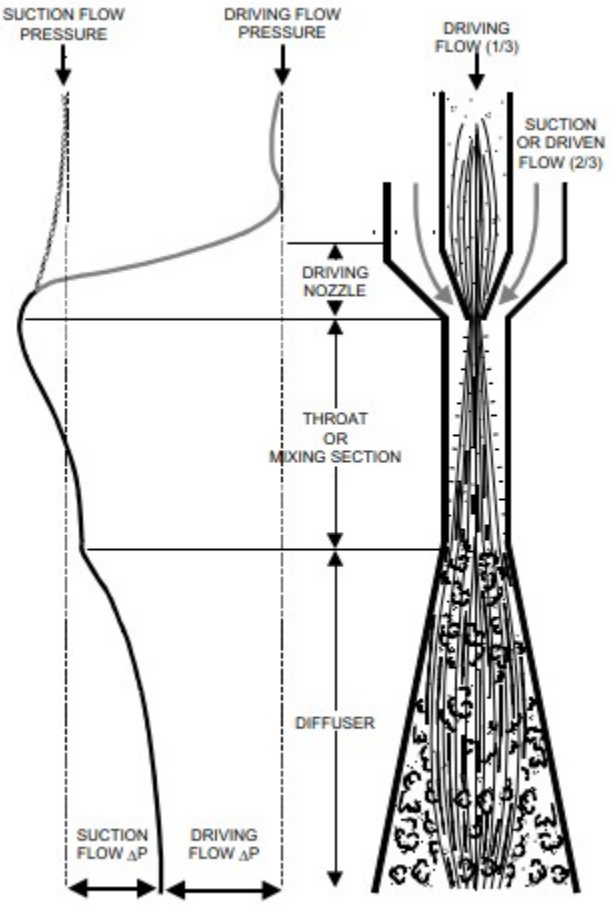


Figure 8-18 Jet Pump Operating Principles

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

293010 Thermodynamics – Brittle Fracture and Thermal Stress
K1.05 (10CFR 55.41.14)
State the effect of fast neutron radiation on reactor vessel materials

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	293010K1.05	
Importance Rating	2.8	------

Proposed Question: **# 75**

_____ is a **MAJOR** contributor to Reactor Vessel embrittlement?

A. High operating temperature

B. High-energy gamma radiation

C. High-energy fission fragments

D. High-energy neutron radiation

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that temperature is a concern with respect to brittle fracture, however it is low temperature that causes an increase in brittle behavior. High temperatures result in metal becoming more ductile.

B INCORRECT: Incorrect but plausible in that high energy gamma radiation will cause temperature to rise in the reactor vessel. Light particles, such as betas and gamma-ray photons, cause little permanent change in metals.

C INCORRECT: Incorrect but plausible in that high energy fission fragments can produce significant changes in the properties of metals, the high energy fission fragments do not interact with the reactor vessel, since they are maintained in the fuel rods.

D **CORRECT**: *(See attached)* Neutrons in excess on 1MeV cause damage to the Reactor Vessel and structural components via direct collisions with the atoms of that material. Metal atoms are knocked out of their lattice positions, resulting in vacancies in the crystalline material of the Reactor Vessel. The lattice structure changes result in increased strength but reduced ductility.

RO Level Justification: Tests the candidate's knowledge of the effects of fast neutron radiation on Reactor Vessel embrittlement and the resulting effects. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to the effect of high energy neutrons on Reactor Vessel embrittlement.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): BWR Thermodynamics Instructor Guide (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: GFES Thermodynamics, Chapter 10, Obj. 7 (As available)

Question Source:

Bank #	GFES Thermodynamics QID B1200
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Copy of Bank Question:

TOPIC: 293010
KNOWLEDGE: K1.05 [2.5/2.8]
QID: B1200 (P1898)

Which one of the following is the major contributor to embrittlement of a reactor vessel?

A. High-energy fission fragments

B. High operating temperature

C. High-energy gamma radiation

D. High-energy neutron radiation

ANSWER: D.

Excerpt from BWR GFES, Thermodynamics, Brittle Fracture Instructor Guide:

3.2 Ductility

Ductility is more commonly defined as the ability of a material to deform easily upon the application of a force, or as the ability of a material to undergo plastic deformation without failure.

Ductile materials show large deformation before fracture. The lack of ductility is often termed brittleness. Usually, if two materials have the same strength and hardness, the one that has the higher ductility is more desirable. The ductility of many metals can change if conditions are altered.

NOTE:
An increase in temperature will increase ductility. A decrease in temperature will cause a decrease in ductility and a change from ductile to ***brittle behavior.***

Irradiation will also increase brittleness

Cold-working also tends to make metals less ductile. Minor additions of impurities to metals, either deliberate or unintentional, can have a marked effect on the change from ductile to brittle behavior.

NOTE:
Ductility is desirable in the high temperature and high pressure applications in reactor plants because of the added stresses on the metals. High ductility in these applications helps prevent brittle fracture

3.3 Hardness

Hardness is the property of a material that enables it to resist plastic deformation, penetration, indentation, and scratching. Therefore, hardness is important from an engineering standpoint because resistance to wear by either friction or erosion by steam, oil, and water generally increases with hardness.

3.4 Toughness

Toughness describes how a material reacts under sudden impacts. It is defined as the ability of a material to absorb energy. ***It is also equivalent to the area under the stress/strain curve.***

INSTRUCTOR NOTES

Excerpt from BWR GFES, Thermodynamics, Brittle Fracture Student Guide:

EFFECTS OF RADIATION ON MATERIALS

The radiation from a nuclear reactor consists of alpha and beta particles, gamma rays, neutrons, fission fragments, and possibly protons. Although fission fragments are not strictly radiation, their behavior is similar to that of alpha particles and protons, both of which are charged particles. In addition, gaseous fission products, such as krypton and xenon, can cause problems in solid fuel materials with swelling caused by the accumulation of these gases.

The effect of radiation in crystalline solids depends on the structural type and the nature of the radiation. Light particles, such as betas and gamma-ray photons, cause little permanent change in metals.

The electrons in the conduction band of metals absorb the small amount of excitation or ionization energy caused by the particles. The electrons rapidly lose their excess energy, are taken up by the atoms of the metals, and manifest themselves as vibration energy, such as heat.

On the other hand, heavier particles, such as protons, alpha particles, neutrons, and fission fragments, produce significant changes in the properties of metals. As a result of elastic collisions, appreciable energy can be transferred from the particle of radiation to the atoms of the metal. If the energy is sufficiently high, these atoms can be displaced from their equilibrium position within the lattice, and permanent physical changes are noted. This effect is known as radiation damage.

If, as a result of an elastic collision between the relatively heavy nuclear particles and an atom, the energy transferred to the atom exceeds a certain minimum value (usually about 25 electron volts for metals), then the atom struck may be displaced from its equilibrium position in the lattice. If several atoms in close proximity are also displaced, some will merely be transferred from one equilibrium position to another in the lattice. However, if a displaced atom is unable to find another equilibrium position, its final position will be a non-equilibrium one. An atom of this type is said to occupy an interstitial site and is called an interstitial.

For each interstitial produced in the lattice, there is a corresponding vacant site in the lattice, called a vacancy, possibly at some distance away. The net result is a more or less permanent defect in the solid that, if sufficiently common, may be accompanied by a change in physical properties.

A fission fragment dissipates essentially all of its energy, and a fast neutron loses a large proportion of its energy within a very small volume. This energy, whether it is removed as atomic translation, vibration, or excitation of electrons, is rapidly converted into heat. The result is the formation of a small region of high temperature, known as a thermal spike. It has been estimated that thermal spikes include about 5,000 to 10,000 atoms and that a temperature of 1,300 to 2,200°F is attained for periods of about 10^{-10} seconds.

At these high temperatures, distortion of the lattice usually occurs. In the rapid cooling from the conduction of heat to the surroundings, a certain amount of lattice distortion is frozen. This is because there will be no time for all atoms to be relocated in equilibrium positions. Thus, thermal spikes permanently change the structures of the materials in which they occur.

When exposed to a neutron flux, many elements undergo a (n, γ) reaction. In the case of metals, no real damage is done since the new substance is an isotope of the irradiated metal. However, if the isotope undergoes radioactive decay, it will usually emit beta particles and, thus, become a new element. This causes an impurity within the metal. If enough of these impurities are found, the

© 2007 GENERAL PHYSICS CORPORATION

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295017 (APE 17) High Off-Site Radioactive Release Rate / 9 **AK1.06** (10CFR 55.41.10)	Tier #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Offsite Radioactive Release Rate: • Filtered vs. nonfiltered release	Group #	2	------
	K/A #	295017AK1.06	
	Importance Rating	3.6	------

Proposed Question: **# 23**

A Loss of Coolant Accident (LOCA) has occurred on Unit 2.

In accordance with 2-EOI-Appendix-13, Emergency Venting Primary Containment, the preferred Containment vent path is from the **(1)**, and it **(2)** permitted to exceed radiological release rate limits when the preferred vent path is used.

A. (1) Drywell
(2) is

B. (1) Drywell
(2) is **NOT**

C. (1) Suppression Chamber
(2) is

D. (1) Suppression Chamber
(2) is **NOT**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 2-EOI Appendix-13 states that if Suppression Chamber vent path is not available, then vent the Drywell. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell allows venting containment, just not exceeding Radiological Release Rate Limits.

C **CORRECT**: (*See attached*) In accordance with EOI-2, Primary Containment Control Bases related to 2-EOI Appendix-13, Emergency Venting Primary Containment, venting from the Suppression Chamber is generally preferred, to obtain the benefits of Suppression Pool scrubbing. For second part, in accordance with 2-EOI Appendix-13, Off-Gas Release Rate Limits will be exceeded and 2-EOI-2, PC/P-3 states that it is OK to exceed Radiological Limits.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

Form 4.2-1 **Written Examination Question Worksheet**

RO Level Justification: Tests the candidate's knowledge of the operational implications and/or cause and effect relationships of filtered vs. nonfiltered release as it applies to High Offsite Radioactive Release Rate. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to radioactive releases.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-EOI-2, Rev.18 (Attach if not previously provided)
2-EOI-APPENDIX-13, Rev.10
EOIPM-Section 0-V(G), Rev. 2
2-AOI-64-1, Rev. 27

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203 Obj. 12 (As available)

Question Source:

Bank #	BFN NRC 1501 #23
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2015

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

Q 23

Which ONE of the following completes the statement below in accordance with 2-EOI-Appendix-13, Emergency Venting Primary Containment?

The preferred containment vent path is from the __ (1) __.

It __ (2) __ permitted to exceed the radiological release rate limits when the preferred vent path is used.

A. (1) Drywell
(2) is

B. (1) Suppression Chamber
(2) is

C. (1) Drywell
(2) is Not

D. (1) Suppression Chamber
(2) is Not

Answer: B

Excerpt from EOIPM 0-V(G):

BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0002 Page 75 of 129

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: PC/P-10, PC/P-11 (cont'd)

Vent lineup

Detailed instructions for selecting and establishing the appropriate vent lineup are provided in the EOI Appendix-13. The following factors are considered in the venting procedures:

- *Pressure capability*. If the containment is pressurized, vent paths with low pressure capabilities should be used only as a last resort. Relieving high primary containment pressures through low pressure systems could damage ductwork, dampers, or filter housings, resulting in the release of radioactivity into the secondary containment. Even if a vent path can be established without damage, it may not be possible to close the vent dampers against high pressures.

- *Radioactivity release*. If the containment atmosphere may be contaminated, vent lineups should be selected so as to minimize the amount of radioactivity released while still achieving the objective of the venting requirement.

- Venting from the suppression chamber is generally preferred, to obtain the benefits of suppression pool scrubbing.

- Filtered vent paths are generally preferred over unfiltered paths. Pressure and temperature limitations, however, may restrict the availability of some filtered paths under accident conditions.

- Elevated release points are generally preferred over ground-level releases.

These considerations are less important if the primary containment radionuclide content is known to be low.

- *Vent flowrate*. The vent flowrate is determined by the differential pressure between the primary containment and atmosphere, the cross-sectional area of the vent line, and the resistance to flow through the line. Larger lines and lower flow resistances permit higher flows for a given differential pressure. Direct vent paths generally present lower resistance to flow than indirect paths and therefore permit higher flowrates.

If primary containment pressure is rapidly approaching the PCPL, it may be necessary to use higher capacity or multiple vent paths to preclude containment failure, even at the risk of increasing the radioactivity release rate. The dampers in larger vent lines, however, may not be operable at high pressures and usually cannot be throttled.

Excerpt from 2-EOI Appendix-13**:**

BFN Unit 2	Emergency Venting Primary Containment	2-EOI Appendix-13 Rev. 0010 Page 3 of 16

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Control Room
ATTACHMENTS	1. Tools and Equipment 2. Vent System Overview 3. Hardened Vent Flow Path 4. HCVS Battery Alignment 5. HCVS Nitrogen Bottle Alignment 6. HCVS Operation from the Remote Operating Station

[1] **NOTIFY** SHIFT MANAGER/SED of the following:

- Emergency Venting of Primary Containment is in progress. ☐
- Off-Gas Release Rate Limits will be exceeded. ☐

NOTES

1) HARDENED CONTAINMENT VENT VALVES 2-FCV-64-221 and 222 may be operated locally with handwheels (U2 RB el. 580, top of clean room, southwest corner).
2) If an alternate DC power source is needed for the HCVS valve solenoids, Att. 4 HCVS Battery Alignment may be performed.
3) If an alternate air supply is needed for the HCVS valves, Att. 5 HCVS Nitrogen Bottle Alignment may be performed.
4) If required, HCVS valves may be operated from the U3 DG building using Att. 6, HCVS Operation from the Remote Operating Station.

[2] **VENT** the Suppression Chamber as follows (Panel 2-9-3):

[2.1] **IF** EITHER of the following exists:

- Suppression Pool water level CANNOT be determined to be below 26 ft., **OR**
- Suppression Chamber CANNOT be vented, **THEN**

CONTINUE in this procedure at Step 1.0[3]. ☐

Supports Distractors A1, B1:

BFN Unit 2	Emergency Venting Primary Containment	2-EOI Appendix-13 Rev. 0010 Page 4 of 16

1.0 INSTRUCTIONS (continued)

[2.2] **PLACE** keylock switch 2-HS-64-222B, HARDENED CONTAINMENT VENT OUTBD PERMISSIVE, in PERM. ☐

[2.3] **CHECK** blue indicating light above 2-HS-64-222B, HARDENED CONTAINMENT VENT OUTBD PERMISSIVE, illuminated. ☐

[2.4] **OPEN** 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBD ISOL VLV. ☐

[2.5] **PLACE** keylock switch 2-HS-64-221B, HARDENED CONTAINMENT VENT INBD PERMISSIVE, in PERM. ☐

[2.6] **CHECK** blue indicating light above 2-HS-64-221B, HARDENED CONTAINMENT VENT INBD PERMISSIVE, illuminated. ☐

[2.7] **OPEN** 2-FCV-64-221, HARDENED CONTAINMENT VENT INBD ISOL VLV. ☐

[2.8] **CHECK** Drywell and Suppression Chamber Pressure lowering. ☐

[2.9] **MAINTAIN** Primary Containment Pressure below 55 psig using 2-FCV-64-222, HARDENED CONTAINMENT VENT OUTBD ISOL VLV, as directed by SRO. ☐

[3] **IF** Suppression Chamber vent path is NOT available, **THEN**

VENT the Drywell as follows:

[3.1] **NOTIFY** SHIFT MANAGER/SED that Secondary Containment integrity failure is possible. ☐

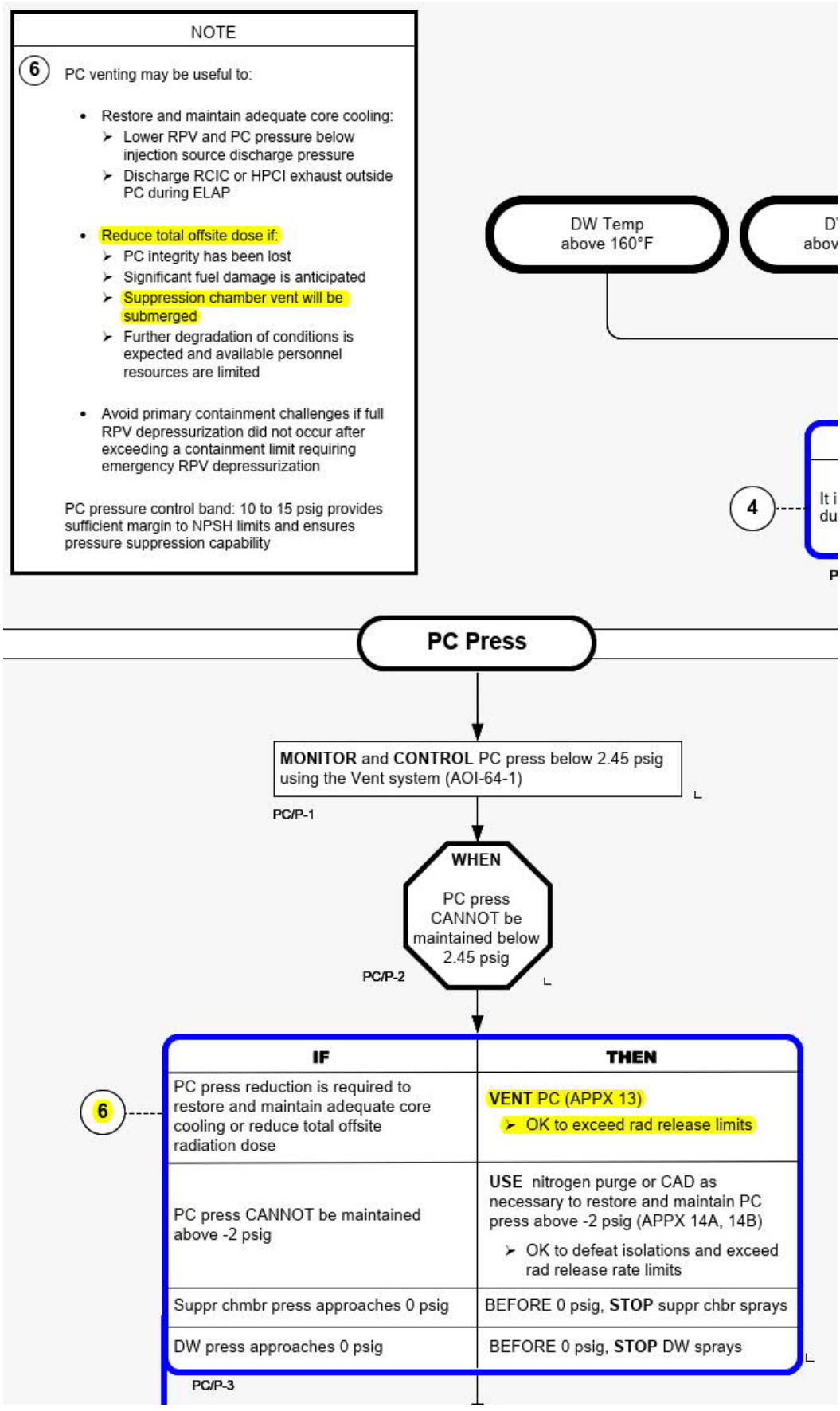
[3.2] **NOTIFY** RADCON that Reactor Building is being evacuated due to imminent failure of Primary Containment vent ducts. ☐

[3.3] **EVACUATE** ALL Reactor Buildings using P.A. System. ☐

[3.4] **START** ALL available SGTS trains. ☐

[3.5] **ENSURE CLOSED** 2-FCV-64-36, DW/SUPPR CHBR VENT TO SGT (Panel 2-9-3). ☐

Excerpt from 2-EOI-2:



Excerpt from 2-AOI-64-1: Supports Distractors B2, D2

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a High Drywell Pressure Condition, and/or High Drywell Temperature Condition, or Drywell Excessive Leakage.

2.0 SYMPTOMS

2.1 Common Symptoms for High Drywell Pressure, High Drywell Temperature and Drywell Excessive Leakage

- DRYWELL ATMOSPHERIC TEMP HIGH (2-XA-55-3B, Window 3)
- PRI CONTAINMENT N_2 PRESS HIGH (2-XA-55-3B, Window 10)
- DRYWELL TEMP HIGH (2-XA-55-3B, Window 16)
- DRYWELL PRESS APPROACHING SCRAM (2-XA-55-3B, Window 30)
- DRYWELL LEAK DETECTION RADIATION HIGH (2-XA-55-3D, Window 12)
- RBCCW PUMP SUCT HDR HIGH (2-XA-55-4C, WINDOW 5)
- DRYWELL FD SUMP PUMP EXCESSIVE OPRN (2-XA-55-4C, Window 11)
- DRYWELL EQPT DR SUMP PUMP EXCESSIVE OPRN (2-XA-5-4C, Window 18)
- DRYWELL PRESSURE ABNORMAL (2-XA-55-5B, Window 31)

2.2 Symptoms for High Drywell Pressure

- SUPPR CHAMBER WATER LEVEL ABNORMAL 2-LA-64-54A (2-XA-55-3B, Window 15)
- Drywell Radiation levels rising, as indicated on DW SUPPR CHBR RAD DIV I and II, 2-RR-90-272 and 273 (Panel 2-9-54 and 55) and Drywell Radiation Monitor, 2-RM-90-256 (Panel 2-9-2)
- Excessive Nitrogen usage, as indicated when performing 2-SI-4.7.A.2.a

Examination Outline Cross-reference:

295004 (APE 4) Partial or Complete D.C. Power / 6

AA2.01 (10CFR 55.43.5 – SRO Only)

Ability to determine and/or interpret the following as they apply to Partial Or Complete Loss of DC Power:

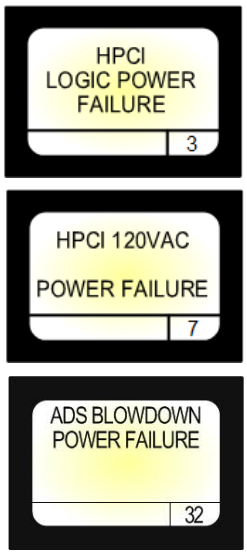
- Partial or complete Loss of DC Power

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295004AA2.01	
Importance Rating	------	4.1

Proposed Question: **# 76**

Unit 2 is operating at 100% RTP when the following conditions occur:

- HPCI LOGIC POWER FAILURE (2-9-3F, Window 3) alarms
- HPCI 120 VAC POWER FAILURE (2-9-3F, Window 7) alarms
- ADS BLOWDOWN POWER FAILURE (2-9-3C, Window 32) alarms



Given the conditions above, the 250V DC RMOV Board **(1)** has been lost.

After manually transferring the affected 250V DC RMOV Board, the Board is considered **(2)** in accordance with Tech Spec Bases.

A. (1) 2A
(2) OPERABLE

B. (1) 2A
(2) INOPERABLE

C. (1) 2B
(2) OPERABLE

D. (1) 2B
(2) INOPERABLE

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that many electrical boards can be placed on alternate and still considered OPERABLE such as 480V Shutdown Boards. The MODE is also a consideration for OPERABILITY with regards to which power source is supplying it. Some boards, such as 480V RMOV Boards can be considered OPERABLE on alternate while in MODES 4 or 5.

B **CORRECT**: (*See attached*) The candidate must recognize all three alarms that are given. Both the 2A and the 2B 250VDC RMOV Boards being lost will cause annunciation in Control Room. Windows 3 and 32 can alarm from either board loss. Window 7 will NOT annunciate from a loss of the 2B 250VDC RMOV Board, but will specifically for a loss of 2A 250VDC since HPCI is a 250VDC RMOV Division II ECCS component. For second part, in accordance with Tech Spec Bases 3.8.7, the OPERABILITY of the 250VDC RMOV Board 2A is based on being energized on its normal power supply in MODES 1, 2, and 3. Therefore, it is considered INOPERABLE in accordance with Tech Spec Bases 3.8.7.

C INCORRECT: First part is incorrect but plausible in that Window 3 alarming can be caused by a loss of either the 2A or 2B 250VDC RMOV Boards. Second part is incorrect but plausible (*See A*).

D INCORRECT**:** First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

SRO Level Justification: Tests the candidate's ability to determine the impacts of a loss or partial loss of 250V DC power as it relates to HPCI. Additionally, Technical Specification Bases knowledge is required from Memory without reference. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution based on Alarm Response Procedures and Technical Specification Bases in relation to HPCI and DC Electrical Distribution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to: (1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): U2 Tech Spec Bases 3.8.7, Rev. 76, 91 (Attach if not previously provided)
2-ARP-9-3C, Rev. 29
2-ARP-9-3F, Rev. 42
OPL171.037, Rev. 18U1
OPL171.042, Rev. 25U1

Proposed references to be provided to applicants during examination: **HPCI LOGIC POWER FAILURE (2-9-3F, Window 3)**
HPCI 120 VAC POWER FAILURE (2-9-3F, Window 7)
ADS BLOWDOWN POWER FAILURE (2-9-3C, Window 32)

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Learning Objective: OPL171.037 Obj. 8 (As available)
OPL171.042 Obj. 6

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.037-12 003 #1346	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content: 55.41
55.43 **X**

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1346. OPL171.037-12 003

With Unit 2 operating at 100% Reactor Power, a Normal Supply Breaker has tripped open AND suffered damage due to arcing. In addition to RHR AND Core Spray logic alarms, the following significant alarms result:

- HPCI LOGIC POWER FAILURE, (2-9-8A, Window 7)
- HPCI 120 VAC POWER FAILURE, (2-9-8A, Window 7)
- ADS BLOWDOWN POWER FAILURE, (2-9-3C, Window 32)

Which ONE of the following completes the statements?

The 250 VDC RMOV Board __**(1)**__ has been lost.

After manually transferring the 250 VDC RMOV Bd, the Board is considered __**(2)**__ in accordance with Tech Spec 3.8.7, "Distribution Systems – Operating."

A. **(1)** 2A
(2) inoperable

B. **(1)** 2B
(2) inoperable

C. **(1)** 2A
(2) operable

D. **(1)** 2B
(2) operable

Excerpts from 2-ARP-9-3F:

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0042 Page 6 of 41

HPCI LOGIC POWER FAILURE

3

(Page 1 of 1)

Sensor/Trip Point:

Relay 23A-K39 (Bus A) Loss of 250V DC

Relay 23A-K44 (Bus B) Control Power

Relay 23A-K44B (Bus B)

Sensor Location: Panel 9-32, Bus A; Aux Instr Rm, El 593' — Panel 9-39, Bus B; Aux Instr Rm, El 593'

Probable Cause:

A. Cleared fuse(s).

B. Loss of 250V DC power supply to panels.

Automatic Action:

A. Logic Bus A failure renders Channel A trip and automatic isolation logic inop. HPCI continues to function.

B. Logic Bus B failure renders Channel B trip, automatic initiation, and automatic isolation logics inop. If HPCI is in service, it can only be shut down by closing the 73-16 Steam Admission valve, however the aux oil pump will not start on coastdown. HPCI becomes inoperable.

Operator Action:

A. **DETERMINE** which logic bus has failed, **REFER TO** automatic action section.

B. **DISPATCH** personnel to check source of power failure:

1. Logic Bus A
   a. Fuses 2-FU2-073-23A-K36 (23A-F19) and 2-FU2-073-23A-K36 (23A-F20), Panel 9-32.
   b. Power supply 250V DC Rx Mov Bd 2B, Breaker 1B1.
2. Logic Bus B
   a. Fuses 2-FU2-073-0039A and 2-FU2-073-0039B, Panel 9-39.
   b. Power supply 250V DC RMOV Bd 2A, Breaker 11D1.

C. **REFER TO** Tech Spec 3.5.1, 3.5.2, 3.3.5.1, 3.3.6.1, and TRM 3.3.3.4.

References: 2-45E620-1; GE 730E928-2-3 and -4.

Technical Specifications 3.3.5.1, 3.3.6.1,3.5.1,3.5.2, TRM 3.3.3.4

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0042 Page 11 of 41

HPCI 120 VAC POWER FAILURE

7

(Page 1 of 1)

Sensor/Trip Point:

Relay 23A-K50

Loss of the 120 VAC from DIV II ECCS ATU inverter and Loss of power to the HPCI Flow IND Controller (2-FIC-73-33)

Sensor Location: Panel 9-19
Aux Instr Rm, El 593'

Probable Cause:
A. Blown fuses, Fuse 2-FU2-073-0033C, Panel 2-9-82 AA1 & AA2,
B. DIV II ECCS ATU inverter failure.
C. Loss of 250V DC power supply to DIV II ECCS ATU inverter (RMOV BD 2A compt 11A1).

Automatic Action:
A. HPCI controller loses power. HPCI becomes inoperable.
B. If HPCI is in service, the HPCI Turbine Stop Valve, 2-FCV-73-18, closes. HPCI controller loses power. HPCI becomes inoperable.
C. 2-PI-064-67B will lose power and become inop.

Operator Action:
A. **DISPATCH** personnel to CHECK the following:
- Fuses 2-FU2-073-0033C, Panel 2-9-82, AA1 & AA2.
- DIV II ECCS ATU inverter.
- DIV II ECCS ATU inverter breaker, RMOV BD 2A, compt 11A1.

B. 2-PI-064-67B will lose power and become inop.
REFER TO TECH SPEC 3.3.3.1, Table 3.3.3.1-1, TRM 3.3.5.
C. **REFER TO**: Tech Spec 3.5.1. Tech Spec 3.3.3.1, Table 3.3.3.1-1.

References:
2-45E620-1 GE 730E928-2 and -4
Technical Specifications 3.5.1, and 3.3.3.1
Technical Specifications Bases 3.3.3.1, TRM 3.3.5

Excerpt from 2-ARP-9-3C:

BFN Unit 2	Panel 9-3 2-XA-55-3C	2-ARP-9-3C Rev. 0029 Page 39 of 42

ADS BLOWDOWN POWER FAILURE

32

(Page 1 of 1)

Sensor/Trip Point:

Relay 2E-K40	Panel 9-33	De-energized
Relay 2E-K1A	Panel 9-30	De-energized
Relay 2E-K1B	Panel 9-33	De-energized
Relay 2E-K12	Panel 9-30	De-energized
Relay 2E-K32	Panel 2-25-32	De-energized
Relay 2E-K33	Panel 2-25-32	De-energized
Relay 2E-K37	Panel 2-25-32	De-energized
Relay 2E-K38	Panel 2-25-32	De-energized

Sensor Location:
Panel 2-25-32
Backup Control Center
El 621', R-13 Q-LINE

Panel 2-9-30 and 2-9-33
Aux Instrument Rm
El 593'

Probable Cause:
A. Cleared Fuse(s)
B. Loss of 250V DC power supply to panels.
C. Auto Xfer of Logic Bus B Power Supplies.

Automatic Action:
Main steam auto relief valves, PCV-1-22 and 30, auto transfer power supply from 250V DC Rx Mov Bd A to 250 DC Rx Mov Bd B (PCV-1-22) & Bd C (PCV-1-30) on loss of normal power supply.
Logic Bus "B" transferred to Alternate supply (250V Rmov Bd 2A) upon Loss of Normal Supply (250V RMOV Bd 2B) or fuse failure.

Operator Action:
A. **CHECK** power is available to PCV-1-22 and -30.
B. **IF** annunciator HPCI LOGIC POWER FAILURE, XA-55-3F Window 3, is in alarm, this is indicative of loss of power to 250V DC Rx Mov Bd 2A/2B. **DISPATCH** personnel to check 250V DC Rx Mov Bd 2A Breaker 11A2 and 250V DC Rx MOV Bd 2B Breaker 1B1.
C. **DISPATCH** personnel(s) to check:
   1. Logic Bus A
      a. 250V DC Rx Mov Bd 2B, Breaker 1F1.
      b. Fuses 2-FU2-001-2E-K3 in Panel 9-30.
   2. Logic Bus B
      a. 250V DC Rx Mov Bd 2A, Breaker 9A1.
      b. Fuses 2-FU1-001-2E/K22A and 2-FU1-001-2E/K22B on Panel 9-33.
      c. Fuses 2-FU2-001-2E-K13 in Panel 9-30.
      d. Fuses 2-FU2-1-2E-K11A and 2-FU2-1-2E-K11) on Panel 9-33 (GG Block).

D. **REFER TO** Tech Spec Section 3.5.1.
E. **REFER TO** TRM 3.3.3.4.

References:
2-45N620-2
2-45E712-1, -2 and -3
GE 730E929 -1, -2 and 3
Technical Specifications 3.5.1
TRM 3.3.3.4.

Excerpts from Unit 2 Tech Spec Bases 3.8.7:

BASES

LCO (continued)

When 480 V Shutdown Board 2B is aligned to the alternate supply 4.16 kV Shutdown Board C, a LOCA/LOOP with a failure of the Shutdown Board D Battery would disable the normal supply 4.16 kV Shutdown Board D, and would also prevent the 480 V Shutdown Board 2B from load shedding its 480 V loads which would overload the alternate supply Diesel Generator D. This would result in the loss of diesel generators C and D, associated 4.16 kV shutdown boards and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 2B is on its alternate supply.

The Unit 2 480 V RMOV boards 2A, 2B, 2D, and 2E have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 2 250 V DC RMOV boards 2A, 2B, and 2C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source, a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

(continued)

Supports Distractors A(2), C(2)

BASES

LCO (continued)

one or more of these boards become inoperable due to a failure not affecting the OPERABILITY of a board listed in Table B 3.8.7-1 (e.g., a breaker supplying a single MCC fails open), the individual loads on the board would be considered inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. However, if one or more of these boards is inoperable due to a failure also affecting the OPERABILITY of a board listed in Table B 3.8.7-1 (e.g., loss of a 4.16 kV shutdown board, which results in de-energization of all boards powered from the 4.16 kV shutdown board), then although the individual loads are still considered inoperable, the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV shutdown board).

The Unit 1 and 2 diesel auxiliary boards can be placed on their alternate feeder breakers and considered OPERABLE as long as the restrictions on the associated drawings are met.

If the 480 V Shutdown Board 2A is placed on the alternate supply 4.16 kV Shutdown Board C, a LOCA/LOOP with a failure of the Shutdown Board B Battery would disable the normal supply 4.16 kV Shutdown Board B, and would also prevent the 480 V Shutdown Board 2A from load shedding its 480 V loads which would overload the alternate supply Diesel Generator C. This would result in the loss of 4.16 kV Shutdown Boards B and C which would impact both divisions ECCS in Units 1 and 2. Therefore, the time limitations and restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 2A is on its alternate supply.

(continued)

Excerpt from OPL171.037 Lesson Plan:

OPL171.037, DC Systems, Rev: 18U1

7. Loss of 250VDC from each unit's 'A' or 'B' 250V RMOV Board results in loss of that divisions ECCS ATU Inverter and the redundant 24VDC converter power supply. This results in a complete loss of voltage to the divisions ECCS ATU logic panels (panel 9-81 or 9-82) and loss of voltage to the other components/systems served by the inverter.	Obj 8 AOI-57-11
8. The Station subsystem (Battery Boards 4, 5, and 6) supply non-1E loads (therefore not considered in the safety analysis.) Including a) Main Turbine EBOP b) Emergency Seal Oil Pump c) RFPT EBOP d) DC lighting e) TB Distribution Boards (control power for electrical equipment in the turbine building) f) Unit PFD inverter DC power supply (U1) g) Unit PFD MMG (U2/U3) DC supply (BB 2&3 Pnl 10) h) Unit PFD (9-9 cab 6) alternate DC supply i) Plant preferred (norm/alt) MG DC supply	Obj 1d Fig-6
II. 250V DC Control Power System (Shutdown Board Batteries) A. Description 1. This system consists of five 120-cell lead-calcium batteries and associated circuitry, switches, indicators and alarms.	Obj 1a
2. There is one battery and charger for A-D and 3EB 4KV Shutdown Boards, and one spare portable charger. NOTE: As part of RHR-LPCI modification (1977) which removed Loop Select Logic, the Unit 3EB Shutdown Board Battery was installed. This battery ensures that a single failure of a Unit 1, 2, or 3 250V battery will not make two U3, 4kV Shutdown Boards inoperable due to the loss of control voltage.	Obj 1b
3. Loads a) Normal control power for the 4KV Shutdown Bus 1 and 2, 4 KV SD Boards A-D and 3EB, and 480v Shutdown Boards. b) ATWS power distribution/ initiation logic (ARI and RPT)	Obj 1d
B. Location 1. The Units 1 and 2 Shutdown Board Batteries are located in the Reactor Building at Elev. 621, adjacent to the A and C 4KV Shutdown Board Rooms. (Primary access is through the Shutdown Board Rooms from Level 3C, Control Bay.) Batteries A and B are at the U1 end (A Shutdown Board Room), and Batteries C and D are at the U2 end (C Shutdown Board Room). The chargers and distribution panels are in adjoining spaces.	Obj 1a Obj 1b
2. 3EB Shutdown Board Battery is located on the south end of U3 Diesel Generator Building at Elev. 583. The charger and	Obj 1a,b,c

Excerpt from OPL171.042 Lesson Plan:

a) Loss of Control Air (1) HPCI Steam line condensate drain valves (FCV-73-6A-6B) (2) HPCI hotwell pump CRW discharge valve (FCV 73-17A and -17B) fails closed.	Obj. ILT 5 Obj. LOR 5.b
b) Loss of Electrical Power (1) All motor-operated isolation valves remain in the last position upon failure of valve power (2) Relay logic power failure (a) If the Logic Bus A (Div I) fails, A Channel (Div I) trips from Isolation logic and A Channel (Div I) isolation logic will be inoperative. (b) If Logic Bus B (Div II) fails, the automatic initiation and trip solenoid systems will not operate. Also, Channel B (Div II) isolation logic will be lost. (73-19 fails closed on Div. II ECCS ATU inverter failure) (3) FCV 73-5 (Units 2 & 3-Solenoid operated) fails closed	Figure 7, 8, 9 Obj. ILT 6 Obj. LOR 6 Obj. NLO 11 Obj. NLO 12 Obj. LOR 6 Obj. ILT 6
c) Power Supplies All motor-operated and solenoid operated valves are 250VDC from 250V Rx MOV Board 1A/ 2A/3A except for 73-2, 73-81, 73-64 which are 480VAC. (73-2 from 480V Rx MOV Board 1A (2A,3A); others from 1B (2B, 3B).	Obj. ILT 5
d) Air-Operated Valves (1) Control air actuated (2) 250VDC solenoid controlled	
e) **The Division II ECCS ATU inverter (powered from 250V RMOV Board A) supplies power to the flow controller and the following instruments on Panel 9-3: (1) PI-73-31A, Pump Discharge (2) PI-73-28A, Booster Pump Suction (3) PI-73-4A, Steam Supply (4) PI-73-21A, Turbine Exhaust (5) FIC-73-33, Flow Ind Controller (6) If Division II ECCS inverter output is lost, HPCI 120VAC FAILURE (XA-55-3F-7) would alarm and flow controller fails downscale, Control valve closes if open. If accompanied by HPCI LOGIC POWER FAILURE (XA-55-3F-3), would indicate a loss of power from 250V RMOV Board A. (7) If Division I ECCS inverter and converter output is lost, HPCI will not initiate from DIV I logic. LIS-3-58A and B will be lost.	Obj. ILT 6 Obj. ILT 6 Obj. LOR 6 Note: on U-1 may indicate a loss of logic bus B **SOER 83-3 (Recommendation 11)
f) Relay Logic Bus A (Div I) is powered from 250V RMOV Board 2B. It supplies power to half of the low level circuit.	

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295006 (APE 6) SCRAM / 1

G2.2.38 (10CFR 55.43.1 - SRO Only)

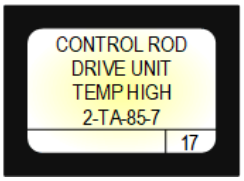
Knowledge of conditions and limitations in the facility license

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295006G2.2.38	
Importance Rating	------	4.5

Proposed Question: **# 77**

In accordance with Tech Spec Bases, the OPERABILITY of an individual Control Rod is based on SCRAM insertion times and the ability to determine **(1)**.

SCRAM insertion times can be adversely affected as indicated by a valid CONTROL ROD DRIVE UNIT TEMPERATURE HIGH (2-9-5A, Window 17) alarming at a setpoint of **(2)**.



A. (1) the Control Rod position **ONLY**
(2) 350 °F

B. (1) the Control Rod position **ONLY**
(2) 281 °F

C. (1) the Control Rod position **AND** Control Rod coupling integrity
(2) 350 °F

D. (1) the Control Rod position **AND** Control Rod coupling integrity
(2) 281 °F

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that if the Control Rod position is one of the 3 individual OPERABILITY requirements and can be verified as full in. If full in, then the coupling integrity would not matter for the Control Rod to perform its intended function. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in that 281°F is the Primary Containment design temperature. Candidates are required to recall, from memory, voluminous numbers of setpoints and values that are easily confused, thus the value is plausible.

C **CORRECT:** (*See attached*) In accordance with Tech Spec Bases 3.1.3, Control Rod OPERABILITY is based primarily upon SCRAM insertion times, Control Rod coupling integrity, and the ability to determine the Control Rod Position. For second part, in accordance with CONTROL ROD DRIVE UNIT TEMPERATURE HIGH (2-9-5A, Window 17), the annunciator alarms at 350 °F.

D INCORRECT**:** First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

SRO Level Justification: Tests the candidate's ability to recall Tech Spec Bases requirements from memory regarding OPERABILITY requirements for Control Rods and alarms associated with those parameters. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as Memory due to the requirement to strictly recall both Tech Spec Bases.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) Information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s):	Tech Spec Bases 3.1.3, Rev. 0	(Attach if not previously provided)
	Tech Spec Bases 3.1.4, Rev. 9	
	2-ARP-9-5A, Rev. 63	
	2-OI-85, Rev. 156	

Proposed references to be provided to applicants during examination: **CONTROL ROD DRIVE UNIT TEMPERATURE HIGH (2-9-5A, Window 17)**

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.005-32 015 #304	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

304. OPL171.005-32 015

In accordance with Technical Specification Bases 3.1.3, the OPERABILITY of an individual control rod is based on ___(1)___.

Slow scramming control rods ___(2)___ be conservatively declared inoperable and not accounted for as "slow" control rods.

A. 1) scram insertion times and ability to determine the control rod position ONLY
   2) can

B✓ 1) scram insertion times, control rod coupling integrity, and ability to determine the control rod position
   2) can

C. 1) scram insertion times and ability to determine the control rod position ONLY
   2) can NOT

D. 1) scram insertion times, control rod coupling integrity, and ability to determine the control rod position
   2) can NOT

Excerpt from Tech Spec Bases 3.1.3:

BASES

APPLICABLE SAFETY ANALYSES (continued)	The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System. Control rod OPERABILITY satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).
LCO	The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.
APPLICABILITY	In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

(continued)

Excerpt from Tech Spec Bases 3.1.4:

BASES (continued)

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6).

To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 185 x 7% ≈ 13) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations.

Table 3.1.4-1 is modified by two Notes, which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

Scram times can be adversely affected by high control rod drive temperatures. Temperatures over 350°F may result in a measurable delay in scram time response times for an otherwise normally performing CRD due to the potential for flashing of the hot water in the drive when the scram valves are opened. As a conservative measure, CRDs which have a

(continued)

BASES

LCO (continued)

temperature of greater than 350°F will either be classified as "slow" rods or an engineering evaluation can be performed.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods can be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analysis. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

Excerpts from 2-ARP-9-5A: Supports either declaring Control Rod as SLOW in accordance with Tech Spec 3.1.4-1 - Note 1: if alarm is valid or not if alarm is NOT valid

CONTROL ROD DRIVE UNIT TEMP HIGH
2-TA-85-7

17

(Page 1 of 2)

Sensor/Trip Point:

TE-85-7 (1 thru 185) 350°F — Alarm comes from recorders, 2-TR-85-7A & 2-TR-85-7A1 2-TR-85-7B & 2-TR-85-7B1

(Alarm Reflashes)

Sensor Location: Located on each control rod drive.

Probable Cause:
A. Insufficient cooling water flow.
B. Malfunction of sensor.
C. Leaking scram discharge valve.
D. Plugged CRD cooling water orifice.

Automatic Action: None

Operator Action:
A. **CHECK** high temp of CRD on recorder 2-TR-85-7A, 2-TR-85-7A1, 2-TR-85-7B, & 2-TR-85-7B1 (Panel 2-9-47) or on ICS.
B. **IF** alarm is valid, **THEN**
PERFORM the following as directed by the Unit SRO:
- **CHECK** cooling water pressure and flow normal on Panel 2-9-5.
- **DISPATCH** personnel to check for HCU scram discharge valve leaking as indicated by elevated discharge piping temperatures for associated CRD.
- **PERFORM** 2-TI-393 for control rods with high temperatures or failed thermocouples.
- **REFER TO** 0-OI-55, 2-OI-85, 2-AOI-85-3.
- **FLUSH** CRD to unblock restricted cooling water flow. **REFER TO** 2-OI-85.
- **DECLARE** the control rod, which is in ALARM, "SLOW" as directed by 2-TI-393 per Tech Spec Table 3.1.4-1 Note 1.
- **RAISE** CRD Flow, as directed by Unit SRO, if required to keep the drives cool per "CRD Pump Operation At Elevated Flow" section of 2-OI-85.

Continued on Next Page

BFN Unit 2	Panel 9-5 2-XA-55-5A	2-ARP-9-5A Rev. 0063 Page 24 of 48

CONTROL ROD DRIVE UNIT TEMP HIGH 2-TA-85-7, Window 17
(Page 2 of 2)

Operator Action: (Continued)

C. **IF** alarm is INVALID, **THEN**
PERFORM the following as directed by the Unit SRO:
- **REFER TO** 0-OI-55.
- **INITIATE** a Condition Report (CR) to determine cause of invalid alarm.

References:

2-45E620-6	2-730E321-10	2-47E610-85-1
0-OI-55	2-OI-85	2-AOI-85-3

Tech Spec Table 3.1.4-1 Note 1 2-TI-393

Excerpts from TS 3.1.4:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

---NOTES---

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds)
	REACTOR STEAM DOME PRESSURE ≥ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

Excerpt from 2-OI-85:

3.3 Tech Specs (continued)

E. During reactor power operation, no more than one control rod in any 5 x 5 array may be inoperable. Tech Specs should be reviewed for actions required when a control rod becomes inoperable.

F. [NER/C] Inoperable control rods should always be inserted (if not stuck) prior to disarming.

Control Rods declared "slow" but NOT inoperable need not be disarmed, but should be included in the control rod exercise test. (2-SR-3.1.3.3) [INPO SER 9-90]

G. If a Control Rod was declared "Inoperable" or "Slow" solely due to high temperatures, the Control Rod may be declared "Operable" or no longer "slow" when the temperatures have lowered back below the High Temp Setpoint. Alarms may have to be re-enabled if the alarm was previously disabled and ICS reset. (Refer to OI-55 and 2-TI-393. Reactor Engineering can reset the ICS Screen)

H. A stuck control rod should be hydraulically isolated. Stuck rods could be severely damaged from ΔP forces on a scram signal if not hydraulically isolated. If a Control Rod is at "00" and has been evaluated to have a stuck Ball Check Valve and not mechanically bound, it may be individually scrammed to unseat the Ball Check Valve. (T.S. 3.1.3 Condition A, Required Action A.2)

I. A CRD HCU Accumulator is Inoperable when either of the following conditions are met:

1. With accumulator pressure less than 940 psig.
2. When isolated and unattended.
3. When isolated for maintenance other than re-charging (Sect. 8.4) and Draining (Sect. 8.5); Reference Tech Specs 3.1.3, 3.1.4 and 3.1.5 (Mode 1 or 2) and 3.9.5, 3.10.3, 3.10.4 and 3.10.5 (Mode 3, 4 or 5).

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295016 (APE 16) Control Room Abandonment / 7

G2.4.45 (10CFR 55.43.5 – SRO Only)

Ability to prioritize and interpret the significance of each annunciator or alarm

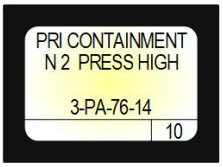
Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295016G2.4.45	
Importance Rating	------	4.3

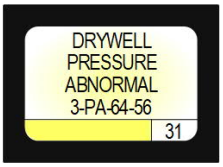
Proposed Question: **# 78**

Unit 3 is operating at 100% RTP when the following conditions occur:

At 0220:

- A **TERRORIST** attack in 1C Hallway requires Units 1, 2, and 3 Main Control Rooms to be evacuated
- 3-AOI-100-2, Control Room Abandonment, is ordered
- PRIMARY CONTAINMENT N2 PRESSURE HIGH (3-9-3B, Window 10) alarms
- DRYWELL PRESSURE ABNORMAL (3-9-5B, Window 31) alarms





At 0245:

- Operators are at 3-Panel-25-32, BACKUP CONTROL PANEL
- Reactor Water Level is (-) 100 inches and lowering
- **NO** injection sources are available

Based **ONLY** on the given **ALARMS** above, the Unit 3 EDGs **(1)** automatically start.

Given the conditions above and in accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification to report is a/an **(2)** EMERGENCY.

Note: SED judgement shall **NOT** be used as a basis for Classification

[REFERENCE PROVIDED]

A. (1) will
(2) SITE AREA

B. (1) will
(2) GENERAL

C. (1) will NOT
(2) SITE AREA

D. (1) will NOT
(2) GENERAL

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible since multiple alarms are received as Drywell Pressure rises to the point of reaching the automatic start setpoint of the EDGs. With only receiving the given alarms, however the EDG auto-start would not be received. Second part is incorrect but plausible in that candidates may assume that the security threat has to directly cause the inability to control a parameter. If the security threat prevents mitigating actions to control the parameter, then the intent is met for the upgrade declaration. The candidate could plausibly select SITE AREA since control of the parameter cannot be established within 20 minutes (HS1 OR HS6).

B INCORRECT: First part is incorrect but plausible *(See A)*. Second part is correct. (*See D*).

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible (*See A*).

D **CORRECT:** (*See attached*). From the given alarms, the highest Drywell Pressure would be 1.96 psig in accordance with DRYWELL PRESSURE ABNORMAL (3-9-5B, Window 31). The EDGs would automatically start at 2.45 psig in accordance with the not given DRYWELL PRESSURE HIGH HALF SCRAM (3-9-4A, Window 8). For second part, in accordance with EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3, a General Emergency (HG1) is declared under the SECURITY leg due to the confirmed security threat in the Control Bay thereby impacting the mitigating actions that could be performed by Operators.

SRO Level Justification: Test the candidate's ability to determine the implications of rising Drywell Pressure and the associated alarms in order to prioritize Emergency Declarations. Additionally, the candidate must integrate those impacts with an ongoing Security event. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome as it relates to Emergency Declarations.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(2) diagnosis that leads to selection of the procedures that should be used to respond to the evolution and
(3) the progression of an event.

Technical Reference(s): EPIP-1, Rev. 63 (Attach if not previously provided)

3-ARP-9-3B, Rev. 25

3-ARP-9-4A, Rev. 51

3-ARP-9-5B, Rev. 33

3-ARP-9-23A, Rev. 30

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3**
PRIMARY CONTAINMENT N2 PRESSURE HIGH (3-9-3B, Window 10)
DRYWELL PRESSURE ABNORMAL (3-9-5B, Window 31)

Learning Objective: OPL171.075 Obj. 2 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 2204 NRC #82	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2022	

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

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 82**

Unit 3 is operating at 100% RTP when the following conditions occur:

At 1000

- A fire is reported in the Unit 3 Main Control Room (MCR)

At 1010

- The Unit 3 MCR is abandoned

At 1025

- RCIC is controlling Reactor Water Level and MSRVs are controlling Reactor Pressure from 3-Panel-25-32, BACKUP CONTROL PANEL

Given the conditions above and in accordance with 3-AOI-100-2, Control Room Abandonment, the responsibility of assessing MCR habitability is performed by the ___**(1)**___.

In accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification to report is a/an ___**(2)**___.

[REFERENCE PROVIDED]

A. (1) Shift Manager
(2) ALERT

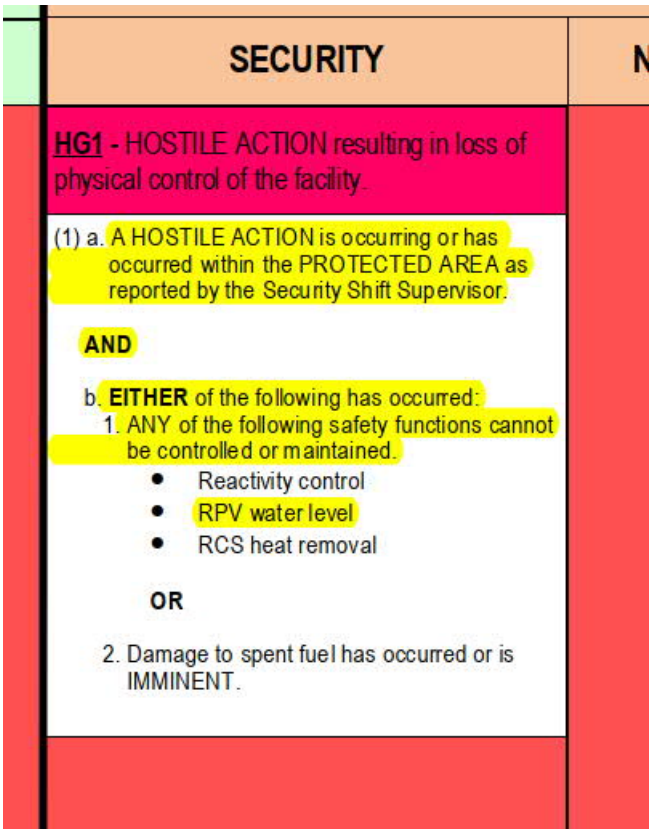
B. (1) Shift Manager
(2) SITE AREA EMERGENCY

C. (1) Incident Commander
(2) ALERT

D. (1) Incident Commander
(2) SITE AREA EMERGENCY

Proposed Answer: **A**

Excerpts from EPIP-1, Attachment -1, Hot Initiating Conditions – MODE 1-2-3:



SECURITY

HG1 - HOSTILE ACTION resulting in loss of physical control of the facility.

(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

AND

b. **EITHER** of the following has occurred:

1. ANY of the following safety functions cannot be controlled or maintained.
   - Reactivity control
   - RPV water level
   - RCS heat removal

OR

2. Damage to spent fuel has occurred or is IMMINENT.

HS6 - Inability to control a key safety function from outside the Control Room.

The SED should declare the Site Area Emergency promptly upon determining that 20 minutes has been exceeded, or will likely be exceeded.

(1) a. An event has resulted in plant control being transferred from the Control Room to the Backup Control Panel.

AND

b. Control of **ANY** of the following key safety functions is not reestablished within 20 minutes.

- Reactivity control
- RPV water level
- RCS heat removal

Excerpt from 3-ARP-9-3B:

BFN Unit 3	Panel 9-3 3-XA-55-3B	3-ARP-9-3B Rev. 0025 Page 13 of 38

PRI CONTAINMENT
N2 PRESS HIGH

3-PA-76-14

10

(Page 1 of 1)

Sensor/Trip Point:

PT-76-14 1.5 psig (alarm from recorder).

Sensor Location: Panel 25-6
Rx Bldg, El 593, R-19 Q-LINE

Probable Cause:
A. Drywell Cooler(s) failure.
B. Steam or water leak inside Drywell.
C. Loss of RBCCW to Drywell Coolers.
D. Low pressure front moving through area.

Automatic Action: None

Operator Action:
A. **CHECK** containment pressure using multiple indications.
B. **CHECK** containment temperature.
C. **REFER TO** 3-OI-64 Venting the Drywell with Standby Gas Treatment Fan.

References: 3-45N620-3 47W600-57 3-47E610-76-1 GE 730E933-1

Excerpt from 3-ARP-9-5B:

BFN Unit 3	Panel 9-5 3-XA-55-5B	3-ARP-9-5B Rev. 0033 Page 37 of 44

DRYWELL
PRESSURE
ABNORMAL
3-PA-64-56

31

(Page 1 of 1)

Sensor/Trip Point:

3-PS-64-56E	1.65 psig rising
3-PS-64-56F	0.1 psig lowering

Sensor Location: Panel 25-5B
Elevation 593

Probable Cause:
A. Drywell DP air compressor failure.
B. Loss of RBCCW.
C. Breach of Primary Containment.
   1. Drywell vent valves open or leaking.
   2. Drywell vacuum breaker open or leaking.

D. LOCA.
E. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** alarm using multiple indications.

B. **IF** RBCCW has been lost, **THEN**
REFER TO 3-AOI-70-1.

C. **REFER TO** 3-AOI-64-1.

References:

3-45E620-6	3-47E610-64-1	3-730E915-17
3-AOI-70-1	3-AOI-64-1	

Excerpt from 3-ARP-9-4A: Supports Distractors A1, B1

BFN Unit 3	Panel 9-4 3-X-55-4A	3-ARP-9-4A Rev. 0051 Page 11 of 47

DRYWELL PRESSURE HIGH HALF SCRAM

8

(Page 1 of 1)

Sensor/Trip Point:

3-PIS-064-0056A	2.45 psi
3-PIS-064-0056B	positive
3-PIS-064-0056C	pressure in
3-PIS-064-0056D	the drywell.

Sensor Location: 3-PNLA-009-0083, 0084, 0085, 0086 U3 Aux Inst Room

Probable Cause:
A. ≥2.45 psig in the drywell.
B. Sensor malfunction.
C. SI or SR in progress.

Automatic Action:
A. Half scram if one sensor actuates.
B. Reactor scram and group 2, 6 and 8 PCIS if one sensor per channel actuates.

Operator Action:
A. **CONFIRM** alarm by multiple indications.
B. **IF** drywell pressure is greater then or equal to 2.45 psig **AND** reactor has NOT scrammed, **THEN**
PERFORM the following:
1. **SCRAM** the reactor.
2. **ENTER** 3-EOI-1 & 3-EOI-2 FLOWCHARTS.

C. **DISPATCH** personnel to the pressure switches to check for abnormal condition.
D. **IF** alarm is NOT valid, **OR** initiating condition is corrected, **THEN**
PERFORM the following:
1. **RESET** Half Scram with SRO permission,
2. **REFER TO** 3-OI-99.

E. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function.
REFER TO NPG-SPP-18.3.5.

References: 3-45N620-5 GE 730E915 FSAR Section 7.2.3.5 and 13.6.2

Excerpt from 3-ARP-9-23A: Supports Distractors A1, B1

BFN Unit 3	Panel 3-9-23A, 3-XA-55-23A	3-ARP-9-23A Rev. 0030 Page 4 of 41

DIESEL GEN 3A RUNNING

1

(Page 1 of 1)

Sensor/Trip Point:

Aux. Relay (DRRA-3) 3-RLY-082-DRRA-3	Relay (ESR 1 & 2) 3-RLY-082-A/ESR1 3-RLY-082-A/ESR2	DG speed ≥ 125 rpm

Sensor Location:

3-RLY-082-DRRA-3
(Relay DRRA-3)
Shutdown Board 3EA
Diesel Generator Bldg.

3-RLY-082-A/ESR1 and ESR2
(Relay ESR 1 & 2)
Electrical Control Cab.
Elevation 565
Diesel Generator Bldg.

Probable Cause:

A. Undervoltage (0 volts after 1.5 sec.) or degraded voltage (< 3920V) on 4160V Shutdown Bd. "EA".
B. Pre Accident Signal Hi Drywell Press at 2.45 psig or Low-Low-Low Rx Water Level at -122 inches (Level 1).
C. Manual start.
D. Sensor malfunction.

Automatic Action:

A. Diesel generator exhaust fan "A or B" starts.
B. RHR service water pumps auto start initiated.

Operator Action:

A. **OBSERVE** alarms and other indications to determine why diesel(s) started:
   1. **IF** diesel start is due to accident signal, **THEN** **PERFORM** the following:
      a. **OBSERVE** proper start of available diesels.
      b. **ENSURE** diesels reach and stay at rated frequency and voltage.
   2. **IF** diesel start is due to loss of voltage, **THEN** **PERFORM** the following:
      a. **ENSURE** proper start of diesel (rated frequency and voltage).
      b. Diesel connects to assigned 4160V Shutdown Bd.

B. As soon as possible after diesel starts, **DISPATCH** personnel to monitor diesel operation locally.
C. **REFER TO** TS 3.8.1 and 3.8.2.

References:

3-45E766-29 3-45E768-1
Technical Specifications 3.8.1 and 3.8.2

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295019 (APE 19) Partial or Complete Loss of Instrument Air / 8

AA2.01 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Instrument Air:

- Instrument air pressure

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295019AA2.01	
Importance Rating	------	3.9

Proposed Question: **# 79**

Following a Reactor SCRAM on Unit 2, the following conditions exist:



- DRYWELL CONTROL AIR PRESSURE LOW, (2-9-3E, Window 35), is in alarm
- Drywell Control Air Pressure is 69 psig
- Reactor Water Level lowered to (-) 10 inches, currently at (+) 30 inches
- **NO** Operator actions have been taken

Given the conditions above, the **INBOARD** MSIVs are currently ___**(1)**___ and the NUSO will direct Pressure to be restored in accordance with ___**(2)**___.

Note: 2-AOI-32-2, Loss of Control Air

2-EOI Appendix-8G, Crosstie CAD to Drywell Control Air

A. (1) OPEN
(2) 2-EOI Appendix-8G

B. (1) OPEN
(2) 2-AOI-32-2

C. (1) CLOSED
(2) 2-EOI Appendix-8G

D. (1) CLOSED
(2) 2-AOI-32-2

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (*See attached*) In accordance with 2-AOI-32-2, Loss of Control Air, MSIVs are expected to close at approximately 45 psig. DRYWELL CONTROL AIR PRESSURE LOW (2-9-3E, Window 35) setpoint is 92 psig lowering, and the given Drywell Control Air Pressure is 69 psig. Thus, a degrading trend in DWCA Pressure is indicated, but pressure remains greater than 45 psig, so MSIVs remain open. For second part, Reactor Water Level was reported to have lowered to (-) 10 inches requiring entry into 2-EOI-1 (SCRAM) at less than (+) 2 inches. Step RC/P-2 provides direction to crosstie CAD or MSRV carts to DWCA in accordance with 2-EOI- Appendix-8G, Crosstie CAD to Drywell Control Air.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible since 2-AOI-32-2, Loss of Control Air, may be used to restore air to the Outboard MSIVs by aligning Drywell Control Air. It does not contain actions to restore Drywell Control Air.

C INCORRECT: First part is incorrect but plausible in that in accordance with 2-AOI-32-2, Loss Of Control Air, MSIVs are expected close at approximately 45 psig, however, as provided, pressure is 69 psig. (Direction to SCRAM would occur prior to 55 psig to mitigate the expected plant transient occurring due to the MSIV impending closure. With no Operator action, the MSIVs would currently be open.) Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible (See C). Second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's ability to determine impact from a loss of Drywell Control Air event and the ability to select the correct procedure to mitigate the transient. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents; (2) Diagnosis that leads to selection of the procedures that should be used to respond to the evolution; and (4) The assessment of the integrated plant response to emergency or abnormal situations crossing several plant systems or safety functions, or both.

Technical Reference(s): 2-ARP-9E, Rev. 34 (Attach if not previously provided)

2-ARP-9-3D, Rev. 34

EOIPM Section 0-V(C), Rev. 3

2-EOI Appendix-8G, Rev. 7

2-AOI-32-2, Rev. 39

OPL171.054, Rev. 18

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **DRYWELL CONTROL AIR PRESSURE LOW, (2-9-3E, Window 35)**

Learning Objective: OPL171.054, Obj. 5, 6 (As available)

Question Source:

Bank #	BFN NRC 19-09 #90	
Modified Bank #		(Note changes or attach parent)
New		

Question History:

Last NRC Exam	1909

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41	
55.43	**X**

Comments:

Copy of Bank Question:

Proposed Question: **# 90**

Following a Reactor SCRAM on Unit 2, the following plant conditions exist:



- DRYWELL CONTROL AIR PRESSURE LOW, (2-9-3E, Window 35), is in alarm
- Reactor Water Level lowered to (-) 10 inches, currently at (+) 30 inches
- Assume **NO** Operator action has been taken

Given the conditions above, which **ONE** of the following completes the statements below?

Currently, the **INBOARD** MSIVs are ___**(1)**___.

The Unit Supervisor will direct Pressure to be restored in accordance with ___**(2)**___ .

Note: 2-AOI-32-2, Loss of Control Air
2-EOI Appendix-8G, Crosstie CAD to Drywell Control Air

A. **(1)** OPEN
(2) 2-EOI Appendix-8G

B. **(1)** OPEN
(2) 2-AOI-32-2

C. **(1)** CLOSED
(2) 2-EOI Appendix-8G

D. **(1)** CLOSED
(2) 2-AOI-32-2

Proposed Answer: **A**

Excerpt from 2-ARP-9-3E:

BFN Unit 2	Panel 9-3 2-XA-55-3E	2-ARP-9-3E Rev. 0034 Page 41 of 41

DRYWELL CONTROL AIR PRESS LOW
2-PA-32-70
35

(Page 1 of 1)

Sensor/Trip Point:
PS-32-70 92 psig

Sensor Location: Panel 25-217
Rx Bldg, El 593', R-11 U-LINE

Probable Cause:
A. Air line break.
B. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **REFER TO** 2-AOI-32A-1.
B. **IF** it is desired to align CAD to the Drywell Control Air System **THEN**:
1. **OPEN** CAD SYSTEM A(B) N2 SHUTOFF VALVE, 0-HS-84-5A(16A), on Unit 1, Panel 9-54(55).
2. **CHECK** N2 VAPORIZER A(B) OUTLET PRESS, 0-PI-84-6(17), reading approximately 100 psig.
3. **PLACE** keylock switch CAD A(B) CROSS TIE TO DW CONTROL AIR, 2-HS-84-48(49), in **OPEN** on Panel 2-9-54(55).
4. **CHECK OPEN** CAD A(B) CROSS TIE TO DW CONTROL AIR, 2-FSV-84-48(49).
5. **REFER TO** 2-OI-84 for CAD cross-tie operability determination and CAD tank level requirements.
6. **IF** alarm cannot be cleared and sustained low Drywell Control Air pressure exists, **THEN**

   NOTIFY Engineering to assess the impact on the MSIV Primary Containment isolation function.
7. **WHEN** Drywell Control Air is available **THEN**:
   REFER TO 2-OI-84 to return CAD system to normal alignment.

References: 2-45E620-1 2-47E610-32-2 2-45E751-1 and 3
2-45E779-5
Technical Specifications 5.4/5.5, Technical Requirements Manual - TRM 3.6.3
FSAR Sections 4.4.5, 10.14.4.2, 13.0

Excerpt from 2-ARP-9-3D as referenced in A(1):

BFN Unit 2	Panel 2-9-3 2-XA-55-3D	2-ARP-9-3D Rev. 0034 Page 23 of 42

MAIN STEAM
RELIEF VLV AIR
ACCUM PRESS LOW

2-PA-32-31

18

(Page 1 of 2)

Sensor/Trip Point:
2-PS-32-31A thru F 82 psig

Sensor Location: El 565', Drywell

Probable Cause:
A. Improper valve alignment.
B. Control air leak.
C. Loss of Drywell Control Air.

Automatic Action: None

Operator Action:
A. **CHECK** Drywell Control Air System, if malfunctioning, **REFER TO** 2-AOI-32A-1. ☐
B. **IF** it is desired to align CAD to the Drywell Control Air System **THEN**: ☐
   1. **OPEN** CAD SYSTEM A(B) N2 SHUTOFF VALVE, 0-HS-84-5A(16A), on Unit 1, Panel 9-54(55). ☐
   2. **CHECK** N2 VAPORIZER A(B) OUTLET PRESS, 0-PI-84-6(17), reading approximately 100 psig. ☐
   3. **PLACE** keylock switch CAD A(B) CROSS TIE TO DW CONTROL AIR, 2-HS-84-48(49), in OPEN on Panel 2-9-54(55). ☐
   4. **CHECK OPEN** CAD A(B) CROSS TIE TO DW CONTROL AIR, 2-FSV-84-48(49). ☐
   5. **REFER TO** 2-OI-84 for CAD cross-tie operability determination and CAD tank level requirements. ☐
   6. **WHEN** Drywell Control Air is available **THEN**: **REFER TO** 2-OI-84 to return CAD system to normal alignment. ☐

Continued on Next Page

Excerpts from EOIPM 0-V(C):

1.0 EOI-1, RPV CONTROL BASES (continued)

RC/P-2

STABILIZE RPV press below 1073 psig using the main turbine bypass vlvs (APPX 8B)

- OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
- Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary

IF	THEN
DW Control Air is or becomes unavailable	**PLACE** the control switch for each MSRV in CLOSE/AUTO AND **PLACE** MSRV auto actuation logic inhibit XS-1-202 to INHIBIT

RC/P-2

DEPRESSURIZE the RPV and maintain cooldown rate below 100°F/hr

- Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary

IF	THEN
ANY MSRV is being used for RPV depressurization AND DW Control Air is or becomes unavailable	Depressurize using sustained MSRV opening and maintain cooldown rate below 100°F/hr

RC/P-3

WHEN

shutdown cooling RPV press interlock clears
AND
further cooldown is required

RC/P-4

COOL DOWN to cold shutdown with shutdown cooling (Table DH-4)

- **MAINTAIN** cooldown rate below 100°F/hr
- Use only RHR pumps NOT required to maintain RPV water lvl above +2 in. by operation in the LPCI mode

IF	THEN
Shutdown cooling CANNOT be operated	**MAINTAIN** RPV pressure below the shutdown cooling RPV pressure interlock using ANY of the systems used for depressurization (Table P-1)

RC/P-5

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/P-2 (cont'd)

- Changing plant conditions may necessitate adjustments to the value at which pressure is initially stabilized. If the pressure control system in use becomes unavailable, the alternate system used in its place may not be capable of maintaining the same control band. Similarly, if an injection system in use becomes unavailable, pressure may need to be reduced to permit use of lower head systems.

Both the rate and the magnitude of RPV pressure changes must be considered. A pressure that is slowly lowering over a relatively wide control band may be more "stable" than short-period oscillations within a narrower control band. In general, the adequacy of steps taken to stabilize RPV pressure must be judged by the effect of any continuing pressure variations on RPV water level and by whether additional actions are possible or likely to afford better control capability. If pressure variations are not interfering with RPV water level or cannot be stopped, pressure may be considered stabilized. If continuing pressure oscillations are complicating efforts to control RPV water level or if the existing pressure prevents use of available injection systems, additional effort is warranted.

The main turbine bypass valves are the preferred means of controlling RPV pressure, since they provide good control capability, are of relatively large capacity, and do not add heat to the suppression pool. The direction to use the bypass valves implicitly permits opening the MSIVs and placing the main condenser in service if such actions are necessary and conditions permit. It does not, however, constitute authorization to defeat any MSIV isolation interlocks.

If the main turbine bypass valves are unavailable or if the steam generation rate exceeds the bypass capacity, the Alternate RPV Pressure Control Systems listed in Table P-1 may be used to stabilize RPV pressure. No prioritization of the listed systems is specified since the appropriate choice of systems is event-specific while the EOIs must address a full spectrum of initial plant conditions and postulated transients.

The action to crosstie CAD to Drywell Control Air should ensure a continuous pneumatic supply to MSRV and inboard MSIV actuators. Without a continuous pneumatic supply to MSRV and inboard MSIV actuators, there is no guarantee: 1) to the number of times that an MSRV can be cycled manually, or 2) that inboard MSIVs will remain open or, if closed, can be reopened.

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/P-2 (cont'd)

Provisional Action

The term "available" is defined as the state or condition of being ready and able to be used (placed into operation) to accomplish the stated (or implied) action or function. As applied to a system, this requires operability of necessary support systems (electrical power supplies, cooling water, lubrication, etc.). Therefore the phrase "becomes unavailable," as it applies to drywell control air means that either the system cannot presently, or in the near future, accomplish its stated function of providing a continuous pneumatic supply to MSRV and inboard MSIV actuators.

Loss of the continuous pneumatic supply would cause the "MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW" annunciator (XA-55-3D-18) and "DRYWELL CONTROL AIR PRESS LOW" (XA-55-3E-35) annunciators to alarm. This limits the number of times that MSRVs can be cycled manually since this mode of valve operation requires pneumatic pressure. While the MSRV accumulators provide a reserve pneumatic supply, leakage through in-line valves, fittings, and actuators may deplete the reserve capacity. There is thus no assurance as to the number of MSRV operating cycles remaining following loss of the continuous MSRV pneumatic supply. For these reasons, if the continuous pneumatic supply is or becomes unavailable when MSRVs must be used to augment RPV pressure control, the valves should be closed to limit the number of cycles and conserve pneumatic pressure so that the MSRVs will be available if emergency RPV depressurization is later required. If other pressure control systems are not capable of maintaining RPV pressure below the lowest MSRV lift setpoint, MSRVs will still open when the lift setpoint is reached.

CLOSE/AUTO is the name for the switch position which causes the MSRV to be in the closed position (assuming that RPV pressure does not exceed the relief mode setpoint). To prevent MSRVs opening in the relief mode, XS-1-202 is placed in "Inhibit" by Appendix 11A.

Excerpt from 2-EOI Appendix-8G:

BFN Unit 2	Crosstie CAD to Drywell Control Air	2-EOI Appendix-8G Rev. 0007 Page 5 of 5

1.0 INSTRUCTIONS (continued)

NOTES

1) CAD System A to Drywell Control Air supplies the following MSIVs:
- 2-FCV-1-14, MSIV LINE A INBOARD
- 2-FCV-1-26, MSIV LINE B INBOARD.

2) CAD System B to Drywell Control Air supplies the following MSIVs:
- 2-FCV-1-37, MSIV LINE C INBOARD
- 2-FCV-1-51, MSIV LINE D INBOARD.

[10.3] **MONITOR** inboard MSIV indication status for valves drifting closed. ☐

[11] IF Drywell Control Air header supplied from CAD System A shows indications of being depressurized, **THEN**

CLOSE the following valves:

- 0-FCV-84-5, CAD SYSTEM A N2 SHUTOFF VALVE (Unit 1, Panel 1-9-54) ☐
- 2-FSV-84-48, CAD A CROSS TIE TO DW CONTROL AIR (Unit 2, Panel 2-9-54). ☐

[12] IF Drywell Control Air header supplied from CAD B shows indications of being depressurized, **THEN**

CLOSE the following valves:

- 0-FCV-84-16, CAD SYSTEM B N2 SHUTOFF VALVE (Unit 1, Panel 1-9-55) ☐
- 2-FSV-84-49, CAD B CROSS TIE TO DW CONTROL AIR (Unit 2, Panel 2-9-55). ☐

LAST PAGE

Excerpts from 2-AOI-32-2 to support Distractors B2, D2:

BFN Unit 2	Loss of Control Air	2-AOI-32-2 Rev. 0039 Page 4 of 25

1.0 PURPOSE

This Abnormal Operating Instruction provides symptoms, automatic action, operator actions and expected system responses for loss of control air.

2.0 SYMPTOMS

A. AIR COMPRESSOR ABNORMAL annunciator, (1-XA-55-20B, Window 29) is in alarm.

B. CONTROL AIR COMP G BKR DE-ENERGIZED (0-XA-55-23B, Window 38) will alarm.

C. CONTROL AIR COMP G MOTOR AMPS, 0-EI-32-2901, on Panel 1-9-20 indicates approximately zero amps.

D. Air Compressor G ICS Display shows Compressor G in an unloaded or shutdown condition.

E. Air Compressor G ICS Display shows lowering Control Air Header Pressure.

F. Control Air Compressor G breaker tripped.

G. SERVICE AIR XTIE VLV OPEN (0-FCV-33-1 Open) annunciator, 0-PA-33-1A/1(3) (Unit 1 and Unit 3) on Panel 1(3)-9-20 is in alarm at (1(3)-XA-55-20B, Window 30).

H. CONTROL AIR PRESS LOW annunciator, 0-PA-32-88 is in alarm (2-XA-55-20B, Window 32).

I. SCRAM PILOT AIR HEADER PRESS LOW annunciator, 2-PA-85-38B on Panel 9-5 is in alarm (2-XA-55-5B, Window 28).

J. MAIN STEAM LINE ISOL VLV POSN HALF SCRAM annunciator is in alarm (2-XA-55-4A, Window 30).

K. DRYWELL CONTROL AIR PRESSURE LOW 2-PA-32-70 annunciator is in alarm (2-XA-55-3E, Window 35).

L. CONDENSER A, B OR C VACUUM LOW 2-PA-47-125 annunciator is in alarm (2-XA-55-7B, Window 17).

M. OG HOLDUP LINE INLET FLOW LOW 2-FA-66-111A annunciator is in alarm (2-XA-55-53, Window 4).

N. HOTWELL A(B)(C) LEVEL ABNORMAL 2-LA-2-3(2-LA-2-6)(2-LA-2-9) is in alarm (2-XA-55-6A, Window 5(6)(7)).

Attachment 1
(Page 1 of 6)

Expected System Responses

1.0 MAIN STEAM

A. If the loss of control air is instantaneous, when control air pressure drops to < 45 psig, the MSIV accumulator air will be routed to close the OutBd MSIVs.

B. If the loss of control air is slow or gradual, a high probability exists for the accumulator air to be vented to atmosphere due to the slow realignment of the 4-way valve. This will prevent accumulator air from assisting in OutBd MSIV closure.

C. STEAM SEAL REGULATOR, 2-PCV-1-147 steam seal regulating valve will fail closed on loss of air. Steam seal pressure can be controlled using STEAM SEAL REG BYPASS VALVE, 2-HS-1-145.

D. Pressure control valve 2-PCV-1-175A(B) for Off Gas Preheater A(B) fails open on loss of air. 2-FCV-1-176A and 2-FCV-1-176B are to be closed after loss of air.

E. Pressure control valves 2-PCV-1-151(153) for SJAE A(B) first and second stage fail closed on loss of air.

F. Pressure control valves 2-PCV-1-166(167) for SJAE A(B) third stage fail closed on loss of air.

G. Main Steam Line Drain valves 2-FCV-1-58, 185, 168, 169, 170, and 171 fail open.

2.0 CONDENSATE

A. COND BSTR PUMP DISCH BYPASS TO COND B & C, 2-FCV-2-29A&B fail closed on loss of air.

B. HOTWELL LVL CONT HIGH LVL BYPASS, 2-LCV-2-3 fails closed on loss of air. 2-FCV-2-4 can be used for Hotwell reject.

C. HOTWELL LVL CONT LOW LVL MAKEUP, 2-LCV-2-6 fails closed on loss of air. FCV-2-7 can be used for Hotwell makeup.

D. Condensate Demineralizer influent, effluent, and drain valves fail as-is.

E. CONDENSATE DEMINERALIZER BYPASS VALVE, 2-FCV-2-130, fails open.

Attachment 2
(Page 1 of 1)

Align Drywell Control Air to Open Outboard MSIV's

The following steps will align the Drywell Control Air System to Outboard MSIVS:

- **CLOSE** 2-32-1878 located at Elev 565 Rx Bldg outside steam vault (this will prevent supplying Drywell Control Air to components outside the Drywell other than the outboard MSIV's).
- **OPEN** 2-32-334 located above the TIP room (aligns Drywell Control Air to the outboard MSIV's).

REOPEN MSIV's.

REFER TO 2-OI-1 Section 5.2 and **ESTABLISH** Main Condenser as heat sink by performing the following:

NOTE
The Main Turbine Turning Gear is to be manually engaged due to loss of air.

- **ENSURE** the Main Turbine is on turning gear.
- **ENSURE** the RFP turbine shaft is rotating.
- **ADJUST** steam seal header pressure by throttling steam seal regulator bypass valve 2-FCV-1-145 (2-PCV-1-147 fails closed on loss of air).
- **PLACE** mechanical vacuum pumps in service. **REFER TO** 2-OI-66.

Excerpt from OPL171.054 Lesson Plan:

E. System Interrelationships

1. Main Steam System
   a) Inboard MSIV will close on a loss of DWCA.
   b) Outboard MSIVs will close on a loss of Control Air.
   c) When control air pressure drops to <45 psig, the MSIV accumulator will route air to the MSIV to assist in closing.
   d) DWCA normally supplies pneumatic pressure to Inboard MSIVs, their accumulators, MSRVs and 6 ADS accumulators. DWCA can also be cross-tied to supply the outboard MSIVs on all three units if plant control air is lost. ADS accumulators contain enough air for a minimum of five operations or maintain the valves open for 30 minutes.

   NLO Objective 10
   NLOR Objective 8
   ILT Objectives 5 and 6
   LOR Objective 5
   Reference ARP-9-3E window 35, AOI-32A-1, and EOI Appendix 8G
   Reference AOI-32-2

2. Condensate the Feedwater Systems
   a) All valves in the condensate and feedwater systems except the following fail closed on a loss of air:
   b) Condensate Demin Inlet, effluent and drain valves fail as is.
   c) RFP minimum flow valves fail open
   d) Feedwater startup bypass valve LCV-3-53 fails as is at 65 psig control air pressure.
3. Raw Cooling Water
   a) All TCVs fail open on loss of Control Air except for the following:
   b) TCV-24-80B and TCV-24-85B (6 inch valves) to the RBCCW heat exchangers fail closed on a loss of CA.
   c) Turbine Lube Oil TCV-24-75B (4 inch valve) fails closed on the loss of CA.
4. Plant Control Air - can be aligned to DWCA to supply pneumatic pressure for the operation of components in the drywell. This should be used as a last resort due to the introduction of oxygen into the drywell.

   Reference print 47E610-70-1

5. Reactor Building Closed Cooling Water (RBCCW) System
   a) DWCA provides pneumatic pressure for control of Drywell Cooler valves and dampers. On a loss of air, the RBCCW valves will fail open while the Cooler fan dampers will fail closed.
   b) The outlet TCV for the RWCU Non-Regenerative Heat Exchanger will fail open on a loss of control air.
6. Containment Atmospheric Dilution (CAD) System - can be aligned to DWCA if a break in the DWCA headers occurs. CAD is a

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295024 (EPE 1) High Drywell Pressure / 5

G2.4.23 (10CFR 55.43.5 - SRO Only)

Knowledge of the bases for prioritizing emergency operating procedures implementation.

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295024G2.4.23	
Importance Rating	------	4.4

Proposed Question: **# 80**

In accordance with EOI-2, Primary Containment Control Bases, the Suppression Chamber Spray Initiation Pressure prevents **(1)** which can occur when 95% of the non-condensibles in the Drywell have been transferred to the Suppression Chamber and replaced by steam during a LOCA.

BEFORE this Suppression Chamber Pressure is reached, the NUSO is required to direct the performance of **(2)**.

Note: EOI APPENDIX-17B, RHR System Operation Drywell Sprays
EOI APPENDIX-17C, RHR System Operation Suppression Chamber Sprays

A. (1) over-pressurizing the Suppression Chamber
(2) EOI APPENDIX-17B

B. (1) over-pressurizing the Suppression Chamber
(2) EOI APPENDIX-17C

C. (1) chugging in the Suppression Chamber downcomers
(2) EOI APPENDIX-17B

D. (1) chugging in the Suppression Chamber downcomers
(2) EOI APPENDIX-17C

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible when 12 psig Suppression Chamber Pressure is compared to 2.45 psig (SCRAM required) Primary Containment Pressure in the EOI-2, PC Pressure leg, the candidate could easily confuse the Bases as the maximum Suppression Chamber Pressure. Second part is incorrect but plausible in that EOI-2, Primary Containment Control, PC/P-6 and 7 direct performance of EOI APPENDIX-17B, RHR System Operation Drywell Sprays, when Suppression Chamber Pressure **exceeds** 12 psig.

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is correct *(See D).*

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT**: *(See attached)* In accordance with EOI-2, Primary Containment Control Bases, the Suppression Chamber Spray Initiation Pressure is the lowest Suppression Chamber Pressure which can occur when 95% of the non-condensibles in the Drywell have been transferred to the Suppression Chamber and replaced by steam during a LOCA. For second part, in accordance with EOI-2, Primary Containment Control, PC/P-4 and 5, **before** Suppression Chamber Pressure rises to 12 psig, initiate EOI APPENDIX-17C, RHR System Operation Suppression Sprays.

SRO Level Justification: Tests the candidate's knowledge of the Bases for prioritizing the Emergency Operating Instructions as it relates to High Drywell Pressure. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as Memory due to the requirement to strictly recall EOI Bases as it relates to Primary Containment Pressures.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-EOI-2, Rev. 18 (Attach if not previously provided)
EOIPM 0-V(G), Rev. 2

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203, Obj. 7 (As available)

Question Source:	Bank #		
	Modified Bank #	ILT EXAM BANK OPL171.016-02 002 #687	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:
55.41
55.43 **X**

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

687. OPL171.016-02 002

Which one of the following completes the statement below in accordance with EOIPM Section 3-VI-L, Suppression Chamber Spray Initiation Pressure Worksheet?

The Suppression Chamber Spray Initiation Pressure (SCSIP) is obtained by calculating the ___(1)___ suppression chamber pressure which results when 95% of the noncondensibles in the containment are in the suppression chamber and suppression pool temperature is at the ___(2)___ normal operating temperature

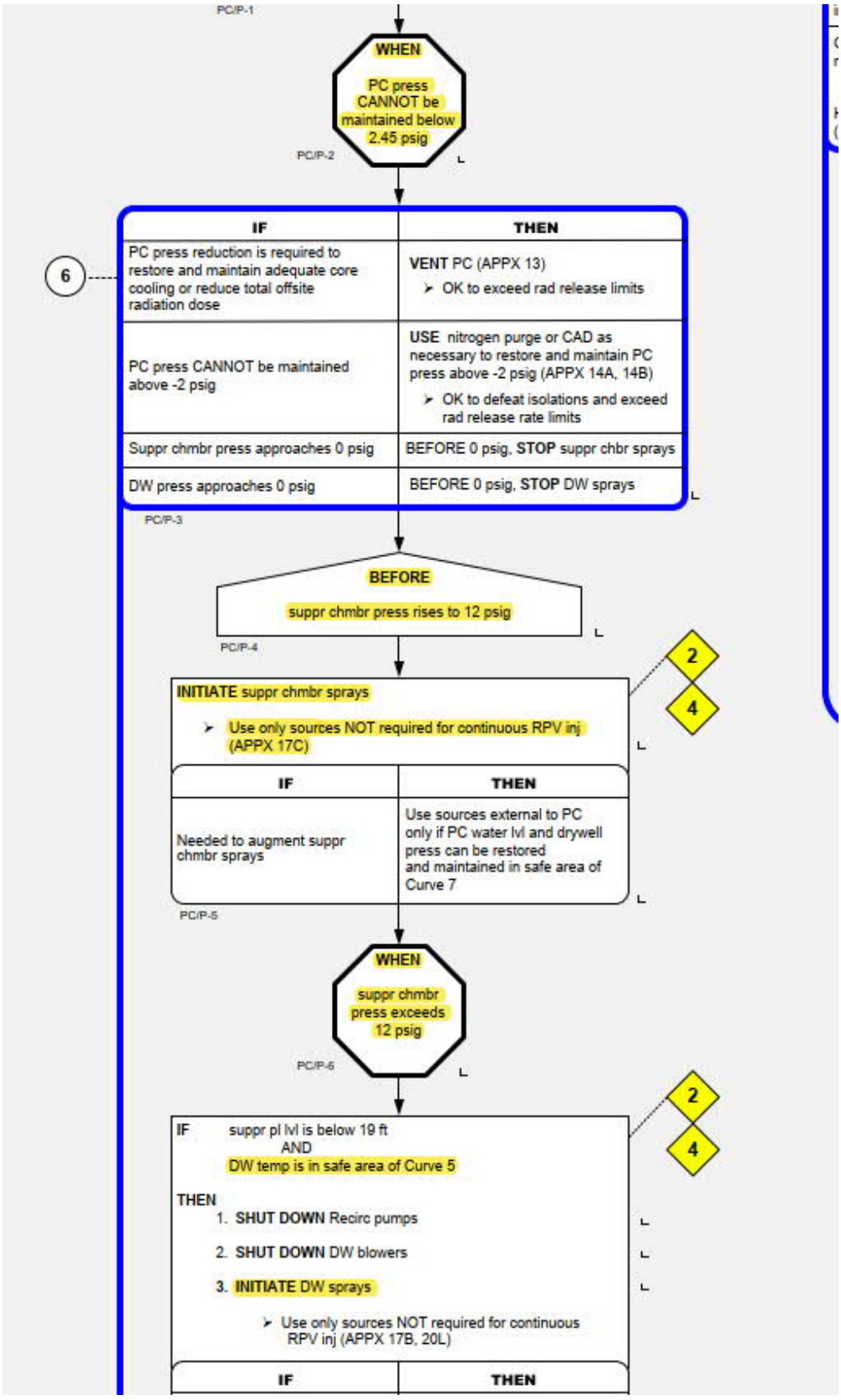
A. (1) lowest
(2) maximum

B. (1) highest
(2) minimum

C. (1) lowest
(2) minimum

D. (1) highest
(2) maximum

Excerpt from 2-EOI-2 PC Pressure leg:



Excerpt from EOIPM 0-V(G):

BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0002 Page 53 of 129

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: Step PC/P-4

Suppression chamber sprays may be initiated any time prior to pressure exceeding the Suppression Chamber Spray Initiation Pressure (SCSIP, **A.65**). The SCSIP is the threshold pressure that signals the possibility of chugging—the cyclic condensation of steam at the downcomer openings. During a LOCA, the drywell atmosphere is purged to the suppression chamber and replaced by steam. When a steam bubble collapses at the exit of the downcomers, the rush of water drawn into the downcomers to fill the void induces stresses at the junction of the downcomers and the vent header in Mark I containments. Repeated application of such stresses could cause fatigue failure of these joints, thereby creating a direct path between the drywell and suppression chamber. Steam discharged through the downcomers could then bypass the suppression pool and directly pressurize the primary containment.

Scale model tests have demonstrated that chugging will not occur if the drywell atmosphere contains at least 1% noncondensibles. As an added conservatism, the SCSIP is determined by assuming the drywell noncondensible content is 5%. The SCSIP is the lowest suppression chamber pressure which can occur when 95% of the noncondensibles in the drywell have been transferred to the suppression chamber. Refer to EOIPM Section 0-V(B)for discussion of the SCSIP.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:
295025 (EPE 2) High Reactor Pressure / 3
EA2.01 (10CFR 55.43.5 - SRO Only)
Ability to determine and/or interpret the following as they apply to High Reactor Pressure:

- Reactor pressure

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295025EA2.01	
Importance Rating	------	4.3

Proposed Question: **# 81**

Unit 2 was operating at 100% RTP when the following conditions occurred:

- A PCIS Group 1 Isolation resulted in a SCRAM
- Reactor Pressure peaked at 1136 psig, currently at 1090 psig
- **NO** Operator actions have been taken

Given the conditions above, ___(1)___ Main Steam Relief Valves (MSRVs) are automatically cycling and the Nuclear Unit Senior Operator (NUSO) will direct the performance of ___(2)___ to control Reactor Pressure.

Note: 2-EOI Appendix-8B, Reopening MSIVs/Bypass Valve Operation
2-EOI Appendix-11A, Alternate RPV Pressure Control Systems MSRVs

A. (1) four (4)
(2) 2-EOI Appendix-8B

B. (1) four (4)
(2) 2-EOI Appendix-11A

C. (1) eight (8)
(2) 2-EOI Appendix-8B

D. (1) eight (8)
(2) 2-EOI Appendix-11A

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that 2-EOI Appendix-8B, Reopening MSIVs/Bypass Valve Operation, cannot be performed due to present PCIS Group 1 isolation signal. The stem states that a PCIS Group 1 signal resulted in a reactor scram, but does NOT indicate that the isolation signal was cleared, thus MSIVs cannot be reopened. (PCIS logic cannot be reset in 2-EOI Appendix-8B, Step 1.0 [4].)

B **CORRECT**: (*See attached*) The Main Steam System contains 13 MSRVs, which all open automatically to control Reactor Pressure to protect the Reactor Pressure Boundary at specific setpoints. They are all assigned to one of 3 groups with each group having a common setpoint to open. Four valves open at 1135 psig, 4 at 1145 psig, and 5 open at 1155 psig. With the maximum given Reactor Pressure peaking at 1136 psig, four MSRVs would have opened. For second part, the candidate must recognize that given a spurious PCIS Group 1 Isolation occurred, the MSIVs will automatically close (not given in the stem) which resulted in a SCRAM. Once the MSIVs are closed, the Main Turbine Bypass Valves are no longer available for use, therefore 2-EOI Appendix 8B, Reopening MSIVs/Bypass Valve Operation, the normal method of Reactor Pressure Control, is not available. This results in the NUSO directing 2-EOI Appendix-11A, Alternate RPV Pressure Control Systems MSRVs for Reactor Pressure Control.

C INCORRECT: First part is incorrect but plausible in that with the given High Reactor Pressure cycling, the candidate could confuse the number of MSRVs and Reactor Pressure setpoint to conclude that 8 MSRVs automatically open which is actually at 1145 psig. For second part, there are multiple Reactor Pressure Control appendices that can be utilized with an MSIV isolation present. With the Group 1 Isolation in MSIV are closed and currently cannot be reopened using 2-EOI Appendix-8B.

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

SRO Level Justification: Tests the candidate's ability to determine and/or monitor Reactor Pressure affects and control systems, and appropriate procedure usage. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(2) Diagnosis that leads to selection of the procedures that should be used to respond to the evolution.

Technical Reference(s): 2-EOI Appendix 8B, Rev. 9 (Attach if not previously provided)
OPL171.009, Rev. 17U1
2-EOI-1, Rev. 21

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.009 Obj. 2 (As available)
OPL171.202 Obj. 18

Form 4.2-1	**Written Examination Question Worksheet**		
Question Source:	Bank #		
	Modified Bank #	BFN NRC 1804 #12	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Copy of Bank Question:

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-reference:

295025 (EPE 2) High Reactor Pressure / 3

EK1.03 (10CFR 55.41.10)

Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE:

- Safety/relief valve tailpipe temperature/pressure relationships

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295025EK1.03	
Importance Rating	3.6	------

Proposed Question: **# 12**

A spurious PCIS Group 1 Isolation occurred on Unit 2 with the following current conditions:

- All Control Rods did NOT insert
- Reactor Power is approximately 28%
- MAIN STEAM RELIEF VALVE OPEN 2-FA-1-1 (2-9-3C, Window 25) in alarm
- Reactor Pressure is cycling between approximately 1090 - 1136 psig
- SRV 1-19 is stuck full open
- NO operator actions have been taken over the last 10 minutes



Given the conditions above, which ONE of the following completes the statements below?

To control Reactor Pressure, __(1)__ SRVs will open.

The expected SRV 1-19 tailpipe temperature on 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE, on Panel 2-9-47 is approximately __(2)__.

A. **(1)** four (4)
(2) 395 °F

B. **(1)** four (4)
(2) 220 °F

C. **(1)** eight (8)
(2) 395 °F

D. **(1)** eight (8)
(2) 220 °F

Proposed Answer: **A**

Excerpt from 2-EOI Appendix-8B:

BFN Unit 2	Reopening MSIVs/Bypass Valve Operation	2-EOI Appendix-8B Rev. 0009 Page 3 of 8

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Control Room
ATTACHMENTS	None

[1] **DISPATCH** personnel to 480V RMOV Board 2C compartment 9B to **PLACE** 2-BKR-001-0057, MAIN STEAM LINE DRAIN VALVE FCV-1-57 (MO 2-78) in ON. ☐

[2] **IF** pressure control with bypass valves is desired and MSIVs are open, **THEN**

PROCEED to step 1.0[14]. ☐

[3] **ENSURE** ALL MSIV control switches in CLOSE position. ☐

[4] **RESET** PCIS logic (Panel 2-9-4). ☐

[5] **DEPRESS** the following pushbuttons to trip RFPTs (Panel 2-9-6):

- 2-HS-3-125A, RFPT 2A TRIP ☐
- 2-HS-3-151A, RFPT 2B TRIP ☐
- 2-HS-3-176A, RFPT 2C TRIP. ☐

[6] **ENSURE CLOSED** the following drain valves (Panel 2-9-3):

NOTE

To prevent auto opening of 2-FCV-1-58 and 2-FCV-1-185, handswitches 2-HS-1-58A and 2-HS-1-185A must be held in the CLOSE position until main turbine speed decreases to below 1700 RPM.

- 2-FCV-1-58, UPSTREAM MSL DRAIN TO CONDENSER ☐
- 2-FCV-1-59, DOWNSTREAM MSL DRAIN TO CONDENSER. ☐
- 2-FCV-1-185, BYPASS UPSTREAM MSL DRAIN TO CONDENSER. ☐

Excerpt from OPL171.009 Lesson Plan:

	(2) Relief functions (a) To provide automatic depressurization for small breaks in the nuclear steam system so that the LPCI mode of RHR and the Core Spray system can operate to protect the fuel barrier. This function is part of the Automatic Depressurization System (ADS). This function can be overridden with the use of two keylock switches on Panel 9-3 (ADS LOGIC INHIBIT SWITCHES) (b) Manual operation by the operator (c) The valves are designed such that 50 psi differential is required (between Reactor Pressure and Drywell Pressure) to open the valves and will auto close with a differential pressure ≤ 20 psi. (d) Pressure switches are utilized to work in the 'relief mode'. The 'relief mode' energizes a solenoid valve to open the SRV. The setpoints are the same as in the safety mode. This is done to compensate for potential setpoint drift in the 'safety relief' mode. (e) This 'relief mode' logic can be defeated by use of a switch on Panel 9-3. This switch "MSRV AUTO ACTUATION LOGIC INHIBIT" (XS-1-202) also brings in an alarm on Panel 9-3.	discussed in OPL171.043. (OF-1 Monitoring) Red light indicates solenoid energized not valve open. Check redundant indications (SRV tail pipe temp, acoustic monitor, MSL flow lowering, Steam flow/feed flow mismatch and Rx power initially goes down, then goes to a higher indicated value) Figure-3 Figure-4
d.	Valve setpoints for safety function (1) 4 valves @ 1135 psig ± 3% (2) 4 valves @ 1145 psig ± 3% (3) 5 valves @ 1155 psig ± 3%	Obj. ILT 2 Obj. LOR 1 Obj. NLOR 1 Obj. NLO 1 Figure-3

Excerpts from 2-EOI-1:

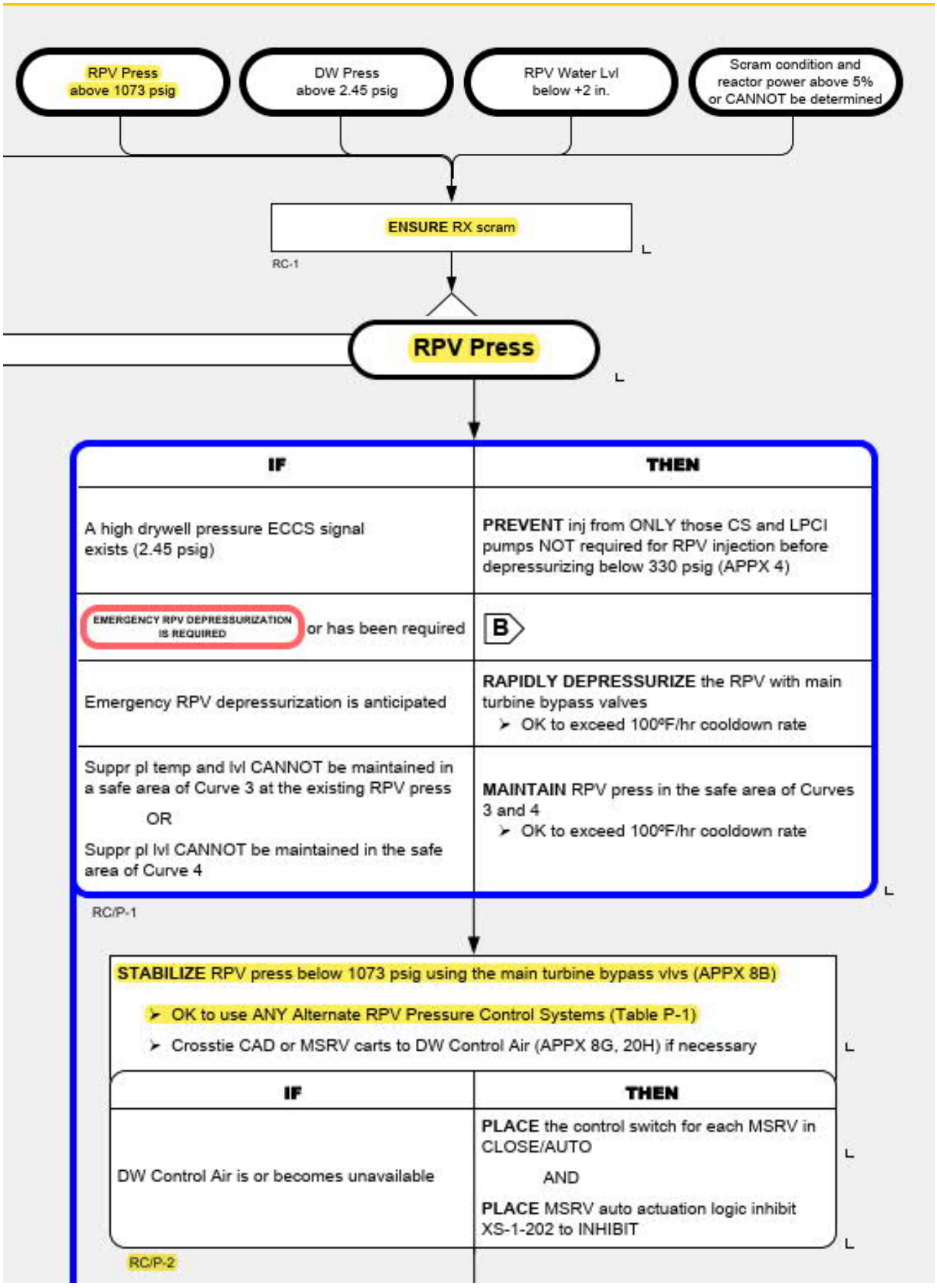


Table P-1

Alternate RPV Pressure Control Systems

Source	APPX
Main turb bypass vlvs	8B
MSRVs only if suppr lvl is above 5.5 ft IF "MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW" annunciator (XA-55-3D-18) is in alarm THEN **MINIMIZE** MSRV cycling by using sustained opening for depressurization	11A
HPCI with CST suction if available 2 5 7	11C
RCIC with CST suction if available 2 3 5	11B
RFPTs on min flow	11F
Main Steam system drains	11D
Steam seals	11G
SJAEs	11G
Off Gas preheater	11G
HPCI and RCIC drains	11J
RWCU if NO boron has been injected into the RPV	11E
MSRV Remote Operation only if suppr lvl is above 5.5 ft	EDMG-24 APPX. D FSI-3E

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
600000 (APE 24) Plant Fire On Site / 8	Tier #	------	1
G2.2.14 (10CFR 55.43.3 - SRO Only)	Group #	------	1
Knowledge of the process for controlling equipment configuration or status	K/A #	600000G2.2.14	
	Importance Rating	------	4.3

Proposed Question: **# 82**

In accordance with 0-FSS-001, Fire Safe Shutdown, the **(1)** will be responsible for the implementation and control of FSS procedures and will assign personnel as necessary.

In accordance with **(2)**, with the exception of Annunciators, plant personnel are allowed to use all available equipment until proven unreliable.

A. (1) Shift Manager
(2) EOIs

B. (1) Shift Manager
(2) 0-FSS-001

C. (1) Fire Brigade Leader
(2) EOIs

D. (1) Fire Brigade Leader
(2) 0-FSS-001

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B)*. Second part is incorrect but plausible in that the EOIs are used to control key plant parameters by controlling system equipment configurations and status and doesn't discount the Annunciators.

B **CORRECT**: *(See attached)* In accordance with 0-FSS-001, the Shift Manager will be responsible for the implementation and control of the FSS and will assign personnel as necessary. For second part, in accordance with 0-FSS-001, NFPA-805 Code plant personnel are allowed to use all available equipment until proven unreliable with the exception of Annunciators.

C INCORRECT: First part is incorrect but plausible in that in accordance with 0-AOI-26-1, Fire Response, the Fire Brigade Leader coordinates changes to the ventilation system alignment. Second part is incorrect but plausible *(See A)*.

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is correct *(See B)*.

Form 4.2-1 **Written Examination Question Worksheet**

SRO Level Justification: Test the candidate's knowledge of the process for controlling equipment configuration or status during a plant fire. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to a plant fire on site.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 0-FSS-001, Rev. 10 (Attach if not previously provided)
OPL171.201, Rev. 10U1

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.501, Obj. 6 (As available)

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	**X**
Question History:	Last NRC Exam	

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

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Excerpts from 0-FSS-001:

BFN	Fire Safe Shutdown	0-FSS-001 Rev. 0010 Page 10 of 135

NOTE

The sub-instructions require precise communication and close coordination of operations department personnel. Portable radios set to channel F2, F4 and "Ops Group" on Harris Radios are common Fire Safe Shutdown channels for communication. Other communications systems (i.e. sound powered phones) can be used, if available.

4.0 PRECAUTIONS AND LIMITATIONS

A. The Shift Manager, or designee, will be responsible for the implementation and control of the FSS and will assign personnel as necessary.

B. The initiation of the RHR system shall be a high priority from the loss of all high pressure make-up to the reactor to minimize the consequences of uncovering the fuel.

C. When monitoring parameters such as reactor water level and reactor pressure, it is preferred to use the credited instrumentation directed by the procedure since other instrumentation may be unreliable due to the fire.

D. Spurious operation of plant equipment may be encountered due to fire induced damage to control circuits and instrumentation. Frequent observation of the control boards by the Operations personnel is required to ensure that spurious equipment operation does not adversely affect nuclear safety.

E. For FSSs, EOIs will be used concurrently if EOI entry conditions have been met. The exception is Fire Area 16 when the Main Control Room is required to be evacuated. There are no EOI procedures at the back up control panel 9-32.

F. Tasks in the fire affected area or adjacent/above the fire affected area may contain smoke/fumes after 60 minutes and the Fire Captain/Incident Commander will be contacted prior to entry to determine if breathing apparatus is still required. SCBA units are stored in Electrical Board Rooms 1A, 2A and 3A. These are stored in the electrical board rooms since Emergency Lights for ingress/egress and equipment operation are not installed in the turbine buildings. Entry to Reactor buildings on each unit is through the Electrical Board Rooms.

1.0 PURPOSE

The purpose of the Fire Safe Shutdown (FSS) is to provide the manual recovery actions needed, in conjunction with the Normal and Emergency operating procedures to ensure safe shutdown of either Unit 1, Unit 2 or Unit 3, or all three units if conditions require, in the event of a disabling fire event. The exception to the rule is the Control Room Abandonment Fire procedure which is a unitized, standalone procedure due to EOIs not being staged at the back up control center . The required actions depend on the location and severity of the fire event.

The Fire Safe Shutdown Procedures are built for fire in one Fire Area.

2.0 DISCUSSION

[1] Implementing Philosophy

[1.1] To initiate an FSS and achieve safe shutdown of the plant during a fire event, **PERFORM** the following:

[1.1.1] **DETERMINE** the location of the fire. (**REFER TO** Table 1 and Illustrations, Figure 1-1A through 30-2, for fire area boundaries.)

[1.1.2] **COMPARE** the required entry conditions provided in Section 3.0 of this instruction with the current plant condition.

[1.2] NFPA-805 Code allows the plant to use all available equipment until proven unreliable with the exception of Annunciators. Credited equipment is equipment which has the highest probability to remain functional in the fire event. Use of the credited equipment is desired in that it will remain in service and not subject to tripping due to a fire related event. So, for example, if the EOIs are telling the operator to place Suppression Pool Cooling in service, the FSS will provide the guidance on which pumps should be used for the best possible outcome and continued operation. If a piece of equipment is in service, it is not necessary to remove it and place the credited, or preferred, piece of equipment in service. But vigilance is required to ensure that initial equipment remains in service. If it trips, subsequent issues like water hammer from voided piping, could occur worsening the fire event.

[1.3] The FSSs will be entered by all three units when entry conditions are met for a unit depending on the fire area (FA). While the fire may not impact the other two units, power supplies and common equipment may be required to mitigate the fire's damage on the impacted unit.

[1.4] Units which are not directly impacted by the fire may keep running at power operations while the impacted unit is involved in the fire. Tech Spec and TRM requirements will be evaluated and applied when common equipment has been damaged.

Excerpts from OPL171.201 Lesson Plan: Supports Distractors A2, C2

OPL171.201, Introduction to the Emergency Operating Instructions, Rev.10U1

Lesson Plan Content

Outline of Instruction	**Instructor Notes and Methods**

I. General Introduction

A. Introduce self/ or guests
B. Take Attendance
C. Solicit student feedback
D. Introduce Topic / Goal
E. What's In It For Me?
F. Learning Objectives
G. Evaluation Method
H. Description of how class will be conducted

II. Presentation

A. Purpose of the EOIs

1. The EOIs and SAMGs function together as an integrated set of instructions. The EOIs protect the principal barriers to radioactivity release through control of key plant parameters. The EOI Contingencies form extensions of the EOIs, providing more detailed instructions for controlling individual parameters under more degraded conditions. The SAMGs extend the EOIs still further, defining strategies for severe accident conditions. SAMGs are used during beyond design basis events, and are beyond the scope of the initial license training program.

B. Overview of the EOIs

1. EOI Flowchart General Structure and Design

a) There are Nine (9) flowchart procedures, three (3) flowchart contingency procedures (C), and one (1) flowchart with Curves/Cautions:

- EOI-1, RPV Control (RC)
- EOI-1A, ATWS RPV Control (ARC)
- EOI-1CSD, RPV Control Mode 4 (RC)
- EOI-1RF, RPV-SFP Control - Mode 5 (DH)
- EOI-2, Primary Containment Control (PCC)
- EOI-2CSD, Containment Control - Mode 4 (CC)
- EOI-2RF, Containment Control - Mode 5 (CR)
- EOI-3, Secondary Containment Control (SCC)
- EOI-4, Radioactivity Release Control (RR)
- EOI-5, Curves and Cautions
- C-1, Alternate Level/Pressure Control
- C-4, RPV Flooding
- C-4A, ATWS RPV Flooding

OPL171.201, Introduction to the Emergency Operating Instructions, Rev.10U1

D. Guidelines for Use of the EOIs — Objective 2

a) Entry into the EOIs

a) Entry condition parameters are selected to provide advance warning of potential emergency conditions, allowing action to be taken sufficiently early to prevent more severe consequences.

b) Entry condition setpoints are chosen so as to be simple, operationally significant, unambiguous, readily identifiable, and familiar to plant operators. In general, the entry conditions are typically one or more of the following:
(1) Scram setpoints
(2) Annunciator alarm setpoints
(3) Trip setpoints
(4) Technical Specification values
(5) ECCS automatic initiation logic setpoints

c) The entry conditions alone do not necessarily mean that an emergency condition exists. The entry conditions are, however, symptomatic of emergency conditions, or conditions that could lead to emergency conditions.

d) Contingency procedures are entered as directed by an EOI, or other contingency procedure.

b) Place-keeping and Note-keeping — Objective 3

a) Place-keeping is very important in cases where several flow paths are being performed concurrently. It can also be valuable information for history purposes after the EOIs are exited.

b) Place-keeping aids are available at each flowchart symbol which requires attention. The operator is required to implement some sort of place-keeping technique (marking mechanism) which he/she feels comfortable with and will satisfy the following requirements:

- The operator can tell at any given time what steps have been executed or addressed, and can tell which steps are presently being executed.
- The operator can tell which flowpaths were re-entered and which steps were executed or addressed again because of the re-entry.
- The place keeping technique does not adversely affect the readability of the procedure steps should the steps have to be revisited because of flowpath re-entry, etc.

c) Each flowchart has "gray space" that the operator may use to write notes or data for future reference. Often this is done for recording the value of some given parameter, such as suppression pool temperature, and the time, so that it may be used to record a rate of change when the next update is given.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295008 (APE 8) High Reactor Water Level / 2

AA2.04 (10CFR 55.43.5 – SRO Only)

Ability to determine and/or interpret the following as they apply to High Reactor Water Level

- Heatup rate

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295008AA2.04	
Importance Rating	------	3.3

Proposed Question: **# 83**

Unit 2 is in MODE 4 with the following conditions:

- Loss of Shutdown Cooling occurred
- NUSO has directed a Reactor Water Level band of (+) 70 to (+) 90 inches

Given the Reactor Water Level band above, in accordance with Tech Specs, the **MAXIMUM** heatup rate allowed is **(1)** per hour.

In accordance with Tech Spec Bases 3.4.8, RHR Shutdown Cooling System – Cold Shutdown, the **MINIMUM** equipment for an OPERABLE RHR Shutdown Cooling **SUBSYSTEM** is **(2)**.

Note: (Assume all piping and valves are OPERABLE and the heat exchangers/RHRSW pumps are associated with RHR pumps)

A. (1) 80°F
(2) 1 RHR Pump, 1 RHR Heat Exchanger and 1 RHRSW Pump

B. (1) 80°F
(2) 2 RHR Pumps, 2 RHR Heat Exchangers and 2 RHRSW Pumps per Heat Exchanger

C. (1) 100°F
(2) 1 RHR Pump, 1 RHR Heat Exchanger, 1 RHRSW Pump

D. (1) 100°F
(2) 2 RHR Pumps, 2 RHR Heat Exchangers and 2 RHRSW Pumps per Heat Exchanger

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 80 °F/Hr (20°F every 15 minutes) is the target heatup/cooldown rate to support a 90 °F/Hr administrative limit when Reactor Coolant System (RCS) is greater than or equal to 215 °F as specified in 2-GOI-100-1A, Unit Startup and Power Operations. Second part is correct (*See C)*.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that Core Spray OPERABILITY requires both pumps in each loop to satisfy safety function for meeting its designed spray pattern. Additionally, OPERABILITY for a LPCI subsystem of the RHR system requires two motor driven pumps and associated piping and valves to transfer water. Lastly, the required components for ECCS subsystem OPERABILITY can vary between MODES adding to the complexity of the information to recall.

C **CORRECT**: (*See attached*) In accordance with Tech Spec 3.4.9 maximum allowed heat up or cool down rate is 100 °F/Hr. For second part, Tech Spec Bases 3.4.8 specifies that an OPERABLE RHR Shutdown Cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, one RHRSW pump capable of providing cooling to the heat exchanger, and the associated piping and valves.

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

SRO Level Justification: Tests the candidate's ability to determine and/or interpret heatup rate as it applies to High Reactor Water Level. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data as it relates to the Tech Spec Bases for RHR SDC subsystem and applicable heatup rates.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(1) information contained in the site's procedures, including alarm response procedures, abnormal operating procedures (AOPs), emergency operating procedures (EOPs), and their associated bases documents.

Technical Reference(s): 2-GOI-100-1A, Rev. 192 (Attach if not previously provided)
2-AOI-74-1, Rev. 43
U2 Tech Spec 3.4.8, Rev. 0
U2 Tech Spec 3.4.9, Amend. 253, 338
U2 Tech Spec 3.5.1, Rev. 0

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.044 Obj. 8 (As available)

Question Source:		
	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	**X**
Question History:	Last NRC Exam	

Form 4.2-1 **Written Examination Question Worksheet**

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

Comments:

Excerpt from 2GOI-100-1A: (Also supports Distractors A1, B1):

BFN Unit 2	Unit Startup and Power Operation	2-GOI-100-1A Rev. 0192 Page 20 of 215

3.2.3 Coolant and Metal Temperatures

A. Lowering Reactor head flange and/or head temperature below the temperature of fully tensioning reactor head bolts may result in bolt relaxation and potential leakage when reactor vessel is pressurized during startup.

B. The following limitations apply to reactor heatup and/or cooldown:

1. When Reactor coolant temperature is less than 215°F, a maximum heatup rate limit of 40°F/hr will reduce the Oxygen and Hydrogen Peroxide content of the coolant, and mitigate the base metal from the reactor flange closure region from exceeding a 100°F/hr maximum Heatup rate limit. In addition, it is recommended that Reactor Criticality from a cold shutdown condition should occur at temperatures greater than 180°F. [PER 558516]

2. Past experience has shown that during startup from cold conditions, RPV flange metal temperature heatup rates of greater than 100°F/hr could occur when the MSIVs are opened later in the startup. To prevent this from occurring, the Reactor should NOT be taken critical until moderator temperature is at least 180°F. This will provide sufficient time for the flange metal temperature to heat up before the MSIVs are open. [PER 558516]

3. During Reactor heatup with reactor coolant temperature greater than or equal to 215°F, and during Reactor Cooldown, the optimum rate of temperature change is 20°F every 15 minutes. This ensures the administrative limit of 90°F/Hr is NOT exceeded.

   Do NOT Attempt to "makeup" for time intervals which fall short of 20°F. If the 20°F is exceeded in any 15 minute period, subtract the amount of heatup/cooldown rate over 20°F from the 20°F for the next 15 minute period. These guidelines will assist in achieving a target heatup/cooldown rate of 80°F/Hr and ensure the administrative limit of 90°F/Hr is NOT exceeded.

4. During Reactor heatup, operators should use metal temperatures as a reminder that as metal heats up, the moderator HEATUP RATE will rise with the same amount of heat input.

Excerpts from Tech Spec 3.5.1: (Supports distractors B2, D2)

BASES

BACKGROUND (continued)

RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with LPCI and CS. In this event, either the vessel would be manually depressurized or the ADS timed sequence would be allowed to time out and open the selected safety/relief valves (S/RVs) depressurizing the RCS, thus allowing the LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly and the LPCI and CS cool the core.

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool may be circulated through a heat exchanger cooled by the RHR Service Water System. Depending on the location and size of the break, portions of the ECCS may be ineffective; however, the overall design is effective in cooling the core regardless of the size or location of the piping break.

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS equipment.

The CS System (Ref. 1) is composed of two independent subsystems. Each subsystem consists of two 50% capacity motor driven pumps, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger. The LOCA analysis (Ref. 13) requires both pumps in a subsystem (loop) to be OPERABLE for the subsystem to be OPERABLE. Failure of one CS pump results in the loss of the associated CS loop for LOCA mitigation. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started (A pump

(continued)

BASES

BACKGROUND (continued)

at 0.2 seconds when offsite power is available and B, C, and D pumps approximately 7, 14, and 21 seconds afterwards and if offsite power is not available all pumps 7 seconds after diesel generator power is available). When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop.

The two LPCI pumps and associated motor operated valves in each LPCI subsystem are powered from separate 4 kV shutdown boards. Both pumps in a LPCI subsystem inject water into the reactor vessel through a common inboard injection valve and depend on the closure of the recirculation pump discharge valve following a LPCI injection signal. Therefore, each LPCI subsystem's common inboard injection valve and recirculation pump discharge valve are powered from one of the two 4 kV shutdown boards associated with that subsystem.

(continued)

Excerpt from 2-AOI-74-1:

4.2 Subsequent Actions (continued)

CAUTION

Accurate coolant temperatures will **NOT** be available if all forced circulation is lost.

[16] [NER/C] **IF** forced circulation has been lost **AND** vessel cavity is less than 80 inches, **THEN** (Otherwise N/A)

PERFORM the following:

[16.1] **RAISE** RPV water level to 80 inches as indicated on RX WTR LEVEL FLOOD-UP, 2-LI-3-55.

[16.2] **MAINTAIN** RPV water level between +70 inches to +90 inches as indicated on RX WTR LEVEL FLOOD-UP, 2-LI-3-55.

[16.3] **INCREASE** frequency of monitoring reactor coolant temperature and pressure, using multiple indications.

[17] **IF** the affected loop of RHR cannot be placed back in Shutdown Cooling, **THEN** (Otherwise N/A)

PLACE the alternate loop of RHR in Shutdown Cooling.
REFER TO 2-OI-74.

[18] **IF** no Unit 2 RHR loop can be placed in Shutdown Cooling, **THEN** (Otherwise N/A)

PERFORM either of the following:

- **OBTAIN** Shift Manager approval and **PLACE** Unit 1 RHR loop in service, CROSS-TIED with Unit 2, for Shutdown Cooling. **REFER TO** 2-OI-74.
- **OBTAIN** Shift Manager approval and **PLACE** Unit 3 RHR loop in service, CROSS-TIED with Unit 2, for Shutdown Cooling. **REFER TO** 2-OI-74

[19] **IF** no RHR loops can be placed in service, **THEN** (Otherwise N/A)

ENSURE a Recirculation Pump in service.
REFER TO 2-OI-68.

Excerpts from Tech Spec 3.4.9:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 shall be completed if this Condition is entered. ---------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1	Restore parameter(s) to within limits.	30 minutes
	AND		
	A.2	Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	----------NOTES---------- 1. Only required to be performed during RCS heatup and cooldown operations or RCS inservice leak and hydrostatic testing when the vessel pressure is > 313 psig. 2. The limits of Figure 3.4.9-2 may be applied during nonnuclear heatup and ambient loss cooldown associated with inservice leak and hydrostatic testing provided that the heatup and cooldown rates are ≤ 15°F/hour. 3. The limits of Figures 3.4.9-1 and 3.4.9-2 do not apply when the tension from the reactor head flange bolting studs is removed. ---------- Verify: a. RCS pressure and RCS temperature are within the limits specified by Curves No. 1 and No. 2 of Figures 3.4.9-1 and 3.4.9-2; and b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.9-1, Curve No. 3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

Excerpt from Tech Spec Bases 3.4.8:

BASES (continued)

APPLICABLE SAFETY ANALYSES	Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. The RHR Shutdown Cooling System meets Criterion 4 of the NRC Policy Statement (Ref. 1).
LCO	Two RHR shutdown cooling subsystems are required to be OPERABLE, and when no recirculation pump is in operation, one RHR shutdown cooling subsystem must be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, one RHRSW pump capable of providing cooling to the heat exchanger, and the associated piping and valves. The subsystems have a common suction source and are allowed to have common discharge piping. Since piping is a passive component that is assumed not to fail, it is allowed to be common to the subsystems. In MODE 4, the RHR cross tie valve (FCV-74-46) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

(continued)

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

259002 (SF2 RWLCS) Reactor Water Level Control

A2.03 (10CFR 55.43.5 – SRO Only)

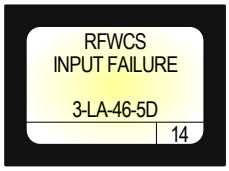
Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- *Loss of Reactor Water Level input*

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	259002A2.03	
Importance Rating	------	3.9

Proposed Question: **# 84**

Unit 3 is operating at 100% RTP with the following conditions:



- Reactor Water Level is as indicated
- RFWCS INPUT FAILURE, (3-9-6C, Window 14) alarms
- The OATC reports REACTOR WATER LEVEL NARROW RANGE 3-LI-3-60 (LEVEL B) and 3-LI-3-206 (LEVEL C) on Panel 3-9-5 indicate BYPASSED
- **NO** Operator Actions have been taken

Given the conditions above, complete the statements below.

3-LIC-46-5, REACTOR WATER LEVEL MASTER CONTROLLER PANEL DISPLAY STATION (PDS), is in **(1)** mode.

The NUSO is required to direct the OATC to bypass 3-LI-3-60 and 3-LI-3-206 in accordance with **(2)**.

Note: 3-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low

A. (1) MANUAL
(2) 3-AOI-3-1

B. (1) MANUAL
(2) 3-ARP-9-6C, Window 14

C. (1) AUTOMATIC
(2) 3-AOI-3-1

D. (1) AUTOMATIC
(2) 3-ARP-9-6C, Window 14

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part incorrect but plausible in that in accordance with 3-OI-3, Reactor Feedwater System, only when all four associated level instruments (3-53, 60, 206 and 253) are declared bad or invalid, will 3-LIC-46-5, REACTOR WATER LEVEL MASTER CONTROLLER PANEL DISPLAY STATION (PDS), trip to MANUAL mode. In this condition, RFWCS FAILED TO MANUAL, (3-9-6C, Window 21), would also be in alarm. Second part is incorrect but plausible in that REACTOR WATER LEVEL ABNORMAL, (3-9-5A, Window 8), alarms when Reactor Water Level reaches ≤ 27 inches or ≥ 39 inches. The Alarm Response Procedure (ARP) states a probable cause could be due to a sensor malfunction from 3-LT-3-53, 60, 206 or 253. The ARP also states that if the alarm is valid then refer to 3-AOI-3-1 or 3-OI-3. These procedures are not provided as references.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible (*See A*).

D **CORRECT**: (*See attached*) In accordance with 3-OI-3, Reactor Feedwater System, RFW Control System will allow up to three Narrow Range Reactor Level instruments to be bypassed at a time. Should the fourth level instrument fail with the other three instruments bypassed, 3-LIC-46-5, REACTOR WATER LEVEL MASTER CONTROLLER PANEL DISPLAY STATION (PDS), will trip to MANUAL mode. For second part, in accordance with the given Alarm Response Procedure (ARP) for RFWCS INPUT FAILURE, (3-9-6C, Window 14), RFWCS automatically bypasses a bad or invalid signal from the system. An amber light atop the respective instrument(s) illuminates when the signal has been automatically bypassed. However, in accordance with ARP for RFWCS INPUT FAILURE, (3-9-6C, Window 14) the NUSO is still required to direct the Operator to bypass the bad/invalid signal using 3-OI-3, Reactor Feedwater System.

SRO Level Justification: Tests the candidate's ability to predict the impact that a loss or faulty Reactor Water Level input instrument has on the Reactor Water Level Control System. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome as it relates to the Reactor Water Level Control System.

Technical Reference(s): 3-OI-3, Rev. 115 (Attach if not previously provided)
3-AOI-3-1, Rev. 17
3-ARP-9-5A, Rev. 56
3-ARP-9-6C, Rev. 30

Proposed references to be provided to applicants during examination: **RFWCS INPUT FAILURE, (3-9-6C, Window 14)**

Learning Objective: OPL171.012 Obj. 8 (As available)

Form 4.2-1 **Written Examination Question Worksheet**

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:
- 55.41
- 55.43 X

Excerpts from 3-ARP-9-6C:

BFN Unit 3	Panel 9-6 3-XA-55-6C	3-ARP-9-6C Rev. 0030 Page 21 of 49

RFWCS
INPUT
FAILURE
3-LA-46-5D

14

(Page 1 of 2)

Sensor/Trip Point:

3-RLY-46-5D
3-XM-46-97/196 Ch. 11

A process input to the RFW Control System declared Bad or Invalid.

A signal is declared bad when it has failed or is offscale. A signal is declared invalid when it fails the validation process described in 3-OI-3 Illustration 8.

Sensor Location: Panel 3-9-97 (Behind Panel 3-9-5).

Probable Cause: Any of the following inputs will cause this annunciator to alarm:

A. RFP A, B, or C Discharge Flow bad or invalid.
B. RFW A or B Line Flow bad or invalid.
C. Main Steam Line Flow A, B, C, or D bad or invalid.
D. Reactor Pressure (Wide Range) A, B, or C bad or invalid.
E. Reactor Pressure (Narrow Range) bad or invalid.
F. Reactor Water Level A, B, C or D bad or invalid.
G. Turbine First Stage Pressure bad.
H. RFW A or B Line Temperature bad.

Automatic Action:

A. RFWCS bypasses a bad or invalid signal from the system.
B. Amber light on the following instruments illuminates when the signal has been automatically bypassed:
   - RFP Discharge Flows, 3-FI-3-20, 13, 6 (Panel 3-9-6).
   - RFW Line Flows, 3-FI-3-78A, 78B (Panel 3-9-5).
   - Main Steam Line Flows, 3-FI-46-1, 2, 3, 4(Panel 3-9-5).
   - WR Reactor Pressure 3-PI-3-54, 61,207 (Panel 3-9-5).
   - Reactor Water Level 3-LI-3-53, 60,206,253 (Panel 3-9-5).

Operator Action:

A. **CHECK** RFWCS continues to maintain Reactor water level.
B. **IDENTIFY** bad/invalid signal by checking Control Room instrumentation and/or ICS. **REFER TO** ATTACHMENT 1 on NEXT PAGE for list of RFWCS instrumentation. **REFER TO** ICS RX FW LVL CONTROL SYS display (FWLCS).
C. **REQUEST** assistance from Site Engineering.
D. **BYPASS** the bad/invalid signal with Unit SRO approval. REFER TO 3-OI-3.

Continued on Next Page

BFN Unit 3	**Panel 9-6 3-XA-55-6C**	**3-ARP-9-6C Rev. 0030 Page 22 of 49**

RFWCS INPUT FAILURE 3-LA-46-5D, Window 14
(Page 2 of 2)

RFWCS CONTROL ROOM INSTRUMENTATION

* Denotes instrumentation with amber light indicating when the associated signal has been bypassed either manually or automatically by RFWCS.

FEEDWATER FLOW
*3A RFP Discharge Flow - 3-FI-3-20 - Panel 3-9-6
*3B RFP Discharge Flow - 3-FI-3-13 - Panel 3-9-6
*3C RFP Discharge Flow - 3-FI-3-6 - Panel 3-9-6
*FW Line A Flow - 3-FI-3-78A - Panel 3-9-5
*FW Line B Flow - 3-FI-3-78B - Panel 3-9-5
Total Feed Flow - 3-XR-3-53 - Panel 3-9-5

MAIN STEAM LINE FLOW
*Main Steam Line A Flow - 3-FI-46-1 - Panel 3-9-5
*Main Steam Line B Flow - 3-FI-46-2 - Panel 3-9-5
*Main Steam Line C Flow - 3-FI-46-3 - Panel 3-9-5
*Main Steam Line D Flow - 3-FI-46-4 - Panel 3-9-5
Total Main Steam Flow - 3-FR-46-5 - Panel 3-9-5

REACTOR PRESSURE
*Reactor Pressure (WR) A - 3-PI-3-54 - Panel 3-9-5
*Reactor Pressure (WR) B - 3-PI-3-61 - Panel 3-9-5
*Reactor Pressure (WR) C - 3-PI-3-207 - Panel 3-9-5
Average WR Pressure - 3-FR-3-53 - Panel 3-9-5
Reactor Pressure (NR) - 3-FR-3-53 - Panel 3-9-5

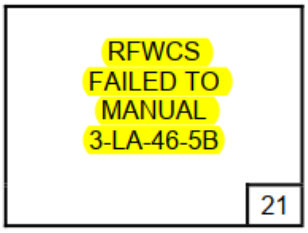
REACTOR WATER LEVEL
*Reactor Water Level A - 3-LI-3-53 - Panel 3-9-5
*Reactor Water Level B - 3-LI-3-60 - Panel 3-9-5
*Reactor Water Level C - 3-LI-3-206 - Panel 3-9-5
*Reactor Water Level D - 3-LI-3-253 - Panel 3-9-5
Average Reactor Water Level - 3-XR-3-53 - Panel 3-9-5

TURBINE FIRST STAGE PRESSURE
Turbine First Stage Steam Flow equivalent - 3-FR-46-5 - Panel 3-9-5

References: 3-729E895-10 3-45E620-7

Supports Distractors A1, B1

BFN Unit 3	Panel 9-6 3-XA-55-6C	3-ARP-9-6C Rev. 0030 Page 33 of 49



(Page 1 of 1)

Sensor/Trip Point:

3-RLY-46-5B
3-XM-46-97/196 Ch. 9

When all four Narrow Range

Reactor water level instruments

3-LI-53, 60,206, and 253 signals are declared bad or invalid by RFWCS.

A signal is declared bad when it fails or is offscale. A signal is declared invalid when it fails the validation process described in 3-OI-3 Illustration 8.

Sensor Location: Panel 3-9-97 (Behind 3-9-5), Unit 3 Main Control Room

Probable Cause: Reactor Water Level Control PDS, 3-LIC-46-5 on Panel 3-9-5 has been automatically transferred to MANUAL by RFWCS.

Automatic Action: When all four of the NR Reactor Water Level signals are declared bad or invalid by RFWCS, Reactor Water Level Control PDS, 3-LIC-46-5, transfers to MANUAL (if in AUTO).

Operation Action:

A. **CHECK** Reactor Water Level is stable.

B. **CONTROL** Reactor Water Level using of any of the following modes:
- Reactor Water Level Control PDS, 3-LIC-46-5 in MANUAL.
- Individual RFPT Speed Control PDS, 3-SIC-46-8 (9) (10), in AUTO or MANUAL.
- Individual RFPT Speed Control Raise/Lower switch in MANUAL GOVERNOR.

C. **REQUEST** assistance from Site Engineering and IMs.

References: 3-729E895-10 3-45E620-7

Excerpt from 3-OI-3:

BFN Unit 3	Reactor Feedwater System	3-OI-3 Rev. 0115 Page 110 of 331

8.2 Bypassing RFWCS Narrow Range Level Instruments

NOTES

1) RFW Control System will allow up to three Narrow Range Reactor Level instruments to be bypassed at a time. Should the fourth level instrument fail with the other three instruments bypassed, REACTOR WATER LEVEL CONTROL PDS, 3-LIC-46-5, will trip to MANUAL mode.

2) Attachment 8 can be referred to for general information on RFWCS instrumentation.

[1] **OBTAIN** approval from Unit SRO.

[2] **BYPASS** desired Narrow Range Reactor Level instrument by performing the following (Panel 3-9-5):

- To bypass LEVEL A, 3-LI-3-53, **DEPRESS** HS-3-53 BYPASS pushbutton.
- To bypass LEVEL B, 3-LI-3-60, **DEPRESS** HS-3-60 BYPASS pushbutton.
- To bypass LEVEL C, 3-LI-3-206, **DEPRESS** HS-3-206 BYPASS pushbutton.
- To bypass LEVEL D, 3-LI-3-253, **DEPRESS** HS-3-253 BYPASS pushbutton.

[3] **ENSURE** amber light illuminated on any bypassed level instrument.

[4] **IF** any level instrument bypassed in Step 8.2[2] was due to signal failure **AND** annunciation was in **NOT** in alarm prior to signal failure, **THEN**

ENSURE RFWCS INPUT FAILURE annunciation, 3-XA-55-6C Window 14, will reset. (Otherwise N/A)

[5] **REQUEST** assistance from Engineering and IMs.

[6] **ENSURE** RFW Control System continues to maintain Reactor water level.

Excerpt from 3-ARP-9A: Supports Distractors A2, C2

BFN Unit 3	Panel 9-5 3-XA-55-5A	3-ARP-9-5A Rev. 0056 Page 13 of 48

REACTOR
WATER LEVEL
ABNORMAL
3-LA-3-53

8

(Page 1 of 1)

Sensor/Trip Point:

RFW Control System

3-RLY-003-0053LL	≤ 27 inches
3-RLY-003-0053LH	≥ 39 inches

Average of valid Narrow Range level signals from RFW Control System. Signals originate from 3-LT-3-53, 3-60, 3-206, 3-253.

Sensor Location:
3-RLY-003-0053LL in Panel 3-9-97
3-RLY-003-0053LH in Panel 3-9-97
3-LT-3-53 on Panel 25-5B (Reactor Bldg. El 593', R17-T)
3-LT-3-60 on Panel 25-6B (Reactor Bldg El 593', R19-Q)
3-LT-3-206 on Panel 3-925-0426 (Reactor Bldg 593',R17-T)
3-LT-3-253 on Panel 25-6D (Reactor Bldg 593", R19-Q))

Probable Cause:
A. Reactor water level high or low.
B. Malfunction of sensor.

Automatic Action:
None
(Reactor scram on low level at +2.0 in)
(Main Turbine/RFPT trip and subsequent scram if ≥ 26% Reactor Power on high level at +55 in.)

Operator Action:
A. **CHECK** Reactor water level hi/low using multiple indications including Average Narrow Range Level on 3-XR-3-53 recorder, 3-LI-3-53, 3-LI-3-60, 3-LI-3-206, and 3-LI-3-253 on Panel 3-9-5.
B. **IF** alarm is valid, **THEN**
REFER TO 3-AOI-3-1 or 3-OI-3.
C. **IF** 3-LI-3-53, 3-LI-3-60, 3-LI-3-206, and 3-LI-3-253 has failed or is invalid, **THEN**
BYPASS with UNIT SRO permission, the affected level instrument.
REFER TO 3-OI-3, Section 8.2.
D. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function.
REFER TO NPG-SPP-18.3.5.

References:

3-45E620-6	3-729E895-10	3-47E610-46-1
3-729E895-1	3-AOI-3-1	3-OI-3

Excerpts from 3-AOI-3-1: Supports Distractors A2, C2:

BFN Unit 3	Loss Of Reactor Feedwater or Reactor Water Level High/Low	3-AOI-3-1 Rev. 0017 Page 3 of 15

1.0 PURPOSE

This instruction provides symptoms, automatic actions, and operator actions for total or partial loss of reactor feedwater or high/low reactor water level.

2.0 SYMPTOMS

A. Annunciators in alarm at greater than or equal to +55" (normal range level instruments 3-3-208A, 208B, 208C and 208D):

1. MAIN TURBINE SHUTDOWN (3-XA-55-8A, Window 11)
2. RFPT TRIPPED (3-XA-55-6C, Window 29)
3. RFPT A ABNORMAL (3-XA-55-6C, Window 1)
4. RFPT B ABNORMAL (3-XA-55-6C, Window 8)
5. RFPT C ABNORMAL (3-XA-55-6C, Window 15)

B. Annunciator HPCI TURBINE TRIPPED (3-XA-55-3F, Window 11) in alarm at greater than or equal to +51" if the steam admission valve is open (normal range level instruments 3-3-208B and 208D).

C. Annunciator REACTOR WATER LEVEL ABNORMAL (3-XA-55-5A, Window 8) in alarm at 27"or 39" (normal range).

D. Annunciators in alarm at 27"(normal range) if a RFP is tripped:

1. RECIRC LOOP A FLOW LIMITER ENFORCING (3-XA-55-4A, Window 35)
2. RECIRC LOOP B FLOW LIMITER ENFORCING (3-XA-55-4B, Window 35)
3. RFP DISCH FLOW LOW (3-XA-55-6C, Window 32)

E. Annunciators in alarm during a Feedwater Control System failure including Feedwater flow, Steam flow, Reactor Water Level input failures.

1. MAIN STEAM Line vs. MAIN TURB STEAM FLOW MISMATCH (3-XA-55-5B, Window 24)
2. RFWCS FAILED TO MANUAL (3-XA-55-6C, Window 21)
3. RFWCS GROSS FAILURE (3-XA-55-6C, Window 7)
4. RFWCS INPUT FAILURE (3-XA-55-6C, 14)
5. RFWCS TROUBLE (3-XA-55-6C, Window 28)

2.0 SYMPTOMS (continued)

F. Rising or lowering Reactor water level as indicated on:

1. RX VESSEL LEVEL (normal range), 3-XR-3-53 (3-9-5)
2. LEVEL A, 3-LI-3-53 (3-9-5)
3. LEVEL B, 3-LI-3-60 (3-9-5)
4. LEVEL C, 3-LI-3-206 (3-9-5)
5. LEVEL D, 3-LI-3-253 (3-9-5)
6. 3-LI-3-208A (3-9-5)
7. 3-LI-3-208B (3-9-3)
8. 3-LI-3-208C (3-9-3)
9. 3-LI-3-208D (3-9-5)

G. Annunciators in alarm on feedwater loss may include:

1. CNDS BSTR PUMP A SUCT PRESS LOW (3-XA-55-6A, Window 19)
2. CNDS BSTR PUMP B SUCT PRESS LOW (3-XA-55-6A, Window 20)
3. CNDS BSTR PUMP C SUCT PRESS LOW (3-XA-55-6A, Window 21)
4. CONDENSATE DEMIN ABNORMAL (3-XA-55-6B, Window 6)
5. RFPT TRIPPED (3-XA-55-6C, Window 29)
6. RFP DISCH FLOW LOW (3-XA-55-6C, Window 32)
7. RFPT A ABNORMAL (3-XA-55-6C, Window 1)
8. RFPT B ABNORMAL (3-XA-55-6C, Window 8)
9. RFPT C ABNORMAL (3-XA-55-6C, Window 15)
10. RFP A, B, OR C NPSH PRESS LOW (3-XA-55-6C, Window 19)
11. MOTOR TRIPOUT (3-XA-55-8C, Window 33)

H. Possible Reactor power reduction on reduced moderation for lowering reactor water level and/or Feedwater flow.

4.2 Subsequent Actions (continued)

[7] **IF** Reactor Water Level signal fails (LI-3-53, 60, 206, 253), **THEN**

OBTAIN Unit SRO's permission, to **PERFORM** the following:

[7.1] **BYPASS** failed level instrument on Panel 3-9-5.
REFER TO 3-OI-3.

[7.2] **IF** four level instruments fail, **THEN**

PERFORM the following:

[7.2.1] **CHECK** level control transfers to MANUAL.

[7.2.2] **MAINTAIN** level in MANUAL mode.

[8] **CHECK** all Safety/Relief Valves closed.

CAUTION

- Failure to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow may result in SJAE isolation.
- Changes in Condensate System flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190, either in the Control Room or locally. Personnel adjusting this valve locally are required to be in direct communication with the Control Room.

[9] **IF** RFPs are in automatic control (blue backlight illuminated), **THEN**

ENSURE 3-LIC-46-5 raises flow of operating RFPs.

[10] **IF** RFPs are in manual control (amber backlight illuminated), **THEN**

RAISE speed of operating RFPs.

Examination Outline Cross-reference:

295014 (APE 14) Inadvertent Reactivity Addition / 1

G2.1.37 (10CFR 55.43.6 - SRO Only)

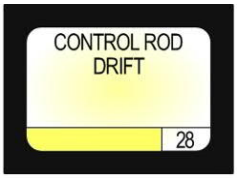
Knowledge of procedures, guidelines, or limitations associated with reactivity management

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295014G2.1.37	
Importance Rating	------	4.6

Proposed Question: **# 85**

Unit 2 is at 80% RTP with the following conditions:

- Control Rod 34-27 drifts out from position 40 to position 48
- CONTROL ROD DRIFT (2-XA-55-5A, Window 28) in alarm



Given the conditions above, the OATC will insert Control Rod 34-27 to position **(1)** and the NUSO will direct entry into **(2)**.

A. (1) 40
(2) 2-AOI-85-6, Rod Drift Out, **AND** 2-AOI-85-7, Mispositioned Control Rod

B. (1) 40
(2) 2-AOI-85-6, Rod Drift Out, **ONLY**

C. (1) 00
(2) 2-AOI-85-6, Rod Drift Out, **AND** 2-AOI-85-7, Mispositioned Control Rod

D. (1) 00
(2) 2-AOI-85-6, Rod Drift Out, **ONLY**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 2-AOI-85-7, Mispositioned Control Rod, directs, with NUSO permission and Shift Manager concurrence, moving a mispositioned Control Rod to its intended position if Operator directed movement resulted in the mispositioning. Second part is correct (*See C*).

B INCORRECT**:** First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that entry into 2-AOI-85-7, Mispositioned Control Rod, is often overlooked by candidates due to the name inferring entry is predicated on Operator action.

C **CORRECT**: *(See attached)* As provided in the stem of the question, Control Rod motion was due to drifting and not from Operator action. In accordance with 2-AOI-85-6, Rod Drift Out, when a Control Rod is moving from its intended position without Operator action, the Control Rod is to be inserted to 00. Likewise, 2-AOI-85-7, Mispositioned Control Rod, directs that a discovered mispositioned Control Rod not being moved (by an Operator) be inserted to 00. The procedures are consistent and ensure the most conservative actions are taken to mitigate the inadvertent reactivity addition.

For second part, entry into both 2-AOI-85-6 and 2-AOI-85-7 is required as partially stated in part (1) above. Entry into 2-AOI-85-6 is required when symptoms, such as those provided in the stem (e.g., CONTROL ROD DRIFT ALARM, changes in Control Rod position number on the Full Core Display) indicate a Control Rod drift out. Additionally, entry into 2-AOI-85-7 is also required when any mispositioned Control Rod criteria is met.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's knowledge of actions required for a drifting Control Rod and the required procedures to mitigate the event. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(2) Diagnosis that leads to selection of the procedures that should be used to respond to the evolution.

Technical Reference(s): 2-ARP-9-5A, Rev. 63 (Attach if not previously provided)
2-AOI-85-6, Rev. 23
2-AOI-85-7, Rev. 25

Proposed references to be provided to applicants during examination: **CONTROL ROD DRIFT (2-XA-55-5A, Window 28)**

Learning Objective: OPL171.005 Obj. 32 (As available)

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	**X**
Question History:	Last NRC Exam	

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

10 CFR Part 55 Content: 55.41
55.43 **X**

Excerpt from 2-ARP-9-5A:

BFN Unit 2	Panel 9-5 2-XA-55-5A	2-ARP-9-5A Rev. 0063 Page 37 of 48

CONTROL ROD DRIFT

28

(Page 1 of 1)

Sensor/Trip Point:

Relays 3A-K37A,B,C, & D Picked up by common relay for each individual rod drift in a given common relay group.

-K37A has 50 rods
-K37B has 43 rods
-K37C has 44 rods
-K37D has 48 rods

Sensor Location:
Panel 2-9-28
El 593'
Aux Instrument Room

Probable Cause:

A. Rod NOT selected, NOT being driven, and leaves even number reed switch.
   1. Leakage thru scram valves.
   2. Excessive cooling water pressure.

B. Rod Settle Timer timed out, and rod passes odd number reed switch.
   1. Drift-in following insert signal:
      a. Failure of insert valve to close.
   2. Drift-out following withdraw signal:
      a. Failure of withdraw valve to close.
      b. Failure of rod to latch after withdraw signal.

C. Defective RPIS alarm circuit.
D. Reactor scram or Rod scram.
E. Loss of 1 or more Rod Position indications.

Automatic Action:
None

Operator Action:

A. **DETERMINE** which rod is drifting from Full Core Display.
B. **IF** no control rod motion is observed, **THEN**
   RESET rod drift as follows.
   1. **PLACE** ROD DRIFT ALARM TEST switch, 2-HS-85-3A-S7, in RESET and **RELEASE**.
   2. **RESET** annunciator.

C. **IF** rod drifting in, **THEN**
   REFER TO 2-AOI-85-5 and 2-AOI-85-7.
D. **IF** rod drifting out, **THEN**
   REFER TO 2-AOI-85-6 and 2-AOI-85-7
E. **REFER TO** Tech Spec 3.1.3, 3.10.8.

References:

2-45E620-6	2-730E321-16	2-AOI-85-5
2-AOI-85-6	Technical Specifications	

Excerpts from 2-AOI-85-6:

BFN Unit 2	Rod Drift Out	2-AOI-85-6 Rev. 0023 Page 4 of 10

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a control rod drift out. This instruction incorporates NRC Commitment NCO-86-0326-021 and INPO Commitment BFC-83-0246-020.

2.0 SYMPTOMS

A. CONTROL ROD DRIFT annunciator, (2-XA-55-5A, Window 28) in alarm.

B. DRIFT status light on Full Core Display in alarm.

C. Control rod position number on Full Core Display changes.

D. If selected, control rod position number on 4-Rod Display changes.

E. If selected, LPRM readings around control rod change.

F. CONTROL ROD SELECT BLOCK TIMER MALFUNCTION annunciator, (2-XA-55-5A, Window 21) in alarm.

G. RBM HIGH/INOP annunciator, (2-XA-55-5A, Window 24) in alarm.

H. APRM HIGH annunciator, (2-XA-55-5A, Window 11) in alarm.

I. LPRM HIGH annunciator, (2-XA-55-5A, Window 12) in alarm.

J. IRM HIGH annunciator, (2-XA-55-5A, Window 26) in alarm.

K. SRM HIGH/INOP annunciator, (2-XA-55-5A, Window 13) in alarm.

L. SRM PERIOD annunciator, (2-XA-55-5A, Window 20) in alarm.

M. CONTROL ROD WITHDRAWAL BLOCK annunciator, (2-XA-55-5A, Window 7) in alarm.

3.0 AUTOMATIC ACTIONS

Rod withdrawal is blocked.

BFN Unit 2	Rod Drift Out	2-AOI-85-6 Rev. 0023 Page 5 of 10

4.0 OPERATOR ACTIONS

CAUTION

[NRC/C] Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. [NCO 940245010]

4.1 Immediate Actions

[1] **IF** multiple control rod drifts are identified, **THEN**

MANUALLY SCRAM the reactor and enter 2-AOI-100-1.

NOTE

If Reactor Power is less than or goes less than 22% RTP during Rod Travel. The RWM system may impose rod blocks.

4.2 Subsequent Actions

[1] **IF** Reactor Power is less than less than 22% RTP, **THEN**

MANUALLY BYPASS the Rod Worth Minimizer as follows:
(N/A this step after rod has been fully inserted.)

[1.1] **PLACE** RWM SWITCH PANEL, 2-XS-85-9025, in BYPASS.

[1.2] **CHECK** the Manual Bypass light is illuminated.

[1.3] **NOTIFY** the Unit SRO that Rod Worth Minimizer has been manually Bypassed. (Reference Tech Spec Sections 3.3.2.1 and 3.1.6)

[1.4] **WHEN** time permits carryout remaining actions of Manual Bypass of the Rod Worth Minimizer section in 2-OI-85.

[2] **IF** a Control Rod is moving from its intended position without operator actions, **THEN**

SELECT the drifting control rod and **INSERT** to the FULL IN (00) position.

Excerpts from 2-AOI-85-7:

BFN Unit 2	Mispositioned Control Rod	2-AOI-85-7 Rev. 0025 Page 4 of 11

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, and operator actions for mispositioned control rod.

Mispositioned Control Rod as defined by NPG-SPP-10.4 does not distinguish between mispositionings caused by equipment failure and those caused by human error since the Reactivity impact is identical. A mispositioned Control Rod is a control rod that meets any of the following criteria:

- Control rod left in a position other than the intended position and not corrected before completion of the confirmation step of the associated Control Rod Movement Instructions.
- Control rod moved more than one notch beyond the intended position.
- Control rod moved in the wrong direction.
- Wrong control rod moved.
- Control rod moved to a position inconsistent with the Reactivity Plan or associated predictions. (found in wrong position)
- Drifting control rod.
- Unexpected single control rod scram.

If the correct Control Rod is being moved from "00" to "08" notch wise, the intended notch position for the first movement is "02", for the second movement is "04", etc. For this example, (00 to 08 notch wise), the following criteria apply to the Control Rod with a single notch signal:

- The Intended Control Rod Position is Notch "02"
- If the Control Rod doubled notched to Notch "04" position the Control Rod would not be considered mispositioned.
- If the Control Rod were to triple notched to Notch "06" position the Control Rod would be mispositioned.

Likewise, if the correct Control Rod was being continuously withdrawn from "00" to "48" using the RONOR Switch, the intended position is the position the Operator intends the Control Rod to go to when the rod position switch is moved. This may be position "48", or if an expected rod block from the RBM occurs causing the rod to stop at an intermediate position, then the intended position would be where it stopped. In that case the rod would NOT be mispositioned and rod movement could continue.

BFN Unit 2	Mispositioned Control Rod	2-AOI-85-7 Rev. 0025 Page 8 of 11

4.2 Subsequent Actions (continued)

NOTE

If Reactor Power is less than or goes less than 22% RTP during Rod Insertion. The RWM system may impose rod blocks if not previously Bypassed.

[3] **IF** the wrong control rod is moved OR a control rod that was not being moved is found to be mispositioned, **THEN**

PERFORM the following: (Otherwise N/A)

[3.1] **INSERT** the mispositioned rod to "00".

[3.2] **IF** a Reactor Startup or Shutdown is not in progress, **THEN**

ENSURE 2-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. (Otherwise N/A)

[3.3] **IF** Reactor Power is less than 22% RTP and the Rod Worth Minimizer is not bypassed, **THEN**

MANUAL BYPASS the Rod Worth Minimizer per 2-OI-85, Manual Bypass of the Rod Worth Minimizer section as directed by the Unit SRO. (Otherwise N/A)

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection

G2.2.42 (10CFR 55.43.2 - SRO Only)

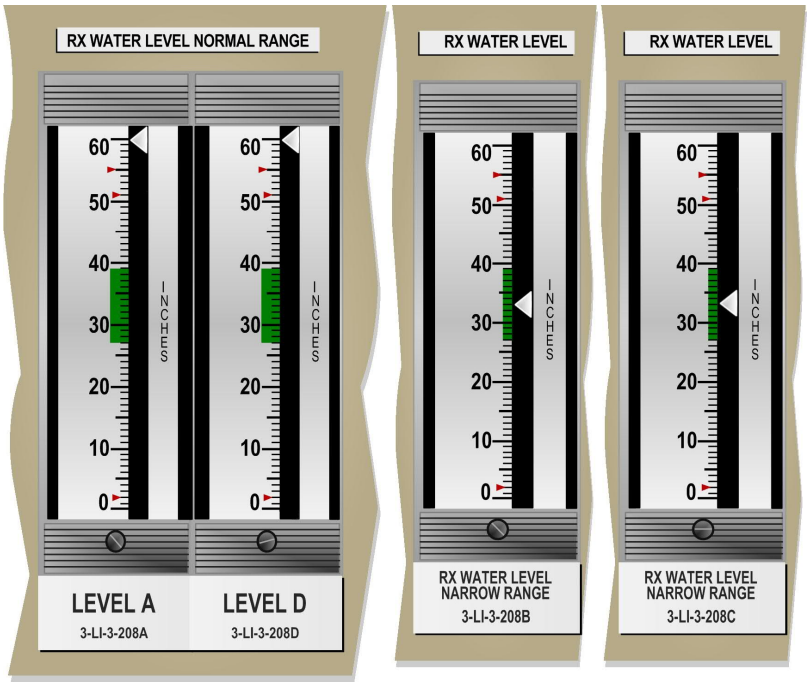
Ability to recognize system parameters that are entry-level conditions for technical specifications

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	206000G2.2.42	
Importance Rating	------	4.6

Proposed Question: **# 86**

Unit 3 is operating at 100% RTP with HPCI operating to support 3-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test, when the following occurs:

- **3-LI-3-208A and D**, RX WATER LEVEL NARROW RANGE, fails **UPSCALE**
- Indicated Reactor Water Level peaked as shown below



Given the conditions above, the Unit 3 HPCI Turbine is **CURRENTLY** **(1)**.

The NUSO is required to complete REQUIRED ACTION(s) **(2)** of the applicable Tech Spec.

[REFERENCE PROVIDED]

A. (1) tripped
(2) C.1 **AND** C.2

B. (1) tripped
(2) C.2 **ONLY**

C. (1) operating
(2) C.1 **AND** C.2

D. (1) operating
(2) C.2 **ONLY**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 3-LI-3-208A and 3-LI-3-208D are both upscale. Throughout ECCS, most trip logic is one-out-of-two taken twice logic (A or C AND B or D). An exception is the 3-LI-3-208s level instrument high Reactor Water Level trip logic (RFP, Main Turbine, HPCI, RCIC). This logic is (A AND C) **OR** (B AND D). For example, RCIC trips upon receipt of 208A AND 208C high Reactor Water Level. Second part is incorrect but plausible in that HPCI high Reactor Water Level trip logic requires both 208B AND 208D at or greater than 51 inches to actuate. From the stem, 208A and 208D are upscale and of these, only 208D affects HPCI. Referencing Tech Spec Table 3.3.5.1-1, upon failure of the 208D (Function 3c, Reactor Vessel Water Level – High, Level 8), Condition C is entered from REQUIRED ACTION A.1. If the candidate misapplies the REQUIRED ACTION C.1 note and Function, then this would result in selecting both REQUIRED ACTIONs C.1 and C.2

B INCORRECT: First part is incorrect but plausible *(See A)* Second part is correct (*See D*).

C INCORRECT: First part is correct *(See D).* Second part is incorrect but plausible *(See A).*

D **CORRECT:** HPCI high Reactor Water Level trip logic requires both **3-LI-3-208B and D**, RX WATER LEVEL NARROW RANGE, instruments (2-out-of-2 once logic) to be greater than or equal to 51 inches to actuate. Given in the stem above, only 208A and 208D are upscale. Since 208B is not greater than or equal to 51 inches, HPCI continues to operate.
For second part, from the stem above, it can be determined that 208A and 208D instruments have failed upscale. Of those, only 208D affects HPCI. In accordance with Tech Spec Table 3.3.5.1 1, Function 3c, Reactor Vessel Water Level – High, Level 8, requires two channels in the current MODE. With 208D failed only one REQUIRED channel is OPERABLE. In this CONDITION, evaluation of Tech Spec 3.3.5.1 concludes that CONDITION 'C' is entered from REQUIRED ACTION A.1. REQUIRED ACTION C.1 is prefaced by a note specifying that C.1 is only applicable for Functions 1.c, 1.e, 2.c, 2.d and 2.f. Thus, C.1 is not applicable to Function 3.c. Therefore, only REQUIRED ACTION C.2 is applicable.

SRO Level Justification: Tests the candidate's ability to recognize system parameters that are entry-level conditions for Technical Specifications associated with HPCI. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

Technical Reference(s): U3 Tech Spec 3.3.5.1, Amend. 213, 294, 305, (Attach if not previously provided)

U3 Tech Spec Bases 3.3.5.1, Rev. 125

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **Unit 3 Tech Spec 3.3.5.1 (No Bases) 3-LI-3-208A, B, C, and D, REACTOR WATER LEVEL NARROW RANGE instrument indications**

Learning Objective: OPL171.275, Obj. 2 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 22-04 #83	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2022	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

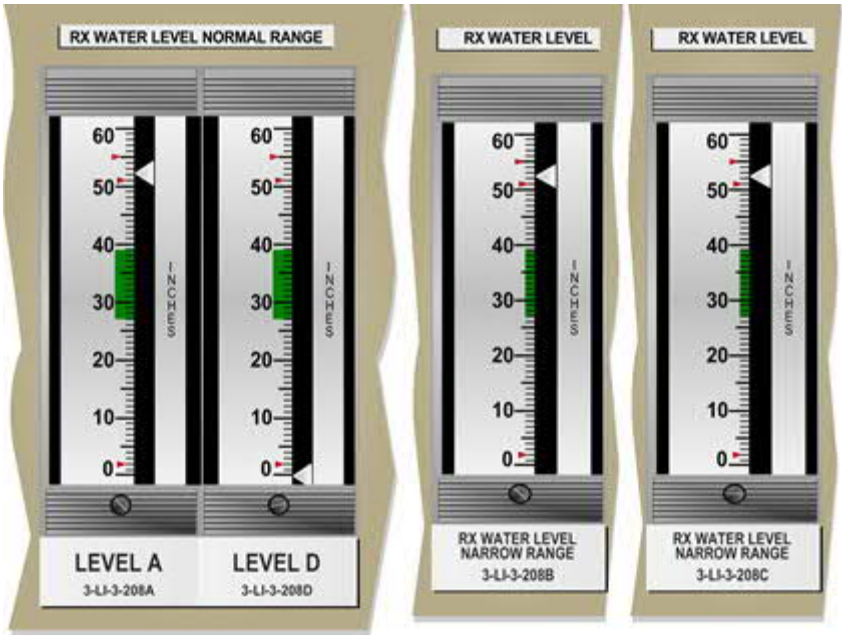
55.43 **X**

Comments:

Copy of Bank Question:

Unit 3 is operating at 100% RTP when a Feedwater Control System malfunction occurs with the following conditions:

- **3-LI-3-208D**, RX WATER LEVEL NARROW RANGE, fails **DOWNSCALE**
- Reactor Water Level peaked as indicated
- Tech Spec 3.3.2.2, Condition A has been entered



Given the conditions above, the Unit 3 Main Turbine is **CURRENTLY** **(1)**.

A few minutes later, **3-LI-3-208C**, RX WATER LEVEL NARROW RANGE, fails **DOWNSCALE** in the next minute, LCO 3.3.2.2 entry into Condition B **(2)** required.

[REFERENCE PROVIDED]

A. (1) tripped
(2) is

B. (1) tripped
(2) is **NOT**

C. (1) operating
(2) is

D. (1) operating
(2) is **NOT**

Excerpts from U3 Tech Spec 3.3.5.1:

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1 ----------NOTE---------- Only applicable for Functions 1.c, 1.e, 2.c, 2.d, and 2.f. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for features in both divisions
	AND	
	C.2 Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1 ----------NOTE---------- Only applicable if HPCI pump suction is not aligned to the suppression pool. ---------- Declare HPCI System inoperable.	1 hour

(continued)

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level — Low Low, Level 2(f)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero
b. Drywell Pressure — High(f)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level — High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level — Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level — High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow—Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level — Low Low Low, Level 1(f)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

(f) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

Excerpts from U3 Tech Spec Bases 3.3.5.1:

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.c. Reactor Vessel Water Level - High, Level 8 (LIS-3-208B and 208D) (continued)

Reactor Vessel Water Level - High, Level 8 signals for HPCI are initiated from two level transmitters from the narrow range water level measurement instrumentation. The Reactor Vessel Water Level - High, Level 8 Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSLs.

Two channels of Reactor Vessel Water Level - High, Level 8 Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.d. Condensate Header Level - Low (LS-73-56A and B)

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally the suction valves between HPCI and the CST are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CST. However, if the water level in the HPCI pump supply header from the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

(continued)

BASES

ACTIONS C.1 and C.2 (continued)

The Note states that Required Action C.1 is only applicable for Functions 1.c, 2.c, 1.e, 2.d, and 2.f. Required Action C.1 is also not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable). The loss of one Function 3.c channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference 4 and considered acceptable for the 24 hours allowed by Required Action C.2.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that redundant features in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

(continued)

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215003 (SF7 IRM) Intermediate-Range Monitor	Tier #	------	2
A2.03 (10CFR 55.43.5 - SRO Only)	Group #	------	1
Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:	K/A #	215003A2.03	
• Stuck detector	Importance Rating	------	3.0

Proposed Question: **# 87**

The following conditions exist on Unit 3:

- 3-HS-99-5A-S1, REACTOR MODE SWITCH, is in START HOT STANDBY
- 3-OI-92A, IRMs, Preparing IRMs for Reactor Startup, is in progress
- IRM 'H' just failed high and has been bypassed

Subsequently, the IRM 'B' IN backlight failed to extinguish when the DRIVE OUT pushbutton was depressed and held, indicating IRM 'B' was stuck.

Given the conditions above, in accordance with Tech Spec 3.3.1.1, RPS Instrumentation, the REQUIRED ACTION that must be performed is based on **(1)** and is completed using **(2)**.

[REFERENCE PROVIDED]

A. **(1)** CONDITION A
(2) 3-OI-99, Reactor Protection System

B. **(1)** CONDITION A
(2) 3-OI-92A, Intermediate Range Monitors

C. **(1)** CONDITION B
(2) 3-OI-99, Reactor Protection System

D. **(1)** CONDITION B
(2) 3-OI-92A, Intermediate Range Monitors

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See Attached)* In accordance with Tech Spec 3.3.1.1, Reactor Protection System (RPS) Instrumentation, in MODE 2, three (3) IRM detectors (A, C, E, G or B, D, F, H) are required to be OPERABLE in each IRM channel (A or B) per each RPS trip system (A or B, respectively). With both the B and H IRM inoperable in the B RPS trip system, only two of the required three required IRM channels are OPERABLE. Therefore, CONDITION A is applicable. For second part, direction for placing a channel of RPS in trip is contained in 3-OI-99, Reactor Protection System – Attachment 3 associated with Tech Spec Table 3.3.1.1 – 1 for IRMs Function 1.

B INCORRECT**:** First part is correct *(See A).* Second part is incorrect but plausible in that 3-OI-92A, Intermediate Range Monitors, provides direction for preparing IRMs for startup, inserting and withdrawing IRMs, and bypassing IRMs.

C INCORRECT: First part is incorrect but plausible in that there are eight IRM channels split between two RPS trip systems. It is easily confused which channels belong to which trip system. Incorrectly determining IRM channel and associated RPS trip system assignments leads to the plausibility that candidate could determine one or more required IRM channels are INOPERABLE in both trip systems (CONDITION B). Second part is correct (*See A*).

D INCORRECT**:** First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's ability to predict the impacts of stuck detectors on the IRM system, and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations. To correctly answer this question, the candidate must understand how IRM channels are arranged in relation to RPS trip systems, recognize that it is not possible to bypass two faulty IRMs in the same RPS trip system, apply Tech Specs to a failure of two IRMs in one RPS trip system, and select the correct procedure for placing an RPS channel in trip.

SRO only because of the links to 10CFR55.43 (2): Facility operating limitations in the technical specifications and their bases, and 10CFR43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency conditions.

This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution as it relates to IRMs and RPS Systems and Tech Specs.

Technical Reference(s): U3 Tech Spec 3.3.1.1, Amend. 221, 283 (Attach if not previously provided)
3-OI-99, Rev. 68
3-OI-92A, Rev. 18

Proposed references to be provided to applicants during examination: **U3 Tech Spec 3.3.1.1 (No Bases)**

Learning Objective: OPL171.020, Obj. 11 (As available)

Form 4.2-1 **Written Examination Question Worksheet**

Question Source:	Bank #	BFN 1804 #88	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2018	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Form 4.2-1 **Written Examination Question Worksheet**

Copy of Bank Question:

Examination Outline Cross-reference:

215003 (SF7 IRM) Intermediate-Range Monitor

A2.06 (10CFR 55.43.5 - SRO Only)

Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- Faulty range switch

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	215003A2.06	
Importance Rating	------	3.2

Proposed Question: **# 88**

The following conditions exist on Unit 3:

- A Reactor Startup is in progress, Reactor Period is 160 seconds
- IRM 'H' failed high on the previous shift and has been bypassed

Subsequently, when the OATC attempts to range IRM 'B' at an indication of 65, the range switch fails and IRM 'B' indication does NOT respond to the range change and continues to rise.

Which ONE of the following completes the statements below?

The required action required by Tech Spec 3.3.1.1, RPS Instrumentation, is **(1)**.

Tech Spec required action is completed using **(2)**.

[REFERENCE PROVIDED]

A. **(1)** Condition A
(2) 3-OI-99, Reactor Protection System

B. **(1)** Condition A
(2) 3-OI-92A, Intermediate Range Monitors

C. **(1)** Condition B
(2) 3-OI-99, Reactor Protection System

D. **(1)** Condition B
(2) 3-OI-92A, Intermediate Range Monitors

Proposed Answer: **A**

Excerpts from Tech Spec 3.3.1.1:

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	OR		
	A.2	----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. --- Place associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 26% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 13% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.55 W + 65.5% RTP and ≤ 120% RTP(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(c) [0.55 W + 65.5% - 0.55 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Excerpt from 3-OI-99:

BFN Unit 3	Reactor Protection System	3-OI-99 Rev. 0068 Page 108 of 108

Attachment 3
(Page 17 of 17)

Actions to Place RPS Instruments in Tripped Condition (Tech Spec Table 3.3.1.1-1)

NOTE
Nuclear Instrumentation Function corresponds to the Tech Spec Table 3.3.1.1 Functions.

NUCLEAR INSTRUMENTATION	TRIP INSERT	REMARKS
IRMS Function: 1	PLACE MODE SWITCH FOR AFFECTED DRAWER IN POSITION OTHER THAN OPERATE	REACTOR MODE SWITCH NOT IN RUN: IRMS RESULTS IN HALF SCRAM OF RPS A (ACEG) OR RPS B (BDFH). REACTOR MODE SWITCH IN RUN: WILL NOT RESULT IN A HALF SCRAM.
APRMS Function: 2	PLACE KEYLOCK SWITCH FOR AFFECTED CHANNEL IN "INOP" POSITION.	ONE CHANNEL RESULTS IN AN APRM HIGH/INOP AND A ROD BLOCK. NO RPS OUTPUT FROM THE VOTERS UNTIL 2 OF THESE CONDITIONS ARE RECEIVED, THEN THE VOTERS INITIATE A FULL SCRAM.

Excerpt from 3-OI-92A: Supports Distractors B2, D2

BFN Unit 3	Intermediate Range Monitors	3-OI-92A Rev. 0018 Page 3 of 20

Table of Contents

1.0 PURPOSE 4

2.0 REFERENCES 4

2.1 Technical Specifications 4

2.2 Technical Requirements Manual-TRM 4

2.3 Final Safety Analysis Report 4

2.4 Plant Instructions 4

2.5 Plant Drawings 5

2.6 Vendor Manuals 5

2.7 Miscellaneous Documents 5

3.0 PRECAUTION AND LIMITATIONS 6

4.0 PRESTARTUP/STANDBY READINESS REQUIREMENTS 9

5.0 STARTUP 10

5.1 Preparing Intermediate Range Monitors for Reactor Startup 10

6.0 SYSTEM OPERATIONS 12

6.1 Bypassing an IRM Channel 12

6.2 Returning an IRM to Service from the Bypassed Condition 13

6.3 Withdrawing IRMs 14

6.4 Inserting IRMs 17

7.0 SHUTDOWN 19

8.0 INFREQUENT OPERATIONS 19

Attachment 1: IRM Trip Outputs 20

EXTERNAL ATTACHMENTS

Attachment 1: None

Attachment 2: Intermediate Range Monitor System Panel Lineup Checklist Unit 3

Attachment 3: Intermediate Range Monitor System Electrical Lineup Checklist Unit 3

Attachment 4: Intermediate Range Monitor System Instrument Inspection Checklist Unit 3

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

295010 (APE 10) High Drywell Pressure / 5

G2.1.20 (10CFR 55.43.5 – SRO Only)

Ability to interpret and execute procedure steps

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295010G2.1.20	
Importance Rating	------	4.6

Proposed Question: **# 88**

Unit 2 was operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual Reactor SCRAM was inserted
- Drywell Pressure is 4.0 psig and rising
- Primary Containment Hydrogen indicates 2.5%

Given the conditions above, 2-EOI-2, PRIMARY CONTAINMENT CONTROL, entry is required based on **(1)** and the NUSO will direct the performance of **(2)**.

Note: 2-EOI Appendix-12, Primary Containment Venting
2-EOI Appendix-13, Emergency Venting Primary Containment

A. (1) Drywell Pressure **ONLY**
(2) 2-EOI Appendix-12

B. (1) Drywell Pressure **ONLY**
(2) 2-EOI Appendix-13

C. (1) Drywell Pressure **AND** Hydrogen Concentration
(2) 2-EOI Appendix-12

D. (1) Drywell Pressure **AND** Hydrogen Concentration
(2) 2-EOI Appendix-13

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Hydrogen concentration EOI entry requirement is often mistaken and missed by candidates. Drywell Pressure of 2.45 psig is significantly more predominant in usage throughout ECCS logic and procedural guidance, and is therefore plausible as the only entry. Lastly, triggers for actions based on Hydrogen concentration are much higher in other charts such as SAMGs as compared to the EOIs. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible if the candidate confuses the given conditions with those that would threaten the core as noted in Note 6 of 2-EOI-2. However, even if the candidate assumed the core was threatened, Drywell Pressure is still well below that of the required band listed Note 6 and no other pressure control mechanisms have been attempted yet.

C **CORRECT**: (*See attached*) In accordance with 2-EOI-2, PRIMARY CONTAINMENT CONTROL, entry is required at 2.45 psig Drywell Pressure AND also at 2.4% Hydrogen concentration. For second part, the direction in 2-EOI-2 to vent using 2-EOI Appendix-12 is contingent upon the expectation that offsite releases will remain within ODCM limits. There is nothing in the stem of the question that indicates that release rates will be exceeded therefore the PC Press specifically directs venting using 2-EOI-Appendix-12 based on Hydrogen concentration.

D INCORRECT: First part is correct *(See C).* Second part is incorrect but plausible *(See B).*

SRO Level Justification: Test the candidate's ability to interpret and execute procedure steps associated with high Drywell pressure. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate at least two different parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome. The candidate must read the multiple indications correctly, integrate them together, and recall the specific facts associated with each to reach a conclusion.

In accordance with NUREG-1021, Revision 12, Section 4.2.B.2.a, this question is related to:
(2) Diagnosis that leads to selection of the procedures that should be used to respond to the evolution.

Technical Reference(s): 2-EOI-2, Rev. 18 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.203 Obj. 2, 17 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

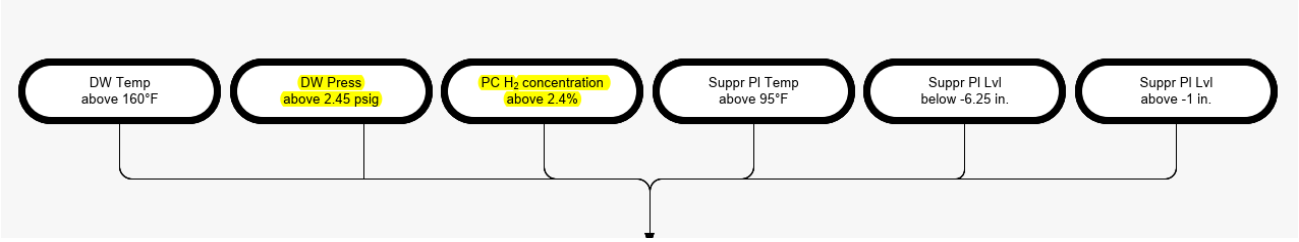
Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Excerpts from 2-EOI-2:

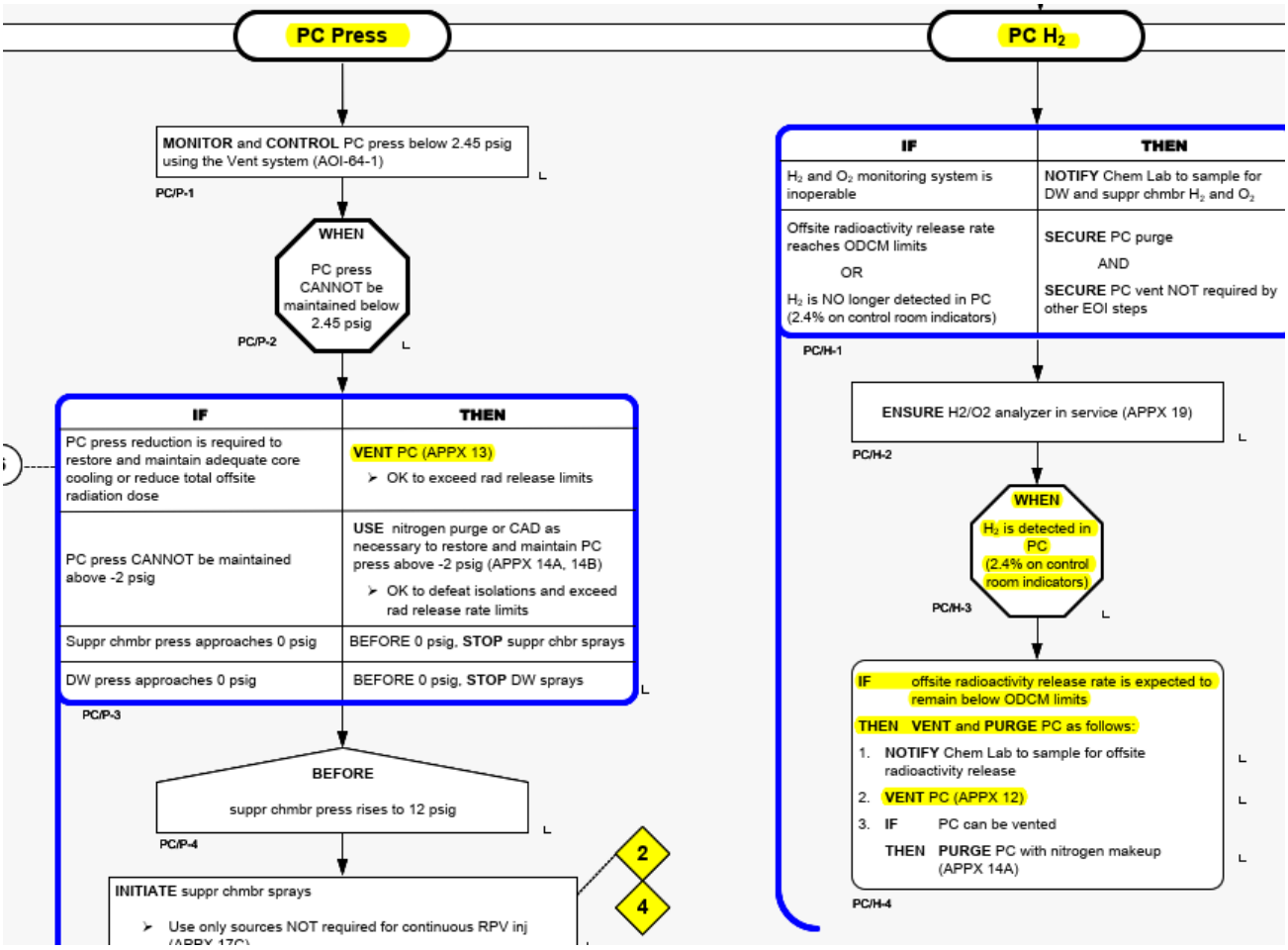


NOTE

6 PC venting may be useful to:

- Restore and maintain adequate core cooling:
  - Lower RPV and PC pressure below injection source discharge pressure
  - Discharge RCIC or HPCI exhaust outside PC during ELAP
- Reduce total offsite dose if:
  - PC integrity has been lost
  - Significant fuel damage is anticipated
  - Suppression chamber vent will be submerged
  - Further degradation of conditions is expected and available personnel resources are limited
- Avoid primary containment challenges if full RPV depressurization did not occur after exceeding a containment limit requiring emergency RPV depressurization

PC pressure control band: 10 to 15 psig provides sufficient margin to NPSH limits and ensures pressure suppression capability


PC Press
MONITOR and CONTROL PC press below 2.45 psig using the Vent system (AOI-64-1)
PC/P-1
WHEN
PC press CANNOT be maintained below 2.45 psig
PC/P-2
IF
THEN
PC press reduction is required to restore and maintain adequate core cooling or reduce total offsite radiation dose
VENT PC (APPX 13)
OK to exceed rad release limits
PC press CANNOT be maintained above -2 psig
USE nitrogen purge or CAD as necessary to restore and maintain PC press above -2 psig (APPX 14A, 14B)
OK to defeat isolations and exceed rad release rate limits
Suppr chmbr press approaches 0 psig
BEFORE 0 psig, STOP suppr chbr sprays
DW press approaches 0 psig
BEFORE 0 psig, STOP DW sprays
PC/P-3
BEFORE
suppr chmbr press rises to 12 psig
PC/P-4
INITIATE suppr chmbr sprays
Use only sources NOT required for continuous RPV inj
2
4
PC H2
IF
THEN
H2 and O2 monitoring system is inoperable
NOTIFY Chem Lab to sample for DW and suppr chmbr H2 and O2
Offsite radioactivity release rate reaches ODCM limits
OR
H2 is NO longer detected in PC (2.4% on control room indicators)
SECURE PC purge
AND
SECURE PC vent NOT required by other EOI steps
PC/H-1
ENSURE H2/O2 analyzer in service (APPX 19)
PC/H-2
WHEN
H2 is detected in PC (2.4% on control room indicators)
PC/H-3
IF offsite radioactivity release rate is expected to remain below ODCM limits
THEN VENT and PURGE PC as follows:
1. NOTIFY Chem Lab to sample for offsite radioactivity release
2. VENT PC (APPX 12)
3. IF PC can be vented
THEN PURGE PC with nitrogen makeup (APPX 14A)
PC/H-4

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

262001 (SF6 AC) AC Electrical Distribution

A2.10 (10CFR 55.43.5 – SRO Only)

Ability to (a) predict the impacts of the following on the AC Electrical Distribution (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- Exceeding current limitations

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	262001A2.10	
Importance Rating	------	3.4

Proposed Question: **# 89**

Unit 1 is operating at 100% RTP with the following conditions:

At 1200:

- 0-SR-3.8.1.1(A), Diesel Generator 'A' Monthly Operability Test, is in progress
- 'A' EDG is loaded with DG 'A' MODE SELECT switch in PARALLEL WITH SYSTEM
- Standby Gas Train 'C' is INOPERABLE

At 1215:

- DIESEL GENERATOR 'A' OVERLOAD (1/2-9-23A, Window 11) alarms



Given the conditions above, this alarm indicates 'A' EDG phase output current has reached a **MINIMUM** of **(1)**.

In accordance with Tech Specs, Unit 1 entry into LCO 3.0.3 is required at **(2)**.

[REFERENCE PROVIDED]

A. (1) 427.8 amps
(2) 1215

B. (1) 427.8 amps
(2) 1615

C. (1) 495 amps
(2) 1215

D. (1) 495 amps
(2) 1615

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible since 427.8 amps is the max motor amps at which a Condensate Booster Pump is allowed to operate. Second part is incorrect but plausible if the candidate fails to accurately apply Tech Spec 3.8.1 CONDITION B.3 which requires declaring required feature(s) ('A' SGT), supported by the INOPERABLE Unit 1 and 2 EDG ('A' EDG), INOPERABLE when the redundant required feature(s) ('C' SGT) are INOPERABLE with a completion time of 4 hours from discovery of the condition concurrent with INOPERABILITY of redundant required features.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is correct (*See D*).

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible *(See A)*

D **CORRECT**: (*See attached*) In accordance with the given Alarm Response Procedure for DIESEL GENERATOR 'A' OVERLOAD (1/2-9-23A, Window 11), the alarm is received at greater than or equal to 495 amps. For second part, as given 'C' Standby Gas Treatment (SGT) is INOPERABLE. At 1215 when the 'A' EDG overload is received, therefore the 'A' EDG should be declared INOPERABLE and Tech Spec 3.8.1 CONDITION B evaluated and entered. REQUIRED ACTION B.3 requires declaring required feature(s) 'A' SGT, supported by the INOPERABLE Unit 1 and 2 EDGs ('A' EDG), INOPERABLE when the redundant required feature(s) are INOPERABLE 'C' SGT with a completion time of 4 hours from discovery of the condition concurrent with INOPERABILITY of redundant required features. The resulting INOPERABILITY time for 'A' SGT would be 1615. Upon declaring 'A' SGT INOPERABLE at 1615, Tech Spec 3.6.4.3 CONDITIONs A and C would be entered with CONDITION 'C' REQUIRED ACTION to enter LCO 3.0.3 immediately (1615).

SRO Level Justification: Tests the candidate's ability to (a) predict the impacts of the current limitations on the AC Electrical Distribution (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to correctly assemble given parameters from abnormal plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

Technical Reference(s): 1-OI-2, Rev. 42 (Attach if not previously provided)

1/2-ARP-9-23A, Rev. 37

U1 Tech Spec 3.8.1, Amend. 249

U1 Tech Spec 3.6.4.3, Amend. 311

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **DIESEL GENERATOR 'A' OVERLOAD (1/2-9-23A, Window 11)**
Unit 1 Tech Spec 3.8.1,
Unit 1 Tech Spec 3.6.4.3
(No Bases)

Learning Objective: OPL171.038 Obj. 14 (As available)

Question Source:

Bank #	
Modified Bank #	ILT EXAM BANK OPL171.036-15 001 #1294
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	**X**

10 CFR Part 55 Content:

55.41	
55.43	**X**

Comments:

Copy of Bank Question:

QUESTIONS REPORT
for ILT Exam Bank 08 22 2018

1294. OPL171.036-15 001

The "A" DG monthly run is currently in progress. "A" DG is in "Parallel with System" and carrying load.

Two minutes later, 1/2-ARP-9-23A Window 11, DIESEL GEN A OVERLOAD, alarms.

This alarm indicates, ___**(1)**___.

Assuming the alarm is valid, ___**(2)**___.

A. 1. DG output voltage ≥ 4400 volts
   2. "A" 4kV Shutdown Board NORMAL and ALTERNATE feeder breakers are tripped open and are **NOT** locked out

B✓ 1. DG phase output current ≥ 495 amps
   2. "A" 4kV Shutdown Board NORMAL and ALTERNATE feeder breakers are tripped open and are **NOT** locked out

C. 1. DG output voltage ≥ 4400 volts
   2. "A" 4kV Shutdown Board NORMAL and ALTERNATE feeder breakers are tripped open and are locked out

D. 1. DG phase output current ≥ 495 amps
   2. "A" 4kV Shutdown Board NORMAL and ALTERNATE feeder breakers are tripped open and are locked out

Excerpt from 1/2-9-23A:

BFN Unit 1 & 2	Panel 0-9-23-7, 0-XA-55-23A	1/2-ARP-9-23A Rev. 0037 Page 16 of 52

DIESEL GEN A OVERLOAD

11

(Page 1 of 1)

Sensor/Trip Point:

Aux Relay:	Relay:	
(51X)	(51)	DG "B" phase output current ≥495 amps
0-51X-082-2547A	051-082-2547A	

Sensor Location: Protective Relay Cabinet (25-47A)
Elevation 565
Diesel Generator Building

Probable Cause:
A. Improper loading of generator.
B. Malfunction of voltage regulator.
C. Malfunction of speed controller.
D. Mode switch failure.
E. Sensor malfunction.

Automatic Action: If Diesel Generator is in Parallel with System, relay 51X1 will actuate and trip Shutdown Bd A normal (1614), 0-BKR-211-000A/003, and alternate (1716), 0-BKR-211-000A/024, feeder breakers.

Operator Action:
A. **CHECK** diesel indication on panel 0-9-23:
- Voltage and amps.
- Reactive and kW loading.
- Mode switch light indication. (More than one mode light illuminated indicates failure of mode switch.)

B. **IF** diesel is **NOT** needed, **THEN**
RETURN Shutdown Board to its normal power source, **UNLOAD** and **SECURE** diesel. **REFER TO** 0-OI-82.

C. **IF** it is necessary to continue diesel operation, **THEN**
1. **IF** diesel is parallel with system, **THEN**
ENSURE auto action occur.
2. **REFER TO** capability curve in 0-OI-82.
3. **REDUCE** load immediately to stay within limits.
4. **REMOVE** all unnecessary loads.

References: 0-45E767-4 0-45E765-22 0-45E765-3

Excerpts from Tech Spec 3.8.1:

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Unit 1 and 2 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Verify power availability from the remaining OPERABLE offsite transmission network. AND	1 hour AND Once per 8 hours thereafter (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one shutdown board concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Restore required offsite circuit to OPERABLE status.	7 days AND 21 days from discovery of failure to meet LCO
B. One required Unit 1 and 2 DG inoperable.	B.1	Verify power availability from the offsite transmission network.	1 hour AND Once per 8 hours thereafter
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Evaluate availability of both temporary diesel generators (TDGs). AND	1 hour AND Once per 12 hours thereafter
	B.3 Declare required feature(s), supported by the inoperable Unit 1 and 2 DG, inoperable when the redundant required feature(s) are inoperable. AND	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	B.4.1 Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure. OR	24 hours
	B.4.2 Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s). AND	24 hours (continued)

Excerpt from Tech Spec 3.6.4.3:

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SGT subsystem inoperable.	A.1	Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours
C. Two or three SGT subsystems inoperable.	C.1	Enter LCO 3.0.3.	Immediately

Excerpt from 1-OI-2: Supports Distractors A1, B1

BFN Unit 1	Condensate System	1-OI-2 Rev. 0042 Page 14 of 114

3.4 Condensate Booster Pumps (continued)

J. There is adequate Flow/NPSH to maintain 100% Power when one Reactor Feedwater Pump and one Condensate Booster Pump (3-2-2) are taken out of service and the following conditions are met:

1. Three Condensate Pumps are in service.
2. The Reactor Feedwater Pump is removed from service and secured (No flow through the minimum flow valve) prior to removing the Condensate Booster Pump from service.
3. Condensate flow above 14.1MLBM/HR (approximately 87.5% reactor power) can result in the CBP motors operating above rated horsepower but within rated service factor.
4. While operating within the rated service power, the CBO motor winding temperatures are not to exceed 266 degrees F and the motor amps are not to exceed 427.8 amps.

3.5 System Requirements

A. [II/C] Condensate flow should always be maintained within the following limits, using CNDS FLOW CONTROL SHORT CYCLE, 1-FIC-2-29 in AUTO if possible, to prevent Condensate Pump damage: [II-B-91-158]

1. One Condensate Pump (only) in operation, greater than 1.5 X 10^6 lbm/hr but less than 7.5 x 10^6 lbm/hr.
2. One Condensate Pump and one Condensate Booster Pump in operation, greater than 2.6 X 10^6 lbm/hr but less than 7.5 x 10^6 lbm/hr.
3. Two Condensate Pump operation, greater than 3.0 X 10^6 lbm/hr but less than 15 x 10^6 lbm/hr.
4. Two Condensate Pump and Two Condensate Booster Pump in operation, greater than 5.3 x 10^6 lbm/hr but less than 15 x 10^6 lbm/hr (An indication that condensate booster pump minimum flow has been achieved is amperage for each operating condensate booster pump at greater than 200 amps).
5. Three Condensate Pump operation, greater than 4.5 X 10^6 lbm/hr but less than 17 x 10^6 lbm/hr.
6. Three Condensate Pump and Three Condensate Booster Pump in operation, greater than 7.8 x 10^6 lbm/hr.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)

G2.4.16 (10CFR 55.43.5 - SRO Only)

Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	262002G2.4.16	
Importance Rating	------	4.4

Proposed Question: **# 90**

Units 1, 2, and 3 are operating at 100% RTP in their normal electrical lineups when:

- 250V DC Battery Board 4 is deenergized
- 480V Shutdown Board 2A is deenergized

Given the conditions above, Unit 2 Panel 9-9 Cabinet 5, Unit Non Preferred,

(1) automatically transfer and the NUSO will direct the crew to take actions in accordance with **(2)**.

Note: 0-AOI-57-6, Loss of Plant Non Preferred
2-AOI-57-4, Loss of Unit Preferred

A. **(1)** will
(2) 2-AOI-57-4

B. **(1)** will
(2) 0-AOI-57-6

C. **(1)** will NOT
(2) 2-AOI-57-4

D. **(1)** will NOT
(2) 0-AOI-57-6

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the BFN DC and 120VAC Systems transfer schemes are complex and often confused. The normal power supply to each Unit's Panel 9-9 Cabinet 5, Unit Non Preferred, is through Battery Board 1 or 2 or 3 Panel 11, Unit Preferred, respectively. The candidate could easily confuse the power supply scheme for Unit 2's Panel 9-9-Cabinet 6, Unit Preferred, which does in fact auto transfer. Second part is correct *(See C).*

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that the candidate could easily confuse the given losses between Unit Preferred, Plant Preferred or Unit Non Preferred. The restoration for Unit's Panel 9-9 Cabinet 5, UNIT Non Preferred, is oddly directed from the 2-AOI-57-4, Loss of UNIT Preferred. This is easily confused with the fact that Battery Board (BB) 2 Panel 13 PLANT Preferred (alternately powered from BB4) is the normal supply to Panel 9-9 Cabinet 4, Plant Preferred. BB2 Panel 14 Plant NON-Preferred is an alternate supply for each Unit's Panel 9-9 Cabinet 5, UNIT NON-Preferred, thus the candidate may incorrectly select 0-AOI-57-6, Loss of PLANT NON-Preferred.

C **CORRECT:** *(See Attached)* With the complexity of BFN's DC and 120VAC Systems, candidates struggle keeping the Units, Batteries and Boards correctly separated. The 250V DC Battery Board (BB) 4 is the normal power supply to the Unit 2 Unit Preferred MMG set **DC** motor. Transfer to alternate is manual. With the given loss of 480V Shutdown Board 2A, the Unit 2 Unit Preferred MMG **AC** motor has lost power and is no longer providing power to 250V DC BB2, Panel 11 Unit Preferred; therefore, it is de-energized from the 1001 breaker. The 1st alternate power supply (Unit Preferred Transformer from 480V RMOV Board 2B from the 1002 breaker) requires manual transfer. The 2nd alternate (Unit 3's MMG set from the 1003 breaker) also requires manual transfer. This in turn causes a loss of primary power supply to Unit 2's Panel 9-9-Cabinet 6, Unit Preferred, (which auto transfers to Unit 3's BB3 Panel 11) **and** Unit 2's Panel 9-9-Cabinet 5, Unit NON-Preferred. Panel 9-9-Cabinet 5 does **NOT** auto transfer, requiring a **MANUAL** transfer to the alternate power supply from BB2 Panel 14, Plant NON-Preferred. For second part, given that now the Unit 2 250V DC BB2, Panel 11, is de-energized, thereby de-energizing Unit 2's Panel 9-9-Cabinet 5 (as stated above), the NUSO will direct the crew to respond in accordance with Unit 2 AOI-57-4, Loss of Unit Preferred. This allows MANUAL transfer to either the 1st or 2nd alternates for Unit 2 250V DC BB2, Panel 11, which will also then resupply 9-9-Cabinet 5. Additionally, 2-AOI-57-4 provides the option to re-energize Unit 2's Panel 9-9-Cabinet 5 from BB2 Panel 14, Plant NON Preferred (see 2-AOI-57-4, Step 4.2[5]).

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible *(See B).*

SRO Level Justification: Tests the candidate's ability to predict the impacts that a loss of a Battery Board (DC) supply as well as an AC supply will have on the Uninterruptable Power Supply System. Based on the given losses, the SRO candidate will determine the correct procedure for this abnormal plant condition. SRO only because of the links to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency conditions. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-ARP-9-8B, Rev. 19 (Attach if not previously provided)

2-AOI-57-4, Rev. 53
0-AOI-57-6, Rev. 20
OPL171.102, Rev. 11U1
0-45E702-3, Rev. 29
0-45E702-2, Rev. 50

Form 4.2-1 **Written Examination Question Worksheet**

0-45E2647-5, Rev. 22

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.102, Obj. 2.e (As available)

Question Source:	Bank #	BFN 19-09 #86	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2019	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 86**

Units 1, 2, and 3 are all operating at 100% RTP in their normal electrical lineups when:

- 250V DC Battery Board 4 is lost

Given the conditions above, which **ONE** of the following completes the statements below?

As a result, if AC power is **SUBSEQUENTLY** lost to the affected Unit's 9-9 Cabinet 5, automatic transfer of Cabinet 5 **(1)** occur.

As the Unit Supervisor, the mitigative actions for the given abnormal conditions would be directed in accordance with **(2)**.

Note: 0-AOI-57-6, Loss of Plant Non Preferred
2-AOI-57-4, Loss of Unit Preferred

A. **(1)** will
(2) 2-AOI-57-4

B. **(1)** will
(2) 0-AOI-57-6

C. **(1)** will NOT
(2) 2-AOI-57-4

D. **(1)** will NOT
(2) 0-AOI-57-6

Proposed Answer: **C**

Excerpt from a Loss of Unit Preferred ARP (not given) directing the use of 2-AOI-57-4 for the given stem indications:

BFN Unit 2	Panel 9-8 2-XA-55-8B	2-ARP-9-8B Rev. 0019 Page 40 of 41

UNIT PFD SUPPLY ABNORMAL

BLUE BAR 35

(Page 1 of 2)

Sensor/Trip Point:

Relay SE - loss of normal DC power source.

Relay TS - DC transfer switch in emergency position.

Relay 2MS-DC motor starts.

Relay 3K-AC generator overvoltage, under frequency, or loss of output voltage.

Relay 13K - difference between battery voltage and open circuit DC motor voltage greater than -7V DC or greater than +12V DC.

Sensor Location: 250V DC Battery Bd 2
El 593'

Probable Cause:

A. Loss of normal DC power source, UNIT PFD DC NORM/EMERG SELECTOR, 2-HS-252-02 in EMERG.
B. DC power transfer.
C. DC motor starts.
D. AC generator overvoltage.
E. AC generator under frequency.
F. AC generator loss of output voltage.
G. Relay failure.
H. Sensor malfunction.
I. Voltage differential between battery and motor is greater than -7V DC or greater than +12V DC.
J. Burned out light bulb on voltage indicating switch which activates Relay 13K.

Automatic Action: None

Operator Action:

A. **IF** 120V AC Unit Preferred is lost, **THEN** **REFER TO** 2-AOI-57-4.

B. **NOTIFY** Unit SRO, Unit 1 and Unit 3.

C. After determining situation, **REFER TO** appropriate portion of 0-OI-57C.

Continued on Next Page

Excerpts from 2-AOI-57-4:

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for Loss of Unit Preferred (Battery Board 2 Panel 11, Control Room Panel 2-9-9 Cabinet 5 Non-preferred and Unit Preferred to Panel 2-9-9 Cabinet #6).

NOTE

A loss of the Unit Preferred power source results in the following boards being de-energized:

- Battery Board 2 Panel 11
- Panel 2-9-9 Cabinet 5 Non-preferred
- Panel 2-9-9 Cabinet 6 Unit Preferred (if panel does **NOT** auto transfer to alternate)

2.0 SYMPTOMS

A. Loss of indication to CRD rod control indicator lights, Panel 2-9-5 (i.e. rod out permissive, timer switch malfunction - select block, rod withdraw, insert and settle, notch override and stabilizing valve indicator lights.)

B. The following recorders fail downscale (Panels 2-9-6, 2-9-7, 2-9-8):

1. TURB EXP AND TEMP MN XFMR Winding Temp (2-XR-47-20/2-TR-47-20)

2. RFPT/RFP VIB and ECC (2-XR-3-177).

C. PANEL 2-9-9 CABINET 1, 2,3 or 6 CONTROL POWER TRANSFER (2-XA-55-7A, Window 15), on Panel 2-9-7, is in alarm on transfer of Unit Preferred Cabinet 6.

D. Unit 2 Panel 2-9-9, Cabinet 6 UNIT PFD 120V AC NORMAL and ALTERNATE SUPPLY lights 2-XI-57-600A and 2-XI-57-600B extinguish.

E. UNIT PFD SUPPLY ABNORMAL (2-XA-55-8B, Window 35) on Panel 2-9-8 is in alarm.

3.0 AUTOMATIC ACTIONS

A. Panel 2-9-9, Cabinet 6, Unit Preferred automatically transfers to the alternate power source.

B. CRD SYS FLOW CONTROL VLV 1A / B, 2-FCV-85-11 A / B CLOSES on loss of Cabinet 6 and OPENS if only Cabinet 5 is de-energized.

C. 2-FCV-1-58, UPSTREAM MSL DRAIN TO CONDENSER and BYPASS VLV UPSTREAM MSL DRAIN TO CONDENSER, 2-FCV-1-185 auto open.

D. Loss of one out of two auctioneered power sources for EHC Control System (This includes the PLU) (ICS is the other power source).

E. Loss of power to Panel 2-9-46.

F. Loss of one out of two auctioneered power sources for RFPT 2C Woodward Governor and Final Driver (ICS is the other power source).

G. Loss of one out of two auctioneered power sources for Recirc Flow Control System (ICS is the other power source).

WARNING

With cabinets 9-9-5 and 9-9-6 de-energized 2-LPNL-925-8635 (TSI Monitoring Panel) has no power. ALL TSI monitoring is lost. **TBWD WILL NOT BE ABLE TO TRIP MAIN TURBINE.** Energizing cabinet 9-9-5 or 9-9-6 shall be Expedited

H. Loss of power to Turbine Supervisory Instrumentation Panel 2-925-8635 (If cabinet 6 fails to transfer)

4.2 Subsequent Actions (continued)

NOTE

CRD SYS FLOW CONTROL VLV 1A/B, 2-FCV-85-11A/B closes on a loss of power to Panel 2-9-9 Cabinet 6 Unit Preferred resulting in a loss of normal cooling water flow to CRD seals. CRD temperatures should be monitored and operation with 2-FCV-85-11A/B closed, limited to less than 1 hour. 2-FCV-85-11A/B can be manually opened, if required. **REFER TO** 2-OI-85.

[4] **PERFORM** the following for the CRD system:

[4.1] **MONITOR** CRD temperatures while 2-FCV-85-11(A/B) are closed.

[4.2] **IF** CRD seal temperatures rise to the alarm setpoint OR the Unit Preferred system cannot be restored within one hour, **THEN**

DISPATCH personnel to MANUALLY OPEN 2-FCV-85-11(A/B). **REFER TO** 2-OI-85 (Otherwise N/A).

[4.3] **IF** Cabinet 5 is de-energized and Cabinet 6 is energized **THEN**

PLACE controller 2-FIC-85-11 in manual and adjust until normal Drive Water Pressure is obtained.

[5] To re-energize PANEL 2-9-9 CABINET 5, **PERFORM** the following from PANEL 2-9-9 CABINET 5 NON-PREFERRED: (Otherwise N/A).

[5.1] **OPEN** the supply breaker (normal or alternate) for the light that is EXTINGUISHED (2-XI-57-500A for normal, 2-XI-57-500B for alternate).

[5.2] **CLOSE** the supply breaker (normal or alternate) for the light that is ILLUMINATED (2-XI-57-500A for normal, 2-XI-57-500B for alternate).

[5.3] **ENSURE** the Off Gas System is lined up and operating properly. **REFER TO** 2-OI-66.

4.2 Subsequent Actions (continued)

NOTE

The lights being extinguished in the following step can indicate a loss of power, or a blown fuse in the indicating light circuit. If replacing the fuse fails to illuminate the lights, a loss of power is the probable cause for the light being extinguished. These lights are wired across two phases of a three phase supply and thus may be lit with a loss of a single phase.

[6] **IF** Unit Preferred Panel 2-9-9 Cabinet 6 has **NOT** auto-transferred, **THEN** (Otherwise N/A).

PERFORM the following:

[6.1] **ENSURE CLOSED** breaker 1102 at Battery Board 3 Panel 11.

[6.2] **CHECK** fuses F1 and F2 for indicating light circuit at Panel 2-9-9 Cabinet 6 and **REPLACE** if cleared (terminal block BC 2-FU1-252-0600BA and 2-FU1-252-0600AA).

NOTES

1) Upon re-energization of the Unit Preferred Bus (Battery Board 2 Panel 11) Panel 2-9-9 Cabinet 6 Unit Preferred will auto-transfer back to the normal source. As this transfer takes place, a momentary loss of UPS, with all associated alarms and the actions may occur.
2) Consideration should be given to only closing the 1002 breaker if Cabinet 6 failed to transfer and cannot be placed on its alternate power supply OR waiting until the unit is in a stable condition.
3) 0-OI-57C will be required to complete the following steps.

CAUTION

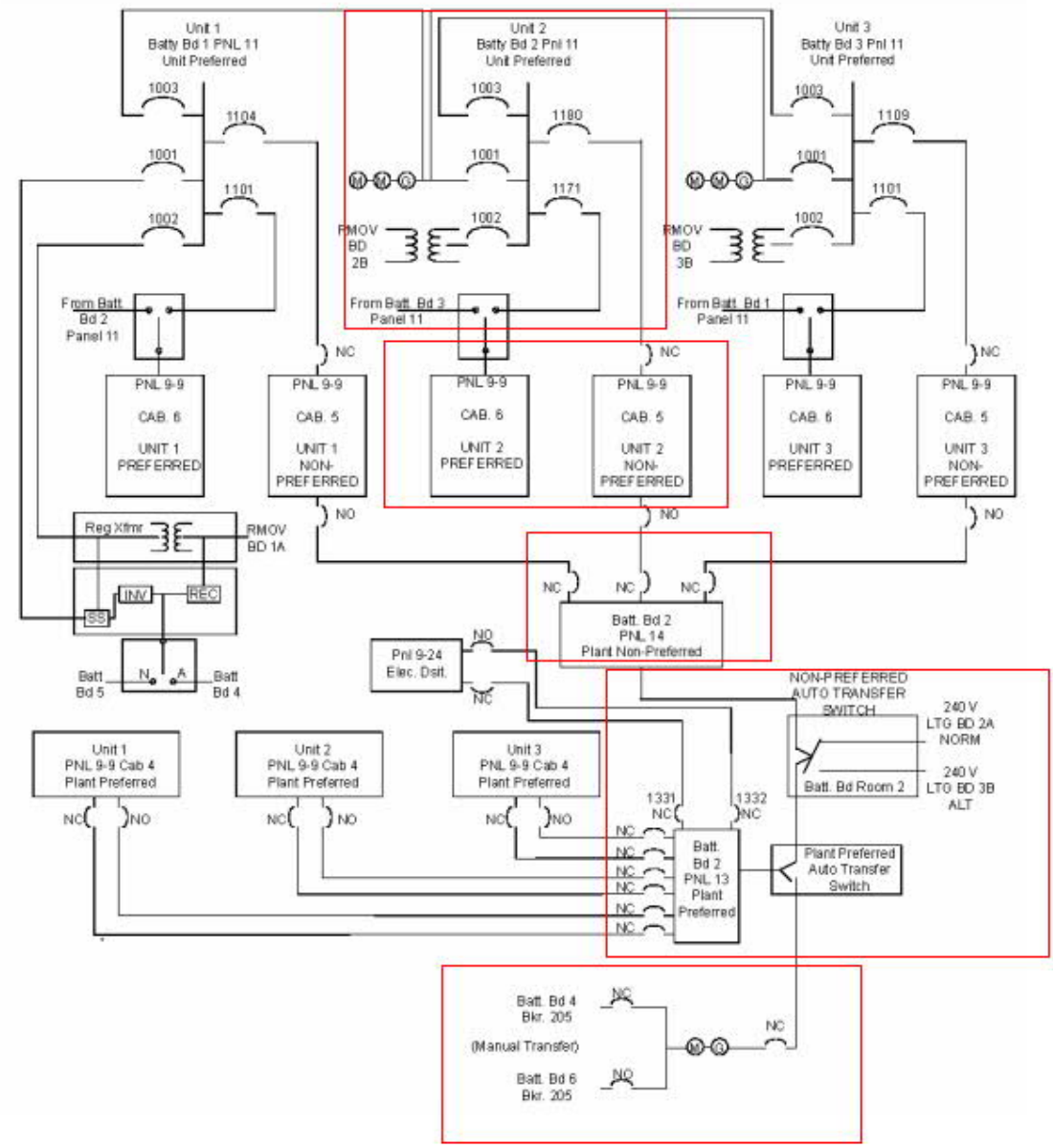
Step 4.2[7] should **NOT** be completed if the Unit Preferred Transformer is in service to RPS. If this condition exists, Step 4.2[8] in required to be performed in order to align the 2nd alternate supply (Unit 3 Unit Preferred MMG) to Unit 2.

[7] **PERFORM** the following at Battery Board 2 Sync and Speed Control Panel (Battery Board 2 Panel 11), to re-energize the Unit Preferred Bus from the transformer: (Otherwise N/A).

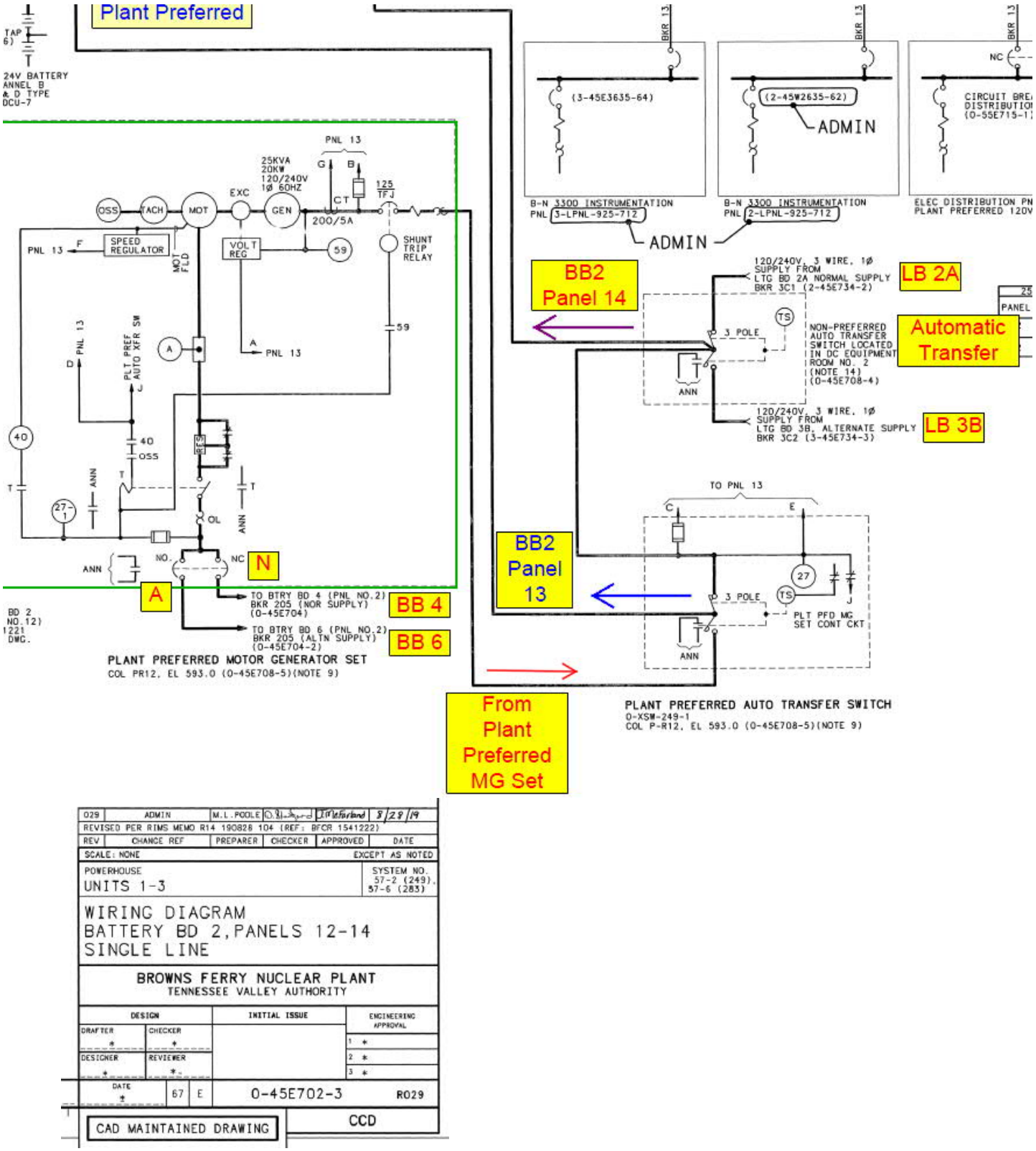
[7.1] **ENSURE** the Unit Preferred transformer is **NOT** in service to RPS.

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0053 Page 18 of 32

Attachment 1
(Page 1 of 1)
Vital 120V AC Distribution



Excerpt from 0-45E702-3 showing BB4 as DC supply to PLANT Preferred MG via auto transfer switch (ABT):



029	ADMIN	M.L.POOLE			8/28/19
REVISED PER RIMS MEMO R14 190828 104 (REF: BFCR 1541222)					
REV	CHANGE REF	PREPARER	CHECKER	APPROVED	DATE
SCALE: NONE				EXCEPT AS NOTED	

POWERHOUSE UNITS 1-3 — SYSTEM NO. 57-2 (249), 57-6 (283)

WIRING DIAGRAM
BATTERY BD 2, PANELS 12-14
SINGLE LINE

BROWNS FERRY NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY

DESIGN		INITIAL ISSUE	ENGINEERING APPROVAL
DRAFTER *	CHECKER *		1 *
DESIGNER *	REVIEWER *		2 *
DATE *			3 *

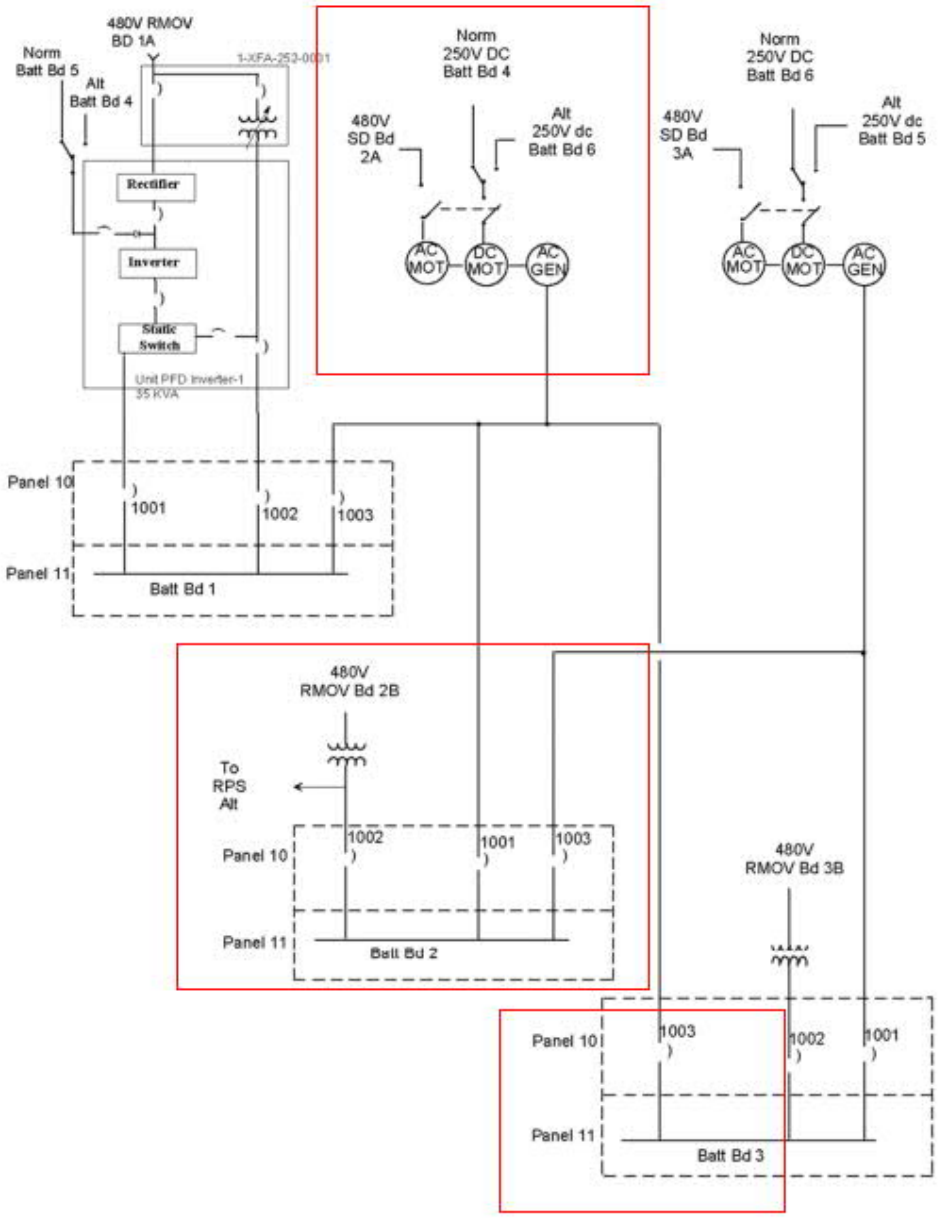
67 E 0-45E702-3 R029

CAD MAINTAINED DRAWING — CCD

Excerpt from OPL171.102 Lesson Plan: Illustrates 250V DC BB4 as normal DC supply for the Unit 2 Unit Preferred MMG:

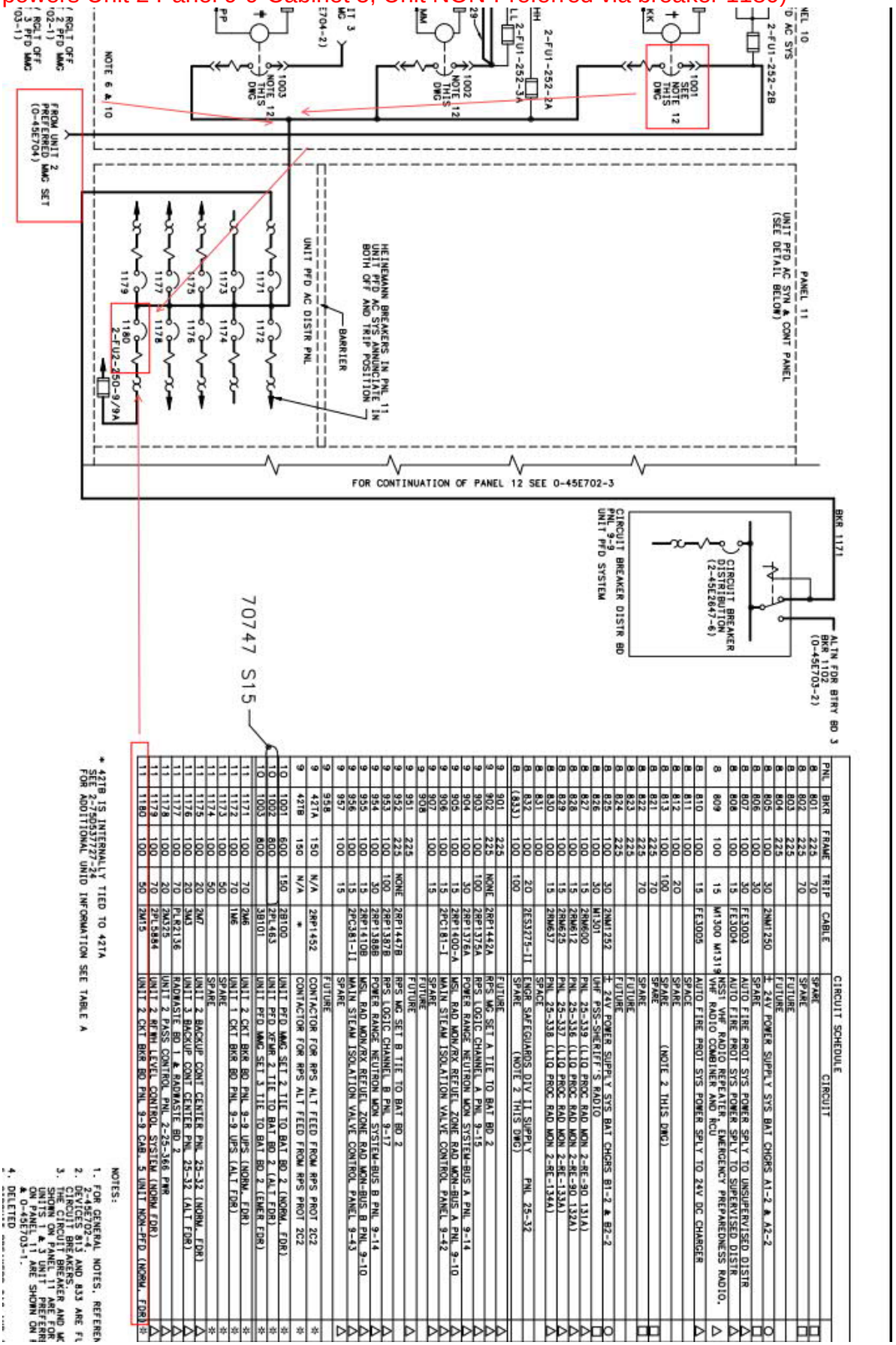
OPL171.102 , 120V AC Power Supplies and Distribution Systems, Rev# 11U2

Appendix A: Illustrations
IL-2: Power Supplies to UPS Battery Board Cabinets



QA Record. Non-RP - Retain in ECM (Lifetime Retention)
RP LPs - Retain in ECM (Life of Nuclear Insurance Policy, plus 10 years)

Excerpt of 0-45E702-2: (Illustrates how 250V DC BB2, Panel 11 Unit Preferred breaker 1001 powers Unit 2 Panel 9-9-Cabinet 5, Unit NON-Preferred via breaker 1180)

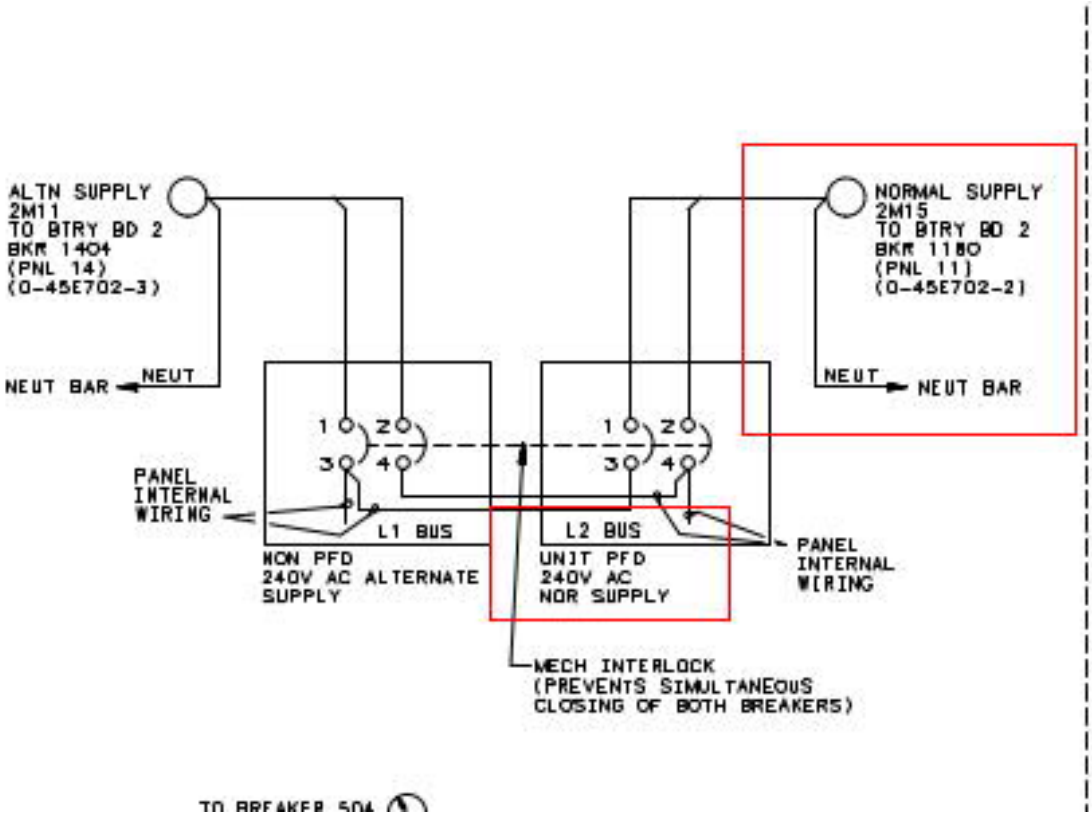


PNL	BKR	FRAME	TRIP	CABLE	CIRCUIT SCHEDULE CIRCUIT
8	801	225	70		SPARE
8	802	225	70		SPARE
8	803	225			FUTURE
8	804	225			FUTURE
8	805	100	30	2NM1250	+ 24V POWER SUPPLY SYS BAT CHGRS A1-2 & A2-2
8	806	100	30		SPARE
8	807	100	30	FE3003	AUTO FIRE PROT SYS POWER SPLY TO UNSUPERVISED DISTR
8	808	100	15	FE3004	AUTO FIRE PROT SYS POWER SPLY TO SUPERVISED DISTR
8	809	100	15	M1300 M1319	NSS1 VHF RADIO REPEATER, EMERGENCY PREPAREDNESS RADIO, VHF RADIO COMBINER AND RCU
8	810	100	15	FE3005	AUTO FIRE PROT SYS POWER SPLY TO 24V DC CHARGER
8	811	100			SPACE
8	812	100	20		SPARE
8	813	100	100		SPARE (NOTE 2 THIS DWG)
8	821	225	70		SPARE
8	822	225	70		SPARE
8	823	225			FUTURE
8	824	225			FUTURE
8	825	100	30	2NM1252	+ 24V POWER SUPPLY SYS BAT CHGRS B1-2 & B2-2
8	826	100	30	M1301	UHF PSS-SHERIFF'S RADIO
8	827	100	15	2RM600	PNL 25-339 (LIQ PROC RAD MON 2-RE-90 131A)
8	828	100	15	2RM612	PNL 25-336 (LIQ PROC RAD MON 2-RE-90 132A)
8	829	100	15	2RM625	PNL 25-337 (LIQ PROC RAD MON 2-RE-133A)
8	830	100	15	2RM637	PNL 25-338 (LIQ PROC RAD MON 2-RE-134A)
8	831	100			SPACE
8	832	100	20	2ES3275-II	ENGR SAFEGUARDS DIV II SUPPLY PNL 25-32
8	(833)	100	100		SPARE (NOTE 2 THIS DWG)
9	901	225			FUTURE
9	902	225	NONE	2RP1442A	RPS MG SET A TIE TO BAT BD 2
9	903	100	100	2RP1375A	RPS LOGIC CHANNEL A PNL 9-15
9	904	100	30	2RP1376A	POWER RANGE NEUTRON MON SYSTEM-BUS A PNL 9-14
9	905	100	15	2RP1400-A	MSL RAD MON/RX REFUEL ZONE RAD MON-BUS A PNL 9-10
9	906	100	15	2PC181-I	MAIN STEAM ISOLATION VALVE CONTROL PANEL 9-42
9	907	100			SPARE
9	908				FUTURE
9	951	225			FUTURE
9	952	225	NONE	2RP1447B	RPS MG SET B TIE TO BAT BD 2
9	953	100	100	2RP1387B	RPS LOGIC CHANNEL B PNL 9-17
9	954	100	30	2RP1388B	POWER RANGE NEUTRON MON SYSTEM-BUS B PNL 9-14
9	955	100	15	2RP1410B	MSL RAD MON/RX REFUEL ZONE RAD MON-BUS B PNL 9-10
9	956	100	15	2PC381-II	MAIN STEAM ISOLATION VALVE CONTROL PANEL 9-43
9	957	100	15		SPARE
9	958				FUTURE
9	42TA	150	N/A	2RP1452	CONTACTOR FOR RPS ALT FEED FROM RPS PROT 2C2
9	42TB	150	N/A	*	CONTACTOR FOR RPS ALT FEED FROM RPS PROT 2C2
10	1001	600	150	2B100	UNIT PFD MMG SET 2 TIE TO BAT BD 2 (NORM. FDR)
10	1002	800		2PL483	UNIT PFD XFMR 2 TIE TO BAT BD 2 (ALT FDR)
10	1003	800		3B101	UNIT PFD MMG SET 3 TIE TO BAT BD 2 (EMER FDR)
11	1171	100	70	2M6	UNIT 2 CKT BKR BD PNL 9-9 UPS (NORM. FDR)
11	1172	100	70	1M6	UNIT 1 CKT BKR BD PNL 9-9 UPS (ALT FDR)
11	1173	100	50		SPARE
11	1174	100	50		SPARE
11	1175	100	20	2M7	UNIT 2 BACKUP CONT CENTER PNL 25-32 (NORM. FDR)
11	1176	100	20	3M3	UNIT 3 BACKUP CONT CENTER PNL 25-32 (ALT FDR)
11	1177	100	70	PLR2136	RADWASTE BD 1 & RADWASTE BD 2
11	1178	100	20	2M325	UNIT 2 PASS CONTROL PNL 2-25-366 PWR
11	1179	100	70	2PL5884	UNIT 2 MFW LEVEL CONTROL SYSTEM (NORM FDR)
11	1180	100	50	2M15	UNIT 2 CKT BKR BD PNL 9-9 CAB. 5 UNIT NON-PFD (NORM. FDR)

* 42TB IS INTERNALLY TIED TO 42TA
SEE 2-750537727-24 FOR ADDITIONAL UNID INFORMATION SEE TABLE A

NOTES:
1. FOR GENERAL NOTES, REFEREN
2. 2-45E702-4 DEVICES 813 AND 833 ARE FU
3. THE CIRCUIT BREAKER AND MC SHOWN ON PANEL 11 ARE FOR UNITS 1 & 3 UNIT PREFERRI ON PANEL 11 ARE SHOWN ON F
4. DELETED

Excerpt of 2-45E2647-5: : (Illustrates how 250V DC BB2, Panel 11 Unit Preferred breaker 1001 powers Unit 2 Panel 9-9-Cabinet 5, Unit NON-Preferred via breaker 1180)



Excerpts from 0-AOI-57-6: (supports Distractors B2 and D2)

BFN Unit 0	Loss of Plant Non- Preferred	0-AOI-57-6 Rev. 0020 Page 4 of 11

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a loss of Nonpreferred (Battery Board 2 Panel 14) resulting from a loss of Lighting Boards 2A and 3B feeds to the Nonpreferred transfer switch or a transfer switch failure.

2.0 SYMPTOMS

NOTES

1) The following symptoms occur simultaneously on a loss of Nonpreferred 120 Vac.
2) Panel 9-9 Cabinet 5 Unit Nonpreferred is normally powered from Unit Preferred and therefore remains energized. Reference X-AOI-57-4, Loss of Unit Preferred if Panel 9-9 Cabinet 5 loses power regardless of the source it is being supplied from.
3) The following annunciators may be in alarm:
   - PNL 9-9 PFD OR NON-PFD BKR TRIPOUT (1-,2-,3-XA-55-8C, Window 6) on Panel 1-,2-,3-9-8 on feeder breaker trip on affected Unit(s) Panel 9-9.
   - BATT BD 2 BKR TRIPOUT/FUSE BLOWN OR GROUND (2-XA-55-8C, Window 7) on Panel 2-9-8 on breaker trip on Battery Board 2 Panel 14.
   - Other alarms associate with loss of Nonpreferred loads (see load list in Attachment 1 of this procedure).

A. Loss of Condenser Circulating Water Vacuum Breaking System. (0-FCV-27-118A and 0-FCV-27-118B open; 0-FCV-27-117 closes).

B. Auxiliary Boilers A, B, and C Trip if in service.

C. RADWASTE BLDG VENT EXH RAD DNSCL annunciator (Unit 1), (1-XA-55-22B, Window 4) is in alarm on loss of power to radiation monitor 0-RM-90-252.

D. Electrical recorder (located at STA desk, Unit 2) prints status of the Nonpreferred and Plant Preferred transfer switches and the Plant Preferred MG Set.

E. Steam Control Valves 0-TCV-12-1, (0-TCV-12-2) respond to controller setpoint with 0-HS-12-1, (0-HS-12-2) in AUTO.).

2.0 SYMPTOMS (continued)

F. Steam Control Valves 0-TCV-12-1, (0-TCV-12-2) OPEN with 0-HS-12-1, (0-HS-12-2) in CLOSE.

G. Heating System Drain to Condenser (Deaerator Bypass) 0-FCV-12-69 CLOSES.

H. Cask Decontamination Shutoff Valves, 0-FCV-12-77A, 0-FCV-12-77B CLOSE.

3.0 AUTOMATIC ACTIONS

A. Plant Preferred MG Set Starts and re-energizes Battery Board 2 Panel 13 (Plant Preferred 120 VAC System) after 6 seconds (Nonpreferred, Battery Board 2 Panel 14 remains deenergized).

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
214000 (SF7 RPIS) Rod Position Information	Tier #	------	2
G2.4.21 (10CFR 55.43.5 - SRO Only)	Group #	------	2
Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions	K/A #	214000G2.4.21	
	Importance Rating	------	4.6

Proposed Question: **# 91**

Unit 1 was operating at 100% RTP when an event occurred resulting in the following conditions:

- An automatic Reactor SCRAM
- 1-AOI-100-1, Reactor SCRAM, ATWS Actions are completed
- 40 Control Rods remain full out
- SLC is NOT injecting
- IRMs are on Range 8 and steady

The Reactor will remain subcritical **WITHOUT** boron under all conditions when all Control Rods are fully inserted except **(1)**.

Given the conditions above, in accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification to report will be a/an **(2)**.

Note: Notification of Unusual Event (NOUE)

[REFERENCE PROVIDED]

A. (1) two Control Rods are at position ‘48’
(2) NOUE

B. (1) two Control Rods are at position ‘48’
(2) ALERT

C. (1) 19 Control Rods are at position ‘02’
(2) NOUE

D. (1) 19 Control Rods are at position ‘02’
(2) ALERT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the candidate must accurately recall and apply EOI-1A Note 1 to the given plant conditions. Note 1 specifies that the Reactor will remain subcritical under all conditions when:

• Any 19 control rods are at notch 02 with all other control rods fully inserted

OR

• All control rods except one are inserted to or beyond position 00

OR

• Determined by Reactor Engineering (0-TI-394)

These conditions are easily and often confused.

Second part is incorrect but plausible in that the candidate could incorrectly determine that the Reactor is shutdown, thus meeting conditions for EPIP-1, Emergency Classification, SU5.

B INCORRECT: First part is incorrect but plausible. *(See A)* Second part is correct. *(See D)*

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible *(See A)*

D **CORRECT**: (*See attached*) In accordance with Note 1 from EOI-1, RPV CONTROL, and EOI-1A, ATWS RPV CONTROL, following a Reactor SCRAM, a Reactor will remain subcritical without boron under all conditions when any 19 Control Rods are at notch '02' with all other Control Rods fully inserted. For second part, two potential classifications are SU5 and SA5, NOUE and ALERT, respectively. SU5 is applicable following an ATWS when the initial automatic or manual Reactor SCRAM isn't successful and manual actions are successful at shutting down the Reactor. SA5 is applicable when both automatic and manual Reactor SCRAMs are unsuccessful and manual actions are **NOT** successful at shutting down the Reactor. EPIP-1 defines shutdown as defined by the EOIs. EOIs consider IRMs on Range 7 or lower as shutdown provided there has been no SLC injected.

SRO Level Justification: Tests the candidate's knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions as it relates to Control Rod position information. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-EOI-1A, Rev. 5 (Attach if not previously provided)
EPIP-1, Rev. 63
EOIPM 0-I(C), Rev. 2
1-GOI-100-1A, Rev. 72

Proposed references to be provided to applicants during examination: **EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3**

Learning Objective: OPL171.075 Obj. 2 (As available)
OPL171.201 Obj. 10i

Form 4.2-1 **Written Examination Question Worksheet**

Question Source:	Bank #		
	Modified Bank #	BFN NRC 2104 #13	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 13**

Which **ONE** of the following completes the statement below in accordance with NOTE #1 from 3-EOI-1, RPV CONTROL?

Following a Reactor SCRAM, the Unit 3 Reactor will remain subcritical **WITHOUT** boron under all conditions when _____.

A. All Control Rods are at position '02'

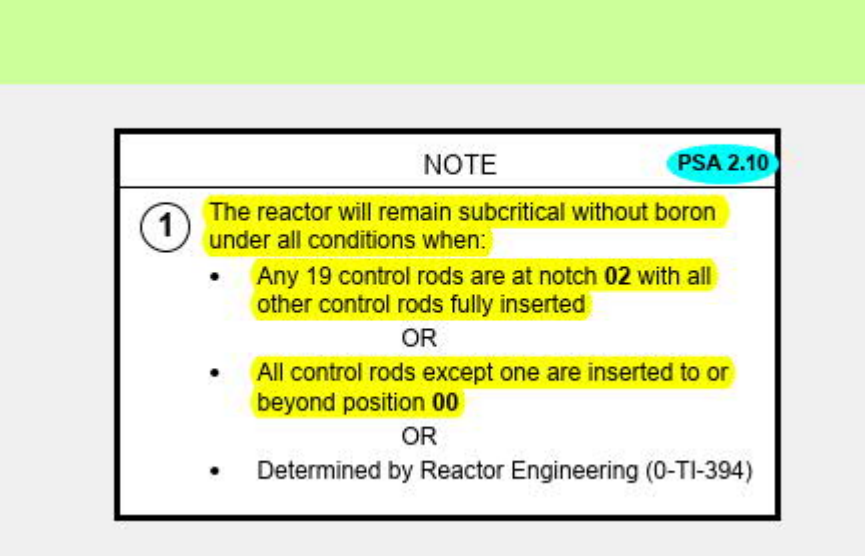
B. Reactor Power is on range 7 of the IRMs and lowering

C. All Control Rods are inserted to or beyond position '00' except two that are fully withdrawn

D. Any 19 Control Rods are at notch '02' with all other Control Rods fully inserted

Proposed Answer: **D**

Excerpts from 1-EOI-1A:



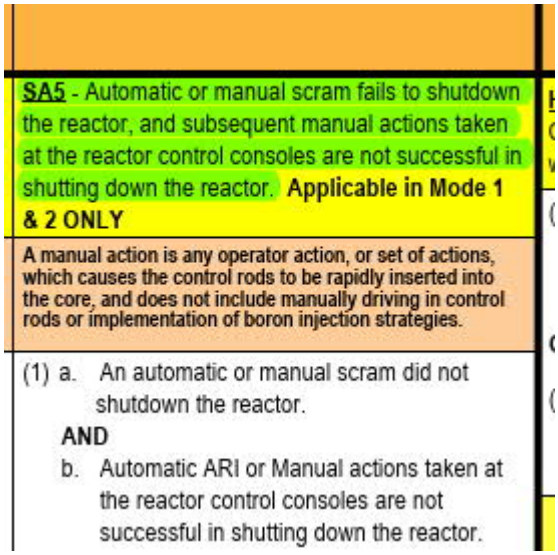
NOTE PSA 2.10

1. The reactor will remain subcritical without boron under all conditions when:
   - Any 19 control rods are at notch **02** with all other control rods fully inserted

   OR
   - All control rods except one are inserted to or beyond position **00**

   OR
   - Determined by Reactor Engineering (0-TI-394)

Excerpt from EPIP-1:



SA5 - Automatic or manual scram fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor. **Applicable in Mode 1 & 2 ONLY**

A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

(1) a. An automatic or manual scram did not shutdown the reactor.

AND

b. Automatic ARI or Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.

SU5

ECL: Unusual Event

Initiating Condition: Automatic or manual scram fails to shutdown the reactor.

Operating Mode Applicability: Power Operation, Startup

Emergency Action Levels: (1 or 2)

Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

(1) a. An automatic scram did not shutdown the reactor.

AND

b. A subsequent automatic ARI or manual action taken at the reactor control consoles is successful in shutting down the reactor.

OR

(2) a. A manual scram did not shutdown the reactor.

AND

b. **EITHER** of the following:

- A subsequent automatic ARI or manual action taken at the reactor control consoles is successful in shutting down the reactor.
- A subsequent automatic scram is successful in shutting down the reactor.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (for example, initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor. Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core. This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to "shutdown" is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA5. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA5 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Instruction criteria.

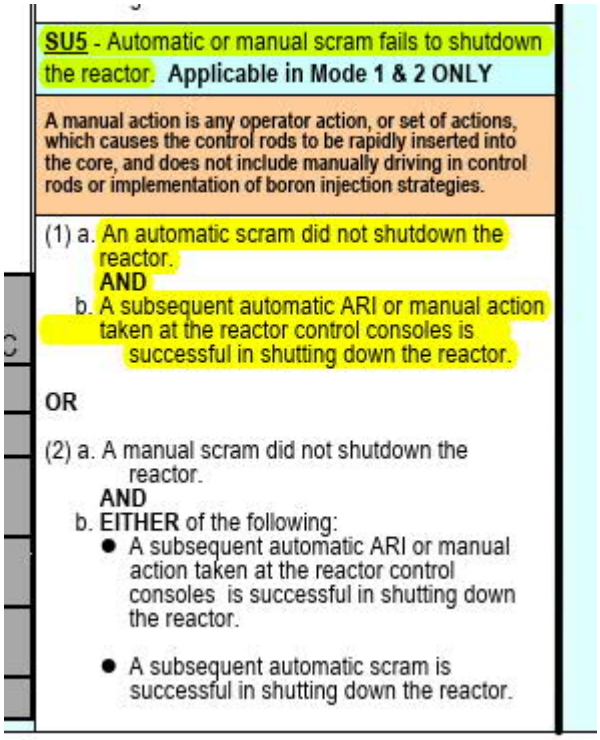
Should a reactor scram signal be generated as a result of plant work (for example, RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.

- If the signal does not cause a plant transient and the scram failure is determined through other means (for example, assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

References
1, 2, 3-AOI-100-1, Reactor Scram
NEI 99-01 R6 SU5

Excerpts from EPIP-1; Supports Distractor A2, C2:



SU5 - Automatic or manual scram fails to shutdown the reactor. **Applicable in Mode 1 & 2 ONLY**

A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

(1) a. An automatic scram did not shutdown the reactor.
AND
b. A subsequent automatic ARI or manual action taken at the reactor control consoles is successful in shutting down the reactor.

OR

(2) a. A manual scram did not shutdown the reactor.
AND
b. **EITHER** of the following:
- A subsequent automatic ARI or manual action taken at the reactor control consoles is successful in shutting down the reactor.
- A subsequent automatic scram is successful in shutting down the reactor.

Excerpt from EOIPM 0-I(C):

3.0 ABBREVIATIONS, ACRONYMS, AND DEFINITIONS (continued)

TERM/PHRASE	MEANING/DEFINITION
	* Denotes NRC/C [RIMS L44 900706 802]
SF/T	Spent Fuel Pool Temperature
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SHUTDOWN	As applied to the reactor, subcritical with reactor power below the heating range.
SLC	Standby Liquid Control
Slowly	Perform a specified action in a careful, controlled manner, making incremental adjustments to avoid undesirable consequences that may result from rapid changes in parameter values. Used specifically to identify the need for caution when restoring or increasing RPV injection under failure-to-scram conditions. If a potential for power excursions exists, RPV injection should be increased gradually while monitoring reactor power. If a sustained increase in reactor power is observed, the injection rate should be stabilized until reactor power is no longer increasing. A direction to "slowly increase RPV injection" moderates the rate of change in injection rate but does not itself limit the final steady state flow. No quantitative limit on the rate of change can be defined since the appropriate rate is dependent on the nature of the event and the operating characteristics of available systems. Injection should be increased as slowly as reasonably achievable while still accomplishing the objectives of the PSTG ATWS strategies. Systems providing effective control capability over a wide range of flows permit more gradual changes than those with no throttling capacity. If flow cannot be throttled, options for increasing injection "slowly" may be limited to initiating systems individually at full flow, starting with those of lowest capacity.
SM	Shift Manager
SP/L	Suppression Pool Level Control
SP/T	Suppression Pool Temperature Control
SRM	Source Range Monitor

Excerpt from 1-GOI-100-1A:

5.4 Withdrawal of Control Rods while in Mode 2 (continued)

NOTE

If in Single Loop Operation, 1-SR-3.4.1(SLO) is required to be completed Prior to Mode change to satisfy Tech Specs and SR-3.0.4.

CAUTIONS

1) [NRC/C] When ranging an IRM from Range 6 to Range 7, power indicated on Range 7 may NOT be consistent with indications observed when ranging from ranges 1 through 6. Should this occur, the Shift Manager or Reactor Engineer should determine if IRM response is acceptable, or if calibrations to ensure adequate gain settings are necessary. [LER 50-260-93006]

2) For Unit 1 Middle of Core Life to End of Core Life, the moderator temperature coefficient of reactivity becomes positive as control rods are withdrawn for startup when moderator temperature is below 350°F. The resulting effect will be for Reactor power to rise until the moderator begins boiling. Exercise additional caution when withdrawing control rods under this condition.

[26] **RAISE** power level by control rod withdrawal until desired rate of heating power is reached. (Usually Range 7 on IRMs.)

Initials _____ Date _____ Time _____

[27] **PERFORM** the following for EHC system:

[27.1] **ENSURE** EHC SETPOINT, 1-PI-47-162 is set at a minimum of 150 psig. (may be set higher depending on plant conditions (actual Reactor pressure))

(R) Initials _____ Date _____ Time _____

[27.2] **ENSURE** EHC in service prior to 150 psig. (N/A if Reactor Pressure is greater than 150 psig prior to startup.)

(R) Initials _____ Date _____ Time _____

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

272000 (SF7, SF9 RMS) Radiation Monitoring

A2.06 (10CFR 55.43.5 - SRO Only)

Ability to (a) predict the impacts of the following on the Radiation Monitoring System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- Downscale trips

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	272000A2.06	
Importance Rating	------	3.1

Proposed Question: **# 92**

Unit 2 is operating at 100% RTP with 0-RM-90-259A, CONTROL ROOM VENTILATION RADIATION MONITOR, indicating **DOWNSCALE** when the following occurs:

- 0-RM-90-259B, CONTROL ROOM VENTILATION RADIATION MONITOR, indicates 230 counts per minute

Given the conditions above, the Control Room Ventilation System **(1)** receive an automatic initiation signal on **ALL** three (3) Units.

In accordance with Tech Specs, part of the applicable REQUIRED ACTIONs are to **(2)**.

[REFERENCE PROVIDED]

A. (1) will
(2) Perform SR 3.3.7.1.2 on the OPERABLE channel **ONCE** per 24 hours

B. (1) will
(2) Place channel in trip **WITHIN** 24 hours

C. (1) will NOT
(2) Perform SR 3.3.7.1.2 on the OPERABLE channel **ONCE** per 24 hours

D. (1) will NOT
(2) Place channel in trip **WITHIN** 24 hours

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with 2-OI-90, Radiation Monitoring System, when one of the Control Room Ventilation Radiation Monitors exceeds the initiation setpoint (221 cpm), all three Units' Control Room ventilation systems will isolate and the selected CREV unit would automatically start after a time delay. For second part, given that 0-RM-90-259A indicates downscale in the stem of the question, then in accordance with Tech Spec 3.3.7.1, CREV System Instrumentation, REQUIRED ACTION – A.1 requires entering the CONDITION referenced in Table 3.3.7.1-1. From the table, Function 5 indicates CONDITION - D is applicable, therefore REQUIRED ACTION D.1 (Perform SR 3.3.7.1.2 on the OPERABLE channel once per 24 hours) is applicable. Since 0-RM-90-259B remains OPERABLE, D.2 (verify alternate monitoring capability) and D.3 (restore one channel to OPERABLE status) are not currently applicable.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that Tech Spec 3.3.7.1, CREV System Instrumentation, Table 3.3.7.1-1, Function 3 and 4 references CONDITION – C for other radiation monitors. In turn, REQUIRED ACTIONs C.1 and C.2 (declare associated CREV subsystem INOPERABLE AND place channel in trip within 24 hour) are plausible.

C INCORRECT: First part is incorrect but plausible if the candidate misapplies the initiation logic for CREV to believe both radiation monitors are required. One radiation detector (0-RM-90-259A) monitors Unit 1 and 2 Control Room while the other detector (0-RM-90-259B) monitors Unit 3 Control Room, however either monitor reaching the setpoint will start the selected CREV unit. Second part is correct *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C).* Second part is incorrect but plausible *(See B).*

SRO Level Justification: Tests the candidate's ability to predict the impacts of downscale trips on the Radiation Monitoring System and any applicable Technical Specifications. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate multiple distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-OI-90, Rev. 73 (Attach if not previously provided)
0-OI-31, Rev. 172
U2 Tech Spec 3.3.7.1, Amend. 253

Proposed references to be provided to applicants during examination: **U2 Tech Spec 3.3.7.1 (No Bases)**

Learning Objective: OPL171.067 Obj. 20 (As available)

Question Source:

Bank #		
Modified Bank #		(Note changes or attach parent)
New	**X**	

Question History:

Last NRC Exam	

Question Cognitive Level: Memory or Fundamental Knowledge

	Comprehension or Analysis	**X**
10 CFR Part 55 Content:	55.41	
	55.43	**X**

Excerpt from 1-OI-90:

BFN Unit 1	Radiation Monitoring System	1-OI-90 Rev. 0072 Page 40 of 45

Illustration 1
(Page 3 of 3)

Radiation Monitoring System Operational Summary

SUBSYSTEM	OPERATION
Ventilation Exhaust Continuous Air Monitors	The following Continuous Air Monitors sample for total gas activity: • 1-RM-90-250; Monitors Reactor/Refueling Zone and Turbine Building exhaust ducts. • 1-RM-90-249 and 251; Monitors Turbine Building Exhaust Roof Ventilators. • 0-RM-90-252; Monitors Radwaste Building Exhaust Ducts.
Drywell Continuous Air Monitor 1-RM-90-256	Continuous Air Monitor, samples for total gas activity. A Primary Containment Isolation signal (high drywell pressure or low reactor water level) will auto close monitor isolation valves, 1-FSV-090-0254A and B, -0255, and -0257A and B. These valves can be reopened when the isolation signal is reset and the reset pushbuttons are depressed.
Control Room Isolation Radiation Monitors 0-RM-90-259A & B	One radiation detector monitors the fresh air supply duct to the Units 1 and 2 Control Room. Unit 3 Control Room Ventilation is monitored by a separate detector. High radiation at either detector initiates Emergency Control Room isolation and pressurization on all units. Panel 25-230
Area Radiation Monitors	Individual detectors monitor the radiation levels in various areas of the plant and provide both local and control room indication.
Air Particulate Monitors	Portable monitors draw air samples from various areas of the plant and monitor the filter paper for particulates.
Drywell High-Range Radiation Monitors 1-RM-90-272A 1-RM-90-273A	Drywell Detectors provide divisionally separate post accident monitoring for gamma radiation. 1-RM-90-272A, 273A provide monitoring to 10^7 R/hr in the drywell. 1-RR-90-272A, 273A provide a record of the radiation monitors and will auto start on accident (LOCA) logic if not in service at the time of the accident.

Excerpt from 0-OI-31:

BFN Unit 0	**Control Bay and Off-Gas Treatment Building Air Conditioning System**	**0-OI-31 Rev. 0164 Page 88 of 242**

5.29 Manual Initiation of Control Room Emergency Ventilation (CREV) System

NOTES

1) The CREV System automatically initiates from:
   - PCIS Group 6

     Reactor vessel water level at "LEVEL 3"

     Drywell pressure at 2.45 psig

     Reactor zone exhaust radiation at 72 mr/hr

     Refuel zone exhaust radiation at 72 mr/hr
   - Control Room High Radiation

     221 CPM above background on U1 & 2 (3) Control Room Radiation-Gas Radiation Recorder, 0-RR-90-259A(B)
2) The CREV System is normally in Standby Readiness.
3) Performance of this instruction requires the use of two (2) Briggs and Stratton keys.
4) Dampers 0-FCO-31-150B, D, E, F, and G close automatically on auto initiation or a start from the control room using CREV TRAIN A (B) INIT/CB ISOL 0-HS-31-150A (0-HS-31-150B) on Panel 2-9-22. The dampers will NOT operate in response to the local fan control switches (0-HS-31-7214B or 0-HS-31-7213B) or to the control room fan control switches (0-HS-31-7214A or 0-HS-31-7213A).

[1] **IF** CREV failed to initiate on a valid initiation signal, or it is desired to place CREV in service with all associated functions and isolations, **THEN**

PERFORM the following at Panel 2-9-22 unless otherwise specified:

[1.1] **ENSURE** CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, (0-LPNL-25-0628, El. 617, CREV room) is in the desired position for the train to be started.

[1.2] **IF** desired to START CREV train "A", **THEN**

PLACE CREV TRAIN A INIT/CB ISOL, 0-HS-31-150A, in the INIT/TEST position.

Excerpts from Tech Spec 3.3.7.1

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

LCO 3.3.7.1 The CREV System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	B.1	Declare associated CREV subsystem inoperable.	1 hour from discovery of loss of CREV initiation capability
	AND		
	B.2	Place channel in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1	Declare associated CREV subsystem inoperable.	1 hour from discovery of loss of CREV initiation capability
	AND		
	C.2	Place channel in trip.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	D.1	Perform SR 3.3.7.1.2 on the OPERABLE channel.	Once per 24 hours
	AND		
	D.2	Verify alternate monitoring capability.	Once per 12 hours from discovery of both channels inoperable
	AND		
	D.3	Restore one channel to OPERABLE status.	30 days from discovery of both channels inoperable

(continued)

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) Deleted.

Form 4.2-1 **Written Examination Question Worksheet**

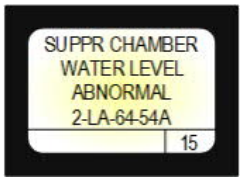
Examination Outline Cross-reference:

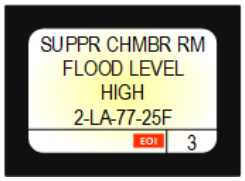
290001 (SF5 SC) Secondary Containment
G2.1.9 (10CFR 55.43.5 – SRO Only)
Ability to direct license personnel activities inside the control room (SRO Only)

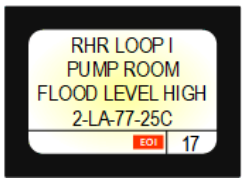
Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	290001G2.1.9	
Importance Rating	------	4.5

Proposed Question: **# 93**

Unit 2 is operating at 100% RTP when the following occurs:







- SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, (2-9-3B, Window 15), alarms
- SUPPRESSION CHAMBER ROOM FLOOD LEVEL HIGH, (2-9-4C, Window 3), alarms
- RHR LOOP I PUMP ROOM FLOOD LEVEL HIGH, (2-9-4C, Window 17), alarms
- Suppression Pool Water Level is 13.5 ft and stable

Subsequently,

- Reactor Building AUO reports 20 inches of water on the 519' elevation floor

Note: 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations
2-EOI-3, Secondary Containment Control

Given the **CURRENT** conditions above and in accordance with the EOIs, HPCI **(1)** available for use and the NUSO is required to direct the crew to **(2)**.

A. (1) is
(2) Shutdown the Reactor in accordance with 2-GOI-100-12A

B. (1) is
(2) Emergency Depressurize in accordance with 2-EOI-3

C. (1) is NOT
(2) Shutdown the Reactor in accordance with 2-GOI-100-12A

D. (1) is NOT
(2) Emergency Depressurize in accordance with 2-EOI-3

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT** (*See attached*) In accordance with 2-EOI-2, Primary Containment Control, SP/L-15 requires securing HPCI if Suppression Pool level cannot be maintained above 12.75 feet only if adequate core cooling is assured. Given current Suppression Pool level of 13.5 ft and stable and no specified or inferred ECCS equipment unavailability, HPCI is available. For second part, given the conditions provided it should be determined that, in accordance with 2-EOI-3, Secondary Containment Control, 2 or more areas have exceeded Max Safe Secondary Containment Water Level and NO Primary System is discharging into Secondary Containment. Therefore, Reactor shutdown is required in accordance with step SC-6. Note that no reference is provided to the candidate.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible if the candidate incorrectly diagnoses conditions and determines that RPV depressurization, rather than Reactor shutdown, is required in accordance with EOI-2, step SC-4.

C INCORRECT: First part is incorrect but plausible if the candidate erroneously recalls the Suppression Pool level below which HPCI must be secured (SP/L-15; 12.75 ft). Other key Suppression Pool Levels that could lead candidate to incorrectly assess the given parameter is 11.5 feet (Emergency Depressurize) or 10 feet (ECCS Pump vortex limit). Second part is correct *(See A).*

D INCORRECT: First part is incorrect but plausible *(See C).* Second part is incorrect but plausible *(See B).*

SRO Level Justification: Tests the candidate's ability to direct license personnel activities inside the Main Control Room as it relates specifically to SRO duties. SRO only because of the link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-ARP-9-3B, Rev. 40 (Attach if not previously provided)
2-ARP-9-4C, Rev. 38
2-EOI-2, Rev. 18
2-EOI-3, Rev. 19

Proposed references to be provided to applicants during examination: **SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, (2-9-3B, Window 15)**
SUPPRESSION CHAMBER ROOM FLOOD LEVEL HIGH, (2-9-4C, Window 3)
RHR LOOP I PUMP ROOM FLOOD LEVEL HIGH, (2-9-4C, Window 17)

Learning Objective: OPL171.203 Obj. 15b (As available)

Form 4.2-1	**Written Examination Question Worksheet**		
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	**X**	
Question History:	Last NRC Exam		

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Excerpt from 2-ARP-9-4C:

SUPPR CHMBR RM
FLOOD LEVEL
HIGH
2-LA-77-25F

3

(Page 1 of 1)

Sensor/Trip Point:
2-LS-77-25F ≥2 inches of water on the floor

Sensor Location: Sensor is located near the floor of the Suppression Chamber room, Column R-11 N-LINE

Probable Cause: Greater than two inches of water on the floor.

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to visually check the suppression chamber room.

B. **IF** alarm is valid, **THEN**
PERFORM the following:
- **CHECK** the floor drain sump pumps running.
- **CHECK** the floor drains for proper drainage.
- **IF** possible, **THEN**
DETERMINE the source of the leak and the leak rate.
- **ENTER** 2-EOI-3 FLOWCHART.

NOTE
The floor drain and equipment drain sump pumps may need to the locked out to prevent Radwaste flooding.

- **NOTIFY** Radwaste Operator to monitor drain collector tank and waste collector tank levels.
- **NOTIFY** Radiation Protection.

References: 0-47E610-77-1 47W600-8 45N620-4
FSAR Sections 13.6.2 and F.7.15

RHR LOOP I
PUMP ROOM
FLOOD LEVEL HIGH
2-LA-77-25C

17

(Page 1 of 1)

Sensor/Trip Point:
2-LS-77-25C ≥ 2 Inches of water on the floor

Sensor Location: Sensor is located near the floor of the southwest RHR pump room.

Probable Cause: Greater than two inches of water on the floor.

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to visually check the RHR pump room.

B. **IF** alarm is valid, **THEN**
PERFORM the following:
- **ENSURE** the floor drain sump pumps running.
- **CHECK** the floor drains for proper drainage.
- If possible, **DETERMINE** the source of the leak and the leak rate.
- **ENTER** 2-EOI-3 FLOWCHART.

References: 0-47E610-77-1 47W600-8 47W600-100
45N620-4 FSAR Sections 13.6.2 and F.7.15

Excerpt from 2-ARP-9-3B:

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0040 Page 19 of 39

SUPPR CHAMBER WATER LEVEL ABNORMAL
2-LA-64-54A

15

(Page 1 of 1)

Sensor/Trip Point:
LT-64-54 ≤ -5.5" H_2O
≥ -1.75" H_2O

Sensor Location: RX Bldg, El 519'
NW corner room just inside door

Probable Cause:
A. Suppression Chamber water level abnormal.
B. Placing Suppression Pool Cooling in service
C. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** Suppression Pool level using multiple indications.

B. **IF** level is low, **THEN**
DISPATCH personnel to check for leaks.

C. **IF** level is high, **THEN**
CHECK for RCIC, HPCI, Core Spray, or RHR draining to Suppression Pool, and **CHECK** 2-TR-64-161 and -162.

D. **REFER TO** 2-OI-74, Section 8.0.

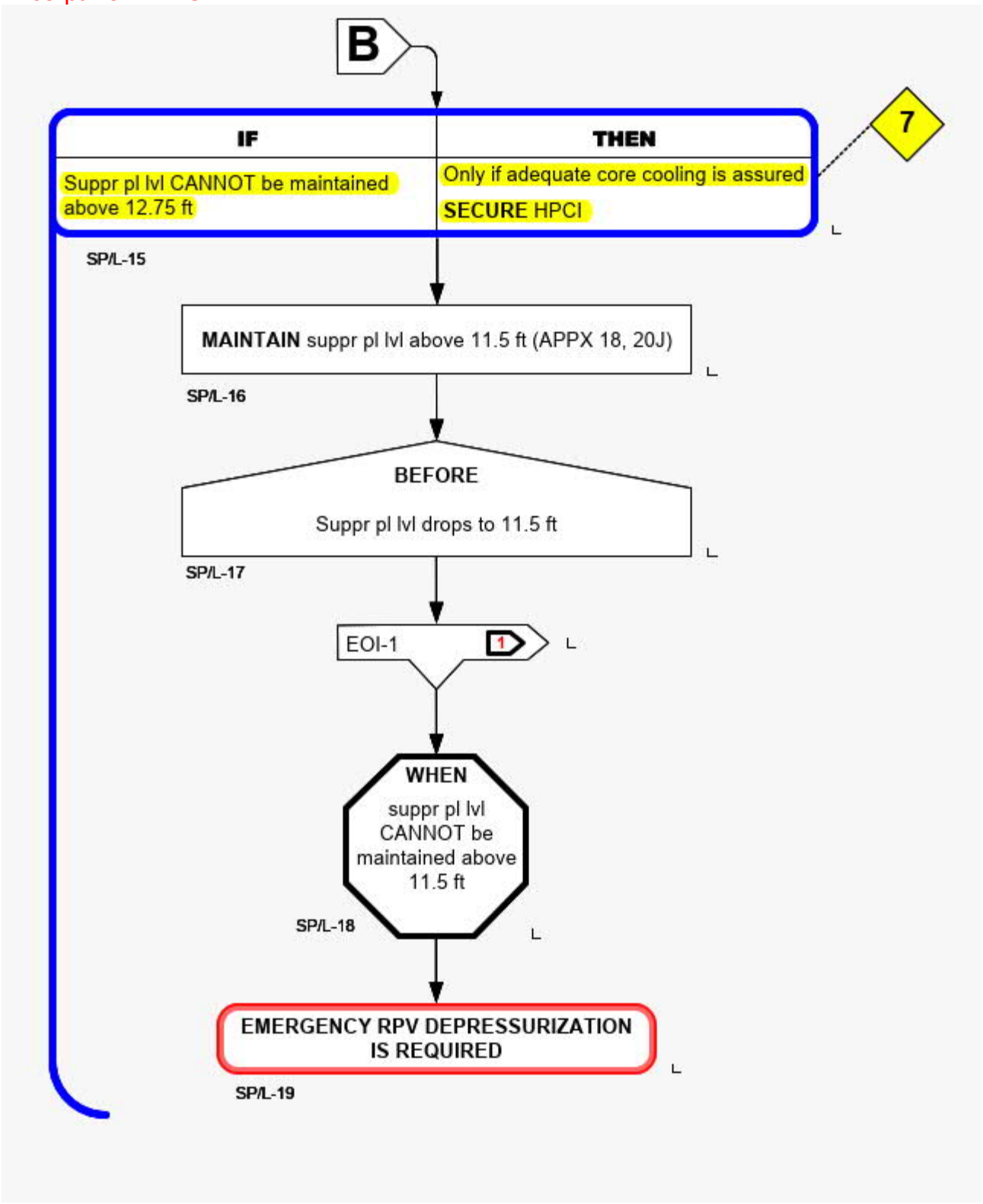
E. **REFER TO** Tech Spec 3.6.2.2.

F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
ENTER 2-EOI-2 Flowchart.

G. **IF** level is above -1" or below -6.25" **AND** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
1. **EVALUATE** plant conditions to **DETERMINE** if 2-EOI-2 entry is appropriate.
2. **RECORD** actions in NOMS log.

References: 2-45E620-3 2-47E610-64-1 GE 730E943-1
Technical Specifications
3.6.2.2

Excerpt from 2-EOI-2:



Excerpts from 2-EOI-3:

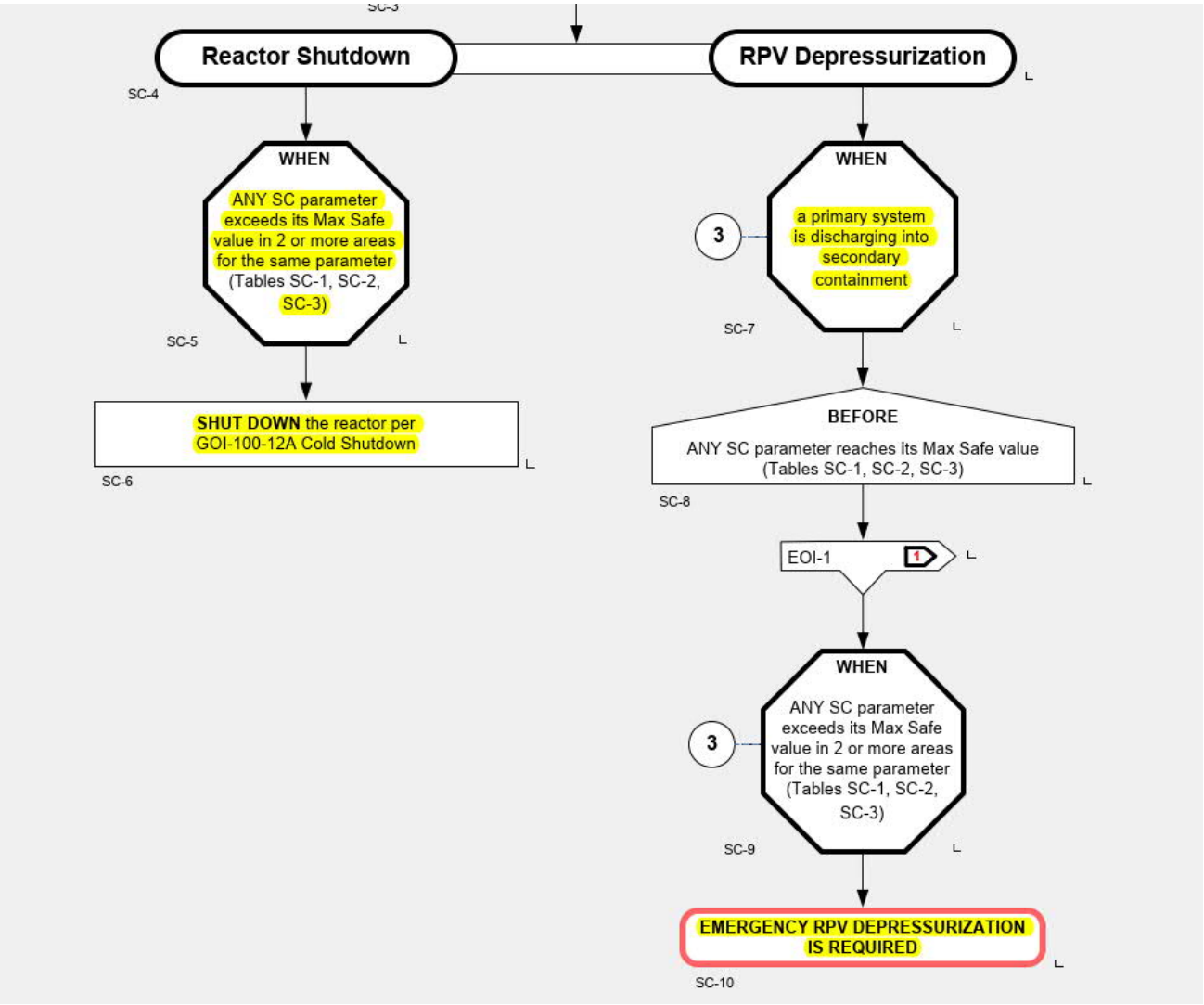


Table SC-3
Secondary Cntmt Water Level

Water Level	Max Normal Value in.	Max Safe Value in.
Floor drain sumps	66	None
Areas	2	20

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

CONDUCT OF OPERATIONS

G2.1.6 (10CFR 55.43.5 - SRO Only)

Ability to manage the control room crew during plant transients (SRO Only)

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.1.6	
Importance Rating	------	4.8

Proposed Question: **# 94**

In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, which **ONE** of the following completes the statements below?

Following a plant transient resulting in a Reactor SCRAM with a PCIS Group (1) Isolation, the NUSO will assign Reactor Water Level Control to the **(1)**.

If HPCI has a sealed in high Reactor Water Level Trip when a high Drywell Pressure signal is received, then the NUSO will initially direct an Operator **(2)** reset the high Reactor Water Level Trip.

A. **(1)** Balance of Plant Operator
(2) to

B. **(1)** Balance of Plant Operator
(2) NOT to

C. **(1)** Operator at the Controls
(2) to

D. **(1)** Operator at the Controls
(2) NOT to

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that HPCI automatically starts on a High Drywell Pressure signal for Reactor Water Level Control (which may indicate that a leak has occurred), candidate could plausibly believe that, given the conditions above, resetting HPCI is correct.

B **CORRECT**: (*See attached*) In accordance with BFN-ODM-4.20, the Nuclear Unit Senior Operator will assign the Balance of Plant Operator (BOP) to maintain Reactor Water Level Control if the Reactor Feedwater System is unavailable. Given that a Group (1) One Isolation has occurred, which causes the Main Steam Isolation Valves (MSIVs) to close, the Reactor Feedwater System is unavailable. For second part, BFN-ODM-4.20, states that the HPCI high Reactor Water Level trip is NOT to be reset when a Drywell Pressure signal is received, this is to allow HPCI to automatically initiate on a future low Reactor Water Level signal if necessary.

C INCORRECT: First part is incorrect but plausible in that in accordance with BFN-ODM-4.20, the Operator at the Controls (OATC) is assigned to maintain Reactor Water Level Control unless Reactor Feedwater Flow (RFW) is lost. Candidate must correctly determine that with the Group 1 isolation, Reactor Feedwater System is unavailable. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is correct (*See B*).

SRO Level Justification: Tests the candidate's knowledge of procedures that govern and are related to managing the crew during Initial Transient Response. Requires assessment of plant conditions and coordination of procedure and selecting the procedure which provides for division of Operator responsibilities during a transient. Requires knowledge of procedural requirements associated with duties unique to the SRO position and SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): BFN-ODM-4.20, Rev. 9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL 171.276 Obj. 3 (As available)

Question Source:

	Bank #	BFN 1804 #95
	Modified Bank #	
	New	

(Note changes or attach parent)

Question History: Last NRC Exam 2018

Question Cognitive Level:

Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	
55.43	**X**

Copy of Bank Question:

Proposed Question: **# 95**

In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, which **ONE** of the following completes the statements below?

Following a plant transient resulting in a Reactor SCRAM with a PCIS Group (1) Isolation, the Unit Supervisor will assign Reactor Water Level Control to the **(1)**.

If HPCI has a sealed in high Reactor Water Level Trip when a high Drywell Pressure signal is received, then ODM 4.20 requires that the Unit Supervisor initially direct an Operator **(2)** reset the high Reactor Water Level Trip.

A. **(1)** Operator at the Controls
(2) to

B. **(1)** Balance of Plant Operator
(2) to

C. **(1)** Operator at the Controls
(2) NOT to

D. **(1)** Balance of Plant Operator
(2) NOT to

Proposed Answer: **D**

Excerpts from BFN-ODM-4.20:

4.7 INITIAL TRANSIENT RESPONSE

NOTE
Procedure references herein are not unit-specific. Select the correct procedure for units 1, 2, or 3, as applicable.

During the initial stages of an event the OATC and UO will respond to the scram by using the AOI-100-1 hard cards. A Post Scram report of key parameters will be provided by the OATC to the NUSO in sufficient detail to allow critical decisions to be made with respect to EOI entry conditions and priorities. During the Post-Scram Report, communication will be minimized. The reactor scram will be announced by the UO as time allows following the scram report. Once the EOIs have been entered and critical parameter control has been established by the use of EOI appendices, the Hard Cards should no longer be used.

4.7.1 Initial Reactor Water Level Control

Reactor water level control will be maintained by the OATC using the reactor feedwater system. If the feedwater system is unavailable, water level control will transition to the BOP Operator. This transition will be vocalized through the use of a crew update. Water level should be maintained at +2" to +51". It is expected that under certain plant conditions that the reactor feedpumps will trip on high reactor water level despite the automatic control of the Scram Response System or prudent operator manual action.

4.7.2 Initial Reactor Pressure Control

Reactor pressure control will be maintained by the UO, following verification that the turbine is tripped, by the use of the turbine bypass valve system or Main Steam Relief Valves. AOI-100-1 provides guidance for pressure control techniques including the use of MSRVs for reactor pressure control.

4.7.3 Initial Reactor Power Control

During the initial transient response Nuclear Instrumentation shall be inserted in a timely manner to allow a rapid determination of subcriticallity. This is imperative to allow the NUSO to direct action during the EOIs and to allow the SM to determine Emergency Plan entry conditions

In general terms, reactor power is primarily controlled in AOI-100-1, Reactor Scram, by the insertion of all rods via a scram to achieve a shutdown condition with reactor power in the source range. When multiple rods fail to fully insert (ATWS), the mitigation strategies of EOI-1A, ATWS RPV Control, are implemented.

4.4 REACTOR WATER LEVEL CONTROL

4.4.1 Generic Expectations for Reactor Water Level Control

In all cases, the operator assigned to control reactor water level is expected to operate systems and take actions to stabilize reactor water level. For example, the UO is expected to minimize the HPCI injection rate to prevent tripping HPCI on high reactor water level. If an operator cannot maintain reactor water level within the prescribed band, immediately notify the NUSO.

If Reactor Water Level Control is being transferred from one injection source to another injection source, the transfer should be discussed and coordinated by the crew so as not to unnecessarily impact the effectiveness of the Reactor Water Level Control. In these instances, The NUSO should provide an update or brief to the crew to facilitate this transfer.

To preclude exceeding the allowed time for operation on minimum flow of RHR pumps (3 minutes), if RHR pumps auto-start or if manually started to spray the Suppression Chamber, the Operator will establish additional flow paths (e.g. SP Cooling). When initiating SC Sprays without previous RHR containment cooling flow paths established, the Operator should start ONLY one RHR pump in the selected loop. When additional flowpaths are required/desired, then start the 2nd RHR pump if needed.

4.4.2 Operator Expectations for Reactor Water Level Control During Transients

Operators are expected to maintain reactor water level within prescribed control bands and to immediately notify the NUSO when unable to do so. When reactor water level is not under operator control, frequent reports are required unless otherwise directed by the NUSO.

If HPCI initiates on an invalid initiation signal, it is expected that the operator report the initiation to the NUSO, verify the initiation signal is not valid, and with concurrence from the NUSO trip HPCI and when the turbine speed is zero, place the Aux Oil Pump (AOP) control switch in Pull- to-Lock (PTL) to prevent HPCI continued operation.

If HPCI initiates on a high drywell pressure signal, and HPCI IS NOT needed for reactor water level control, then trip HPCI and when the turbine speed is zero, place the AOP control switch in PTL to prevent HPCI continued operation.

If RCIC initiates on an invalid signal, it is expected that the operator report the initiation to the NUSO, verify the initiation signal is not valid, and with concurrence from the NUSO, trip RCIC. The preferred method is to use the RCIC Turbine Trip pushbutton.

If HPCI has a sealed in high reactor water level trip when a high drywell pressure signal is received, DO NOT reset the high reactor water level trip signal, this will allow the HPCI system to auto initiate on a future lower reactor water level signal, if necessary.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

Conduct of Operations
G2.1.40 (10CFR 55.43.5 – SRO Only)
Knowledge of refueling administrative requirements

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.1.40	
Importance Rating	------	3.9

Proposed Question: **# 95**

Unit 2 is in MODE 5 performing a **NON-SPIRAL** core reload during a scheduled refueling outage with the following conditions:

- The next fuel bundle is to be moved to Reactor Cavity position 26-43
- As the respective fuel bundle is grappled, SRM A and D fail downscale
- **ALL** other SRMs are **OPERABLE**

NOTE: CONSIDER EACH STATEMENT BELOW SEPARATELY

Which **ONE** of the following completes the statements below in accordance with Tech Specs and 0-GOI-100-3C, Fuel Movement Operations During Refueling?

Given the conditions above, fuel movements **(1)** continue.

Three-direction (X, Y and Z planes) movements are allowed in **(2)**.

[REFERENCE PROVIDED]

A. (1) can
(2) Reactor Vessel area **ONLY**

B. (1) can
(2) Spent Fuel Pool **AND** Reactor Vessel area

C. (1) can NOT
(2) Reactor Vessel area **ONLY**

D. (1) can NOT
(2) Spent Fuel Pool **AND** Reactor Vessel area

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 0-GOI-100-3C, Fuel Movement Operations During Refueling, and Tech Spec 3.3.1.2, SRM Instrumentation, specify requirements for non-spiral unloading/reloading. Two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the Reactor core. Additionally, requirements are that one SRM be OPERABLE in the quadrant of the Reactor core where CORE ALTERATIONS are being performed and the other SRM to be OPERABLE in an adjacent quadrant containing fuel. In the given conditions, fuel is to be loaded into Quadrant A when a failure of both the A SRM (located in Quadrant A) and the D SRM (located in adjacent Quadrant D) occurs. Candidate could plausibly incorrectly determine the quadrant undergoing CORE ALTERATIONS and/or GOI and/or Tech Spec requirements for location of required OPERABLE SRMs. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible *(See A).* Second part is incorrect but plausible in accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling, three-direction movements are NOT allowed in the Spent Fuel Pool. Two-direction horizontal movements (X and Y directions only) are allowed; however, extreme caution shall be used when utilizing two-direction horizontal movements in the Spent Fuel Pool or three-direction movements in the Reactor vessel area. Candidate must correctly recall procedural limitations for movements within the Spent Fuel Pool and within the Reactor vessel.

C **CORRECT:** 0-GOI-100-3C, Fuel Movement Operations During Refueling, and Tech Spec 3.3.1.2, SRM Instrumentation, specify requirements for non spiral unloading/reloading. Two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Additionally, requirements are that one SRM be OPERABLE in the quadrant of the Reactor Core where CORE ALTERATIONS are being performed and the other SRM to be OPERABLE in an adjacent quadrant containing fuel. In the given conditions, fuel is to be loaded into Quadrant A when a failure of both the A SRM (located in Quadrant A) and the D SRM (located in adjacent Quadrant D) occurs, thus the required SRM in the quadrant undergoing CORE ALTERATIONS is no longer met and fuel movements can NOT continue. For second part, in accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling, three direction movements (X, Y and Z planes) are only allowed in the Reactor Vessel area.

D INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's knowledge of refueling administrative requirements. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U2 Tech Spec 3.3.1.2, Amend. 345 (Attach if not previously provided)

U2 Tech Spec Bases 3.3.1.2, Amend. 130

0-GOI-100-3C, Rev. 105

Form 4.2-1 **Written Examination Question Worksheet**

Proposed references to be provided to applicants during examination: **0-GOI-100-3C, Attachment 14 (Page 1 of 1) Core Quadrants**

Learning Objective: OPL171.019 Obj. 12 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN NRC 2204 #88	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2022	

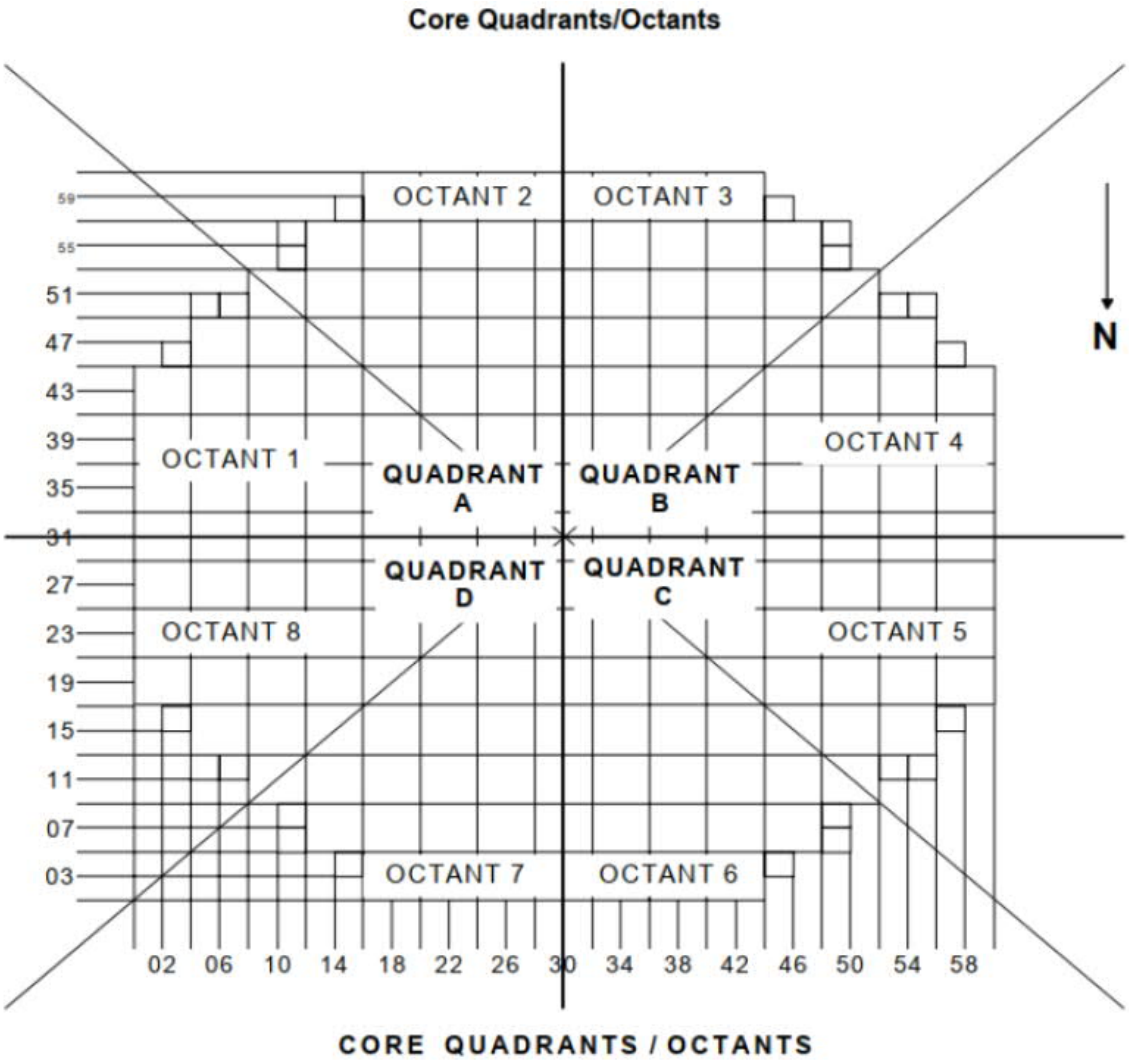
Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

REFERENCE PROVIDED (to candidate):



Copy of Bank Question:

Proposed Question: **# 88**

Unit 2 is in MODE 5 during a scheduled refueling outage. A **NON-SPIRAL** core reload is in progress with the following conditions:

- SRM B is **INOPERABLE**
- The next fuel bundle to be moved is currently located in the Spent Fuel Pool and is designated for Reactor Cavity position **10-15**
- As the respective fuel bundle is grappled, SRM C fails downscale and is declared **INOPERABLE**
- **ALL** other SRMs are **OPERABLE**

Given the conditions above, fuel moves _____ in accordance with Tech Specs and 0-GOI-100-3C, Fuel Movement Operations During Refueling.

[REFERENCE PROVIDED]

A. **CAN** continue since the SRM in the **AFFECTED** core quadrant is **OPERABLE**

B. **CANNOT** continue since the SRM in the **AFFECTED** core quadrant is **INOPERABLE**

C. **CANNOT** continue since the SRMs in the **ADJACENT** core quadrants are **INOPERABLE**

D. **CAN** continue since the SRMs in the **AFFECTED AND AN ADJACENT** core quadrant are **OPERABLE**

Proposed Answer: **D**

Excerpts from Tech Spec 3.3.1.2: (Not provided to candidate)

3.3 INSTRUMENTATION

3.3.1.2 Source Range Monitor (SRM) Instrumentation

LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	A.1	Restore required SRMs to OPERABLE status.	4 hours
B. Three required SRMs inoperable in MODE 2 with IRMs on Range 2 or below.	B.1	Suspend control rod withdrawal.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more required SRMs inoperable in MODE 3 or 4.	D.1	Fully insert all insertable control rods.	1 hour
	AND		
	D.2	Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Source Range Monitor	2(a)	3	SR 3.3.1.2.1 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	3,4	2	SR 3.3.1.2.3 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7
	5	2(b)(c)	SR 3.3.1.2.1 SR 3.3.1.2.2 SR 3.3.1.2.4 SR 3.3.1.2.5 SR 3.3.1.2.7

(a) With IRMs on Range 2 or below.

(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

Excerpts from Tech Spec Bases 3.3.1.2:

BASES

LCO (continued)

In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

In MODE 5, during a spiral offload or reload, an SRM outside the fueled region will no longer be required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant provided the Table 3.3.1.2-1, footnote (b), requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edge of a continuous fueled region (the cell can be reloaded or offloaded in any sequence).

In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate coverage is provided by requiring one SRM to be OPERABLE in the quadrant of the reactor core where CORE ALTERATIONS are being performed, and the other SRM to be OPERABLE in an adjacent quadrant containing fuel. These requirements ensure that the reactivity of the core will be continuously monitored during CORE ALTERATIONS.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

With one or more required SRM inoperable in MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded. CORE ALTERATIONS must be immediately suspended and action must be immediately initiated to insert all insertable control rods in core cells containing one or more fuel assemblies. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position.

Action (once required to be initiated) to insert control rods must continue until all insertable rods in core cells containing one or more fuel assemblies are inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

SR 3.3.1.2.1 and SR 3.3.1.2.3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

(continued)

Excerpts from 0-GOI-100-3C:

3.5 Fuel or Fuel Related Component (FRC) Handling (continued)

4. [NRC/C] Loss of Neutron Monitoring. During CORE ALTERATIONS, except as specified in TS 3.3.1.2, two Operable SRMs shall be inserted to normal level and located in and adjacent to the quadrant where core alterations are being performed. Each fueled region shall be monitored by at least one SRM. [NRC Insp. Report 89-18]. Only one SRM channel is required to be Operable during spiral offload or reload when the fueled region includes only that SRM detector. Tech Spec Table 3.3.1.2-1, Note b.
5. A Refuel Floor radiation alarm (10 mr/hr).
6. At least two communications links to the Control Room cannot be established.
7. At least two communication lines to the ROV Monitoring Station cannot be established (if ROV Cameras utilized).
8. Control rod not fully inserted with fuel in the cell.
9. Control rod position is unknown with fuel in the cell.
10. When an operable SRM has been completely surrounded by fuel and is not reading greater than or equal to 3 cps.
11. Air leakage on the fuel grapple is noted.
12. If any errors are suspected on the Fuel Assembly Transfer Form (FATF), until the error is corrected.
13. If an item is lost in the Reactor Cavity, until the item is dispositioned per NPG-SPP-06.5.
14. If the Fuel Handling Supervisor must leave the refuel floor (after the step in progress is completed) without being properly relieved.

N. The Fuel Pool Cooling System is only required to be in service to maintain fuel pool water within specifications listed in CI-13.1 and water temperature less than 125°F but greater than 72°F.

1. [NER/C] Temperature in the SFSP below 68°F exceeds the temperature reactivity assumed in the criticality analysis. [INPO SER 90-017]
2. [NER/C] Maintaining water temperature below 100°F minimizes the release of soluble activity. [GE SIL 541]

3.7 Neutron Monitoring

A. With fuel in the Reactor Vessel the following neutron monitoring must be operable for refueling.

1. At least two operable SRMs, one located in the quadrant where core alterations are being performed and one adjacent to the quadrant where core alterations are being performed, except as specified in Tech Spec 3.3.1.2.

2. If a complete core off-load is being performed, the SRMs must be initially operable.

3. SRM count rate is NOT required to be greater than 3.0 CPS with three or less adjacent fuel assemblies and no other fuel assemblies in the associated quadrant.

B. If a control rod is being withdrawn from a fuel cell containing one or more fuel bundles, then in addition to the Tech Spec requirement for IRM operability, the SRMs must be aligned to provide a coincident high flux scram. If a shutdown margin test **OR** a subcriticality test of the strongest control rod has been performed, the SRMs do not need to be aligned to provide a scram function.

C. [GE Recommendation] During fuel handling operations in the reactor vessel, the Reactor Protection System shall be capable of initiating a reactor scram based upon inputs from the SRMs by performing the following, as applicable:

1. If all rods are fully inserted and rod withdrawal is inhibited (Electrically/Hydraulically), then placing SRMs/FLCs in coincident scram mode is not required. However, if the Shift Manager so desires he can have the SRMs placed in the coincident mode. (NSAC/164L)

2. If all rods associated with cells which contain fuel are NOT fully inserted AND rod withdrawal is NOT inhibited (Electrically/Hydraulically), then placing the SRMs/FLCs in the Coincident Scram Mode will be sufficient whenever four SRMs/FLCs are operable or for 12 hours when one SRM/FLC is inoperable for maintenance or testing if the other three SRMs/FLCs remain operable. The SRMs/FLCs shall be placed in the Non-coincident Scram Mode if more than one SRM/FLC is inoperable or one SRM/FLC is inoperable for greater than 12 hours. [RIMS R62 900119 809]

D. [NRC/C] SRM/FLC Operable and Responding is defined in Technical Specification (TS) 3.3.1.2 as having a neutron count rate of ≥ 3 cps while monitoring the core and a minimum signal to noise ratio of ≥ 3:1 except as specified in TS 3.3.1.2. [NRC Insp. Report 89-18]

E. When fuel is not in the Reactor Vessel, the RPS Neutron Monitoring shorting links may be reinstalled to prevent erroneous Neutron Monitoring related spikes or faults from causing ESF actuation's.

4.0 PREREQUISITES (continued)

[7] During CORE ALTERATIONS, except as specified in Tech Spec 3.3.1.2, two operable SRMs shall be inserted to the normal level and located in and adjacent to the quadrant where core alterations are being performed. Each fueled region shall be monitored by at least one SRM/FLC.

- When four or more fuel assemblies are adjacent to an SRM (FLC), it must be reading ≥ 3 cps and have a signal-to-noise ratio of ≥ 3:1 except as specified in Tech Spec 3.3.1.2.
- Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fuel region includes only that SRM detector. Tech Spec Table 3.3.1.2-1
- The following is provided to clarify quadrant locations.

Fuel Moved in Quadrant	Required Operable SRM/FLC Quadrant Locations
A	A&B or A&D
B	A&B or B&C
C	B&C or C&D
D	C&D or A&D

(8) _____ _____ _____
1st Initials Date Time

_____ _____ _____
IV Initials Date Time

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

Equipment Control

G2.2.19 (10CFR 55.43.5 – SRO Only)

Knowledge of maintenance work order requirements.

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.2.19	
Importance Rating	------	3.4

Proposed Question: **# 96**

In accordance with NPG-SPP-07.3, Work Activity Risk Management Process, Operations **____(1)____** **REQUIRED** to review Attachment 4, BWR Operational Risk Review – RED SHEET for a normally scheduled performance of 0-SR-3.8.1.1(A) – Diesel Generator 'A' Monthly Operability Test.

In accordance with NPG-SPP 6.1, Work Order Process, the **____(2)____** has the responsibility of assigning a task as a Priority 1 emergent work order.

A. (1) is NOT
(2) Shift Manager

B. (1) is NOT
(2) Work Week Manager

C. (1) is
(2) Shift Manager

D. (1) is
(2) Work Week Manager

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with NPG-SPP-07.3, Work Activity Risk Management Process, Section 3.6, periodic instructions performed more than once per refueling cycle requires the completion of the High Risk Management Plan (Form 41218) at least once per cycle to assess risk. The RED SHEET, ATTACHMENT 4, states that it is completed for activities that are not normally scheduled activities. For second part, in accordance with NPG-SPP-06.1, Work Order Process, the Shift Manager has the responsibility of assigning the Priority to a Work Order.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible in that the Work Week Manager (WWM) owns the process and ensuring the work is risk reviewed for the responsible WWM's week. In accordance with NPG-SPP-07.3, Work Activity Risk Management Process, the WWM responsibilities include monitoring/managing all risk for his work week, ensures all emergent work has risk reviews, schedules critical evolutions are scheduled and completed, and tracking high risk for week.

C INCORRECT: First part is incorrect but plausible in that in accordance with NPG-SPP-07.3, the RED SHEET is required for risk evaluation typically when work must be completed, but has not been through the normal risk screening process. It can be used at the discretion of Operations when an activity is performed as additional RISK identification methods, but not required. Since the test involves EDGs and the grid, the risk involved could be assumed to be at a point where the candidate thinks the RED SHEET is required. Second part is correct *(See A).*

D INCORRECT: First part is incorrect but plausible (*See C*). Second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's knowledge of work order development and the risk review process. SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question is rated as memory due to strictly recalling facts related to work order development.

Technical Reference(s): NPG-SPP-07.3, Rev. 37 (Attach if not previously provided)
NPG-SPP-06.1, Rev. 12

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.239 Obj. 3 (As available)

Question Source:	Bank #	BFN NRC 2204 #96
	Modified Bank #	(Note changes or attach parent)
	New	
Question History:	Last NRC Exam	2022

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

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 96**

In accordance with NPG-SPP-07.3, Work Activity Risk Management Process, Operations ___**(1)**___ **REQUIRED** to review Attachment 4, BWR Operational Risk Review – RED SHEET for a normally scheduled performance of 0-SR-3.8.1.1(A) – Diesel Generator 'A' Monthly Operability Test.

In accordance with NPG-SPP 6.1, Work Order Process, the ___**(2)**___ has the responsibility of assigning a task as a Priority 1 emergent work order.

A. (1) is NOT
(2) Shift Manager

B. (1) is NOT
(2) Work Week Manager

C. (1) is
(2) Shift Manager

D. (1) is
(2) Work Week Manager

Proposed Answer: **A**

Excerpts from NPG-SPP-07.3:

3.6 Repetitive Work Activity Risk Assessment

NOTE

For purposes of this section, repetitive work is defined as surveillances (SR, SI, PI), preventive maintenance (PM) activities, SOI alignments, or supporting clearances that are performed at least every refueling cycle.

A. For periodic surveillances, periodic instructions or preventive maintenance activities performed more frequent than a refueling cycle, the Risk Characterization (and, if required, Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan (Form 41218)), shall be completed in its entirety at least once.

B. Individual departments are responsible to maintain repetitive work risk characterizations. Review environmental conditions and current equipment issues prior to use.

C. Repetitive work risk characterizations and any required High Risk Management Plans should be reviewed by Critical Evolutions initially and if the job plan changes significantly to ensure accuracy.

D. Any changes to work activity plans / instructions other than editorial require a new risk review.

E. The assigned Work Group has the responsibility to ensure that Risk Characterizations are completed and documented in Maximo for repetitive work activities.

1. For repetitive work activities previously classified as HIGH Risk,

   a. Previously approved Attachment 3 should be added to the job plan so they can be reviewed by the SRO during final risk review. If the previously approved Attachment 3 is not attached then the work activity owner should add it to the work order and update the job plan.

   b. The Work Activity Owner shall review the approved High Risk Management Plan to verify the risk of the work is understood and mitigating actions are still appropriate for the work being performed. A copy of the approved High Risk Management Plan shall be included in the work package and attached electronically in Maximo. Review to determine if changes to job plan affect the High Risk Plan, if so re-perform Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan

3.1.7 Manager - Online Work Management (continued)

D. Monitors work week development for low risk activities to bring to Critical Evolutions. This is intended to be an opportunity for the team to better understand routine low risk work or as a development tool for the site by exercising the process. This selection practice is discretionary and not required by this procedure.

E. Ensures all scheduled high risk activities are presented to Critical Evolutions.

3.1.8 Operations SRO

A. Performs Operational Impact review for planned work and documents any Operational Impacts on the Operations tab in Maximo.

B. Review the Maintenance Impacts tab or lognotes in Maximo for any component failure modes that the planner may have annotated under plant response.

C. Responsible for documenting the Final Risk Characterization for the work activity in the RISK ASSESSMENT field in Maximo using Attachment 6, Use of Electronic Documentation. This Final Risk Characterization and Operational Impact review should be performed by the SRO advancing the Work Order status. The SRO is the final authority for all risk classifications. Final risk shall be based on using a consequence bias by setting probability to 1 (Risk = Probability x Consequence).

D. Performs review for pre-authorization of work using Attachment 9, Work Activities that are Eligible for Pre-Authorization and documents required level of Operations review of work during the execution week in the work order.

E. Evaluates combination of LOW risk activities planned by one or more workgroups that may create a HIGH risk for the site (use of discretionary High risk classification on Site Aggregate Risk Assessment) due to the nature of the work or system interrelationships. This is not an objective evaluation, but a subjective evaluation using the skills and knowledge inherent in the qualifications of an SRO.

F. Assists with maintaining Attachment 8, Site Aggregate Risk Assessment with the help of Work Week Manager.

G. When required by step 3.7 or the Shift Manager, completes an Operational risk review for Emergent, High priority work using Attachment 4, NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET / Attachment 5, NPG-SPP-07.3-4 PWR Operational Risk Review - RED SHEET.

3.1.9 Operations Shift Manager

A. Responsible for implementation of the site schedule and ensuring the Site Aggregate Risk Assessment is current based upon plant conditions and Emergent Work.

B. Responsible for managing risk changes based on plant conditions, schedule changes, and Emergent Work.

3.1.9 Operations Shift Manager (continued)

C. For Emergent work that does not undergo the normal risk review process applies a graded approach to manage risk to the site using the Attachment 4, NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET or Attachment 5, NPG-SPP-07.3-4 PWR Operational Risk Review - RED SHEET.

3.2 Program Elements

3.2.1 Work Activity Risk Management Principles/Process Overview

A. Work Activity Risk Management is a tool to enhance the preparation, execution, and oversight of High Risk work activities.

B. Work Activity Risk is a three-phase process used to evaluate the risk associated with work activities.

1. Risk characterization of the work activities.

2. Development and approval of High Risk Management Plans for High Risk activities along with corresponding risk mitigation strategies.

3. Site Aggregate Risk Assessment and execution of work.

C. Work activity risk characterization and evaluation should be performed as early in the work control/planning process as possible. Emergent, support, and repetitive activities are addressed individually with specific actions for risk management.

D. Related work activities may be grouped and considered as a single work activity when completing risk evaluations. For grouped items, the highest risk activity should be considered as overall risk for the grouped activities.

E. LOW Risk activities are those that do not meet the criteria of Attachment 2, Risk Characterization and Operational Impact for High Risk. These activities will be controlled through normal plant processes.

F. Medium risk only applies to site aggregate risk assessment (step 3.5) and not to individual work activities.

G. For HIGH Risk work activities, the Work Activity Owner is responsible for completing and receiving approval of Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan (Form 41218). For all scheduled work activities, this analysis and approval shall be completed prior to the end of T-3 weeks, or removed from the week. Exceptions are per the NPG-SPP-07.1 Managed Exceptions process, however, still require the full risk process described in step 3.3 prior to execution.

H. Activities that result in Fire Protection Impairments and non-green fire PRA are an elevated risk condition. Risk Mitigation Actions (RMAs) controlled by NPG-SPP-18.4.6 serve to satisfy the High Risk Management Plan.

Supports Distractors C1, D1:

3.6 Repetitive Work Activity Risk Assessment (continued)

c. Operations SRO must approve the use of previously approved High Risk Management Plans as a log note in Maximo and "Critical Evolution Approved" box is changed to YES. If no longer appropriate or expired, the High Risk Management Plan must be re-performed and approved.

d. Operational Vulnerabilities and Conditional Single Point Vulnerabilities are not eligible to reuse a prior Operational Vulnerability/Conditional Single Point Vulnerability Risk Mitigation Checklist (Attachment 10, Form 41918) and must be re-performed regardless if it is repetitive or not to ensure system health of redundant components prior to work activity.

3.7 Emergent Work Risk Assessment

A. Emergent work risk evaluations should be performed based on the components current status. When components fail or must be emergently removed from service, subsequent work to restore the equipment to service may be graded low risk even though the component that is removed from service may be a high risk condition. Example: A pump fails that places the plant in a Conditional SPV. A subsequent WO to restore the failed component is graded low risk since the conditional SPV was created due to a failed component and not the implementation of the WO itself.

B. Emergent work that meets Priority 1 or Priority 2 criteria, and needs to be worked emergently within the next 24 hours does not require full implementation of the risk review process. The following actions are required:

1. For activities that meet Priority 1 criteria, the Shift Manager may waive risk assessment activities prescribed by this procedure.

2. The approval must be based on a thorough understanding of the potential consequences of the work.

3. The Shift Manager determines what actions are required to manage the risk.

4. At a minimum, the Operational Risk Review process (RED SHEET) described in section 3.8 is required, except as 3.7B.1 allows. [C.7]

5. Attachment 8, Site Aggregate Risk should be updated prior to the next scheduled Site Aggregate Risk sheet publication for emergent conditions that present a High Risk to the station, such as Conditional SPVs, and Operational Vulnerabilities.

6. The Shift Manager is responsible to coordinate and evaluate ongoing scheduled activities against the emergent issue condition and work with the WWM to resolve conflicts which may include postponing existing scheduled work or aborting existing work to restore plant equipment to service.

C. All other work added to the schedule after the T-3 Schedule is reviewed via the process described in steps 3.3, 3.4 (including Critical Evolutions if required) and 3.5. This risk evaluation should be performed at the earliest opportunity.

3.8 Operational Risk Review - RED SHEET [C.7]

A. An Operational Risk review using Attachment 4, NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET / Attachment 5, NPG-SPP-07.3-4 PWR Operational Risk Review - RED SHEET is required for Emergent Work that meets the Priority 1, Priority 2 criteria and needs to be worked within the next 24 hours. RED SHEET process may be used for emergent outage high risk work activities to protect critical path at discretion of SM.

B. The RED SHEET concentrates on the Operational risk attributes at the point of execution for work that did not follow the normal risk process. RED SHEETs are also used to re-evaluate previously approved activities based upon different plant conditions and assumptions determined in the original risk management plan (if not re-presented to Critical Evolutions) as directed by the SM.

C. Red SHEETs are required in Modes 1-4 for PWR and Modes 1-3 for BWR. However, if the potential exists for outage unit Emergent Work (not in outage schedule) to impact the other unit(s) in Modes 1-4 for PWR or Modes 1-3 for BWR, then the RED SHEET must still be processed through the non-outage unit(s).

D. The Work Activity Owner will complete sections 1 and 2 of the applicable RED SHEET (BWR or PWR).

E. The RED SHEET will be provided with the work package for Operations Review.

F. An Operation's SRO will complete section 3 and additional sections as required of applicable RED SHEET that is provided with the Work Activity.

G. Work Activity Owner obtains additional required approvals (Operations Director and Plant Managers) as directed by the RED SHEET.

H. Equipment removed from service will be entered in EOOS (or Phoenix).

I. Applicable RED SHEET will be re-evaluated following any substantial break in work or plant configuration change prior to resuming work, when in use. This re-evaluation is applicable only to a RED SHEET that has a YES in section 2. A substantial break in work is defined as greater than 24 hours from the end of the last shift where work was performed on that activity.

3.9 Outage Aggregate Risk

Shutdown risk assessment is described in NPG-SPP-07.2.11, Shutdown Risk Management.

Attachment 4
(Page 1 of 2)

NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET

BWR Operational Risk Review - **RED SHEET**

The RED SHEET shall be completed when work orders, testing, significant plant configuration changes, or plant maneuvers in support of maintenance are required and the normal risk review process is not performed (Step 3.8) or as directed by the SM. The RED SHEET will be completed for each shift while active work in progress. Attach completed RED SHEET to the work package until work completion.

1. Brief Description of Work or Activity: Procedure# / WO# - _____/_____
   _____
   _____

2. Will this work affect (or have the potential to affect, if in close proximity) any of the following systems/components:

2.1	Reactor Protection System/ Reactor Recirc / Rod Control	☐ Yes ☐ No	*DIV ___
2.2	PCIS/Containment/Containment Vacuum Breakers	☐ Yes ☐ No	*DIV ___
2.3	Safeguards (ESF, ECCS, HPCI, RCIC)/Shutdown Cooling	☐ Yes ☐ No	*DIV ___
2.4	Control Rod Drive	☐ Yes ☐ No	
2.5	Electrical Penetrations	☐ Yes ☐ No	
2.6	Main Steam/MSIVs/COND (Pumps, Xfer Sw., Cond-Demins) Reactor Feedwater / Heater Drains and Vents / Circ Water Pumps	☐ Yes ☐ No	*DIV ___
2.7	Turbine Trip Systems / EHC or Turbine Control System/ Generator and support systems (stator cooling water, bus duct cooling, voltage regulator, protective relays, lube oil, etc.)	☐ Yes ☐ No	*DIV ___
2.8	Diesel Generators/any 4.16KV/any 480V Distribution	☐ Yes ☐ No	*DIV ___
2.9	Vital 120V AC Buses, MG sets & Inverters	☐ Yes ☐ No	*DIV ___
2.10	Any 125VDC/250VDC (incl. battery chargers)	☐ Yes ☐ No	*DIV ___
2.11	Switchyard, transformer or Cap bank yard and relay room	☐ Yes ☐ No	
2.12	Back up Control Panel	☐ Yes ☐ No	
2.13	Spent Fuel Pool Inventory Control and Cooling	☐ Yes ☐ No	*DIV ___
2.14	Cable Raceways / Underground Cable	☐ Yes ☐ No	*DIV ___
2.15	Welding (Power Block Critical Areas - All)	☐ Yes ☐ No	
2.16	Reactor Coolant System Inventory or Cooling	☐ Yes ☐ No	*DIV ___
2.17	ODMI / ACMP issues	☐ Yes ☐ No	

NOTE: A Separate RED SHEET is required for each DIVISION

3. If YES to any of the above, Block 2, the following potential problems have been reviewed by Operations and any other group involved in the activity for applicability: (YES = applicable and Comp measures may be req'd) (Check Block) Initials

3.1	Errors could cause loss of load/unit trip / transient	☐ Yes ☐ No	____
3.2	Valving errors could cause hydraulic perturbations	☐ Yes ☐ No	____
3.3	Instrument, P.C. board/component - removal/installation / PLC work Activity is performed on "In Service" or not "Bypassed" equipment / component.	☐ Yes ☐ No	____
3.4	Physical limitations of work area could affect other equipment in the area that cannot be protected when work activity is in progress.	☐ Yes ☐ No	____
3.5	Placing of jumpers (without banana jacks or other permanent connection), boots on relays, or lifted leads cannot be performed without contingency actions to limit risk established.	☐ Yes ☐ No	____
3.6	Work on sensitive equipment or could affect sensitive equipment (inverters, Neutron Monitoring, Protection Racks, etc.) is being performed with Sensitive equipment "In Service" or Contingencies cannot be established to limit effects to "non-consequential".	☐ Yes ☐ No	____

EXAMPLE

Attachment 4
(Page 2 of 2)

NPG-SPP-07.3-3 BWR Operational Risk Review - RED SHEET

BWR Operational Risk Review - **RED SHEET**

	(Check Block)	Initials
3.7 Activity has potential adverse effects on common unit systems. (50.59 evaluation required if 50.59 screening did NOT screen out)	☐ Yes ☐ No	_____
3.8 Instrumentation components/systems will be worked while energized (including low voltage systems).	☐ Yes ☐ No	_____
3.9 Redundant equipment NOT available OR equipment can NOT be protected to limit potential adverse effects.	☐ Yes ☐ No	_____
3.10 If an SPV or Operational vulnerability created, complete Form 41918.	☐ Yes ☐ No	_____

IF any question in section 3.0 is answered YES, **THEN** DO NOT perform the activity until contingencies mitigating actions are established to limit risk to the satisfaction of the SM/SRO. The Operations Director and Plant Manager shall be informed of the risk, and concur with proceeding, as applicable.

Initials

MODES 1 AND 2

1. If YES to items 2.1, 2.2, 2.3, 2.6, 2.9 or 2.14, stop all work in the other Div; RPS/ESF actuation is probable. _____
2. If YES to items 2.3, 2.4, 2.6, 2.7, 2.8, 2.9, 2.10 or 2.11 considerations must be given to validate the impact of work or activity on the simulator. _____
3. If YES to any item 2.1 through 2.15 **AND** that activity that poses a threat to nuclear safety or nuclear generation as evaluated in section 3, the activity MUST be approved by the Operations Director and then by the Plant General Manager and signify by signing Lines 4.4 and 4.5. _____
4. If any question in 3.0 is answered YES and contingencies will not totally resolve the issue with OPS SRO reviewer, then Plant General Manager and OPS Director approval is required, otherwise N/A signature lines 4.4 and 4.5. _____

MODES 3 THROUGH 5

1. If any item 2.2, 2.3, 2.5, 2.6, 2.9, 2.10 or 2.14 is YES, stop all work in the other Div; An RPS/ESF actuation is probable. _____
2. If any item 2.3, 2.8 through 2.11 or 2.14 or 2.15 are YES, appropriate reviews are to be made to determine the effect on the opposite unit (Tech Specs, Procedures, Dept. instructions). If any question in 3.0 is answered YES and contingencies will not totally resolve the issue with OPS SRO reviewer, then Plant General Manager and OPS Director approval is required. _____
3. If items 2.15 and 2.16 are marked YES, consider evaluating the condition on the simulator. _____
4. If items 1, 2, and 3 above are NO, then signature lines 4.4 and 4.5 may be marked N/A. _____

4. Signatures:

4.1 Supervisor in charge of work/activity _____ Date _____
4.2 SRO EOOS (What if?) Evaluation _____ Date _____
4.3 SM / SRO _____ Date _____
4.4 OPS Director _____ Date _____
4.5 Plant General Manager _____ Date _____

EXAMPLE

Supports Distractors B2, D2:

3.1.5 Work Group Superintendent (continued)

G. Responsible to ensure package walkdowns and reviews are complete and workers understand the scope and their role in risk mitigation prior to starting work.

3.1.6 Work Week Manager (WWM)

A. Responsible to monitor their applicable weeks, tracking that activities being characterized as High Risk are being properly dispositioned, while coordinating with the Operations Shift Managers.

B. Responsible for monitoring and managing work week aggregate risk, including work activity planning and preparation.

C. Ensure that Attachment 8, Site Aggregate Risk, is maintained for each work week. (WM to maintain similar to Attachment 8, Site Aggregate Risk Work Management may make enhancements as necessary to this template, however the threshold values for risk/color changes should be determined by the peer teams and processed through a procedure revision).

D. Responsible to schedule Critical Evolutions meeting for their work week and coordinate scheduling emergent Critical Evolutions which may be required after Schedule with other WWM's.

E. At the T-3 Schedule meeting, the Work Week Manager should facilitate discussions of High Risk work by day, to ensure site awareness of High Risk work scheduled in the week.

F. Ensures activities without completed risk reviews or HIGH Risk work activities without approved Attachment 3, NPG-SPP-07.3-2 High Risk Management Plan (Form 41218) are removed from the schedule. Exceptions are per the NPG-SPP-07.1 Managed Exceptions process, however, still require the full risk process described in step 3.3 prior to execution.

G. Ensures Emergent Work identified after the T-3 Schedule meeting have risk reviews completed, and for high risk work, Critical Evolutions approval, prior to merging into the schedule.

H. Ensures HIGH risk activities are communicated to Operations SRO / SM for listing on the Daily Plant Status Report under "Daily Aggregate Risk Activities".

3.1.7 Manager - Online Work Management

A. Responsible for ensuring the WWMs monitor their applicable weeks, tracking that activities being characterized as High Risk are being properly dispositioned, while coordinating with the Operations Shift Managers.

B. Verifies HIGH risk activities are communicated to Operations SRO / SM for listing on the Daily Plant Status Report.

C. Responsible to communicate HIGH risk activities to site during T-week process approaching execution and ensuring site engagement into preparation, mitigation of risk and execution of the scheduled work.

Excerpt from NPG-SPP-06.1:

3.2.2 Reviewing of Work (continued)

B. If wires are landed / lifted, systems vented, or components drained, the system response must be understood and evaluated by a supervisor, prior to starting that job.

C. All minor Maintenance will be risk reviewed in accordance with NPG-SPP-07.3, Work Activity Risk Management Process.

3.3 Evaluating and Approving the Work Orders (WOs)

(See NPG-SPP-22.300, Corrective Action Program for Review of Condition Reports)

A. Emergency WOs

If the SM deems an emergency and assigns the Priority to a WO in accordance with NPG-SPP-07.1.4, Work Management Prioritization, the following steps are performed:

1. All personnel are to act under the command and control of the SM or designee. Actions taken shall be documented by the persons performing the actions including testing to demonstrate that the maintenance was performed correctly and that post-maintenance testing was adequate.

2. Planning shall be done on a WO in parallel with the work performance if needed, or work performance documented as soon as possible.

3. All work/troubleshooting activities shall be documented in the approved plant instruction such as, PM, procedure, SI, work order, or SR prior to the completion of the activity (for traceability of the work)..

4. All material used in Safety Related/Quality Related activities shall be documented (Form TVA 575N or equivalent).

5. If work activities involve QC hold points, the appropriate supervisor or designee (the on shift representative) shall notify site QC to arrange QC support in the field.

6. All configuration changes shall be documented in the WO or approved plant instruction.

3.4 Performing Work Orders

The Implementing Organization perform processes for planning, executing, testing, and closeout of WOs in accordance with MMDP-1, Maintenance Management System .

Attachment 1 contains guidance for performing toolpouch and minor maintenance.

4.0 RECORDS

4.1 QA Records

WO packages, as described in MMDP-1 for Safety Related or Quality Related equipment, are QA records and are processed in accordance with NPG-SPP-31.2, Records Management.

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:

Equipment Control

G2.2.38 (10CFR 55.43.1 – SRO Only)

Knowledge of conditions and limitations in the facility license

Level	RO	SRO
Tier #	----	3
Group #	----	----
K/A #	G2.2.38	
Importance Rating	----	4.5

Proposed Question: **# 97**

In accordance with Tech Spec 3.0, LCO Applicability, LCO _____, upon discovery of a failure to meet an LCO, the REQUIRED ACTIONS of the associated CONDITIONS shall be met, except as provided in other specified LCOs.

A. 3.0.1

B. 3.0.2

C. 3.0.5

D. 3.0.6

Proposed Answer: B

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that LCO 3.0.1 states that LCOs shall be met during the MODES or other specified CONDITIONS in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

B **CORRECT:** *(See attached)* In accordance with Tech Spec Bases 3.0, LCO Applicability, LCO 3.0.2 applies when upon discovery of a failure to meet an LCO, the REQUIRED ACTIONS of the associated CONDITIONS shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the REQUIRED ACTION(S) is not required, unless otherwise stated.

C INCORRECT: Incorrect but plausible in that LCO 3.0.5 applies when equipment removed from service or declared INOPERABLE to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment.

D INCORRECT: Incorrect but plausible in that LCO 3.0.6 applies when a supported system LCO is not met solely due to a support system LCO not being met, the CONDITIONS and REQUIRED ACTIONS associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system.

SRO Level Justification: Tests the candidate's knowledge of conditions and limitations based on Tech Spec Bases LCO Applicability. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as memory due to strictly recalling facts related to Tech Spec LCO Applicability.

Form 4.2-1 **Written Examination Question Worksheet**

Technical Reference(s): U2 Tech Spec 3.0, Amend. 286 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.087 Obj. 15 (As available)

Question Source:

	Bank #	
	Modified Bank #	BFN NRC 2104 #94
	New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2021

Question Cognitive Level:

Memory or Fundamental Knowledge	**X**
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	
55.43	**X**

Comments:

Copy of Bank Question:

Proposed Question: **# 94**

In accordance with Tech Spec Bases 3.0, LCO Applicability, which **ONE** of the following completes the statement below?

Upon entry into LCO _____, an evaluation shall be made to determine if a Loss of Safety Function exists.

A. 3.0.4

B. 3.0.5

C. 3.0.6

D. 3.0.7

Proposed Answer: **C**

Excerpts from Tech Spec 3.0:

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 10 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

(continued)

3.0 LCO APPLICABILITY (continued)

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." If a loss of safety function is

(continued)

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
Radiation Control	Tier #	------	3
G2.3.12 (10CFR 55.43.4 – SRO Only)	Group #	------	------
Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters.	K/A #	G2.3.12	
	Importance Rating	------	3.7

Proposed Question: **# 98**

In accordance with 2-GOI-200-2A, Primary Containment Initial Entry, Plant Manager approval **(1)** required for initial Drywell entry with the Reactor at Power.

A member of **(2)** will remain at the Personnel Airlock in continuous communication with the Control Room **AND** with the persons in the Drywell.

A. **(1)** is
(2) Operations

B. **(1)** is
(2) Radiation Protection

C. **(1)** is NOT
(2) Operations

D. **(1)** is NOT
(2) Radiation Protection

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT** (*See attached*) In accordance with 2-GOI-200-2A, Primary Containment Initial Entry, Plant Manager authorization is required for Drywell entry when the Reactor is at Power. For second part, with Primary Containment required, a member of Operations will remain at the Personnel Airlock during Drywell entry and will be in continuous communication with the Control Room and with personnel in the Drywell.

B INCORRECT**:** First part is correct *(See A)*. Second part is incorrect but plausible in that Rad Protection personnel are intimately involved in the Drywell entry process. Specifically, Radiation Protection provides entry and backup teams, establishes containment access point, maintains airlock door keys, unlocks airlock, places and secures SCBA kits in the airlock, etc.

C INCORRECT: First part is incorrect but plausible in that Shift Manager permission is required to perform any initial Drywell entry. However, the question specifically addresses an initial Drywell entry “with the Reactor at Power.” In this case, the additional permission of the Plant Manager is required. Second part is correct (*See A*).

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is incorrect but plausible (*See B*).

SRO Level Justification: Tests the candidate's knowledge concerning control of radiation releases related to the Primary Containment entry requirements associated with decision making duties unique to the SRO position. SRO only because of link to 10CFR55.43 (4): Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. This question is rated as Memory due the strict recall of specific select procedural requirements related to Primary Containment entry.

Technical Reference(s): 2-GOI-200-2A, Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.016, Obj. 3 (As available)

Question Source:			
	Bank #	BFN NRC 11-02 #98	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2011	

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Copy of Bank Question:

Proposed Question: **# 98**

Which ONE of the following completes the statements below in accordance with 1-GOI-200-2, "Primary Containment Initial Entry and Closeout?"

Initial Drywell Entry with the Reactor at Power must be approved by the __**(1)**__ .

A member of __**(2)**__ will remain at the Personnel Airlock in continuous communication with the Control Room **AND** with the persons in the Drywell.

A. **(1)** Shift Manager **ONLY**
(2) Rad Protection

B. **(1)** Shift Manager **AND** Plant Manager
(2) Rad Protection

C. **(1)** Shift Manager **ONLY**
(2) Operations

D. **(1)** Shift Manager **AND** Plant Manager
(2) Operations

Proposed Answer: **D**

Excerpts from 2-GOI-200-2A: (Also supports Distractors A2,B2)

BFN Unit 2	Primary Containment Initial Entry	2-GOI-200-2A Rev. 0004 Page 9 of 57

3.2 Initial Primary Containment Entry

A. Shift Manager approval is required for all initial entries.

B. At least one of the airlock doors is required to be closed whenever Primary Containment is required.

C. If Primary Containment is required, a member of Operations will remain at the Personnel Airlock during drywell entry. This person will be in continuous communication with the Control Room and with the persons in the Drywell.

D. Initial entries will be preplanned with the best available data, including dose rates external to the inner airlock door, remote radiation monitor inside of containment and primary containment CAM readings to ensure the safety of the entry team. Any indication of unusual conditions will be further investigated prior to initial entry.

E. Initial entries are permitted only when the Reactor Mode switch is in SHUTDOWN, REFUEL, or STARTUP/HOT STANDBY position, unless drywell entry at power has been authorized by the Plant Manager.

F. If a Drywell Entry is performed at power and the Hydrogen Water Chemistry in service, higher radiation may be observed in the Drywell. Therefore Radiation Protection should be contacted to determine if the Hydrogen Water Chemistry should be removed prior to entry.

G. Entry team personnel are required to meet the requirements set forth in NPG-SPP-05.10 for SCBA use, when required. Determination for usage of SCBA will be made by the Shift Manager, Radiation Protection Manager and Safety Manager.

H. Both the Suppression Chamber AND the Drywell are required to be purged UNTIL the O_2 concentration is 19.5%-23.5% as verified by Industrial Safety with Chemistry obtaining a grab sample IAW CI-403. This will be performed prior to entry into the Drywell OR Suppression Chamber, UNLESS otherwise approved by the Plant Manager or authorized representative.

I. SCBAs will be required until O_2 concentrations are known to be 19.5%-23.5% as verified by Industrial Safety with Chemistry obtaining a grab sample IAW CI-403.

4.1 Primary Containment Initial Entry Prerequisites (continued)

[13] Radiation Protection,

PERFORM the following:

[13.1] **ESTABLISH** an entry team and backup team.

Initials Date Time
Radiation Protection

[13.2] **ENSURE** an RWP is ready and all personnel included are assembled for Radiation Protection / Operations briefing.

Initials Date Time
Radiation Protection Shift Supervisor

[13.3] **IF** Reactor Power is greater than 40%, **THEN**

REQUEST Radiation Protection Manager approval for strongback removal.

Initials Date Time
Radiation Protection Manager

[13.4] **ESTABLISH** Radiation Protection at containment access point.

Initials Date Time
Radiation Protection

[13.5] **ENSURE** entry team has keys to locks on outside airlock door, as applicable. (N/A for Torus entry.)

Initials Date Time
Radiation Protection

4.1 Primary Containment Initial Entry Prerequisites (continued)

[19] **IF** Drywell entry at Power is required, **THEN**

PERFORM the following: (Otherwise N/A)

[19.1] **REQUEST** Plant Manager AUTHORIZE Drywell Entry at Power.

Initials Date Time
Plant Manager

[19.2] **IF** entry above Elevation 563' is required, **THEN**

REQUEST Radiation Protection Manager **AND** Plant Manager AUTHORIZE entry above Elevation 563.

Initials Date Time
Radiation Protection Manager

Initials Date Time
Plant Manager

[19.3] **IF** entry above Elevation 563' is **NOT** authorized, **THEN**

Radiation Protection **BRIEF** Entry Team on NO ENTRY above Elevation 563'.

Initials Date Time
Radiation Protection

[19.4] **ENSURE** Nitrogen to Primary Containment is isolated and a clearance is placed on Containment inerting valves, DWCA Nitrogen Supply, CAD Valves, and TIP Nitrogen Purge Valves.

Initials Date Time
OPS

4.1 Primary Containment Initial Entry Prerequisites (continued)

[21] **WHEN** above steps 4.1[1] through 4.1[20.6] are completed, **THEN**

CONTINUE in this procedure.

Initials _____ Date _____ Time _____
Unit SRO

[22] **ESTABLISH** FME control. **REFER TO** SPP-6.5, Foreign Material Control.

Initials _____ Date _____ Time _____

NOTE

When the Entry Team and Backup Team are directed to standby at the containment access, they are required to be dressed according to the RWP with SCBAs donned (if required) and hoses connected but face masks need not be in place until they are directed to enter the containment.

[23] Entry Team,

ENTER and **SURVEY** the Drywell Access Room.

Initials _____ Date _____ Time _____
Unit SRO

[24] Radiation Protection Entry Team Members,

UNLOCK the outer airlock door, as applicable.

Initials _____ Date _____ Time _____
Unit SRO

4.1 Primary Containment Initial Entry Prerequisites (continued)

CAUTION

One of the airlock doors is required to be closed whenever Primary Containment integrity is required. Operations should keep vessel (inner) door open unless shuttling entry team members into containment.

[25] OPS Entry Team Member,

OPEN AIRLOCK TO ATMOSPHERE VALVES to equalize pressure between the clean room and the airlock.

Initials	Date Unit SRO	Time

[26] OPS Entry Team Member,

REFER TO Attachment 3 and **OPEN** the outer airlock door.

Initials	Date Unit SRO	Time

[27] Industrial Safety,

ENSURE Airlock O_2 concentration is at least 19.5%.

Initials	Date Unit SRO	Time

[28] Radiation Protection,

PLACE and **SECURE** two charged SCBA kits in the airlock. (N/A if not applicable)

Initials	Date Unit SRO	Time

5.0 INSTRUCTION STEPS

5.1 Primary Containment Initial Entry

NOTE

1) Any member of the Entry Team has the authority to stop the initial entry if abnormal or unforeseen condition exists that could cause an unsafe condition during the entry. This includes but is not limited to equipment malfunction, procedural deficiencies, unusual radiation levels, and personnel error.
2) If a Drywell Leak Inspection is to be performed by Operations, the point of entry for this inspection should be given on the associated preplan developed for the inspection, as stated on Attachment 5, Drywell Leak Inspection.

[1] **ENSURE** applicable Section 4.0 prerequisites have been completed.

Initials	Date Unit SRO	Time

[2] **ENSURE** applicable signoffs complete.

Initials	Date Unit SRO	Time

[3] **OBTAIN** Shift Manager approval to enter the Drywell, and **NOTIFY** entry/back-up teams that approval has been given to enter Drywell.

Initials	Date Unit SRO	Time

Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures / Plan	Tier #	------	3
G2.4.3 (10CFR 55.43.5 – SRO Only)	Group #	------	------
Ability to identify post-accident instrumentation	K/A #	G2.4.3	
	Importance Rating	------	3.9

Proposed Question: **# 99**

LI-64-159A and XR-64-159, Suppression Pool Water Level Post-Accident Monitoring (PAM), instrumentation is identified by a/an **(1)** label.

The Tech Spec Bases for the Suppression Pool Water **UPPER** level limit is to prevent **(2)** during a DBA LOCA.

A. (1) black
(2) inoperability of the Suppression Chamber-to-Drywell Vacuum Breakers

B. (1) black
(2) excessive clearing loads from MSRV discharges and excessive pool swell loads

C. (1) orange
(2) inoperability of the Suppression Chamber-to-Drywell Vacuum Breakers

D. (1) orange
(2) excessive clearing loads from MSRV discharges and excessive pool swell loads

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that the Suppression Chamber-to-Drywell Vacuum Breakers limit the negative differential pressure (DP) across the Drywell and Suppression Chamber Primary Containment boundary. This DP has a direct impact upon the containment vent system waterleg. The basis for containment vent system internal vacuum breakers, waterleg level, MSRV vacuum breakers, MSRV relief line water level, and associated impacts of Suppression Pool water level are easily confused.

B **CORRECT**: (*See attached*) In accordance with BFN-ODM-4.5, Operator Aids and Operator Information Systems, the process is set forth for implementation and control of Operator Aids, Operations Information Tags, etc., and references 0-TI-414, Component Labeling, Signs, Operator Aids and Permanent Convention. 0-TI-414 defines the process for standardized labeling of plant components, along with the implementation and control of operator aids and other permanent information postings. Specifically, 0-TI-414 states that 'equipment required for post-accident monitoring should have a black background with white letters.' For second part, per Tech Spec Bases 3.6.2.2, 'if the Suppression Pool Water Level is too high, it could result in excessive clearing loads from MSRV discharges and excessive pool swell loads during a DBA LOCA.'

C INCORRECT: First part is incorrect but plausible BFN-ODM-4.5, Operator Aids and Operator Information Systems, describes the process set forth for implementation and control of Operator Aids, Operations Information Tags, etc., and references 0-TI-414, Component Labeling, Signs, Operator Aids and Permanent Convention. 0-TI-414 defines the process for standardized labeling of plant components, along with the implementation and control of operator aids and other permanent information postings. Specifically, 0-TI-414 states that 'EOI equipment should be identified by an EOI tag with white letters on orange background.' BFN uses various colors to denote issues such as protected equipment, Shutdown Cooling alignments, process radiation monitoring alarms and failures, common system alarms, EOI entry conditions and components, etc. The various colors and meanings are easily confused, thus, 'orange' is plausible. Second part is incorrect but plausible (*See A*).

D INCORRECT: First part is incorrect but plausible *(See C)*. Second part is correct *(See B).*

SRO Level Justification: Test candidate's ability to identify Post-Accident instrumentation. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): BFN-ODM-4.5, Rev. 6 (Attach if not previously provided)

0-OI-55, Rev. 51

0-TI-414, Rev. 15

U2 Tech Spec Bases 3.6.1.6, Rev. 0

U2 Tech Spec Bases 3.3.3.1, Rev. 0

U2 Tech Spec Bases 3.6.2.2, Rev. 0

Proposed references to be provided to applicants during examination: **NONE**

Learning Objective: OPL171.003, Obj. 11h (As available)

Form 4.2-1 **Written Examination Question Worksheet**

Question Source:	Bank #		
	Modified Bank #	BFN NRC 2104 #85	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2021	

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

10 CFR Part 55 Content:
- 55.41
- 55.43 **X**

Form 4.2-1 **Written Examination Question Worksheet**

Copy of Bank Question:

Examination Outline Cross-reference:

295029 (EPE 6) High Suppression Pool Water Level / 5

G2.2.22 (10CFR 55.43.2 - SRO Only)

Knowledge of limiting conditions for operations and safety limits.

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295029G2.2.22	
Importance Rating	------	4.7

Proposed Question: **# 85**

Unit 2 is in MODE 1 at 100% RTP when a leak into the Suppression Pool has resulted in the following condition:

- At 0200, Suppression Pool Level is (-) 3 inches and rising at 1 inch per hour

Given the condition above, which **ONE** of the following completes the statements below?

The Tech Spec Limit for 3.6.2.2, Suppression Pool Level, will be reached at ___**(1)**___.

The Bases of the Tech Spec Suppression Pool upper level limit is to prevent ___**(2)**___ during a DBA LOCA.

A. (1) 0315
(2) inoperability of the Drywell-Suppression Chamber Vacuum Breakers

B. (1) 0315
(2) excessive clearing loads from MSRV discharges and excessive pool swell loads

C. (1) 0400
(2) inoperability of the Drywell-Suppression Chamber Vacuum Breakers

D. (1) 0400
(2) excessive clearing loads from MSRV discharges and excessive pool swell loads

Proposed Answer: **D**

Excerpt from BFN-ODM-4.5:

1.0 ODM-4.5, OPERATOR AIDS AND OPERATOR INFORMATION SYSTEMS

INTENT: This directive describes the process set forth for the implementation and control of Operator Aids, Operations Information Tags, Identification of "Out of Service" Instrumentation and Indications, and the Light Lens Cover Convention.

1.1 ODM-4.5 DIRECTIVE

A. Operator Aids

Refer to 0-TI-414 Component Labeling, Signs, Operator Aids, and Permanent Information Postings.

B. Operations' Information Tags

1. In order to enhance Operator's awareness of off-normal equipment conditions or system alignments, Operations Information Tags have been provided to provide a highly visible indication on affected components or systems. The types of tags are as follows:

   - Aligned for Shutdown Cooling (Green - U1, Orange - U2, Blue - U3)
   - Temporary Lift (White)
   - Aligned Per S.I. (SR) (Manila)
   - Information (General Use) (Manila)
   - Protected Equipment (Hot Pink)|
   - Blank (Fluorescent Green)

2. These tags are to be used judiciously at the discretion of the Unit Supervisor or at direction of plant procedures in those situations where a system is aligned differently from normal standby alignment. Information such as the applicable procedure section or WO number is to be entered on the tag as appropriate.

3. These tags are not to be used to provide directives associated with equipment/component operation or to track official system status. The tags may be used to display information to the operator of unusual conditions which exist on the system or component. They are to be used to convey information ONLY. These type tags are designed for a short duration, but may be used to convey information which will remain throughout the operating cycle.

Excerpt from 0-TI-414:

Attachment 1

(Page 8 of 13)

Labeling Standard

3.0 VALVE AND DAMPER LABELING

Plant valves and dampers should be labeled regardless of function or proximity to other equipment. This section applies to plant system valves, valves installed on vendor supplied skids, instrument root and isolation valves, and ventilation system dampers.

3.1 Label Contents

A. Line 1: Component UNID.

B. Line 2: Tag Legend.

C. Line 3: May include valve or damper position during operations for safety system major flow path valves and locked valves or dampers.

3.2 Label Coding

Different colored labels may be used. See Attachment 2.

4.0 INSTRUMENT LABELING

4.1 Label Contents

A. Line 1: Component UNID.

B. Line 2: Tag Legend.

C. Line 3: Units of measurement may be included.

4.2 Label Coding

Instrument labels in the Control Room, Relay Rooms, or Remote Shutdown Panels should be white background with black letters. Equipment required for post accident monitoring should have a black background with white letters.

Supports Distractors C1, D1:

Attachment 1

(Page 11 of 13)

Labeling Standard

7.1.2 Labeling Considerations

Non-safety-related electrical equipment should be labeled with white letters on a black background including spared safety related equipment. Safety related electrical component should be label according to Section 2.1 or this attachment. EOI equipment should be identified by an EOI tag with white letters on orange background. FLEX equipment should be identified by a FLEX tag with white letters on a lime green background.

7.2 Switches and Breakers

7.2.1 Switches

Label contents.

A. Line 1: Component UNID.

B. Line 2: Tag Legend.

C. Line 3 and 4: May include switch positions, power supply, or EQ if environmentally qualified.

7.2.2 Breakers

A. Label contents for MCC and switchyard breakers.

1. Line 1: Component UNID.

2. Line 2: Tag Legend.

3. Line 3: May include EQ if environmentally qualified.

B. When only one breaker is contained in a compartment, then the breaker label should be located on the compartment door.

Excerpt from 0-OI-55:

BFN Unit 0	Annunciator System	0-OI-55 Rev. 0051 Page 11 of 44

3.0 PRECAUTIONS AND LIMITATIONS

A. Breakers tripped by protective relays should **NOT** be reclosed except as directed by the Unit SRO.

B. Prior to entry into a Battery Room, the ventilation fan for the Battery Room should be in service.

C. Extreme care should be used when de-energizing equipment while locating grounds to prevent interruption of power to vital or safeguards equipment. **REFER TO** 0-GOI-300-2.

D. All safety requirements concerning smoking, fires, or sparks should be observed when in the Battery Rooms or Battery Board Rooms because of potential accumulation of hydrogen in flammable concentrations.

E. Annunciator windows are color coded to reflect type of response required as follows:

1. Critical - Red Bar/Solid Magenta - Immediate operator action required to prevent or mitigate significant damage to equipment or property and/or to avert conditions leading to imminent loss of load.

2. Urgent - Yellow Bar - Unusual or serious operational or maintenance situation requiring prompt operator action.

3. Radiation Monitor Downscale/Inoperable - Magenta Bar - Indicates loss of process radiation monitoring, compensatory sampling may be required.

4. Operational - White - Operational or maintenance situation requiring operator attention.

5. Common System - Blue - Elevated awareness required and notification, communication, and/or coordination of activities with other affected units may be necessary.

F. For CRD High Temperature Alarms, TI-393 should be performed to evaluate the operability for that specific Control Rod.

G. When Disabling/Enabling digital recorder alarms, Attachment 3 or Attachment 4 will be used in conjunction with OPDP-4-1. Attachment 3 or Attachment 4 pages will be attached with the OPDP-4-1 paperwork and referenced in the disabling/Enabling steps in OPDP-4-1.

H. For Annunciators generated by digital paperless recorders, alarm must first be acknowledged on the recorder before the annunciator can be reset.

Excerpt from Tech Spec Bases 3.3.3.1:

BASES

LCO (continued)

3. Suppression Pool Water Level (LI-64-159A and XR-64-159)

Suppression pool water level is a Category 1 variable provided to detect a breach in the reactor coolant pressure boundary (RCPB). This variable is also used to verify and provide long term surveillance of ECCS function. The wide range suppression pool water level measurement provides the operator with sufficient information to assess the status of both the RCPB and the water supply to the ECCS. The wide range water level indicators monitor the suppression pool water level from two feet from the bottom of the pool to five feet above normal water level. Two wide range suppression pool water level signals are transmitted from separate differential pressure transmitters and are continuously recorded and displayed on one recorder and one indicator in the control room. The recorder and indicator are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

4. Drywell Pressure
(PI-64-67B, XR-64-50, PI-64-160A, and XR-64-159)

Drywell pressure is a Category 1 variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two different ranges of drywell pressure channels (normal and wide range) receive signals that are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders and two control room indicators. These recorders and indicators are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

(continued)

Excerpt from Tech Spec Bases 3.6.2.2:

BASES

BACKGROUND (continued)

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the S/RV quenchers, main vents, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from S/RV discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to S/RV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 remains valid.

Suppression pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 2).

(continued)

Excerpt from Tech Spec Bases 3.6.1.6: Supports Distractor A2, C2:

BASES

BACKGROUND (continued)

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, air in the drywell is purged into the suppression chamber free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused in two possible ways, namely, Emergency Core Cooling Systems flow from a recirculation line break, or drywell spray actuation following a loss of coolant accident (LOCA). These two cases determine the maximum depressurization rate of the drywell.

In addition, the waterleg in the Mark I Vent System downcomer is controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is less than the suppression chamber pressure, there will be an increase in the vent waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The internal vacuum breakers limit the height of the waterleg in the vent system during normal operation.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are presented in Reference 1 as part of the accident response of the primary containment systems. Internal (suppression chamber-to-drywell) and external (reactor building- to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary.

(continued)

Examination Outline Cross-reference:

G2.4.43 (10CFR 55.43.5 – SRO Only)

Knowledge of emergency communications systems and techniques.

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.4.43	
Importance Rating	------	3.8

Proposed Question: **# 100**

Unit 1 was operating at 100% RTP when an event occurred requiring the insertion of a manual Reactor SCRAM resulting in the following conditions:

- At 1000, the Main Control Room was abandoned
- At 1016, Reactor Water Level injection was established at 1-PNL-25-32, Backup Control Panel, with no other complications

Given the conditions above, which **ONE** of the following completes the statements below?

In accordance with 1-AOI-100-2, Control Room Abandonment Immediate Actions, AUOs are notified by **(1)** that Unit 1 Main Control Room is being evacuated.

In accordance with EPIP-1, Emergency Classification Procedure, the **HIGHEST** required Emergency Classification Action Level (EAL) for this event would be a/an **(2)**.

[REFERENCE PROVIDED]

A. (1) PA announcement
(2) SITE AREA EMERGENCY

B. (1) PA announcement
(2) ALERT

C. (1) hand-held radios
(2) SITE AREA EMERGENCY

D. (1) hand-held radios
(2) ALERT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** First part is correct (*See B*). Second part is incorrect but plausible in that in accordance with EPIP-1, if control of ANY key safety functions is not re-established at the Backup Control Panel within 20 minutes, the Emergency Classification Level is a Site Area Emergency (HS6).

B **CORRECT**: (*See attached*) In accordance with 1-AOI-100-2, Control Room Abandonment Immediate Actions, the crew will announce to all plant personnel that Unit 1 Main Control Room is being evacuated by dialing 7687 to conduct a PA announcement. For second part, Main Control Room was abandoned at 1000 and Reactor Water Level injection was established at 1-PNL-25-32, Backup Control Panel, with no other complications at 1016. In accordance with EPIP-1 Attachment-1, HOT Conditions, control of Reactor Water Level was established less than 20 minutes at the Backup Control Panel which meets the HIGHEST Emergency Classification Action Level of an ALERT (HA6).

C INCORRECT**:** First part is incorrect but plausible in that in accordance with 0-GOI-300-1, Operator Round Logs, AUOs assigned to Emergency Responder positions are required to carry radios, thus notification could be expected to be performed via hand-held radios. Second part is incorrect but plausible (*See A*).

D INCORRECT**:** First part is incorrect but plausible (*See C*). Second part is correct *(See B).*

SRO Level Justification: Test the candidate's knowledge of communication systems and techniques during Emergency Conditions. SRO only because of the link to 10CFR55.43 (5): Assessment of required actions in appropriate procedures during normal, abnormal, and emergency situations. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome as it relates to Emergency and Abnormal Procedures.

Technical Reference(s): 1-AOI-100-2, Rev. 25 (Attach if not previously provided)
EPIP-1, Rev. 63
0-GOI-300-1, Rev. 212

Proposed references to be provided to applicants during examination: **EPIP-1, Attachment 1, HOT INITIATING CONDITIONS-MODES 1-2-3**

Learning Objective: _OPL171.075 Obj. 16_ (As available)

Question Source: Bank # ____ Modified Bank # ____ (Note changes or attach parent) New **X**

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Excerpt from 1-AOI-100-2:

BFN Unit 1	Control Room Abandonment	1-AOI-100-2 Rev. 0025 Page 7 of 79

4.1 Immediate Action (continued)

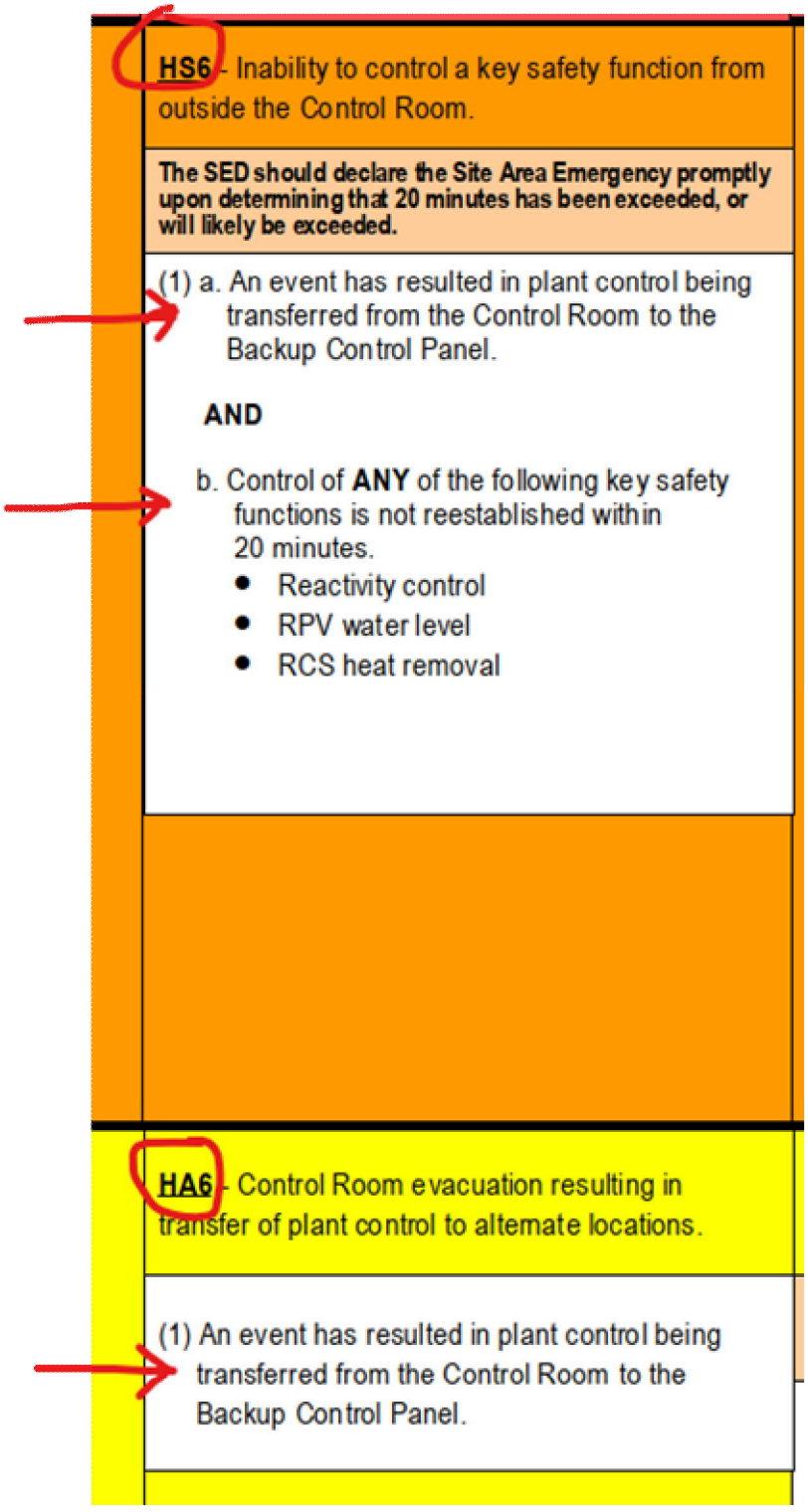
[12] **ANNOUNCE** to all plant personnel

- Unit 1 Main Control Room is being evacuated
- All operations personnel report to your assigned backup control stations

[13] **OBTAIN** hand-held radios from Unit 2 Control Room.

[14] **PROCEED** to Backup Control Panel 1-25-32.

Excerpt from EPIP-1:



EPIP-1 R63, Attachment 1
BFN Hot Condition ICs/EALs

Excerpt from 0-GOI-300-1: Supports Distractors C1, D1

BFN Unit 0	Operator Round Logs	0-GOI-300-1 Rev. 0212 Page 11 of 21

3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. The following guidance shall be used when installing and updating Instrument Scale Zone Coding.

1. Color zone coding should be restricted to instruments where zone coding provides the operator with meaningful information. Indiscriminate application of color zone coding can reduce its effectiveness.

2. Reference conditions (ranges of operation) have been determined for each component based upon Operating Instructions, vendor/engineering recommendations and reliable system/component operation. These reference conditions are listed in the affected attachment for each component. In general, these reference conditions are based upon full power operations and are conservative to both Operating Instructions and Technical Specifications.

3. The application of color zone coding material shall **NOT** interfere with the operation or obscure indication of the instrument.

4. Reliance on the indication being within the "GREEN BAND" will **NOT** be the sole check performed to ensure the indication is correct for current plant conditions. All readings obtained during Operator Rounds will be verified for current plant conditions to ensure the current indication is valid for current plant conditions.

5. In instances where color zone coding may interfere with instrument indication or operation, or where face plate cover removal is difficult, a green label with the UNID (where applicable) and normal operating range will be placed on or near the instrument face plate.

F. The night shift and day shifts are defined by the day the shift begins. (Sunday day shift is Sunday 0700-1900, Sunday night shift is Sunday 1900 to Monday 0700.)

G. Hourly monitoring of the Forebay Temperature is required anytime Forebay Temperature is ≥91°F, as indicated on 0-TI-27-144, on Panel 2-9-20, per Technical Specification 3.7.1.2.

H. AUOs assigned to Emergency Responder positions are required to carry a radio.

4.0 PREREQUISITES

None

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-1
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 80% Reactor Power, 'B' Reactor/Refuel Zone Exhaust Fans are running. Control Rod 30-43 is at position 00 for Directional Control Valve Maintenance.

Turnover: Perform 2A RFPT Overspeed Trip Exerciser Test. Reduce Reactor Power to < 75% using Core Flow in preparation for recovering Control Rod 30-43.

Critical Tasks:
1. When Drywell Sprays are required, initiate sprays while in the safe region of the Drywell Spray Initiation Limit Curve to preclude Primary Containment failure.
2. Following a loss of high pressure injection, restore high pressure injection to the Reactor Vessel to preclude Emergency Depressurization due to low Reactor Water Level.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Reactor Feedwater Pump (RFPT) 2A Overspeed Trip Exerciser Test
2.	N/A	R-OATC	Reduce Reactor Power for Control Rod Recovery and Testing
3.	NMAPRMGAIN(1)	C-OATC C-NUSO	Average Power Range Monitor (APRM) 1 Failure
4.	RD22	I-OATC MC-OATC I-NUSO	Control Rod Drive (CRD) Flow Transmitter Failure
5.	XA-55-3A_26	C-BOP C-NUSO	PSC Head Tank Low Level Alarm
6.	NMAPRMGAIN(3)	TS-NUSO	Average Power Range Monitor (APRM) 3 Failure
7.	HP01	C-BOP MC-BOP TS-NUSO	Inadvertent High Pressure Coolant Injection (HPCI) Initiation
8.	TC07 TH33A	M-ALL	EHC Piping Leak / Drywell Steam Leak
9.	TH30E TH30G	C-BOP C-NUSO	LT-3-208A and C Fail High

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will perform the Overspeed Trip Exerciser Test on Reactor Feedwater Pump (RFPT) 2A in accordance with 2-OI-3, Section 8.10.
2. The crew will reduce Reactor Power using Core Flow in preparation for recovering Control Rod 30-43 and performing Scram Time Testing.
3. APRM 1 will experience a critical fault, requiring the crew to respond in accordance with the appropriate Alarm Response Procedure. The crew will bypass APRM 1 in accordance with 2-OI-92B, Average Power Range Monitoring.
4. The CRD Flow Transmitter will fail high, requiring the crew to respond in accordance with Alarm Response Procedures and 2-OI-85, Control Rod Drive System, to restore proper CRD System flow.
5. The crew will receive the PSC Head Tank Low Level Alarm (Panel 2-9-3A, Window 26) and will respond in accordance with the Alarm Response Procedure to manually start the PSC Head Tank Pumps.
6. APRM 3 will fail high. Since APRM 1 is already bypassed due to a failure, APRM 3 cannot be bypassed, requiring the Nuclear Unit Senior Operator to address the Technical Specifications.
7. HPCI will inadvertently initiate, requiring the crew to trip and lock out the HPCI Turbine. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
8. A leak will develop in the EHC System, eventually requiring the crew to insert a manual Reactor SCRAM. When the Reactor SCRAM is inserted, a steam leak will occur in the Drywell, requiring the crew to respond in accordance with the Abnormal and Emergency Operating Procedures.
9. Following the Reactor SCRAM, Reactor Water Level Instruments LT-3-208A and LT-3-208C will fail high, causing a trip of Reactor Feedwater Pumps and Reactor Core Isolation Cooling (RCIC). The crew will be required to line up other high pressure injection methods to provide for Reactor Water Level Control to prevent Emergency Depressurization.

The Scenario ends when the crew has sprayed the Drywell, and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available high-pressure injection system.

Critical Tasks 2

1. **When Drywell Sprays are required, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve to preclude Primary Containment failure.**

   a. **Safety Significance:**

   Precludes failure of Primary Containment.

   b. **Cues:**

   Procedural compliance.
   High Drywell and Suppression Chamber Pressure.
   High Drywell Temperature.

   c. **Measured by:**

   Observation – the crew verifies Primary Containment parameters prior to initiating Drywell Sprays.

   d. **Feedback:**

   Drywell Pressure and Temperature lowering.
   Suppression Chamber Pressure lowering.
   RHR Flow to Containment Cooling and Sprays.

   e. **Critical Task Failure Criteria:**

   Drywell Sprays are initiated with Drywell Pressure and Temperature outside the safe area of the Drywell Spray Initiation Curve (Curve 5).

2. **Following a loss of high pressure injection, restore high pressure injection to the Reactor Vessel to preclude Emergency Depressurization due to low Reactor Water Level.**

   a. **Safety Significance:**

   Prevents challenging plant safety by preventing the initiation of Emergency Depressurization.

   b. **Cues:**

   Procedural compliance.
   Loss of Reactor Feedwater Pump.
   Lowering Reactor Water Level.

   c. **Measured by:**

   Observation – the crew restores high pressure injection to the Reactor Vessel.

   d. **Feedback:**

   Rising Feedwater Flow indication.
   Rising Reactor Water Level.

   e. **Critical Task Failure Criteria:**

   Emergency Depressurization is required based on low Reactor Water Level.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 1 Page 1 of 3

Event Description: Reactor Feedwater Pump (RFPT) 2A Overspeed Trip Exerciser Test

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**NOTE: The crew has been pre-briefed, but may elect to hold a re-focus brief.**
	NUSO	Directs the Balance of Plant Operator (BOP) to conduct the Overspeed Trip Exerciser Test on 2A Reactor Feedwater Pump (RFPT) in accordance with 2-OI-3, Reactor Feedwater System, Section 8.10.
	BOP	2-OI-3, Reactor Feedwater System Section 8.10, Overspeed Exerciser Test **NOTES** 1) The following steps will test the circuitry associated with the overspeed trip test and mechanical overspeed trip lockout. The test may be performed at speed or stopped; it should be performed prior to rolling the turbine off the turning gear and periodically with the applicable RFP in operation. This test will normally be performed during scheduled reductions in power with reactor power less than 90%. 2) The success of this test depends on the mechanical overspeed trip bolt actuating as a result of oil pressure. The system was designed for this test to be performed with the RFPT at normal operating speed. The test may not function properly at less than 75% speed (4125 rpm) due to the trip mechanism being sluggish at speeds well below rated. If this is the case, no action is required except to reperform the test at a higher speed. 3) The following steps are performed at Panel 2-9-6. [1] **OBTAIN** Unit SRO approval to perform this test. [2] For RFPT being tested, **ENSURE** Main Oil Pump running: • RFPT 2A 2A1 MAIN OIL PMP • RFPT 2A 2A2 MAIN OIL PMP

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 1 Page 2 of 3

Event Description: Reactor Feedwater Pump (RFPT) 2A Overspeed Trip Exerciser Test

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **PERFORM** the following for the RFPT to be tested. [3.1] **PLACE** 2-HS-3-109A, RFPT 2A OVERSPEED TEST TRIP LOCKOUT, in MECH and **CHECK** indications: • Green (normal) light extinguished at switch • Amber (mechanical lockout) light to the right of the green light is illuminated [3.2] N/A [3.3] N/A [4] **PERFORM** the following RFPT Overspeed Test. [4.1] **DEPRESS** and **HOLD** 2-HS-3-110, RFPT 2A OVERSPEED TEST on Panel 2-9-6 and **CHECK** indications: • White (trip) light illuminated at pushbutton • Green (normal) light extinguished at pushbutton [4.2] N/A [4.3] N/A [5] **RELEASE** 2-HS-3-110, RFPT 2A OVERSPEED TEST pushbutton. [6] **PERFORM** the following RFPT Overspeed Test Reset. [6.1] **DEPRESS** and **HOLD** 2-HS-3-106, RFPT 2A OVERSPEED TEST RESET, and **CHECK** the following on Panel 2-9-6: • 2-HS-3-106, RFPT 2A OVERSPEED TEST RESET, white (trip) light illuminated. • 2-HS-3-110, RFPT 2A OVERSPEED TEST, white (trip) light extinguished [6.2] N/A [6.3] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 1 Page 3 of 3

Event Description: Reactor Feedwater Pump (RFPT) 2A Overspeed Trip Exerciser Test

Time	Position	Applicant's Actions or Behavior
	BOP	[7] **RELEASE** Overspeed Test Reset on Panel 2-9-6 and **CHECK** the following. [7.1] RFPT 2A • 2-HS-3-106, RFPT 2A OVERSPEED TEST RESET, white (trip) light extinguished. • 2-HS-3-110, RFPT 2A OVERSPEED TEST, green (normal) light illuminated [7.2] N/A [7.3] N/A **NOTE** Waiting 30 seconds before placing Overspeed Test Trip Lockout switch in normal position in Step 8.10[8]. allows ample time for Overspeed test trip logic to reset. [8] **PERFORM** the following, when 30 seconds have elapsed for the RFPT to be tested.
	BOP	Allows 30 seconds to elapse before proceeding with Step [8.1].
	BOP	[8.1] **PLACE** 2-HS-3-109A, RFPT 2A OVERSPEED TEST TRIP LOCKOUT, in NORM and CHECK indications: • Green (normal) light illuminated at switch • Amber (mechanical lockout) light to the right of the green light is extinguished [8.2] N/A [8.3] N/A [9] **NOTIFY** Unit SRO when test is complete. [10] **RECORD** results in Narrative Log.
	NRC	**End of Event 1. Proceed to Event 2, Reduce Reactor Power for Control Rod Recovery and Testing.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 2 Page 1 of 2

Event Description: Reduce Reactor Power for Control Rod Recovery and Testing

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew has pre-briefed Event 2, but may elect to conduct a re-focus brief.**
	NUSO	Directs the Operator at the Controls (OATC) to reduce Reactor Power to ≤ 75% in accordance with 2-OI-68, Reactor Recirculation System and Reactivity Control Plan (RCP) RCP-NRC2304-U2S1.
	NRC	**The crew may elect to reduce Reactor Power using any method below.**
	OATC	2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown in 0-TI-248, Unit 2 Core Operating Limits Report (COLR) and 2-SR-2. 2) Recirc Flow changes made during the latter part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor Pump Speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing core flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pumps 2A and/or 2B speed with Recirc Individual Control, **THEN PERFORM** the following: • LOWER Recirc Pump 2A using 2-HS-96-17A (17B) (17C), LOWER SLOW (MEDIUM)(FAST) AND/OR • LOWER Recirc Pump 2B using 2-HS-96-8A (18B) (18C), LOWER SLOW(MEDIUM)(FAST)

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 2 Page 2 of 2

Event Description: Reduce Reactor Power for Control Rod Recovery and Testing

Time	Position	Applicant's Actions or Behavior
	NUSO	[2] **WHEN** desired to control Recirc Pumps 2A and/or 2B Speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 2A & 2B using the following pushbuttons as required: 2-HS-96-33, LOWER SLOW 2-HS-96-34, LOWER MEDIUM 2-HS-96-35, LOWER FAST
	NRC	**End of Event 2. Request that the driver insert Event 3, Average Power Range Monitor (APRM) 1 Failure**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 3 Page 1 of 2

Event Description: Average Power Range Monitor (APRM) 1 Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. A critical fault APRM 1 will occur.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to insert a critical fault on APRM 1.**
	OATC	Acknowledges and reports the following alarms to the NUSO: • CONTROL ROD WITHDRAWAL BLOCK, 2-9-5A, Window 7 • APRM HIGH/INOPOR OPRM TRIP, 2-9-5A, Window 25
	NUSO	Directs the Balance of Plant Operator (BOP) and the OATC to respond in accordance with the Alarm Response Procedures.
	OATC	2-ARP-9-5A, Alarm Response Procedure CONTROL ROD WITHDRAWAL BLOCK, Window 7 Operator Actions: A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s) and **REFER TO** operator action for alarm(s). B. N/A C. N/A D. N/A E. N/A
	OATC	2-ARP-9-5A, Alarm Response Procedure APRM HIGH/INOPOR OPRM TRIP, Window 25 A. **CHECK** alarm by multiple indications. B. With NUSO permission, **BYPASS** initiating channel to reset the alarm. **REFER TO** 2-OI-92B, Average Power Range Monitoring. C. N/A D. N/A E. N/A F. N/A G. **NOTIFY** Reactor Engineer H. **CHECK** thermal limits on RUNMON.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 3 Page 2 of 2

Event Description: Average Power Range Monitor (APRM) 1 Failure

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Reactor Engineer, acknowledge any report or direction given.**
	NUSO	I. **REFER** TO Tech Spec Table 3.3.1.1-1, TRM Table 3.3.4-1. J. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5. Directs the OATC to bypass APRM 1 in accordance with 2-OI-92B, Average Power Range Monitoring.
	NRC	**EXAMINER NOTE: The NUSO may reference Tech Specs and the TRM. There are no required Tech Spec or TRM actions, but the NUSO may enter an Information Only LCO.**
	OATC	2-OI-92B, Average Power Range Monitoring Section 6.1, Bypassing APRM/OPRM Channel **CAUTION** NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with APRMs bypassed unless bypassing is specifically allowed within approved procedures [1] **REVIEW** all Precautions and Limitations. **REFER TO** Section 3.0. [2] **PLACE** 2-HS-92-7B/S3, APRM BYPASS, to desired channel to be bypassed. [3] **CHECK** BLUE BYPASSED lights illuminated on Panel 2-9-14, Voters. [4] **CHECK** white bypass light on Panel 2-9-5 is illuminated.
	NRC	**End of Event 3. Request that the driver insert Event 4, Control Rod Drive (CRD) Flow Transmitter Failure.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 4

Event Description: Control Rod Drive (CRD) Flow Transmitter Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The CRD Flow Transmitter will fail high, causing the CRD Flow Control Valves to close.

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, insert Event 4, Control Rod Drive (CRD) Flow Transmitter Failure.**
	OATC	Acknowledges and reports the following alarm to the Nuclear Unit Senior Operator (NUSO): • CRD ACCUMULATOR CHARGING WATER HEADER PRESSURE HIGH, 2-9-5A, Window 10
	NUSO	Acknowledges the OATC's alarm report. Directs the OATC to respond in accordance with applicable Alarm Response Procedures and subsequently 2-OI-85, Control Rod Drive System.
	OATC	2-ARP-9-5A, Alarm Response Procedure CRD ACCUM CHG WTR HDR PRESS HIGH, 2-9-5A, Window 10 A. **CHECK** pressure high on 2-PI-85-13A, CRD ACCUMULATOR CHARGING WATER HEADER on Panel 2-9-5. B. **CHECK** 2-FCV-85-11A (B), CRD LINE A(B) FLOW CONTROL VALVE, in service.
	NRC	**The crew may attempt to switch Flow Control Valves in accordance with 2-AOI-85-3, CRD System Failure. However, as long as the Flow Transmitter is failed high, neither set of Flow Control Valves will operate in automatic. 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, must be placed in MANUAL.**
	OATC	C. **IF** in-service controller has failed, **THEN REFER TO** 2-OI-85, Control Rod Drive System. D. N/A
	OATC	Determines that the CRD Flow Transmitter has failed high, causing 2-FCV-85-11A, CRD LINE A FLOW CONTROL VALVE to CLOSE. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, takes manual control of 2-FIC-85-11, CRD SYSTEM FLOW CONTROL to restore CRD Parameters back to normal.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 4

Event Description: Control Rod Drive (CRD) Flow Transmitter Fails High

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted by the crew as the Outside/Work Control NUSO, Reactor Engineer, or Instrument Mechanics, acknowledge any information or direction given.**
	OATC	2-OI-85, Control Rod Drive System Section 8.34, AUTOMATIC/MANUAL operation of 2-FIC-85-11 [1] **REVIEW** all Precautions and Limitations in Section 3.6. [2] **IF** transferring 2-FIC-85-11 from AUTO to MANUAL **THEN**: [2.1] **PLACE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL in **BALANCE.** [2.2] **BALANCE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL by turning Manual Control Pot inside Control Selector Wheel until red deviation pointer is in the Green Band. [2.3] **PLACE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL in MANUAL. [2.4] **ADJUST** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL manual potentiometer to establish the desired system flow. Refer to Section 5.1 or 6.10. [3] N/A
	NRC	**End of Event 4. Request that the Driver insert Event 5, PSC Head Tank Low Level Alarm.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 5 Page 1 of 1

Event Description: PSC Head Tank Low Level Alarm

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The PSC Head Tank Low Level Alarm will be received on Panel 2-9-3.

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, insert Event 4, PSC Head Tank Low Level Alarm.**
	BOP	Acknowledges and reports the following alarm to the Nuclear Unit Senior Operator (NUSO): • PSC HEAD TANK LEVEL LOW, 2-9-3A, Window 26
	NUSO	Acknowledges the OATC's alarm report. Directs the OATC to respond in accordance with the applicable Alarm Response Procedure.
	BOP	2-ARP-9-3A, Alarm Response Procedure PSC HEAD TANK LEVEL LOW, 2-9-3A, Window 26 A. **ENSURE** both PSC Head Tank Pumps are running.
	NRC	**15 seconds after the PSC Head Tank Pumps are started, the PSC Head Tank Low Level Alarm can be cleared and the Head Tank Pumps will automatically stop as expected.**
	OATC	Starts 2-HS-75-75A, 2A PSC Head Tank Pump, and 2-HS-75-76A, 2B PSC Head Tank Pump.
	Driver	**If contacted by the crew as the Outside/Work Control NUSO, Electrical Maintenance, or Instrument Mechanics, acknowledge any information or direction given.**
	OATC	B. **ENSURE** power available to the PSC Head Tank Pumps. C. **CHECK** 2-FCV-75-57, PSC PUMP SUCTION INBOARD ISOLATION VALVE and 2-FCV-75-58, PSC PUMP SUCTION OUTBOARD ISOLATION VALVE, open. D. N/A E. N/A F. N/A
	NRC	**End of Event 5. Request that the Driver insert Event 6, Average Power Range Monitor (APRM) 3 Failure.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 6 Page 1 of 3

Event Description: Average Power Range Monitor (APRM) 3 Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. APRM 3 will fail low.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause APRM 3 to fail low.**
	OATC	Acknowledges and reports the following alarms to the NUSO: • APRM DOWNSCALE / OPRM INOP, 2-9-5A, Window 4 • CONTROL ROD WITHDRAWAL BLOCK, 2-9-5A, Window 7
	NUSO	Directs the BOP and the OATC to respond in accordance with the Alarm Response Procedures.
	OATC	2-ARP-9-5A, Alarm Response Procedure CONTROL ROD WITHDRAWAL BLOCK, Window 7 Operator Actions: A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s) and **REFER** TO operator action for alarm(s). B. N/A C. N/A D. N/A E. N/A
	OATC	2-ARP-9-5A, Alarm Response Procedure APRM DOWNSCALE / OPRM INOP, Window 4 A. **DETERMINE** which APRM/OPRM channel is downscale / inoperable. B. **IF** APRM failed downscale, **THEN BYPASS** channel. **REFER TO** 2-OI-92B.
	OATC	Reports to the NUSO that because APRM 1 failed previously and is bypassed on Panel 2-9-5 that APRM 3 cannot be bypassed.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 6 Page 2 of 3

Event Description: Average Power Range Monitor (APRM) 3 Failure

Time	Position	Applicant's Actions or Behavior	
	OATC	C. N/A D. N/A E. **REFER TO** Tech Spec Tables 3.3.1.1-1, TRM Table 3.3.4-1.	
	Driver	**If contacted as the Reactor Engineer or an Instrument Mechanic, acknowledge any information or direction given.**	
	NUSO	Technical Specification 3.3.1.1 – Reactor Protection System (RPS) Instrumentation LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE. Applicability: According to Table 3.3.1.1-1. NOTE: Separate Condition entry is allowed for each channel. **CONDITION:** One or more required channels inoperable.	
	NUSO	**REQUIRED ACTION:** A.1 Place channel in trip. OR A.2 ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------- A.2 Place associated trip system in trip.	**COMPLETION TIME:** A.1 – 12 hours A.2 N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 6 Page 3 of 3

Event Description: Average Power Range Monitor (APRM) 3 Failure

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Requirements Manual 3.3.4 - Control Rod Block Instrumentation LCO 3.3.4 The control rod block instrumentation for each Function in Table 3.3.4-1 shall be OPERABLE. APPLICABILITY: According to Table 3.3.4-1 NOTE: Separate Condition entry is allowed for each Function. **CONDITION:** One or more required channels inoperable.	
	NUSO	**REQUIRED ACTION:** A.1 Enter the Condition referenced in Table 3.3.4-1 for the Function. B.1 Place at least one inoperable channel in the tripped condition.	**COMPLETION TIME:** A.1 – Immediately B.1 – 1 hour
	NRC	**End of Event 6. Request that the Driver insert Event 7, Inadvertent High Pressure Coolant Injection (HPCI) Initiation.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 7 Page 1 of 3

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. HPCI will start and inject to the Reactor Vessel.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause an inadvertent initiation of HPCI.**
	BOP	Recognizes and reports that HPCI has initiated; acknowledges and reports the following alarm on Panel 2-9-3F: • HPCI TURB EXH DRAIN POT LEVEL HIGH, Window 33 Verifies that no Emergency Core Cooling System (ECCS) initiation signal exists and informs the NUSO.
	NUSO	Verifies that there no valid ECCS initiation signals and directs the BOP to trip and lock out HPCI in accordance with ODM 4.20, Strategies for Successful Transient Mitigation, Section 4.4.2, Operator Expectations for Reactor Water Level Control During Transients.
	BOP	Trips and locks out HPCI.
	OATC	Monitors Reactor Water Level, Reactor Power, and Reactor Pressure. Reports that Reactor Water Level Control has switched to Single Element Control (if applicable).
	Driver	**When contacted by the crew as the Outside/Work Control NUSO, acknowledge any direction given.** **If contacted as an AUO to go to 2A 250V DC RMOV Board to open the breaker for the HPCI Minimum Flow Valve, acknowledge the direction. 2 minutes after being contacted, call the crew back and state that you are ready to open the breaker, and when directed to open the breaker insert Event 17. Inform the crew that the breaker for the HPSI Minimum Flow Valve is open.** **If contacted as the Reactor Engineer, acknowledge any direction given.**
	NRC	**If Reactor Water Level Control transfers to Single Element Control, the crew may elect to transfer Reactor Water Level Control back to Three Element Control.** **If the crew does not decide to place Reactor Water Level Control in Three Element Control, end of Event 7; request that the driver insert Event 8 after the Tech Spec call is complete.**

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 7 Page 2 of 3

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Time	Position	Applicant's Actions or Behavior
	NUSO	If necessary, directs the OATC to transfer Reactor Water Level Control to Three Element Control in accordance with 2-OI-3, Reactor Feedwater System, Section 8.31, Transferring RFWCS Between Single Element and Three Element Control.
	OATC	2-OI-3, Reactor Feedwater System Section 8.31, Transferring RFWCS Between Single Element and Three Element Control **NOTES** 1) This section is performed at Panel 2-9-5. 2) The RFW Control System will not allow THREE ELEMENT operation until ALL of the following permissives are met: • Total Steam Flow greater than 16% • Reactor Water Level Control (PDS), 2-LIC-46-5, in AUTO and at least one RFPT Speed Control (PDS) in AUTO • A Valid Total Steam Flow signal exists (Four good Steam Line Flows, or at least two good Steam Line Flow signals and a valid Turbine First Stage Pressure signal) • A Valid Total Feedwater Line Flow signal exists (Two good Feedwater Line Flow signals, or one good Feedwater Line Flow and three valid individual RFP Flow signals) • At least one valid Narrow Range Level signal [1] **IF** desired to transfer level control from Single Element to Three Element, **THEN** (Otherwise N/A): [1.1] **ENSURE** conditions in Note 2 are met for placing level control in Three Element. [1.2] **CHECK** stable Steam Flow and Feedwater Flow. [1.3] **DEPRESS** 2-HS-46-6/3, THREE ELEMENT pushbutton, and **CHECK** green backlight for pushbutton illuminates. [1.4] **CHECK** green backlight for 2-HS-46-6/1, SINGLE ELEMENT pushbutton, is extinguished. [1.5] **CHECK** Reactor Water Level stable. [2] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 7 Page 3 of 3

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.5.1 – ECCS Operating LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE. Applicability: MODE 1, MODES 2 and 3, except High Pressure Coolant Injection (HPCI) and ADS valves are not required to be OPERABLE with Reactor Steam Dome Pressure ≥150 PSIG. NOTE: LCO 3.0.4.b is not applicable to HPCI. Condition: C. HPCI System inoperable.	
	NUSO	**REQUIRED ACTION:** C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	**COMPLETION TIME:** C.1 – Immediately C.2 – 14 days
	Driver	**If contacted as the Work Control/Outside NUSO to prepare Protected Equipment tags for RCIC, acknowledge the direction.** **If contacted as the Work Control/Outside NUSO to investigate the failure of HPCI to isolate acknowledge the direction.**	
	NRC	**End of Event 7. Request that the driver insert Event 8, EHC Piping Leak / Drywell Steam Leak.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 1 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Symptoms/Cues: When requested by the Chief Examiner, an EHC Piping leak will develop when initiated by the simulator booth. A Drywell Steam leak will occur when the Reactor SCRAM is inserted.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to start the EHC Leak.**
	NRC	**Once the EHC Leak is inserted by the Driver, it takes approximately 2 ½ minutes to receive the EHC Reservoir Low Alarm (Panel 2-9-7B, Window 9).**
	BOP	2-ARP-9-7B, Alarm Response Procedure EHC RESERVOIR LEVEL LOW, Window 9 A. **IMMEDIATELY ASSIGN** a Licensed Operator to continuously monitor the operating EHC pump's amperage and pressure for fluctuations and continue with this procedure. B. **IF** while performing this procedure, OR during the EHC Reservoir Level Low transient, the operating pump's amperage or pressure begins to fluctuate (indicative of pump cavitation), **THEN IMMEDIATELY PERFORM** the following: 1. **INITIATE** Reactor SCRAM. 2. **ENSURE** Main Turbine trip. 3. **TRIP** and **LOCK** OUT EHC Pumps. C. **DISPATCH** personnel to verify level on local gauge.
	Driver	**If contacted as the Turbine Building Assistant Unit Operator (AUO) or Outside/Work Control NUSO, acknowledge any information or direction given.**
	NUSO	D. **ADD** fluid. **REFER TO** 2-OI-47A, EHC System. E. **INITIATE** search for cause of low level such as leaking servo, leaking line, or open valve. F. **NOTIFY** Radwaste to LOCKOUT all 3 Units' Turbine Building Equipment and Floor Drain Sump Pumps.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 2 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	Driver	**If dispatched to investigate the EHC problem and look for leaks wait approximately 5 minutes and report there is a leak in the EHC system, but you cannot tell where it is coming from.** **If contacted as the Radwaste Operator, acknowledge any information or direction given.**
	NUSO	G. **IF** the source of the leak is identified to be a leaking CV or MSV servo valve, **THEN IMMEDIATELY REFER TO** 2-OI-47. H. IF unable to restore and maintain minimum EHC tank level (above -4"), **THEN INITIATE** load reduction. I. **REFER TO** Spill Prevention Control and countermeasure plan RWI-007.
	NRC	**EHC Pressure will reach 1300 psig (lowering) in approximately 6 minutes after inserting Event 8.**
	BOP	Acknowledges and reports the following alarms to the NUSO: • EHC HYD FLUID HDR PRESS LOW, 2-9-7B, Window 1 • STANDBY EHC PUMP RUNNING, 2-9-7B, Window 8
	BOP	2-ARP-9-7B, Alarm Response Procedure EHC RESERVOIR LEVEL LOW, Window 8 A. **ENSURE** 2-HS-47-2A(1A), Standby EHC PUMP 2B(2A), running. B. **CHECK** EHC HEADER PRESSURE indicator, 2-PI-47-7 between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC Pump unit and associated piping. D. **REFER TO** 2-OI-47A, EHC System.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 3 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	BOP	2-ARP-9-7B, Alarm Response Procedure EHC HYD FLUID HDR PRESS LOW, Window 1 A. **ENSURE** Standby EHC PUMP 2B(2A), 2-HS-47-2A(1A) running. B. **CHECK** EHC HEADER PRESSURE indicator, 2-PI-47-7 between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC pump unit. **NOTE** On EHC Hydraulic System failure, accumulator and check valve arrangement will provide approximately one minute bypass valve operation. D. **IF** EHC Hydraulic System fails, **THEN ENSURE** Turbine trips at or below 1100 psig.
	NUSO	2-OI-47A, EHC System Precautions and Limitations I. Loss of the EHC System will result in a Turbine-Generator trip. L. EHC Hydraulic Pumps should NOT be operated when fluid level in the reservoir is below the low level alarm setpoint. N. EHC Hydraulic Fluid Pump Motor Current of 140 amps relates to a system flow of 100 gpm (maximum set pump flow). If motor current approaches this value, the EHC System should be checked for leaks, both internal and external.
	NUSO	When there are indications of an EHC leak, directs the OATC to insert a manual Reactor SCRAM.
	OATC	Inserts a manual Reactor SCRAM.
	NRC	**When the Reactor MODE SWITCH is placed in SHUTDOWN, the following Events will occur:** • **LT-3-208A and C Fail High (See Event 9 on page 28 of 36), tripping RCIC and RFPTs** • **One (1) minute later the Drywell Steam Leak will begin ramping in over a 15 minute period.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 4 of 11

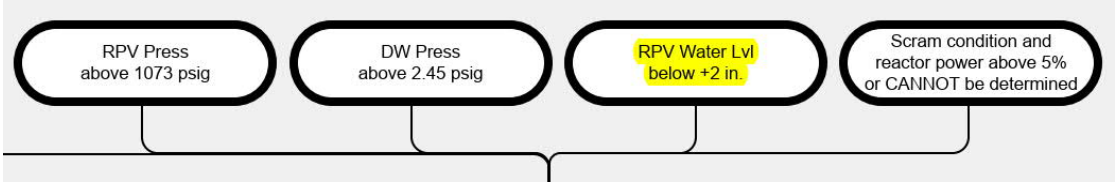
Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	BOP	Stops and locks out the EHC Pumps.
	OATC	2-AOI-100-1, Reactor SCRAM / Hard Card 2.1 IMMEDIATE ACTIONS [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A and 2-HS-99-5A/S3B, REACTOR SCRAM B on Panel 2-9-5. [2] **PLACE** 2-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] N/A [4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the Unit SRO: • Reactor SCRAM • MODE SWITCH is in SHUTDOWN • All rods in • Reactor Water Level and trend (recovering or lowering) • Reactor pressure and trend • MSIV position (Open or Closed) • Power level [5] N/A
	BOP	2.2 SUBSEQUENT ACTIONS: PANELS 2-9-7 & 2-9-8 [1] **WHEN** ≤50 MWe, or as directed by the Unit SRO, **THEN ENSURE** TRIPPED the Main Turbine as follows: [1.1] **DEPRESS** 2-HS-47-67D, TRIP pushbutton, on Panel 2-9-7. [1.2] **ENSURE** OPEN GENERATOR PCB 224. [1.3] **ENSURE** TRIPPED VOLTAGE REGULATOR using 2-HS-57-24. [2] **ANNOUNCE** Reactor SCRAM over PA system.
	OATC / BOP	2.3 SUBSEQUENT ACTIONS: PANEL 2-9-3 & 2-9-5 **NOTE** The following steps are not required to be performed in order, but only as required to maintain stable conditions.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 5 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[1] **MONITOR** and **CONTROL** RPV Pressure to keep below 1073 psig and stable, or as directed by Unit SRO. [1.1] N/A [1.2] N/A [1.3] **IF** MSRVs are cycling and bypass valves are NOT available, **THEN ENSURE OPEN** MSRVs on Panel 2-9-3 until RPV Pressure is controlled between 800 and 1000 psig. [2] **IF** any PCIS Isolation Signal is received, **THEN ENSURE** PCIS Isolations using any of the following: [2.1] Containment Isolation Status System on Panel 2-9-4. [2.2] PCIS Mimic and individual control switch indications. [2.3] ICS. [2.4] 2-OI-64, Primary Containment System. [3] **IF** HPCI and/or RCIC are in service and injecting to the vessel, **THEN MONITOR** and **CONTROL** Reactor Water Level as necessary.
	NRC	**NOTE: See Event 9 for Reactor Water Level Control Actions.**
	OATC / BOP	[4] **DRIVE** in all IRM's and SRM's from Panel 2-9-5 as time and conditions permit. [5] **ENSURE** SCRAM DISCH VOL VENT & DR VLVS closed by green **indicating** lights at SDV Display on Panel 2-9-5 [6] **RETURN** to 2-AOI-100-1, Subsequent Actions Section, **AND CONTINUE** with actions as required.
	NUSO	Enters 2-EOI-1, RPV Control Modes 1-3 following the Reactor SCRAM based on Reactor Water Level below +2". 

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 6 of 11

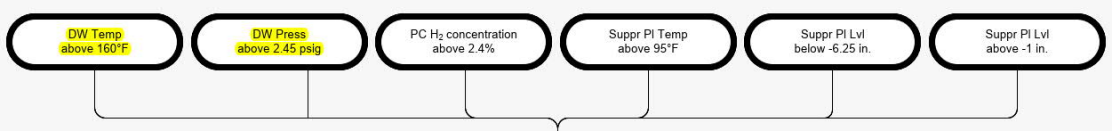
Event Description: EHC Piping Leak / Drywell Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


RC/P-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>A high Drywell Pressure ECCS Signal exists (2.45 psig)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Emergency RPV Depressurization is required or has been required</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Emergency RPV Depressurization is anticipated</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Pool Temp and Level CANNOT be maintained in a safe area of Curve 3 at the existing RPV Pressure
OR
Suppression Pool Level CANNOT be maintained in the safe area of Curve 4</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-2
<table>
<tr><td colspan="2">STABILIZE RPV Pressure below 1073 psig using the Main Turbine Bypass Valves (APPX 8B)
➢ OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
➢ Crosstie CAD or MSRV carts to Drywell Control Air (APPX 8G, 20H) if necessary</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the OATC or BOP to maintain Reactor Pressure in a band below 1073 psig.</td></tr>
</table>

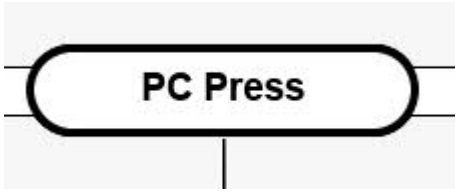
Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 7 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	CREW	Acknowledges and reports the following Drywell Pressure and Temperature alarms when received: • PRI CONTAINMENT N2 PRESS HIGH, 2-9-3B, Window 10 • DRYWELL NORM OPERATING PRESS HIGH, 2-9-3B, Window 19 • DRYWELL PRESS APPROACHING SCRAM, 2-9-3B, Window 30 • DRYWELL ATMOSPHERIC TEMP HIGH, 2-9-3B, Window 3 • DRYWELL PRESSURE ABNORMAL, 2-9-5B, Window 31 • DRYWELL TEMP HIGH, 2-9-3B, Window 16
	NRC	**NOTE: Due to the multitude of alarms received following the Reactor SCRAM and the rate of rise of Drywell Pressure, the crew may not have time to address 2-OI-64, Primary Containment System, or 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell. The crew may just use the EOI Appendices to address the steam leak into the Drywell.**
	NUSO	Re-enters 2-EOI-1, RPV Control Modes 1-3 and enters 2-EOI-2, Primary Containment Control when Drywell Pressure exceeds 2.45 psig. 2-EOI-1  2-EOI-2 (NOTE: 2-EOI-2 will be re-entered when Drywell Temperature exceeds 160º F). 

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 8 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	

PC/P-1

MONITOR and **CONTROL** PC press below 2.45 psig using the Vent system (AOI-64-1)

PC/P-2

WHEN Primary Containment Pressure CANNOT be maintained below 2.45 psig

PC/P-3

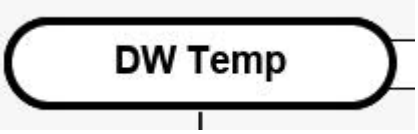
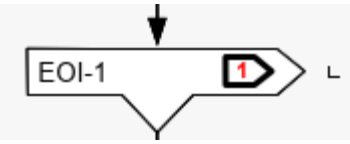
IF	THEN
Primary Containment Pressure reduction is required to restore and maintain Adequate Core Cooling or reduce total offsite radiation dose	**NO ACTION REQUIRED**
Primary Containment Pressure CANNOT be maintained above (-)2 psig	**NO ACTION REQUIRED**
Suppression Chamber Pressure approaches 0 psig	**NO ACTION REQUIRED**
Drywell Pressure approaches 0 psig	**NO ACTION REQUIRED**

PC/P-4

BEFORE Suppression Chamber Pressure rises to 12 psig

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 9 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>PC/P-5
INITIATE Suppression Chamber Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17C)
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Needed to augment Suppression Chamber Sprays</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>When Drywell Temperature exceeds 160º F, re-enters 2-EOi-2, Primary Containment Control.
 
DW/T-1
MONITOR and CONTROL Drywell Temperature below 160° F using all available Drywell cooling
DW/T-2
WHEN Drywell Temperature CANNOT be maintained below 160° F
DW/T-3
BEFORE Drywell Temperature rises to 280º F
  </td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 10 of 11

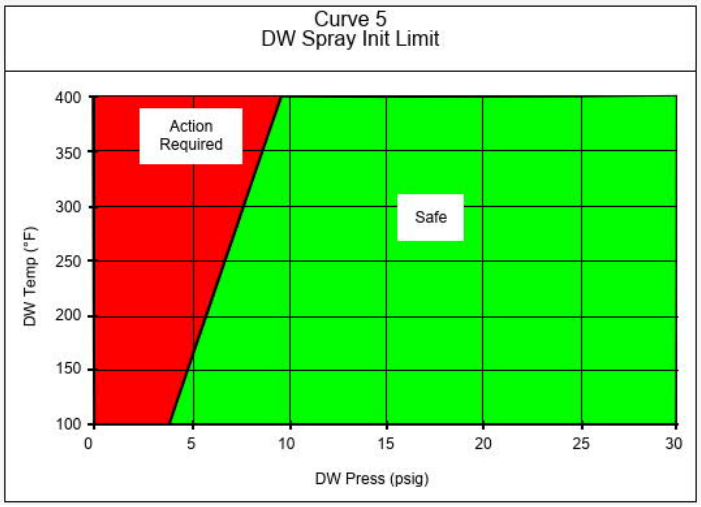
Event Description: EHC Piping Leak / Drywell Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
DW/T-4
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
DW/T-5
<table>
<tr><td colspan="2">IF Suppression Pool Water Level is below 19 ft
AND
Drywell Temperature is in the safe area of Curve 5
THEN
1. SHUT DOWN Recirc Pumps
2. SHUT DOWN Drywell Blowers
3. INITIATE Drywell Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B, 20L)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to Augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>Crew</td><td>Critical Task:
When Drywell Sprays are required, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve to preclude Primary Containment failure.

Critical Task Failure Criteria:
Drywell Sprays are initiated with Drywell Pressure and Temperature outside the safe area of the Drywell Spray Initiation Curve (Curve 5).</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 11 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	Crew	
	NRC	**When the crew is spraying the Drywell and has control of Reactor Water Level using High Pressure Makeup sources (See Event 9 on page 28 of 36), end of Event 8 and end of Scenario. Request that the Driver place the Simulator in FREEZE.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 1 of 7

Event Description: LT-3-208A and C Fail High

Symptoms/Cues: Event is automatically entered on Simulator setup; LT-3-208A and C will fail high when the Reactor MODE SWITCH is placed in SHUTDOWN.

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Driver</td><td>Event 9 is automatically entered when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the Driver to cause LT-3-208A and LT-3-208C to fail high and therefore a loss of RCIC and RFPTs.</td></tr>
<tr><td></td><td>OATC</td><td>Reports that RFPTs have tripped.</td></tr>
<tr><td></td><td>BOP</td><td>Recognizes and reports that RCIC has a sealed in high level trip.</td></tr>
<tr><td></td><td>Driver</td><td>If contacted as the Work Control/Outside SRO or an AUO to investigate, acknowledge the direction.</td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-1, RPV Control Modes 1-3

RC/L-1
ENSURE each as required:
• PCIS isolations (Groups 1, 2 and 3)
• ECCS
• RCIC

RC/L-2
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Raising RPV Water Level above +51 in. will facilitate use of shutdown cooling, steam-driven injection systems, or Alternate Injection Subsystems (Table L-2)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 2 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	NUSO	RC/L-3 **RESTORE** and **MAINTAIN** RPV Water Level between +2 in. and +51 in. with ANY Preferred Injection Systems (Table L-1) ➢ OK to use ANY Alternate Injection Subsystems (Table L-2) Directs the BOP to maintain Reactor Water Level using any available methods in Table L-1 or L-2. **NOTE: RCIC and Reactor Feedwater are not available due to the High Water Level Trip. HPCI may be used if necessary, even though it was previously locked out due to an inadvertent initiation.**

Table L-1
Preferred Injection Systems

SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available 2 3 5	5C, 20M	1200 psig
HPCI with CST suction if available 2 5 7	5D, 20N	1200 psig
CNDS	6A	470 psig
CS 2	6D, 6E	330 psig
LPCI 2	6B, 6C	320 psig

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 3 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	NUSO	Table L-2 Alternate Injection Subsystems

SOURCE	APPX	INJ PRESS
EHPM Pump	7L	1210 psig
SLC (test tank)	7B	1450 psig
SLC (boron tank)	7B	1450 psig
CNDS transfer pumps to RHR and CS	7A	110 psig
RHR crosstie to other units	7C	320 psig
Stby coolant	7D	160 psig
RHR drain pumps	7E, 7F	50 psig
PSC head tank pumps	7G	30 psig
RCIC (aux boiler steam) with CST suction if available (2) (3) (5)	7H	1200 psig
RCIC manual start	20A	1200 psig
HPCI (aux boiler steam) with CST suction if available (2) (5) (7)	7J	780 psig
Fire Protection system	7K	150 psig
FLEX Pump Sys (CILRT/CS)	20D	150 psig
FLEX Pump Sys (Standby Coolant)	20B	150 psig
FLEX Pump Sys (CILRT/CRD)	20C	150 psig

	NRC	**NOTE: If the crew decides to use HPCI for Reactor Water Level Control, some valves will already be in the OPEN position due to the inadvertent initiation. They are noted on the procedure below. Additionally, as soon as the HPCI Aux Oil Pump is taken out of the PULL TO LOCK position, the HPCI Turbine will start and inject to the Reactor.**

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 4 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI-Appendix-5D, Injection System Lineup HPCI [1] N/A [2] N/A [3] N/A [4] **VERIFY** at least one SGTS train in operation. **CAUTIONS** 1) Operating HPCI Turbine below 2400 rpm may result in unstable system operation and equipment damage. 2) Operating HPCI Turbine with suction temperatures above 140º F may result in equipment damage. [5] **VERIFY** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection [6] **IF** high Reactor Water Level trip logic is actuated, **THEN** [6.1] **DEPRESS** HPCI TURBINE TRIP REACTOR LEVEL HIGH RESET pushbutton. [6.2] **CHECK** HPCI TURBINE TRIP LEVEL HIGH amber light has extinguished. [7] **PLACE** HPCI AUXILIARY OIL PUMP handswitch in START. **HPCI WILL START AND INJECT** [8] **PLACE** HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 2-FCV-73-30, HPCI PUMP MINIMUM FLOW VALVE. **CLOSED AND DE-ENERGIZED DURING EVENT 7** [10] **OPEN** 2-FCV-73-44, HPCI PUMP INJECTION VALVE. **ALREADY OPEN** [11] **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine. **ALREADY OPEN**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 5 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	BOP	[12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 2-CKV-73-45, HPCI SYSTEM CHECK VLV, opens by observing 2-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 gpm **CLOSED AND DE-ENERGIZED DURING EVENT 7** [13] **ADJUST** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO.
	BOP	If directed to use the EHPM instead of HPCI 2-EOI-APPENDIX-7L, Alternate RPV Injection System Lineup EHPM System [1] **CHECK** 4KV EHPM BD 2 is energized from normal power supply on 2-EI-7-410A, EHPM NORMAL SOURCE VOLTAGE. [2] N/A [3] N/A [4] N/A

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 6 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTIONS** 1) Pump minimum continuous stable operating flow is 431 gpm and maximum continuous stable operating flow is 1250 gpm. Above 1250 gpm, adequate NPSH is not assured. 2) Prolonged EHPM pump operation on minimum flow or test return mode will add heat to the CST. As decay heat dissipates, it may be necessary to cycle EHPM pump to minimize extended run times. Ref motor start limitations per 2-OI-7. 3) If power is being supplied to more than one EHPM Board by the Supplemental Diesel Generators, then coordination between units is required to ensure a duration of at least 10 seconds between starting EHPM Pumps. [5] **ESTABLISH** Unit 2 RPV injection in BATCH mode as follows: [5.1] **START** EHPM PUMP by placing 2-HS-7-1A, to START. [5.2] **THROTTLE** 2-FCV-007-0008, EHPM PUMP INJECTION VALVE, as necessary to establish flow 950-1250 gpm as indicated on EHPM SYS NORMAL FLOW 2-FI-7-403A. [5.3] **MONITOR** Unit 2 RPV Level on 2-LI-3-148AA, RPV LEVEL 'A', and 2-LI-3-148BA, RPV LEVEL 'B'. [5.4] **WHEN** desired RPV level is reached, **THEN CLOSE** 2-FCV-007-0008, EHPM PUMP INJECTION VALVE. [5.5] **REPEAT** steps 1.0[5.2] through 1.0[5.4] as necessary to maintain RPV level. [6] N/A [7] N/A [8] **MONITOR** the Unit 2 CST level and as refill as necessary per 2-OI-7.
	CREW	**Critical Task:** **Following a loss of high pressure injection, restore high pressure injection to the Reactor Vessel to preclude Emergency Depressurization due to low Reactor Water Level.** **Critical Task Failure Criteria:** **Emergency Depressurization is required based on low Reactor Water Level.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 7 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	NRC	**When the crew is spraying the Drywell and has control of Reactor Water Level using High Pressure Makeup sources, end of Event 8 and end of Scenario. Request that the Driver place the Simulator in FREEZE.**

Scenario Setup
UNIT 2

IC	28
Exam IC	285

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-3	163	AOI-85-3	26		
OI-47A	53	AOI-100-1	121		
OI-68	168	TRM 3.3.4	0		
OI-85	156	T.S. 3.3.1.1	345		
OI-92B	44	T.S. 3.5.1	334		
ARP-3A	58	OPDP-1	55		
ARP-3B	40	EOI-1	21		
ARP-3F	42	EOI-2	18		
ARP-5A	63	APPX-5D	10		
ARP-7B	39	APPX-7L	6		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	ILT 2304 NRC Scenario 1 UNIT 2.sch
Event Files(s):	ILT 2304 NRC Scenario 1 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ON LINE	Drywell Leakage (GPM)	Protected Equipment
	230		
		1.89	
Rx Power	PRA (Phoenix) -Green	Floor Drain (GPM)	
80%	500Kv GRID - Qualified	0.31	
MWe	161Kv Grid -Qualified	Equipment Drain (GPM)	
1025		1.58	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

Control Rod 30-43 Directional Control Valve Maintenance completed.

Alternated Reactor/Refuel Zone Exhaust Fans, 'B' fans in service, 'A' fans tagged for maintenance

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Perform 2A RFPT Overspeed Trip Exerciser Test.

Reduce Reactor Power to < 75% using Core Flow in preparation for recovering Control Rod 30-43.

Wait for Reactor Engineer to brief Control Rod 30-43 recovery.

OPERATOR WORK AROUNDS **OWAs – U0-1/U2-0** **Burdens - U0 – 0/U2-0** **Operator Challenges – U0 – 12/U2 - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-1
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 80% Reactor Power, 'B' Reactor/Refuel Zone Exhaust Fans are running. Control Rod 30-43 is at position 00 for Directional Control Valve Maintenance.

Turnover: Perform 3A RFPT Overspeed Trip Exerciser Test. Reduce Reactor Power to < 75% using Core Flow in preparation for recovering Control Rod 30-43.

Critical Tasks:
1. When Drywell Sprays are required, initiate sprays while in the safe region of the Drywell Spray Initiation Limit Curve to preclude Primary Containment failure.
2. Following a loss of high pressure injection, restore high pressure injection to the Reactor Vessel to preclude Emergency Depressurization due to low Reactor Water Level.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Reactor Feedwater Pump (RFPT) 3A Overspeed Trip Exerciser Test
2.	N/A	R-OATC	Reduce Reactor Power for Control Rod Recovery and Testing
3.	NMAPRMGAIN(1)	C-OATC C-NUSO	Average Power Range Monitor (APRM) 1 Failure
4.	RD22	I-OATC MC-OATC I-NUSO	Control Rod Drive (CRD) Flow Transmitter Failure
5.	XA-55-3A_26	C-BOP C-NUSO	PSC Head Tank Low Level Alarm
6.	NMAPRMGAIN(3)	TS-NUSO	Average Power Range Monitor (APRM) 3 Failure
7.	HP01	C-BOP MC-BOP TS-NUSO	Inadvertent High Pressure Coolant Injection (HPCI) Initiation
8.	TC07 TH33A	M-ALL	EHC Piping Leak / Drywell Steam Leak
9.	TH30E TH30G	C-BOP C-NUSO	LT-3-208A and C Fail High

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will perform the Overspeed Trip Exerciser Test on Reactor Feedwater Pump (RFPT) 3A in accordance with 3-OI-3, Section 8.10.
2. The crew will reduce Reactor Power using Core Flow in preparation for recovering Control Rod 30-43 and performing Scram Time Testing.
3. APRM 1 will experience a critical fault, requiring the crew to respond in accordance with the appropriate Alarm Response Procedure. The crew will bypass APRM 1 in accordance with 3-OI-92B, Average Power Range Monitoring.
4. The CRD Flow Transmitter will fail high, requiring the crew to respond in accordance with Alarm Response Procedures and 3-OI-85, Control Rod Drive System, to restore proper CRD System flow.
5. The crew will receive the PSC Head Tank Low Level Alarm (Panel 3-9-3A, Window 26) and will respond in accordance with the Alarm Response Procedure to manually start the PSC Head Tank Pumps.
6. APRM 3 will fail high. Since APRM 1 is already bypassed due to a failure, APRM 3 cannot be bypassed, requiring the Nuclear Unit Senior Operator to address the Technical Specifications.
7. HPCI will inadvertently initiate, requiring the crew to trip and lock out the HPCI Turbine. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
8. A leak will develop in the EHC System, eventually requiring the crew to insert a manual Reactor SCRAM. When the Reactor SCRAM is inserted, a steam leak will occur in the Drywell, requiring the crew to respond in accordance with the Abnormal and Emergency Operating Procedures.
9. Following the Reactor SCRAM, Reactor Water Level Instruments LT-3-208A and LT-3-208C will fail high, causing a trip of Reactor Feedwater Pumps and Reactor Core Isolation Cooling (RCIC). The crew will be required to line up other high pressure injection methods to provide for Reactor Water Level Control to prevent Emergency Depressurization.

The Scenario ends when the crew has sprayed the Drywell, and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available high-pressure injection system.

Critical Tasks 2

1. **When Drywell Sprays are required, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve to preclude Primary Containment failure.**

   a. **Safety Significance:**

   Precludes failure of Primary Containment.

   b. **Cues:**

   Procedural compliance.
   High Drywell and Suppression Chamber Pressure.
   High Drywell Temperature.

   c. **Measured by:**

   Observation – the crew verifies Primary Containment parameters prior to initiating Drywell Sprays.

   d. **Feedback:**

   Drywell Pressure and Temperature lowering.
   Suppression Chamber Pressure lowering.
   RHR Flow to Containment Cooling and Sprays.

   e. **Critical Task Failure Criteria:**

   Drywell Sprays are initiated with Drywell Pressure and Temperature outside the safe area of the Drywell Spray Initiation Curve (Curve 5).

2. **Following a loss of high pressure injection, restore high pressure injection to the Reactor Vessel to preclude Emergency Depressurization due to low Reactor Water Level.**

   a. **Safety Significance:**

   Prevents challenging plant safety by preventing the initiation of Emergency Depressurization.

   b. **Cues:**

   Procedural compliance.
   Loss of Reactor Feedwater Pump.
   Lowering Reactor Water Level.

   c. **Measured by:**

   Observation – the crew restores high pressure injection to the Reactor Vessel.

   d. **Feedback:**

   Rising Feedwater Flow indication.
   Rising Reactor Water Level.

   e. **Critical Task Failure Criteria:**

   Emergency Depressurization is required based on low Reactor Water Level.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 1 Page 1 of 3

Event Description: Reactor Feedwater Pump (RFPT) 3A Overspeed Trip Exerciser Test

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**NOTE: The crew has been pre-briefed, but may elect to hold a re-focus brief.**
	NUSO	Directs the Balance of Plant Operator (BOP) to conduct the Overspeed Trip Exerciser Test on 3A Reactor Feedwater Pump (RFPT) in accordance with 3-OI-3, Reactor Feedwater System, Section 8.9.
	BOP	3-OI-3, Reactor Feedwater System Section 8.10, Overspeed Trip Exerciser Test **NOTES** 1) The following steps will test circuitry associated with overspeed trip test and mechanical overspeed trip lockout. The test may be performed at speed or stopped; it should be performed prior to rolling turbine off turning gear and periodically with applicable RFP in operation. This test will normally be scheduled during reductions in power with reactor power less than 90%. 2) The success of this test depends on mechanical overspeed trip bolt actuating as a result of oil pressure. The system was designed for this test to be performed with the RFPT at normal operating speed. The test may NOT function properly at less than 75% speed (4125 rpm) due to trip mechanism being sluggish at speeds well below rated. If this is the case, then no action is required except to perform test at a higher speed. 3) The following steps are performed at Panel 3-9-6. [1] **OBTAIN** Unit SRO approval to perform test. [2] **ENSURE** Main Oil Pump running for RFPT being tested: • RFPT 3A 3A1 MAIN OIL PUMP 3-HS-3-103A • RFPT 3A 3A2 MAIN OIL PUMP 3-HS-3-250

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 1 Page 2 of 3

Event Description: Reactor Feedwater Pump (RFPT) 3A Overspeed Trip Exerciser Test

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **PERFORM** the following for the RFPT to be Tested. [3.1] **PLACE** 3-HS-3-109A, RFPT 3A OVERSPEED TEST TRIP LOCKOUT, in MECH, and **CHECK** indications: • Green (normal) light extinguished at switch • Amber (mechanical lockout) light to the right of the green light is illuminated [3.2] N/A [3.3] N/A [4] **PERFORM** the following RFPT Overspeed Test: [4.1] **DEPRESS** and **HOLD** 3-HS-3-110, RFPT 3A OVERSPEED TEST, on Panel 3-9-6, and **CHECK** indications: • White (trip) light illuminated at pushbutton • Green (normal) light extinguished at pushbutton [4.2] N/A [4.3] N/A [5] **RELEASE** RFPT OVERSPEED TEST pushbutton. [6] PERFORM the following RFPT Overspeed Test Reset. [6.1] **DEPRESS** and **HOLD** 3-HS-3-106, RFPT 3A OVERSPEED TEST RESET, and **CHECK** the following on Panel 3-9-6: • 3-HS-3-106, RFPT 3A OVERSPEED TEST RESET, white (trip) light illuminated • 3-HS-3-110, RFPT 3A OVERSPEED TEST, white (trip) light extinguished [6.2] N/A [6.3] N/A [7] **RELEASE** Overspeed Test Reset on Panel 3-9-6, and CHECK the following: [7.1] RFPT 3A • 3-HS-3-106, RFPT 3A OVERSPEED TEST RESET, white (trip) light extinguished. • RFPT 3A OVERSPEED TEST, 3-HS-3-110, green (normal) light illuminated. [7.2] N/A

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 1 Page 3 of 3

Event Description: Reactor Feedwater Pump (RFPT) 3A Overspeed Trip Exerciser Test

Time	Position	Applicant's Actions or Behavior
	BOP	[7.3] N/A **NOTE** Waiting 30 seconds before placing Overspeed Test Trip Lockout switch in normal position in Step 8.10[8] allows ample time for Overspeed test trip logic to reset. Waits 30 seconds before continuing.
	BOP	[8] **PERFORM** the following, when 30 seconds have elapsed for the RFPT to be Tested. [8.1] **PLACE** 3-HS-3-109A, RFPT 3A OVERSPEED TEST TRIP LOCKOUT, in NORM, and CHECK indications: • Green (normal) light illuminated at switch • Amber (mechanical lockout) light to the right of the green light is extinguished [8.2] N/A [8.3] N/A [9] **NOTIFY** Unit SRO of test completion. [10] **RECORD** results in Narrative Log.
	NRC	**End of Event 1. Proceed to Event 2, Reduce Reactor Power for Control Rod Recovery and Testing.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 2 Page 1 of 2

Event Description: Reduce Reactor Power for Control Rod Recovery and Testing

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew has pre-briefed Event 2, but may elect to conduct a re-focus brief.**
	NUSO	Directs the Operator at the Controls (OATC) to reduce Reactor Power to ≤ 75% in accordance with 3-OI-68, Reactor Recirculation System and Reactivity Control Plan (RCP) RCP-NRC2304-U3S1.
	NRC	**The crew may elect to reduce Reactor Power using any method below.**
	OATC	3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown on 0-TI-248, Unit 3 Core Operating Limits Report (COLR) and 3-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the latter part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor Pump Speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pump 3A Speed with Recirc Individual Control, **THEN PERFORM** the following: [1.1] N/A [1.2] Lower Recirc Pump 3A Speed using 3-HS-96-17A(17B)(17C), SLOW (MEDIUM) (FAST)

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 2 Page 2 of 2

Event Description: Reduce Reactor Power for Control Rod Recovery and Testing

Time	Position	Applicant's Actions or Behavior
	OATC	[2] **IF** desired to control Recirc Pump 3B Speed with Recirc Individual Control, **THEN PERFORM** the following: [2.1] N/A [2.2] Lower Recirc Pump 3B Speed using 3-HS-96-18A(18B)(18C), SLOW (MEDIUM) (FAST). [3] **WHEN** desired to control Recirc Pumps 3A and/or 3B Speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 3A & 3B using the following push buttons as required: 3-HS-96-33, LOWER SLOW 3-HS-96-34, LOWER MEDIUM 3-HS-96-35, LOWER FAST
	NRC	**End of Event 2. Request that the driver insert Event 3, Average Power Range Monitor (APRM) 1 Failure**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 3

Event Description: Average Power Range Monitor (APRM) 1 Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. A critical fault APRM 1 will occur.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to insert a critical fault on APRM 1.**
	OATC	Acknowledges and reports the following alarms to the NUSO: • CONTROL ROD WITHDRAWAL BLOCK, 3-9-5A, Window 7 • APRM HIGH/INOPOR OPRM TRIP, 3-9-5A, Window 25
	NUSO	Directs the Balance of Plant Operator (BOP) and the OATC to respond in accordance with the Alarm Response Procedures.
	OATC	3-ARP-9-5A, Alarm Response Procedure CONTROL ROD WITHDRAWAL BLOCK, Window 7 Operator Actions: A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s) and **REFER TO** operator action for alarm(s). B. N/A C. N/A D. N/A E. N/A
	OATC	3-ARP-9-5A, Alarm Response Procedure APRM HIGH/INOPOR OPRM TRIP, Window 25 A. **CHECK** alarm by multiple indications. B. **BYPASS** with UNIT SRO permission, initiating channel to reset the alarm and Voters. **REFER TO** 3-OI-92B, Average Power Range Monitoring. C. N/A D. N/A E. N/A F. N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 3 Page 2 of 2

Event Description: Average Power Range Monitor (APRM) 1 Failure

Time	Position	Applicant's Actions or Behavior
	OATC	G. **NOTIFY** Reactor Engineer H. **CHECK** thermal limits on RUNMON.
	Driver	**If contacted as the Reactor Engineer, acknowledge any report or direction given.**
	NUSO	I. **REFER** TO Tech Spec Table 3.3.1.1-1, TRM Table 3.3.4-1. J. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5. Directs the OATC to bypass APRM 1 in accordance with 3-OI-92B, Average Power Range Monitoring.
	NRC	**EXAMINER NOTE: The NUSO may reference Tech Specs and the TRM. There are no required Tech Spec or TRM actions, but the NUSO may enter an Information Only LCO.**
	OATC	3-OI-92B, Average Power Range Monitoring Section 6.1, Bypassing APRM/OPRM Channel **CAUTION** NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with APRMs bypassed unless bypassing is specifically allowed within approved procedures [1] **REVIEW** all Precautions and Limitations. **REFER TO** Section 3.0. [2] **PLACE** 3-HS-92-7B/S3, APRM BYPASS, to desired channel to be bypassed. [3] **CHECK** BLUE BYPASSED lights illuminated on Panel 3-9-14, Voters. [4] **CHECK** white bypass light on Panel 3-9-5 is illuminated.
	NRC	**End of Event 3. Request that the driver insert Event 4, Control Rod Drive (CRD) Flow Transmitter Failure.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 4 Page 1 of 2

Event Description: Control Rod Drive (CRD) Flow Transmitter Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The CRD Flow Transmitter will fail high, causing the CRD Flow Control Valves to close.

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, insert Event 4, Control Rod Drive (CRD) Flow Transmitter Failure.**
	OATC	Acknowledges and reports the following alarm to the Nuclear Unit Senior Operator (NUSO): • CRD ACCUMULATOR CHARGING WATER HEADER PRESSURE HIGH, 3-9-5A, Window 10
	NUSO	Acknowledges the OATC's alarm report. Directs the OATC to respond in accordance with applicable Alarm Response Procedures and subsequently 3-OI-85, Control Rod Drive System.
	OATC	3-ARP-9-5A, Alarm Response Procedure CRD ACCUM CHG WTR HDR PRESS HIGH, 3-9-5A, Window 10 A. **CHECK** pressure high on 3-PI-85-13A, CRD ACCUMULATOR CHARGING WATER HEADER on Panel 3-9-5. B. **CHECK** 3-FCV-85-11A (B), CRD LINE A(B) FLOW CONTROL VALVE, in service.
	NRC	**The crew may attempt to switch Flow Control Valves in accordance with 3-AOI-85-3, CRD System Failure. However, as long as the Flow Transmitter is failed high, neither set of Flow Control Valves will operate in automatic. 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, must be placed in MANUAL.**
	OATC	C. **IF** in-service controller has failed, **THEN REFER TO** 3-OI-85, Control Rod Drive System. D. N/A
	OATC	Determines that the CRD Flow Transmitter has failed high, causing 3-FCV-85-11A, CRD LINE A FLOW CONTROL VALVE to CLOSE. In accordance with OPDP-1, Conduct of Operations, Section 3.5.3, Manual Control of Automatic Systems, takes manual control of 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, to restore CRD Parameters back to normal.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 4 Page 2 of 2

Event Description: Control Rod Drive (CRD) Flow Transmitter Failure

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted by the crew as the Outside/Work Control NUSO, Reactor Engineer, or Instrument Mechanics, acknowledge any information or direction given.**
	OATC	3-OI-85, Control Rod Drive System Section 8.33, AUTOMATIC/MANUAL operation of 3-FIC-85-11 [1] **REVIEW** all Precautions and Limitations in Section 3.6. [2] **IF** transferring 3-FIC-85-11 from AUTO to MANUAL, **THEN**: [2.1] **PLACE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, in **BALANCE**. [2.2] **BALANCE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, by turning Manual Control Pot inside Control Selector Wheel until red deviation pointer is in the Green Band. [2.3] **PLACE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, in MANUAL. [2.4] **ADJUST** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, manual potentiometer to establish the desired system flow. Refer to Section 5.1 or Section 6.11. [3] N/A
	NRC	**End of Event 4. Request that the Driver insert Event 5, PSC Head Tank Low Level Alarm.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 5 Page 1 of 1

Event Description: PSC Head Tank Low Level Alarm

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The PSC Head Tank Low Level Alarm will be received on Panel 3-9-3.

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, insert Event 4, PSC Head Tank Low Level Alarm.**
	BOP	Acknowledges and reports the following alarm to the Nuclear Unit Senior Operator (NUSO): • PSC HEAD TANK LEVEL LOW, 3-9-3A, Window 26
	NUSO	Acknowledges the OATC's alarm report. Directs the OATC to respond in accordance with the applicable Alarm Response Procedure.
	BOP	3-ARP-9-3A, Alarm Response Procedure PSC HEAD TANK LEVEL LOW, 3-9-3A, Window 26 A. **CHECK** both PSC Head Tank Pumps are running.
	NRC	**15 seconds after the PSC Head Tank Pumps are started, the PSC Head Tank Low Level Alarm can be cleared and the Head Tank Pumps will automatically stop as expected.**
	BOP	Starts 3-HS-75-75A, 3A PSC Head Tank Pump, and 3-HS-75-76A, 3B PSC Head Tank Pump.
	Driver	**If contacted by the crew as the Outside/Work Control NUSO, Electrical Maintenance, or Instrument Mechanics, acknowledge any information or direction given.**
		B. **CHECK** power available to pumps. C. **CHECK** 3-FCV-75-57, PSC PUMP SUCTION INBOARD VALVE and 3-FCV-75-58, OUTBOARD ISOLATION VALVE, open. D. N/A E. N/A F. N/A
	NRC	**End of Event 5. Request that the Driver insert Event 6, Average Power Range Monitor (APRM) 3 Failure.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 6 Page 1 of 3

Event Description: Average Power Range Monitor (APRM) 3 Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. APRM 3 will fail low.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause APRM 3 to fail low.**
	OATC	Acknowledges and reports the following alarms to the NUSO: • APRM DOWNSCALE / OPRM INOP, 3-9-5A, Window 4 • CONTROL ROD WITHDRAWAL BLOCK, 3-9-5A, Window 7
	NUSO	Directs the BOP and the OATC to respond in accordance with the Alarm Response Procedures.
	OATC	3-ARP-9-5A, Alarm Response Procedure CONTROL ROD WITHDRAWAL BLOCK, Window 7 Operator Actions: A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s) and **REFER** TO operator action for alarm(s). B. N/A C. N/A D. N/A E. N/A
	OATC	3-ARP-9-5A, Alarm Response Procedure APRM DOWNSCALE / OPRM INOP, Window 4 A. **DETERMINE** which APRM/OPRM channel is downscale / inoperable. B. **IF** APRM failed downscale, **THEN BYPASS** channel. **REFER TO** 3-OI-92B.
	OATC	Reports to the NUSO that because APRM 1 failed previously and is bypassed on Panel 3-9-5 that APRM 3 cannot be bypassed.
	OATC	C. N/A D. N/A E. **REFER TO** Tech Spec Tables 3.3.1.1-1, TRM Table 3.3.4-1.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 6 Page 2 of 3

Event Description: Average Power Range Monitor (APRM) 3 Failure

Time	Position	Applicant's Actions or Behavior	
	Driver	**If contacted as the Reactor Engineer or an Instrument Mechanic, acknowledge any information or direction given.**	
	NUSO	Technical Specification 3.3.1.1 – Reactor Protection System (RPS) Instrumentation LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE. Applicability: According to Table 3.3.1.1-1. NOTE: Separate Condition entry is allowed for each channel. **CONDITION:** One or more required channels inoperable.	
	NUSO	**REQUIRED ACTION:** A.1 Place channel in trip. OR A.2 ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------- A.2 Place associated trip system in trip.	**COMPLETION TIME:** A.1 – 12 hours A.2 N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 6 Page 3 of 3

Event Description: Average Power Range Monitor (APRM) 3 Failure

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Requirements Manual 3.3.4 - Control Rod Block Instrumentation LCO 3.3.4 The control rod block instrumentation for each Function in Table 3.3.4-1 shall be OPERABLE. APPLICABILITY: According to Table 3.3.4-1 NOTE: Separate Condition entry is allowed for each Function. **CONDITION:** One or more required channels inoperable.	
	NUSO	**REQUIRED ACTION:** A.1 Enter the Condition referenced in Table 3.3.4-1 for the Function. B.1 Place at least one inoperable channel in the tripped condition.	**COMPLETION TIME:** A.1 – Immediately B.1 – 1 hour
	NRC	**End of Event 6. Request that the Driver insert Event 7, Inadvertent High Pressure Coolant Injection (HPCI) Initiation.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 7 Page 1 of 3

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. HPCI will start and inject to the Reactor Vessel.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause an inadvertent initiation of HPCI.**
	BOP	Recognizes and reports that HPCI has initiated; acknowledges and reports the following alarm on Panel 3-9-3F: • HPCI TURB EXH DRAIN POT LEVEL HIGH, Window 33 Verifies that no Emergency Core Cooling System (ECCS) initiation signal exists and informs the NUSO.
	NUSO	Verifies that there no valid ECCS initiation signals and directs the BOP to trip and lock out HPCI in accordance with ODM 4.20, Strategies for Successful Transient Mitigation, Section 4.4.2, Manual Control of Automatic Systems.
	BOP	Trips and locks out HPCI.
	OATC	Monitors Reactor Water Level, Reactor Power, and Reactor Pressure. Reports that Reactor Water Level Control has switched to Single Element Control (if applicable).
	Driver	**When contacted by the crew as the Outside/Work Control NUSO, acknowledge any direction given.** **If contacted as an AUO to go to 3A 250V DC RMOV Board to open the breaker for the HPCI Minimum Flow Valve, acknowledge the direction. 2 minutes after being contacted, call the crew back and state that you are ready to open the breaker, and when directed to open the breaker insert Event 17. Inform the crew that the breaker for the HPSI Minimum Flow Valve is open.** **If contacted as the Reactor Engineer, acknowledge any direction given.**
	NRC	**If Reactor Water Level Control transfers to Single Element Control, the crew may elect to transfer Reactor Water Level Control back to Three Element Control.** **If the crew does not decide to place Reactor Water Level Control in Three Element Control, end of Event 7; request that the driver insert Event 8 after the Tech Spec call is complete.**

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 7 Page 2 of 3

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Time	Position	Applicant's Actions or Behavior
	NUSO	If necessary, directs the OATC to transfer Reactor Water Level Control to Three Element Control in accordance with 3-OI-3, Reactor Feedwater System, Section 8.31, Transferring RFWCS Between Single Element and Three Element Control.
	OATC	3-OI-3, Reactor Feedwater System Section 8.31, Transferring RFWCS Between Single Element and Three Element Control **NOTES** 1) This section is performed at Panel 3-9-5. 2) The RFW Control System will not allow THREE ELEMENT operation until all of the following permissives are met: • Total Steam Flow > 16%. • Reactor Water Level Control (PDS), 3-LIC-46-5, in AUTO (blue light illuminated) and at least one RFPT Speed Control (PDS) in AUTO (blue light illuminated). • A Valid Total Steam Flow Signal exists (Four good Steam Line Flows, or at least two good Steam Line Flow signals and a valid Turbine First Stage Pressure signal). • A Valid Total Feedwater Line Flow signal exists (Two good Feedwater Line Flow signals, or one good Feedwater Line Flow and three valid individual RFP Flow signals). • At least one valid Narrow Range Level signal. [1] **IF** desired to transfer level control from Single Element to Three Element, **THEN PERFORM** the following: [1.1] **ENSURE** conditions in Note 2 are met for placing level control in Three Element. [1.2] **OBSERVE** stable Steam Flow and Feedwater Flow. [1.3] **DEPRESS** THREE ELEMENT pushbutton, 3-HS-46-6/3. • **ENSURE** green backlight for pushbutton illuminates. [1.4] **ENSURE** extinguished green backlight for SINGLE ELEMENT pushbutton, 3-HS-46-6/1. [1.5] **CHECK** Reactor Water Level stable.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 7 Page 3 of 3

Event Description: Inadvertent High Pressure Coolant Injection (HPCI) Initiation

Time	Position	Applicant's Actions or Behavior	
	OATC	[2] N/A	
	NUSO	Technical Specification 3.5.1 – ECCS Operating LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE. Applicability: MODE 1, MODES 2 and 3, except High Pressure Coolant Injection (HPCI) and ADS valves are not required to be OPERABLE with Reactor Steam Dome Pressure ≥150 PSIG. NOTE: LCO 3.0.4.b is not applicable to HPCI. Condition: C. HPCI System inoperable.	
	NUSO	**REQUIRED ACTION:** C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	**COMPLETION TIME:** C.1 – Immediately C.2 – 14 days
	Driver	**If contacted as the Work Control/Outside NUSO to prepare Protected Equipment tags for RCIC, acknowledge the direction.** **If contacted as the Work Control/Outside NUSO to investigate the failure of HPCI to isolate acknowledge the direction.**	
	NRC	**End of Event 7. Request that the driver insert Event 8, EHC Piping Leak / Drywell Steam Leak.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 1 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Symptoms/Cues: When requested by the Chief Examiner, an EHC Piping leak will develop when initiated by the simulator booth. A Drywell Steam leak will occur when the Reactor SCRAM is inserted.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to start the EHC Leak.**
	NRC	**Once the EHC Leak is inserted by the Driver, it takes approximately 2 ½ minutes to receive the EHC Reservoir Low Alarm (Panel 3-9-7B, Window 9).**
	BOP	3-ARP-9-7B, Alarm Response Procedure EHC RESERVOIR LEVEL LOW, Window 9 A. **IMMEDIATELY DEDICATE** a Licensed Operator to continuously monitor the operating EHC Pump's Amperage and Pressure for fluctuations and continue with this procedure. B. **IF**, while performing this procedure, OR during the EHC Reservoir Level low transient, the operating pump's amperage or pressure begins to fluctuate (indicative of pump cavitation), **THEN IMMEDIATELY**: 1. **INITIATE** Reactor Scram. 2. **ENSURE** Main Turbine trip. 3. **TRIP AND LOCK OUT** the EHC Pumps. C. **DISPATCH** personnel to check level on local gauge.
	Driver	**If contacted as the Turbine Building Assistant Unit Operator (AUO) or Outside/Work Control NUSO, acknowledge any information or direction given.**
	NUSO	D. **ADD** fluid. **REFER TO** 3-OI-47A, EHC System, Section 8.1. E. **INITIATE** search for cause of low level such as leaking servo, leaking line, open valve, etc. F. **NOTIFY** Radwaste to LOCKOUT all 3 Units' Turbine Building Equipment and Floor Drain Sump Pumps.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 2 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	Driver	**If dispatched to investigate the EHC problem and look for leaks wait approximately 5 minutes and report there is a leak in the EHC system, but you cannot tell where it is coming from.** **If contacted as the Radwaste Operator, acknowledge any information or direction given.**
	NUSO	G. **IF** the source of the leak is identified to be a leaking CV or MSV Servovalve, **THEN IMMEDIATELY REFER TO** 3-OI-47, Turbine-generator System. H. **IF** unable to restore and maintain minimum EHC tank level (above -4"), **THEN INITIATE** load reduction. I. **REFER TO** RWI-007, Integrated Pollution Prevention Plan.
	NRC	**EHC Pressure will reach 1300 psig (lowering) in approximately 6 minutes after inserting Event 8.**
	BOP	Acknowledges and reports the following alarms to the NUSO: • EHC HYD FLUID HDR PRESS LOW, 3-9-7B, Window 1 • STANDBY EHC PUMP RUNNING, 3-9-7B, Window 8
	BOP	3-ARP-9-7B, Alarm Response Procedure EHC RESERVOIR LEVEL LOW, Window 8 A. **ENSURE** 3-HS-47-2A(1A), Standby EHC PUMP 3B(3A), running. B. **CHECK** EHC HEADER PRESSURE indicator, 3-PI-47-7 between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC Pump unit and associated piping. D. **REFER TO** 3-OI-47A, EHC System.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 3 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
		3-ARP-9-7B, Alarm Response Procedure EHC RESERVOIR LEVEL LOW, Window 1 A. **ENSURE** Standby EHC PUMP 3B(3A), 3-HS-47-2A(1A) running. B. **CHECK** EHC HEADER PRESSURE indicator, 3-PI-47-7 between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC pump unit. **NOTE** On EHC Hydraulic System failure, accumulator and check valve arrangement will provide approximately one minute bypass valve operation. D. **IF** EHC Hydraulic System fails, **THEN ENSURE** Turbine trips at or below 1100 psig.
		3-OI-47A, EHC System Precautions and Limitations I. Loss of the EHC System will result in a Turbine-Generator trip. L. EHC Hydraulic Pumps should NOT be operated when fluid level in the reservoir is below the low level alarm setpoint. N. EHC Hydraulic Fluid Pump Motor Current of 140 amps relates to a system flow of 100 gpm (maximum set pump flow). If motor current approaches this value, the EHC System should be checked for leaks, both internal and external.
	NUSO	When there are indications of an EHC leak, directs the OATC to insert a manual Reactor SCRAM.
	OATC	Inserts a manual Reactor SCRAM.
	NRC	**When the Reactor MODE SWITCH is placed in SHUTDOWN, the following Events will occur:** • **LT-3-208A and C Fail High (See Event 9 on page 28 of 36), tripping RCIC and RFPTs** • **One (1) minute later the Drywell Steam Leak will begin ramping in over a 15 minute period.**

Form 3.3-2 Required Operator Actions

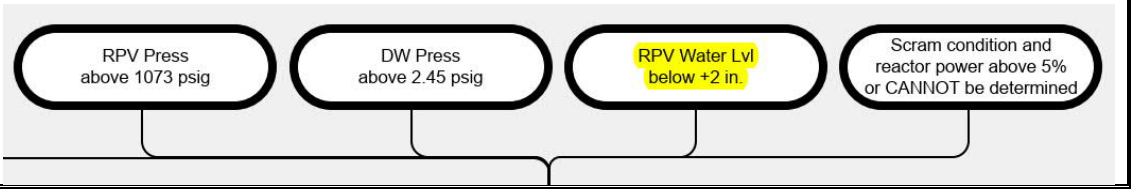
Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 4 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	BOP	Stops and locks out the EHC Pumps.
	OATC	3-AOI-100-1, Reactor SCRAM / Hard Card 2.1 IMMEDIATE ACTIONS [1] **DEPRESS** 3-HS-99-5A/S3A, REACTOR SCRAM A and 3-HS-99-5A/S3B, REACTOR SCRAM B on Panel 3-9-5. [2] **PLACE** 3-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] N/A [4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the Unit SRO: • Reactor SCRAM • MODE SWITCH is in SHUTDOWN • All rods in • Reactor Water Level and trend (recovering or lowering) • Reactor pressure and trend • MSIV position (Open or Closed) • Power level [5] N/A
	BOP	2.2 SUBSEQUENT ACTIONS: PANELS 3-9-7 & 3-9-8 [1] At ≤50 MWe, or as directed by the Unit SRO, ENSURE TRIPPED the Main Turbine as follows: [1.1] **DEPRESS** the TRIP pushbutton, 3-HS-47-67D on Panel 3-9-7. [1.2] **ENSURE OPEN** GENERATOR PCB 234. [1.3] **ENSURE TRIPPED** 3-HS-57-24, VOLTAGE REGULATOR. [2] **ANNOUNCE** Reactor SCRAM over PA system.
	OATC / BOP	2.3 SUBSEQUENT ACTIONS: PANEL 3-9-3 & 3-9-5 **NOTE** The following steps are not required to be performed in order, but only as required to maintain stable conditions.

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 5 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[1] **MONITOR** and **CONTROL** RPV Pressure to keep below 1073 psig and stable, or as directed by Unit SRO. [1.1] N/A [1.2] N/A [1.3] **IF** MSRVs are cycling and bypass valves are NOT available, **THEN ENSURE OPEN** MSRVs on Panel 3-9-3 until RPV Pressure is controlled between 800 and 1000 psig. [2] **IF** any PCIS Isolation Signal is received, **THEN ENSURE** PCIS Isolations using any of the following: [2.1] Containment Isolation Status System on Panel 3-9-4. [2.2] PCIS Mimic and individual control switch indications. [2.3] ICS. [2.4] 3-OI-64, Primary Containment System. [3] **IF** HPCI and/or RCIC are in service and injecting to the vessel, **THEN MONITOR** and **CONTROL** Reactor Water Level as necessary.
	NRC	**NOTE: See Event 9 for Reactor Water Level Control Actions.**
	OATC / BOP	[4] **DRIVE** in all IRM's and SRM's from Panel 3-9-5 as time and conditions permit. [5] **ENSURE** SCRAM DISCH VOL VENT & DR VLVS closed by green **indicating** lights at SDV Display on Panel 3-9-5 [6] **RETURN** to 3-AOI-100-1, Subsequent Actions Section, **AND CONTINUE** with actions as required.
	NUSO	Enters 3-EOI-1, RPV Control Modes 1-3 following the Reactor SCRAM based on Reactor Water Level below +2". 

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 6 of 11

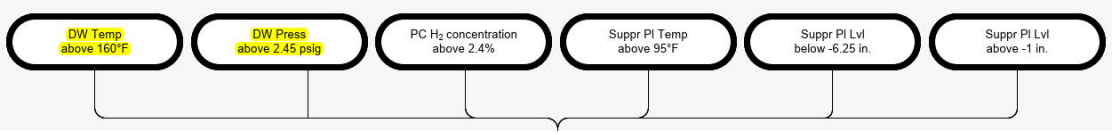
Event Description: EHC Piping Leak / Drywell Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


RC/P-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>A high Drywell Pressure ECCS Signal exists (2.45 psig)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Emergency RPV Depressurization is required or has been required</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Emergency RPV Depressurization is anticipated</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Pool Temp and Level CANNOT be maintained in a safe area of Curve 3 at the existing RPV Pressure
OR
Suppression Pool Level CANNOT be maintained in the safe area of Curve 4</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-2
<table>
<tr><td colspan="2">STABILIZE RPV Pressure below 1073 psig using the Main Turbine Bypass Valves (APPX 8B)
➢ OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
➢ Crosstie CAD or MSRV carts to Drywell Control Air (APPX 8G, 20H) if necessary</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the OATC or BOP to maintain Reactor Pressure in a band below 1073 psig.</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 7 of 11

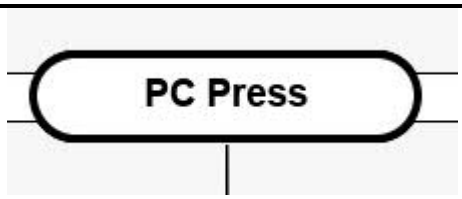
Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	CREW	Acknowledges and reports the following Drywell Pressure and Temperature alarms when received: • PRI CONTAINMENT N2 PRESS HIGH, 3-9-3B, Window 10 • DRYWELL NORM OPERATING PRESS HIGH, 3-9-3B, Window 19 • DRYWELL PRESS APPROACHING SCRAM, 3-9-3B, Window 30 • DRYWELL ATMOSPHERIC TEMP HIGH, 3-9-3B, Window 3 • DRYWELL PRESSURE ABNORMAL, 3-9-5B, Window 31 • DRYWELL TEMP HIGH, 3-9-3B, Window 16
	NRC	**NOTE: Due to the multitude of alarms received following the Reactor SCRAM and the rate of rise of Drywell Pressure, the crew may not have time to address 3-OI-64, Primary Containment System, or 3-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell. The crew may just use the EOI Appendices to address the steam leak into the Drywell.**
	NUSO	Re-enters 3-EOI-1, RPV Control Modes 1-3 and enters 3-EOI-2, Primary Containment Control when Drywell Pressure exceeds 2.45 psig. 3-EOI-1  3-EOI-2 (NOTE: 3-EOI-2 will be re-entered when Drywell Temperature exceeds 160º F). 

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 8 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	 PC/P-1 **MONITOR** and **CONTROL** PC press below 2.45 psig using the Vent system (AOI-64-1) PC/P-2 **WHEN** Primary Containment Pressure CANNOT be maintained below 2.45 psig PC/P-3

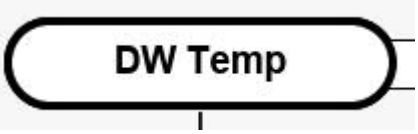
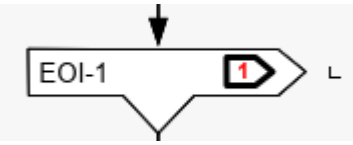
IF	THEN
Primary Containment Pressure reduction is required to restore and maintain Adequate Core Cooling or reduce total offsite radiation dose	**NO ACTION REQUIRED**
Primary Containment Pressure CANNOT be maintained above (-)2 psig	**NO ACTION REQUIRED**
Suppression Chamber Pressure approaches 0 psig	**NO ACTION REQUIRED**
Drywell Pressure approaches 0 psig	**NO ACTION REQUIRED**

PC/P-4

BEFORE Suppression Chamber Pressure rises to 12 psig

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 9 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>PC/P-5
INITIATE Suppression Chamber Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17C)
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Needed to augment Suppression Chamber Sprays</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>When Drywell Temperature exceeds 160º F, re-enters 3-EOi-2, Primary Containment Control.
 
DW/T-1
MONITOR and CONTROL Drywell Temperature below 160° F using all available Drywell cooling
DW/T-2
WHEN Drywell Temperature CANNOT be maintained below 160° F
DW/T-3
BEFORE Drywell Temperature rises to 280º F
  </td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 10 of 11

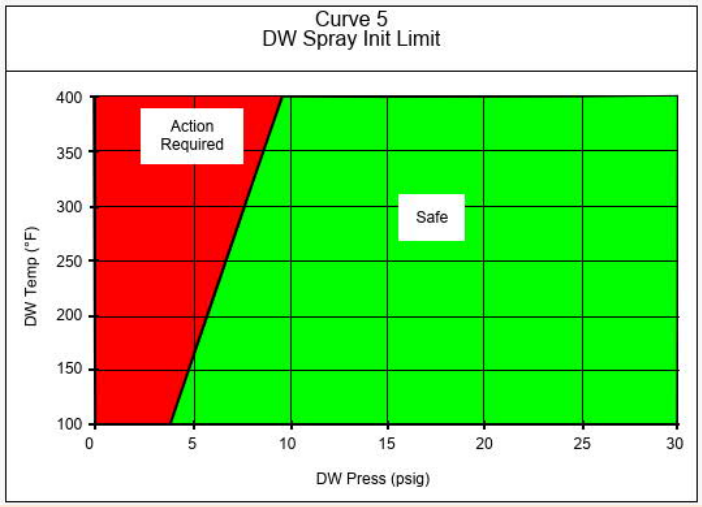
Event Description: EHC Piping Leak / Drywell Steam Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
DW/T-4
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
DW/T-5
IF Suppression Pool Water Level is below 19 ft
AND
Drywell Temperature is in the safe area of Curve 5
THEN
1. SHUT DOWN Recirc Pumps
2. SHUT DOWN Drywell Blowers
3. INITIATE Drywell Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B, 20L)
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to Augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>Crew</td><td>
Critical Task:
When Drywell Sprays are required, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit Curve to preclude Primary Containment failure.

Critical Task Failure Criteria:
Drywell Sprays are initiated with Drywell Pressure and Temperature outside the safe area of the Drywell Spray Initiation Curve (Curve 5).
</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 8 Page 11 of 11

Event Description: EHC Piping Leak / Drywell Steam Leak

Time	Position	Applicant's Actions or Behavior
	Crew	
	NRC	**When the crew is spraying the Drywell and has control of Reactor Water Level using High Pressure Makeup sources (See Event 9 on page 28 of 36), end of Event 8 and end of Scenario. Request that the Driver place the Simulator in FREEZE.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 1 of 7

Event Description: LT-3-208A and C Fail High

Symptoms/Cues: Event is automatically entered on Simulator setup; LT-3-208A and C will fail high when the Reactor MODE SWITCH is placed in SHUTDOWN.

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Driver</td><td>Event 9 is automatically entered when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the Driver to cause LT-3-208A and LT-3-208C to fail high and therefore a loss of RCIC and RFPTs.</td></tr>
<tr><td></td><td>OATC</td><td>Reports that RFPTs have tripped.</td></tr>
<tr><td></td><td>BOP</td><td>Recognizes and reports that RCIC has a sealed in high level trip.</td></tr>
<tr><td></td><td>Driver</td><td>If contacted as the Work Control/Outside SRO or an AUO to investigate, acknowledge the direction.</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-1, RPV Control Modes 1-3

RC/L-1
ENSURE each as required:
• PCIS isolations (Groups 1, 2 and 3)
• ECCS
• RCIC

RC/L-2
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Raising RPV Water Level above +51 in. will facilitate use of shutdown cooling, steam-driven injection systems, or Alternate Injection Subsystems (Table L-2)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 2 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	NUSO	RC/L-3 **RESTORE** and **MAINTAIN** RPV Water Level between +2 in. and +51 in. with ANY Preferred Injection Systems (Table L-1) ➢ OK to use ANY Alternate Injection Subsystems (Table L-2) Directs the BOP to maintain Reactor Water Level using any available methods in Table L-1 or L-2. **NOTE: RCIC and Reactor Feedwater are not available due to the High Water Level Trip. HPCI may be used if necessary, even though it was previously locked out due to an inadvertent initiation.**

Table L-1
Preferred Injection Systems

SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available 2 3 5	5C, 20M	1200 psig
HPCI with CST suction if available 2 5 7	5D, 20N	1200 psig
CNDS	6A	470 psig
CS 2	6D, 6E	330 psig
LPCI 2	6B, 6C	320 psig

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 3 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	NUSO	

Table L-2
Alternate Injection Subsystems

SOURCE	APPX	INJ PRESS
EHPM Pump	7L	1210 psig
SLC (test tank)	7B	1450 psig
SLC (boron tank)	7B	1450 psig
CNDS transfer pumps to RHR and CS	7A	110 psig
RHR crosstie to other units	7C	320 psig
Stby coolant	7D	160 psig
RHR drain pumps	7E, 7F	50 psig
PSC head tank pumps	7G	30 psig
RCIC (aux boiler steam) with CST suction if available (2 3 5)	7H	1200 psig
RCIC manual start	20A	1200 psig
HPCI (aux boiler steam) with CST suction if available (2 5 7)	7J	780 psig
Fire Protection system	7K	150 psig
FLEX Pump Sys (CILRT/CS)	20D	150 psig
FLEX Pump Sys (Standby Coolant)	20B	150 psig
FLEX Pump Sys (CILRT/CRD)	20C	150 psig

	NRC	**NOTE: If the crew decides to use HPCI for Reactor Water Level Control, some valves will already be in the OPEN position due to the inadvertent initiation. They are noted on the procedure below. Additionally, as soon as the HPCI Aux Oil Pump is taken out of the PULL TO LOCK position, the HPCI Turbine will start and inject to the Reactor.**

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 4 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	BOP	3-EOI-Appendix-5D, Injection System Lineup HPCI [1] N/A [2] N/A [3] N/A [4] **VERIFY** at least one SGTS train in operation. **CAUTIONS** 1) Operating HPCI Turbine below 2400 rpm may result in unstable system operation and equipment damage. 2) Operating HPCI Turbine with suction temperatures above 140º F may result in equipment damage. [5] **VERIFY** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection [6] **IF** high Reactor Water Level trip logic is actuated, **THEN** [6.1] **DEPRESS** HPCI TURBINE TRIP REACTOR LEVEL HIGH RESET pushbutton. [6.2] **CHECK** HPCI TURBINE TRIP LEVEL HIGH amber light has extinguished. [7] **PLACE** HPCI AUXILIARY OIL PUMP handswitch in START. **HPCI WILL START AND INJECT** [8] **PLACE** HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 3-FCV-73-30, HPCI PUMP MINIMUM FLOW VALVE. **CLOSED AND DE-ENERGIZED DURING EVENT 7** [10] **OPEN** 3-FCV-73-44, HPCI PUMP INJECTION VALVE. **ALREADY OPEN** [11] **OPEN** 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine. **ALREADY OPEN**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 5 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	BOP	[12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 3-CKV-73-45, HPCI SYSTEM CHECK VLV, opens by observing 3-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 gpm **CLOSED AND DE-ENERGIZED DURING EVENT 7** [13] **ADJUST** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO.
	BOP	If directed to use the EHPM instead of HPCI 3-EOI-APPENDIX-7L, Alternate RPV Injection System Lineup EHPM System [1] **CHECK** 4KV EHPM BD 2 is energized from normal power supply on 3-EI-7-410A, EHPM NORMAL SOURCE VOLTAGE. [2] N/A [3] N/A [4] N/A

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 6 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
		CAUTIONS 1) Pump minimum continuous stable operating flow is 431 gpm and maximum continuous stable operating flow is 1250 gpm. Above 1250 gpm, adequate NPSH is not assured. 2) Prolonged EHPM pump operation on minimum flow or test return mode will add heat to the CST. As decay heat dissipates, it may be necessary to cycle EHPM pump to minimize extended run times. Ref motor start limitations per 3-OI-7. 3) If power is being supplied to more than one EHPM Board by the Supplemental Diesel Generators, then coordination between units is required to ensure a duration of at least 10 seconds between starting EHPM Pumps.
	BOP	[5] **ESTABLISH** Unit 3 RPV injection in BATCH mode as follows: [5.1] **START** EHPM PUMP by placing 3-HS-7-1A, to START. [5.2] **THROTTLE** 3-FCV-007-0008, EHPM PUMP INJECTION VALVE, as necessary to establish flow 950-1250 gpm as indicated on EHPM SYS NORMAL FLOW 3-FI-7-403A. [5.3] **MONITOR** Unit 3 RPV Level on 3-LI-3-148AA, RPV LEVEL 'A', and 3-LI-3-148BA, RPV LEVEL 'B'. [5.4] **WHEN** desired RPV level is reached, **THEN CLOSE** 3-FCV-007-0008, EHPM PUMP INJECTION VALVE. [5.5] **REPEAT** steps 1.0[5.2] through 1.0[5.4] as necessary to maintain RPV level. [6] N/A [7] N/A [8] **MONITOR** the Unit 3 CST level and as refill as necessary per 3-OI-7.
	CREW	**Critical Task:** **Following a loss of high pressure injection, restore high pressure injection to the Reactor Vessel to preclude Emergency Depressurization due to low Reactor Water Level.** **Critical Task Failure Criteria:** **Emergency Depressurization is required based on low Reactor Water Level.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-1 Event No.: 9 Page 7 of 7

Event Description: LT-3-208A and C Fail High

Time	Position	Applicant's Actions or Behavior
	NRC	**When the crew is spraying the Drywell and has control of Reactor Water Level using High Pressure Makeup, end of Event 8 and end of Scenario. Request that the Driver place the Simulator in FREEZE.**

Form 3.3-2 Required Operator Actions

Scenario Setup
UNIT 3

IC	28
Exam IC	263

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-3	115	AOI-85-3	14		
OI-47A	40	AOI-100-1	80		
OI-68	106	TRM 3.3.4	2		
OI-85	99	T.S. 3.3.1.1	221		
OI-92B	23	T.S. 3.5.1	244		
ARP-3A	60	OPDP-1	55		
ARP-3B	25	EOI-1	16		
ARP-3F	38	EOI-2	15		
ARP-5A	56	APPX-5D	8		
ARP-7B	31	APPX-7L	5		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	ILT 2304 NRC Scenario x UNIT 3.sch
Event Files(s):	ILT 2304 NRC Scenario x UNIT 3.evt

UNIT 3 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ON LINE** **230**	**Drywell Leakage (GPM)** **1.89**	**Protected Equipment**
Rx Power **80%**	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (GPM)** **0.31**	
MWe **1026**		**Equipment Drain (GPM)** **1.58**	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

Control Rod 30-43 Directional Control Valve Maintenance completed.

Alternated Reactor/Refuel Zone Exhaust Fans, 'B' fans in service, 'A' fans tagged for maintenance

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Perform 3A RFPT Overspeed Trip Exerciser Test.

Reduce Reactor Power to < 75% using Core Flow in preparation for recovering Control Rod 30-43.

Wait for Reactor Engineer to brief Control Rod 30-43 recovery.

OPERATOR WORK AROUNDS **OWAs – U0-1/U3-0** **Burdens - U0 – 0/U3-0** **Operator Challenges – U0 – 12/U3 - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-2
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 70% Reactor Power, Reactor Shutdown in Progress. A3 EECW Pump is Tagged for maintenance.

Turnover: To support preparations for Reactor SCRAM, continue with power reduction and planned equipment shutdowns in accordance with 2-GOI-100-12A, Section 5.3.1 [5.3], and 2-OI-2, Condensate System, Section 7.1 [3.4]. Support maintenance on A3 EECW Pump.

Critical Tasks:

1. With a failure of an Emergency Diesel Generator (EDG) to automatically start and tie on to its respective 4KV Shutdown Board, the crew takes manual action to restore power to 2A 480V Shutdown Board to exit a Station Blackout, or commences a plant cooldown within 20 minutes of the Station Blackout.

2. The crew restores EECW Flow with at least 1 EECW Pump within 8 minutes from the time the EDGs start to prevent damage to and a loss of the Emergency Diesel Generators.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Shutdown 2C Condensate/Condensate Booster Pumps
2.	N/A	R-OATC	Lower Reactor Power Using Control Rods
3.	RD12	C-OATC C-NUSO	Control Rod Difficult to Insert
4.	SW03C	C-BOP TS-NUSO	Emergency Equipment Cooling Water (EECW) Pump Trip
5.	PMP-76-110	TS-NUSO	H2/O2 Analyzer Failure
6.	ED08C	C-OATC MC-OATC C-NUSO	2C 4KV Unit Board Trip
7.	XA-55-7A_32	C-BOP C-NUSO	Generator Bus Duct High Temperature
8.	ED01 ED01A 52_BKR DG B	M-ALL	2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie
9.	DG01C	C-BOP MC-BOP C-NUSO	DG Fails to Auto Start
10.	SW07A SW07B SW07C SW07D	C-BOP MC-BOP C-NUSO	Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will shut down 2C Condensate/Condensate Booster Pumps in accordance with 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, and 2-OI-2, Condensate System.
2. The crew will continue the Reactor Shutdown by inserting Control Rods in accordance with 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, and 2-OI-85, Control Rod Drive System.
3. A Control Rod will be difficult to insert during the Reactor Shutdown. The crew will take action in accordance with 2-OI-85, Control Rod Drive System, to insert the Control Rod.
4. C3 Emergency Equipment Cooling Water (EECW) Pump will trip. The crew will start another pump to support the North EECW Header in accordance with 0-OI-67, Emergency Equipment Cooling Water. The Nuclear Unit Senior Operator (NUSO) will reference Technical Specifications.
5. The H2/O2 Analyzer will fail, requiring the NUSO to address Technical Specifications.
6. 2C 4KV Unit Board will trip and lock out due to an electrical fault. The crew will respond in accordance with Alarm Response Procedures and Abnormal Operating Instructions.
7. The crew will receive an annunciator that Generator Bus Duct Temperature is high, requiring action in accordance with Alarm Response Procedures.
8. 2B 480V Shutdown Board Normal Feeder Breaker will trip, followed by a loss of all Offsite Power. Coupled with the loss of 2B 480V Shutdown Board and the failure of 'B' Emergency Diesel Generator (DG) to tie onto the 'B' 4KV Shutdown Board, the Loss of Offsite Power will result in a Station Blackout. The crew will respond in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) / Station Blackout, until the crew transfers 2A 480V Shutdown Board to its alternate power supply.
9. 'C' DG will fail to automatically start on the Loss of Offsite Power, requiring the crew to manually start 'C' DG.
10. Emergency Equipment Cooling Water (EECW) Pumps will fail to auto start when the DGs restore power to the 4160V Shutdown Boards. The crew will restart the EECW pumps within 8 minutes as required by 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout.

The Scenario ends when the crew has restored power to 2A 480V Shutdown Board, has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) with high pressure makeup sources, and has restored EECW Flow to the Emergency Diesel Generators.

Critical Tasks 2

1. **With a failure of an Emergency Diesel Generator (EDG) to automatically start and tie on to its respective 4KV Shutdown Board, the crew takes manual action to restore power to 2A 480V Shutdown Board to exit a Station Blackout, or commences a plant cooldown within 20 minutes of the Station Blackout.**

   a. **Safety Significance**
   Mitigate Station Blackout.

   b. **Cues**
   Procedural compliance.
   Loss of power to both 480V Shutdown Boards.
   Loss of power to Low Pressure ECCS system valves.

   c. **Measured by:**
   Observation – Power restored to 2A 480V Shutdown Board or a plant cooldown is commenced.

   d. **Feedback**
   Power is restored to 2A 480 Shutdown Board.
   Power available to Low Pressure ECCS system valves.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to restore power to 2A Shutdown Board or commence a plant cooldown within 20 minutes of the Station Blackout on Unit 3.

2. **The crew restores EECW Flow with at least 1 EECW Pump within 8 minutes from the time the EDGs start to prevent damage to and a loss of the Emergency Diesel Generators.**

   a. **Safety Significance**
   Diesel Generator operation without EECW flow will cause EDG failure.

   b. **Cues**
   Procedural compliance.
   EECW Header Flow.
   EECW Pump Run Indications.

   c. **Measured by:**
   Observation – The crew verifies at least one EECW Pump is restarted as required when power is restored to the 4KV Shutdown Busses.

   d. **Feedback**
   EECW Header Flow.
   EECW Pump Run indications.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to restore EECW Flow with at least one EECW Pump to the EDGs within 8 minutes following EDG start upon a Loss of Offsite Power.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 1 Page 1 of 2

Event Description: Shutdown 2C Condensate/Condensate Booster Pumps

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**NOTE: The crew has been pre-briefed, but may elect to hold a re-focus brief.**
	NUSO	2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, Section 5.3.1, Reducing Reactor Power to 40% [5] **WHEN** Reactor is at less than 76% power (12 M lbm/hr) **THEN** [5.3] **SHUTDOWN** from service one of the three Condensate Booster Pumps, as directed by the NUSO, per 2-OI-2, Condensate System. [5.4] **SHUTDOWN** from service one of the three Condensate Pumps, as directed by the NUSO, per 2-OI-2, Condensate System.
	BOP	2-OI-2. Condensate System Section 3.1, General Precautions D. To remove a Condensate pump along with any other pump, then additional restrictions apply: 1. Pumps must be removed from high pressure to low 2. Condensate Flow should be reduced to 76% (12.0 M lbm/hr) prior to removing the Condensate Pump. 3. Condensate Flow can be restored to 87.5% (14.1 M lbm/hr). 4. The Condensate/Feedwater System is susceptible to perturbations from other evolutions. Any evolutions that have the potential to affect Condensate/Feedwater System flow/NPSH should be considered High Risk. Section 7.1, Condensate System Shutdown [1] **CHECK** Condensate system in operation in accordance with this instruction. [2] **REVIEW** the Precautions and Limitations in Section 3.0.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 1 Page 2 of 2

Event Description: Shutdown 2C Condensate/Condensate Booster Pumps

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **IF** removing the first two Condensate Booster Pumps/Condensate Pumps for power reductions or Condensate System shutdown, **THEN PERFORM** the following: [3.1] **ENSURE** Reactor Power is less than or equal to 76% (12.0 M lbm/hr) prior to removing a Condensate Pump. (Ref. P&L 3.1D.) [3.2] **ENSURE** Hydrogen Injection is secured to the Condensate Booster Pump(s) to be stopped in Section 7.1[3.4]. **REFER TO** 2-OI-4, Hydrogen Water Chemistry System.
	Driver	**If contacted as the Turbine Building Assistant Unit Operator to shut down Hydrogen Injection, report that Hydrogen Injection has been stopped to Condensate Booster Pump 2C.**
	BOP	[3.3] **ENSURE** the number of Condensate/Condensate Booster Pumps to be left in service will supply adequate suction pressure to running RFP(s). [3.4] **WHEN** directed by 2-GOI-100-12, 2-GOI-100-12A, or the Unit 2 NUSO, **THEN STOP** CONDENSATE BOOSTER PUMP using one of the following: • 2-HS-2-68A, CONDENSATE BOOSTER PUMP 2C [3.5] **REMOVE** Condensate Demins from service as required, and PLACE in standby. **REFER TO** 2-OI-2A, Condensate Demineralizer System.
	Driver	**When contacted as the Turbine Building AUO to remove Demins from service, acknowledge any direction given.**
	BOP	[3.6] **NOTIFY** Chemistry which Condensate Pump(s) will be stopped and to realign oxygen injection to Condensate Pump(s).
	Driver	**When contacted as Chemistry to realign oxygen injection, report that realignment has been completed.**
	BOP	[3.7] **ENSURE** the number of Condensate/Condensate Booster Pumps to be left in service will supply adequate suction pressure to running RFP(s). [3.8] **WHEN** directed by 2-GOI-100-12, 2-GOI-100-12A, or the Unit 2 NUSO, **THEN STOP** CONDENSATE PUMP using one of the following: • 2-HS-2-15A, CONDENSATE PUMP 2C
	NRC	**End of Event 1. Continue to Event 2, Lower Reactor Power Using Control Rods.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 2 Page 1 of 2

Event Description: Lower Reactor Power Using Control Rods

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: During Event 2, Event 3 (Control Rod Difficult to Insert) is automatically entered on Simulator setup. No Action is required by the Driver to insert Event 3.**
	NUSO	2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, Section 5.3.1, Reducing Reactor Power to 40% [6] **CONTINUE** to **REDUCE** Reactor Power by a combination of Control Rod insertions and Core Flow changes, as recommended by Reactor Engineer.
	NUSO	Directs the Operator at the Controls (OATC) to continue the Reactor Shutdown by inserting Control Rods in accordance with 2-OI-85, Control Rod Drive System, and Reactivity Control Plan (RCP) RCP-NRC2304-U2S2.
	OATC	2-OI-85, Control Rod Drive System, Section 6.7, Control Rod Insertion Section 6.7.1, Initial Requirements [1] **REVIEW** Precautions and Limitations in Sections 3.7 and 3.8. [2] **OBSERVE** the following during Control Rod repositioning: • Control Rod reed switch position indicators (four rod display) agree with the indication on the Full Core Display. • Nuclear Instrumentation responds as Control Rods move through the core (This ensures Control Rod is following drive during Control Rod movement.) [3] **ENSURE** the following prior to Control Rod movement: • CRD POWER, 2-HS-85-46 in ON • When Rod Worth Minimizer is enforcing, the ROD WORTH MINIMIZER is operable and LATCHED in to the correct ROD GROUP. [4] **PERFORM** the following to insert the Control Rod as appropriate. • Control Rod Notch Insertion per Section 6.7.2 • Control Rod Continuous Insertion per Section 6.7.3

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 2 Page 2 of 2

Event Description: Lower Reactor Power Using Control Rods

Time	Position	Applicant's Actions or Behavior
	OATC	Section 6.7.3, Continuous Insertion of Control Rod [1] **CHECK** Section 6.7.1 has been performed. [2] **SELECT** the desired Control Rod by depressing the appropriate 2-XS-85-40, CRD ROD SELECT pushbutton.
	NRC	**EXAMINER NOTE: Control Rod insertion order per RCP-NRC2304-U2S2.**
	OATC	[3] **OBSERVE** the following for the selected Control Rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED • White light on the Full Core Display ILLUMINATED [4] **PLACE** and HOLD 2-HS-85-48, CRD CONTROL SWITCH, in ROD IN. [5] **WHEN** Control Rod notch reaches the even rod notch position prior to the desired final Control Rod notch position, **THEN RELEASE** 2-HS-85-48, CRD CONTROL SWITCH. [6] **OBSERVE** the Control Rod settles into desired position and the ROD SETTLE light extinguishes.
	NRC	**When Event 3 is complete and satisfied with Control Rod insertion, end of Events 2 and 3.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 3 Page 1 of 2

Event Description: Control Rod Difficult to Insert

Symptoms/Cues: Event is initiated by Control Rod position. While driving in Control Rod 06-39, it will slow and stick at Position 38.

Time	Position	Applicant's Actions or Behavior
	NRC	**Event is inserted on simulator startup and is initiated by Control Rod position. No action is required by the Driver to insert Event 3.**
	OATC	While inserting Control Rods in accordance with 2-OI-85, Control Rod Drive System, determines that Control Rod 06-39 is difficult to insert. Informs the Nuclear Unit Senior Operator (NUSO).
	NUSO	Directs the OATC to take action in accordance with 2-OI-85.
	OATC	2-OI-85, Control Rod Drive System Section 8.17, Control Rod Difficult to Insert [1] **ENSURE** the Control Rod will NOT notch in, in accordance with Section 6.7 or Section 8.20. [2] **REVIEW** all Precautions and Limitations in Section 3.0. [3] N/A [4] **CHECK** CRD SYSTEM FLOW is between 40 gpm and 65 gpm, indicated by 2-FIC-85-11. [5] **ENSURE** 2-PDI-85-17A, CRD DRIVE WATER HEADER DP is between 250 -270 psid on by throttling 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE, as necessary. [6] N/A [7] **IF** Control Rod motion is observed, but the CRD fails to notch-in (example: settles back to the original position) with normal operating drive water pressure, **THEN**: [7.1] **NOTIFY** Reactor Engineer to determine what parameters should be recorded for further evaluation.
	Driver	**When contacted as the Reactor Engineer, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 3 Page 2 of 2

Event Description: Control Rod Difficult to Insert

Time	Position	Applicant's Actions or Behavior
	OATC	[7.2] **RAISE** 2-PDI-85-17A, CRD DRIVE WATER HEADER DP, NOT to exceed 300 psid, using 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE.
	NRC	**EXAMINER NOTE: Control Rod 06-39 will insert when CRD Drive Water Pressure is raised about 285 psig.**
	OATC	[7.3] **INSERT** Control Rod as directed in Section 6.7. [7.4] **LOWER** 2-PDI-85-17A, CRD DRIVE WATER HEADER DP, to between 250 psid and 270 psid using 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE.
	NRC	**End of Event 3. When satisfied with Control Rod insertion, end of Events 2 and 3. Request that the Driver insert Event 4.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 1 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. C3 EECW Pump will trip.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause C3 Emergency Equipment Cooling Water (EECW) Pump to trip.**
	NRC	**EXAMINER NOTE: A3 EECW Pump is tagged out as part of the initial conditions – therefore, when C3 EECW Pump Trips, there will be no EECW Pumps running on the North Header.**
	BOP	Acknowledges and reports the following alarms to the NUSO: • MOTOR TRIPOUT, 2-9-8C, Window 33
	BOP	Determines that C3 EECW Pump has tripped and informs the NUSO.
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure and 0-OI-67, Emergency Equipment Cooling Water.
	BOP	2-ARP-9-8C, Alarm Response Procedure MOTOR TRIPOUT, Window 33 A. **CHECK** Control Room for white disagreement light illuminated for affected equipment. B. **CLEAR** disagreement light. C. **DISPATCH** personnel to CHECK: 1. Relays at associated electrical board. 2. Equipment for abnormal conditions. 3. Safe-stop locally reset, if necessary. D. **REFER TO** 0-GOI-300-2 if relay targets are present or for motor starting limits. E. **REFER TO** appropriate OI for recovery or realignment of equipment.
	NRC	**The crew may elect to start either A1 or C1 RHRSW Pump and align to EECW in accordance with 0-OI-67, Emergency Equipment Cooling Water. Both procedures are shown below, but only one of the two will be used. (Procedure for C1 starts on page 10 of 38)**

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 2 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**EXAMINER NOTE: The crew may discuss filling and venting the EECW North Header – if so, direct the driver to inform the NUSO that time compression has been used, and that the North Header has been filled and vented.**
	BOP	0-OI-67, Emergency Equipment Cooling Water **IF THE CREW ELECTS TO USE A1 EECW PUMP:** Section 8.1, Operation of RHRSW Pump A1 (for EECW in place of A3) **NOTES** 1) RHRSW Pump A1 may be aligned for service by this section when: • It is used to meet the minimum number of Tech. Spec. operable pumps; or • At the discretion of the NUSO, it is needed to replace another pump's operation; or • At the discretion of the NUSO, it is needed to assist in supplying header flow/pressure demand 2) If used to meet EECW requirements, RHRSW Pump A1 must be aligned to EECW, the pump started, and should remain running. RHRSW Pump A1 does NOT have the same auto start signals as RHRSW Pump A3. 3) The RHRSW pump control switches and amp meters are located at Control Room Panel 9-3, Unit 1, 2, and 3. 4) When RHRSW Pump A1 is aligned for EECW, its RHRSW function required by the Safe Shutdown Program (NFPA 805) is inoperable. NFPA 805 FPR or FPRM requirement shall be addressed. 5) Minimum flow for A1 RHRSW/EECW Pump is 1700 gpm. **CAUTION** Only one RHRSW Pump in a given RHRSW Pump Room may be counted toward meeting Technical Specification 3.7.2 requirements for EECW Pump operability.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 3 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[1] To line up RHRSW Pump A1 for EECW System operation, **PERFORM** the following: [1.1] **ENSURE** EECW System is in prestartup/standby readiness alignment in accordance with Section 4.0. [1.2] **REVIEW** all precautions and limitations in Section 3.0. [1.3] **ENSURE** RHRSW Pump A1 is in standby readiness in accordance with 0-OI-23, Residual Heat Removal Service Water System. [1.4] through [1.10] – OUTSIDE ACTIONS
	Driver	**When contacted as the Outside AUO to perform Steps [1.4] – [1.9], acknowledge the direction and insert Event 14. Report that Steps [1.4] – [1.10] are complete, and A1 RHRSW Pump has been aligned for EECW.**
	BOP	[1.11] **REQUEST** a caution order be issued to tag RHRSW Pump A1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the A1 Pump should remain running to be operable for EECW.
	Driver	**When contacted as Work Control / Outside NUSO for a clearance on RHRSW Pump A1 and its associated crosstie valves, acknowledge the direction.**
	BOP	[2] To start RHRSW (EECW) Pump A1, PERFORM the following: [2.1] **START** RHRSW Pump A1 using one of the following: • 0-HS-23-1A/2, RHRSW PUMP A1 on Unit 2 [2.2] **CHECK** RHRSW Pump A1 running current is less than 61 amps using one of the following: • 0-EI-23-1/2, RHRSW PUMP A1 AMPS on Unit 2 [2.3] AND [2.4] – OUTSIDE ACTIONS
	Driver	**When contacted as the Outside AUO to complete Steps [2.3] and [2.4], report that the steps are complete.**

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 4 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[2.5] **NOTIFY** Chemistry of running RHRSW (EECW) Pump(s). [2.6] **ENSURE** a caution order has been issued to tag RHRSW Pump A1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the A1 pump should remain running to be operable for EECW.
	Driver	**When contacted as Chemistry acknowledge any information or direction given.** **When contacted as Work Control / Outside NUSO concerning the caution order, report that the caution order is in progress.**
	BOP	**IF THE CREW ELECTS TO USE C1 EECW PUMP:** Section 8.3, Operation of RHRSW Pump C1 (for EECW in place of C3) **NOTES** 1) RHRSW Pump C1 may be aligned for service by this section when: • It is used to meet the minimum number of Tech. Spec. operable pumps; or • At the discretion of the NUSO, it is needed to replace another pump's operation; or • At the discretion of the NUSO, it is needed to assist in supplying header flow/pressure demand 2) If used to meet EECW requirements, RHRSW pump C1 must be aligned to EECW, the pump started, and should remain running. RHRSW Pump C1 does NOT have the same auto start signals as RHRSW Pump C3. 3) The RHRSW pump control switches and amp meters are located on Panel 9-3, Unit 1, 2, and 3. 4) When RHRSW Pump C1 is aligned for EECW, its RHRSW function required by the Safe Shutdown Program (NFPA 805) is inoperable. NFPA 805 FPR or FPRM requirement shall be addressed. 5) Minimum flow for C1 RHRSW/EECW pump is 1700 gpm.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 5 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTION** Only one RHRSW Pump in a given RHRSW Pump Room may be counted toward meeting Technical Specification 3.7.2 requirements for EECW Pump operability. [1] To line up RHRSW Pump C1 for EECW System operation, **PERFORM** the following: [1.1] **ENSURE** EECW System is in prestartup/standby readiness alignment in accordance with Section 4.0. [1.2] **REVIEW** all precautions and limitations in Section 3.0. [1.3] **ENSURE** RHRSW Pump C1 is in standby readiness in accordance with 0-OI-23, Residual Heat Removal Service Water System. [1.4] through [1.10] – OUTSIDE ACTIONS
	Driver	**When contacted as the Outside AUO to perform Steps [1.4] – [1.9], acknowledge the direction and insert Event 30. Report that Steps [1.4] – [1.10] are complete, and C1 RHRSW Pump has been aligned for EECW.**
	BOP	[1.11] **REQUEST** a caution order be issued to tag RHRSW Pump C1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the C1 Pump should remain running to be operable for EECW.
	Driver	**When contacted as Work Control / Outside NUSO for a clearance on RHRSW Pump C1 and its associated crosstie valves, acknowledge the direction.**
	BOP	[2] To start RHRSW (EECW) Pump C1, PERFORM the following: [2.1] **START** RHRSW Pump C1 using one of the following: • 0-HS-23-8A/2, RHRSW PUMP C1, on Unit 2 [2.2] **CHECK** RHRSW Pump C1 running current is less than 61 amps using one of the following: • 0-EI-23-8/2, RHRSW PUMP C1 AMPS, on Unit 2 [2.3] AND [2.4] – OUTSIDE ACTIONS

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 6 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as the Outside AUO to complete Steps [2.3] and [2.4], report that the steps are complete.**
	BOP	[2.5] **NOTIFY** Chemistry of running RHRSW (EECW) Pump(s). [2.6] **ENSURE** a caution order has been issued to tag RHRSW Pump C1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the C1 pump should remain running to be operable for EECW.
	Driver	**When contacted as Chemistry acknowledge any information or direction given.** **When contacted as Work Control / Outside NUSO concerning the caution order, report that the caution order is in progress.**
	NUSO	Technical Specification 3.7.1, Residual Heat Removal Service Water (RHRSW) System LCO 3.7.1 ---------- NOTE ---------- The number of required RHRSW Pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for ≥ 24 hours. ---------- Four RHRSW subsystems shall be OPERABLE with the number of OPERABLE Pumps as listed below: 1. 1 unit fueled - four OPERABLE RHRSW Pumps. 2. 2 units fueled - six OPERABLE RHRSW Pumps. 3. 3 units fueled - eight OPERABLE RHRSW Pumps. **CONDITION:** One required RHRSW Pump inoperable.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 7 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** A.1 ----------NOTES---------- 1. Only applicable for the 2 units fueled condition. 2. Only four RHRSW Pumps powered from a separate 4 kV Shutdown Board are required to be OPERABLE if the other fueled unit has been in MODE 4 or 5 for > 24 hours. ---------- Verify five RHRSW Pumps powered from separate 4 kV Shutdown Boards are OPERABLE. OR A.2 Restore required RHRSW Pump to OPERABLE status.	**COMPLETION TIME:** A.1 – Immediately A.2 – 30 days
	NRC	**EXAMINER NOTE: SRO may elect to momentarily review Tech Spec 3.7.2 before moving to Tech Spec 3.7.1.**	
	NRC	**End of Event 4. Request that the Driver insert Event 5, H2O2 Analyzer Failure.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 1 of 4

Event Description: H2O2 Analyzer Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The H2O2 Analyzer will lose control power.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a loss of the H2O2 Analyzer.**
	NRC	**2-9-7C, Window 22 will alarm about 1 minute after Event 5 is entered by the Driver.**
	BOP	Acknowledges and reports the following alarms: • DRYWELL/SUPPR CHAMBER H2O2 ANALYZER FAILURE, 2-9-7C, Window 22
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the Alarm Response Procedure.
	BOP	2-ARP-9-7C, Alarm Response Procedure DRYWELL/SUPPR CHAMBER H2O2 ANALYZER FAILURE, Window 22 Operator Action: A. **CHECK** Panel 2-9-54 and 2-9-55 for abnormal indicating lights such as low flow, H2 or O2 downscale, pump off, etc. B. **IF** sample pump is NOT running, **THEN ATTEMPT** to start pump. Reference 3-OI-76, Placing H2/O2 Analyzer in Service. C. **IF** sample pump will NOT start OR H2/O2 analyzer malfunction, **THEN ENSURE** CR initiated to address issue. D. **REFER** TO Technical Specification 3.6.3.2, TRM 3.3.11 and TRM 3.6.2.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 2 of 4

Event Description:

Time	Position	Applicant's Actions or Behavior
	BOP	2-OI-76, Containment Inerting System Section 5.5, Placing the H2/O2 Analyzer in Service **NOTES** 1) In order to ensure accurate readings, H2/O2 ANALYZER is required to have been energized (power supply breakers closed) for at least 1 hour to allow temperatures to stabilize following cold start. 2) Performance of this section can result in DRYWELL/SUPPR CHAMBER H2/O2 ANALYZER FAILURE, Panel 2-9-7C window 22, to alarm when sampling suction paths are alternated. 3) This section is performed at Panels 2-9-54 & 2-9-55. [1] **DEPRESS** 2-HS-76-91, H2/O2 ANALYZER ISOLATION RESET. [2] **PLACE** H2/O2 ANALYZER DRYWELL/ SUPPRESSION CHAMBER SELECT, 2-HS-76-110, in either SUPPRESSION CHAMBER or DRYWELL. [3] **IF** 2-HS-76-110, H2/O2 ANALYZER DRYWELL/ SUPPRESSION CHAMBER SELECT, is in SUPPRESSION CHAMBER, **THEN CHECK** the following red indicating lights illuminated: • 2-IL-76-49-1, SUPPRESSION CHAMBER SAMPLE VALVES 2-FSV-76-55/56 OPEN • 2-IL-76-49-3, SAMPLE RETURN VALVES 2-FSV-76-57/58 OPEN [4] **IF** 2-HS-76-110, H2/O2 ANALYZER DRYWELL/ SUPPRESSION CHAMBER SELECT, is in DRYWELL, **THEN CHECK** the following red indicating lights illuminated: • 2-IL-76-49-2, DRYWELL SAMPLE VALVES 2-FSV-76-49/50 OPEN • 2-IL-76-49-3, SAMPLE RETURN VALVES 3-FSV-76-57/58 OPEN [5] AT 2-MON-76-110 A. **TOUCH** 2-MON-76-110 display screen B. **DEPRESS** Go to Panel PROCESS VALUES soft key C. **DEPRESS** Go to Panel MAINT MENU soft key D. **DEPRESS** LOG ON soft key

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 3 of 4

Event Description: H2O2 Analyzer Failure

Time	Position	Applicant's Actions or Behavior	
	BOP	E. **ENTER** password 1915 on soft keypad F. **DEPRESS** ENT soft key on keypad G. **DEPRESS** STANDBY MODE ON soft key to enable sample pump operation H. **ENSURE** soft key reads STANDBY MODE OFF I. **DEPRESS** Go To Panel PROCESS VALUES soft key J. **DEPRESS** Go To Panel MAIN soft key K. **ENSURE** STANDBY MODE not displayed	
	BOP	Determines that the sample pump will not restart and informs the NUSO.	
	NUSO	Technical Requirement 3.3.11 – Hydrogen Monitoring Instrumentation LCO 3.3.11 The Primary Containment Hydrogen Analyzer shall be OPERABLE Applicability: MODE 1 during the time period a. From 24 hours after THERMAL POWER is > 15% RTP following startup, to b. 24 hours prior to reducing THERMAL POWER to < 15% RTP prior to the next scheduled Reactor shutdown. **CONDITION:** Primary Containment Hydrogen Analyzer inoperable.	
	NUSO	**REQUIRED ACTION:** A. Restore Primary Containment Hydrogen Analyzer to OPERABLE status.	**COMPLETION TIME:** A. – 7 days

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 4 of 4

Event Description: H2O2 Analyzer Failure

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Requirement 3.6.2 – Oxygen Concentration Monitor LCO 3.6.2 The Primary Containment Oxygen Concentration Monitor shall be OPERABLE Applicability: MODES 1 and 2. **CONDITION:** Primary Containment Oxygen Concentration Monitor inoperable.	
	NUSO	**REQUIRED ACTION:** A. Begin alternate sampling and analyze results.	**COMPLETION TIME:** A. Immediately AND Once per 7 days thereafter
	NRC	**End of Event 5. Request that the Driver insert Event 6, 2C 4kV Unit Board Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 6 Page 1 of 2

Event Description: 2C 4kV Unit Board Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 4kV Unit Board 2C will trip, causing a loss of the running CRD Pump.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to trip 2C 4kV Unit Board.**
	Crew	Acknowledges and reports the following alarms: • 4KV UNIT BOARD 2C UNDERVOLTAGE, 2-9-8B, Window 14 • CONDENSATE BOOSTER PUMP C AUX OIL PRESSURE LOW, 2-9-6A, Window 14 • MOTOR TRIPOUT, 2-9-8C, Window 33
	OATC	Reports a loss of CRD System Flow due to 2A CRD Pump being de-energized.
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures and direct the OATC to respond in accordance with 2-AOI-85-3, CRD System Failure.
	OATC	2-AOI-85-3, CRD System Failure Immediate Actions [1] **IF** operating CRD PUMP has failed **AND** the standby CRD Pump is available, **THEN PERFORM** the following at Panel 2-9-5: (Otherwise N/A) [1.1] **PLACE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL in MAN at minimum setting. [1.2] **START** associated standby CRD Pump using the following: 2-HS-85-2A, CRD PUMP 1B [1.3] **IF** CRD Pump 1B was started, **THEN OPEN** CRD PUMP 1B DISCH TO U2, using 2-HS-85-8A [1.4] **ADJUST** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, to establish the following conditions: • 2-PDI-85-18A, CRD COOLING WATER HEADER DP, between 10 psid and 20 psid • 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, between 40 and 65 gpm [1.5] **BALANCE** CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, and **PLACE** in AUTO or BALANCE.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 6 Page 2 of 2

Event Description: 2C 4kV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	2-ARP-9-8B, Alarm Response Procedure 4KV UNIT BOARD 2C UNDERVOLTAGE, Window 14 Operator Action: A. **CHECK** Unit in stable condition by checking: • Condensate Pump 2C • Condensate Booster Pump 2C • RCW Pump 2C • CCW Pump 2C • CRD Pump 2A B. **IF** undervoltage has occurred, **THEN** 1. **CLEAR** disagreement lights on breakers. 2. **REDUCE** load as necessary to maintain stable operating conditions. 3. Condenser discharge may need to be throttled for two CCW pump operation. **REFER TO** 2-OI-27, Condenser Circulating Water System. C. **CHECK** Unit Bd 2C for abnormal conditions: relay targets, smoke, burned paint, etc. D. **REFER TO** 0-OI-57A, Switchyard and 4160V AC Electrical System, to re-energize board. E. **REFER TO** appropriate OI for recovery or realignment of equipment.
	NRC	**End of Event 6.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 7 Page 1 of 2

Event Description: Generator Bus Duct High Temperature

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Generator Bus Duct High Temperature alarm, Panel 2-9-7A, Window 32, will annunciate.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause Generator Bus Duct High Temperature annunciation.**
	BOP	Acknowledges and reports the following alarm: • GENERATOR BUS DUCT TEMPERATURE HIGH, 2-9-7A, Window 32
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	2-ARP-9-7A, Alarm Response Procedure GENERATOR BUS DUCT TEMPERATURE HIGH, Window 32 Operator Action: A. **IF** both fans are not operating, **THEN PERFORM** the following: • **START** all available fans GEN BUS DUCT HX FAN A(B) using 2-HS-262-1A(2A), on panel 2-9-8 • **ENSURE** system operation in accordance with 2-OI-47, Turbine-Generator System
	BOP	Starts 2B Bus Duct Cooling Fan.
	BOP	B. N/A C. N/A D. **CHECK** the status of alarm GEN BUS DUCT FAN FAILURE, 2-EA-262-161 (2-XA-55-7A, Window 31). E. **DISPATCH** personnel to the main Bus Cooling Fans on elevation 586' to check fans, cooler condition, and temperatures. F. **CHECK** temperature alarm is valid by available means including using a portable temperature monitor device locally at the 14 in-service thermostats. **REFER** to Figure 1.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 7 Page 2 of 2

Event Description: Generator Bus Duct High Temperature

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Turbine Building AUO to monitor Generator Temperatures locally, acknowledge the direction. Once the second Bus Duct Fan is started, wait 2 minutes and report that Main Bus Temperature is 73 C and steady.**
	BOP	**NOTE** With two fans operating at the design airflow of 45,000 scfm, the maximum allowable Bus Duct Cooler discharge temperature is 114ºF. G. N/A H. N/A I. **ENSURE** the system is operating in accordance with 2-OI-47, Turbine-Generator System.
	Driver	**Clear alarm as required for GENERATOR BUS DUCT TEMPERATURE HIGH, 2-9-7A, Window 32.**
	NRC	**End of Event 7. Request that the Driver insert Event 8, 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 1 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to cause a trip of 2B 480V Shutdown Board and a Loss of Offsite Power.**
	NRC	**EXAMINER NOTES:** **2B 480V Shutdown Board will trip when Event 8 is entered by the driver; the Loss of Offsite Power will occur 3 minutes after Event 8 is entered.** **'B' Diesel Generator (EDG) will not tie onto 'B' 4kV Shutdown Board due to an electrical fault. This event is entered upon Simulator setup – no action is required by the driver for this malfunction.** **Event 9, DG Fails to Start, and Event 10, Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start, will occur during Event 8. No action is required by the Driver to insert Events 9 or 10.**
	NRC	**Numerous alarms will be received when 2B 480V Shutdown Board is lost. Only some are covered in this Scenario Guide.**
	BOP	Acknowledges and reports the following alarms: • 480V SHUTDOWN BOARD 2B UV OR XFR, 2-9-8B, Window 30 • 480V REACTOR MOV BOARD 2B OR 2E UV, 2-9-8C, Window 24
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures and 2-AOI-99-1, Loss of Power to one RPS Bus.
	NRC	**Only the immediate action of 2-AOI-99-1, Loss of Power to one RPS Bus is covered below.**
	CREW	2-AOI-99-1, Loss of Power to one RPS Bus 4.1 Immediate Action [1] **STOP** all testing with potential RPS half-scrams or PCIS logic isolation signals.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 2 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 2-ARP-9-8B 480V SHUTDOWN BOARD 2B UV OR XFR, Window 30 Operator Action: A. **CHECK** for indication of 480V Shutdown Board 2B loss: • RWCU Pump 2B shutdown • Fuel Pool Cooling Pump 2B shutdown • 480V Shutdown Board 2B voltage (2-EI-57-30) • RBCCW Pump 2B failure • Half SCRAM B. N/A C. N/A D. **IF** manual transfer is **NOT** accomplished, **THEN REFER TO** Tech Spec 3.8.7.and 3.8.8
	Driver	**When contacted as the Reactor Building AUO to investigate the reason for the loss of 2B 480V Shutdown Board, acknowledge the direction. Wait 2 minutes and report to the crew that 2B 480V Shutdown Board overcurrent relay flags indicate an overcurrent condition has occurred.**
	NRC	**Due to time constraints, it is not expected that the NUSO reference Technical Specifications for this Event. Technical Specification evaluation should not be used to evaluate the applicant's Technical Specification competency.**
	BOP	Alarm Response Procedure, 2-ARP-9-8C 480V REACTOR MOV BD 2B OR 2E UV, Window 24 A. **CHECK** light indications for loss of any 480V equipment. B. **CHECK** 480V Rx MOV Bd 2B & 2E for abnormal conditions: relay targets, smoke, burned paint, breaker position, etc. **NOTE** If the mechanism resets (hear click and feel resistance when pushing in), this indicates that an overcurrent condition tripped the breaker.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 3 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	C. **IF** Normal or Alternate feeder breaker tripped, **THEN MANUALLY DEPRESS** mechanical trip/reset mechanism on breaker face to reset Bell Alarm lockout device. [NER/C II-B-92-069] D. **IF** undervoltage or transfer has occurred: 1. **REFER TO** Technical Specification Section 3.8.7. 2. **RESET** possible half-scram. **REFER TO** 2-OI-99, Reactor Protection System, E. **REFER TO** 0-OI-57B to re-energize or transfer board. F. **REFER TO** appropriate OI for recovery or realignment of equipment.
	BOP	Reports to the NUSO that Loop II Valves of RHR and Core Spray have lost power. Reports to the NUSO that a half SCRAM has occurred.
	Driver	**If contacted as the Control Bay AUO or Work Control / Outside NUSO to transfer the 2B 480V RMOV Board to Alternate, acknowledge the direction.**
	Crew	Determines that a Loss of Offsite Power has occurred.
	NRC	**Station Blackout conditions start 3 minutes and 30 seconds after Event 8 is inserted.**
	NUSO	Directs the crew to respond in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout and 2-AOI-100-1, Reactor SCRAM.
	NRC	**The subsequent actions contained in 2-AOI-100-1, Reactor SCRAM, are too numerous to list here, and only immediate actions are listed below.**
	OATC	2-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A and 2-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 2-9-5. [2] **PLACE** REACTOR MODE SWITCH, 2-HS-99-5A/S1, in SHUTDOWN. [3] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 4 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the NUSO: • Reactor SCRAM • MODE SWITCH is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Reactor Power level
	OATC	**NOTES** 1) Perform steps 4.1[5.3] and 4.1[5.4] in parallel. 2) Step 4.1[5.9] may be performed before step 4.1[5.7] if Reactor Water Level is slowly lowering. [5] N/A
	OATC	Provides a Reactor SCRAM report to the NUSO. Responds in accordance with 2-AOI-100-1, Reactor SCRAM.
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout **NOTE** Performing this instruction, in conjunction with an earthquake, may require resetting the individual Diesel Generator's 86G Lockout Relay and the Field Breaker (both locally at the Diesel Generator electrical cabinet). 1] **ENSURE** Diesel Generators have started and tied to respective 4kV Shutdown Boards, **THEN DISPATCH** personnel to Diesel Generators.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 5 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	Determines that 'B' DG will not tie onto 'B' 4 kV Shutdown Board. Determines that 'C' DG failed to automatically start.
	NRC	**See Event 9, DG Fails to Auto Start, on page 35 of 38.**
	BOP	[4] **PERFORM** the following to ensure at least one train of Diesel Generator Room Fans are energized: • **CHECK** 480V DSL Aux Board A or B energized
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout 4.2 Subsequent actions **NOTES** 1) The SBO Unit does only Attachment 12 while the other two units perform only the subsequent actions of the AOI. 2) The following is the preferred pump combinations of Unit 1 and Unit 2 RHR pumps used in SDC: 1B/2D, 1A/2C, 1A/2D, 1B/2C. By using these pumps, it ensures that a loss of a Diesel during a station blackout would not result in a loss of Shutdown cooling for both Unit 1 and Unit 2. [1] **IF** ANY EOI entry condition is met, **THEN REFER TO** the appropriate EOI(s).
	NUSO	Enters 2-EOI-1, RPV Control Directs the crew to maintain Reactor Water Level using any available Table L-1 or Table L-2 Injection Subsystem (Normally HPCI and/or RCIC).

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 6 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	NUSO	

Table L-1
Preferred Injection Systems

SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available ◇2 ◇3 ◇5	5C, 20M	1200 psig
HPCI with CST suction if available ◇2 ◇5 ◇7	5D, 20N	1200 psig
CNDS	6A	470 psig
CS ◇2	6D, 6E	330 psig
LPCI ◇2	6B, 6C	320 psig

Table L-2
Alternate Injection Subsystems

SOURCE	APPX	INJ PRESS
EHPM Pump	7L	1210 psig
SLC (test tank)	7B	1450 psig
SLC (boron tank)	7B	1450 psig
CNDS transfer pumps to RHR and CS	7A	110 psig
RHR crosstie to other units	7C	320 psig
Stby coolant	7D	160 psig
RHR drain pumps	7E, 7F	50 psig
PSC head tank pumps	7G	30 psig
RCIC (aux boiler steam) with CST suction if available ◇2 ◇3 ◇5	7H	1200 psig
RCIC manual start	20A	1200 psig
HPCI (aux boiler steam) with CST suction if available ◇2 ◇5 ◇7	7J	780 psig
Fire Protection system	7K	150 psig
FLEX Pump Sys (CILRT/CS)	20D	150 psig
FLEX Pump Sys (Standby Coolant)	20B	150 psig
FLEX Pump Sys (CILRT/CRD)	20C	150 psig

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 7 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	2-EOI-Appendix-5D, Injection System Lineup HPCI [1] N/A [2] N/A [3] N/A [4] **VERIFY** at least one SGT train in operation.
	Driver	**Standby Gas Trains (SGT) should already be running, but if not and if contacted to start an SGT, insert Event 18 to start 'A' SGT.**
	OATC / BOP	**CAUTIONS** 1) Operating HPCI Turbine below 2400 RPM may result in unstable system operation and equipment damage. 2) Operating HPCI Turbine with suction temperatures above 140º F may result in equipment damage. [5] **VERIFY** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection **NOTE** HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open. [6] N/A [7] **PLACE** HPCI AUXILIARY OIL PUMP handswitch in START. [8] **PLACE** HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE. [10] **OPEN** 2-FCV-73-44, HPCI PUMP INJECTION VALVE. [11] **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 8 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 2-CKV-73-45, HPCI SYSTEM CHECK VLV, opens by observing 2-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 GPM. **CAUTION** 2-FCV-073-0030, HPCI PUMP MIN FLOW VALVE, automatically opens when system flow is at or below 900 GPM (lowering) only if a system initiation signal is present. Manually opening the Minimum Flow Valve may be required for pump min flow protection. [13] **ADJUST** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO. [16] N/A [17] N/A
	OATC / BOP	2-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 2-XS-71-51A(B) RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and **OPEN** 2-FCV-71-9, RCIC TURB TRIP/THROT VALVE. [4] **ENSURE** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 GPM.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 9 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[5] **OPEN** 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE. [6] **OPEN** 2-FCV-71-39, RCIC PUMP INJECTION VALVE. [7] **OPEN** 2-FCV-71-25, RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 2-HS-71-31A, RCIC VACUUM PUMP, in START. **CAUTIONS** 1) Operating RCIC Turbine below 2100 RPM may result in unstable system operation and equipment damage. 2) High Suppression Chamber pressure may trip RCIC. 3) Operating RCIC Turbine with suction temperatures above 240º F may result in equipment damage. [9] **OPEN** 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine. [10] **CHECK** proper RCIC operation by observing the following: A. RCIC turbine speed accelerates above 2100 RPM. B. Flow to RPV controlled automatically at 620 GPM. C. 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 GPM. [11] **ADJUST** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] **IF** BOTH of the following exist: • RCIC Initiation signal is NOT present, **AND** • RCIC flow is below 60 GPM, **THEN ENSURE OPEN** 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 10 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>OATC / BOP</td><td>[13] N/A
[14] N/A</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to maintain Reactor Pressure below 1073 psig using 2-EOI-Appendix-11A, Alternate RPV Pressure Control Systems MSRVs.</td></tr>
<tr><td></td><td></td><td>2-EOI-APPENDIX-11A, Alternate RPV Pressure Control Systems MSRVs

[1] N/A
[2] N/A
[3] OPEN MSRVs using the following sequence to control RPV Pressure as directed by SRO:

<table>
<tr><td>1</td><td>2-PCV-1-179</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>2</td><td>2-PCV-1-180</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>3</td><td>2-PCV-1-4</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>4</td><td>2-PCV-1-31</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>5</td><td>2-PCV-1-23</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>6</td><td>2-PCV-1-42</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>7</td><td>2-PCV-1-30</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>8</td><td>2-PCV-1-19</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>9</td><td>2-PCV-1-5</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>10</td><td>2-PCV-1-41</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>11</td><td>2-PCV-1-22</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>12</td><td>2-PCV-1-18</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>13</td><td>2-PCV-1-34</td><td>MN STM LINE C RELIEF VALVE</td></tr>
</table>

[4] N/A
[5] N/A
[6] N/A</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 11 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	Continuing 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout 4.2 Subsequent actions [2] **IF** any Unit is under a Station Blackout, **THEN PERFORM** the following only for the Unit in Station Blackout • Perform Attachment 12 for that Unit
	NRC	**The Station Blackout time starts when power is lost to all 480V Shutdown Boards.**
	CREW	**Critical Task: START:_____ STOP:_____** **With a failure of an Emergency Diesel Generator (EDG) to automatically start and tie on to its respective 4KV Shutdown Board, the crew takes manual action to restore power to 2A 480V Shutdown Board to exit a Station Blackout, or commences a plant cooldown within 20 minutes of the Station Blackout.** **Critical Task Failure Criteria:** **The operating crew fails to restore power to 2A Shutdown Board or commence a plant cooldown within 20 minutes of the Station Blackout on Unit 2.**
	BOP	Determines that Unit 2 is in a Station Blackout condition due to the following plant conditions: • 2B 480V Shutdown Board has tripped and locked out • 'B' DG will not tie onto 'B' 4 kV Shutdown Board • 4KV Shutdown Board 'C' is de-energized until 'C' DG is manually started
	SRO	Acknowledges that Unit 2 is in a Station Blackout condition. Once the 'C' DG is started and tied to 'C' 4 kV Shutdown Board, directs the BOP to transfer 2A 480V Shutdown Board to the alternate power supply to exit Station Blackout.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 12 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	[3] N/A [4] **CHECK** automatic actions and **PERFORM** any that failed to occur. **NOTE** The following subsequent actions may be performed out of order, depending on plant conditions. [5] **REFER TO** 3-AOI-78-1, Fuel Pool Cooling System Failure for a complete Loss of AC POWER, as necessary. **NOTES** 1) EECW supply valves to the Control Air Compressors and Reactor Building Closed Cooling Water (RBCCW) are air operated. If initial air pressure is low, air compressors may trip on high temperature, until cooling water flow is established. 2) At US discretion, 0-FCV-67-53, EECW NORTH HDR SUPPLY VLV TO AIR COMP, can be placed in the open position with hand switch. The valve will automatically come open once EECW pressure is above setpoint. **REFER TO** 0-OI-67, Emergency Equipment Cooling Water System, for valve operation. 3) The North header supply to Unit 1 RBCCW, the North header supply to Unit 2 RBCCW and the South header supply to Unit 3 RBCCW are normally isolated with a manual valve; therefore, no flow will occur when either 1-FCV-67-50, EECW NORTH HEADER SUPPLY TO RBCCW, 2-FCV-67-50 EECW NORTH HEADER SUPPLY TO RBCCW, or 3-FCV-67-51, EECW SOUTH HEADER SUPPLY TO RBCCW, opens.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 13 of 13

Event Description: 2B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	6] **WHEN** EECW Header Pressure is restored above the reset pressure setpoint (psig) for the valves listed below, **THEN** Common / Unit 2 0-FCV-67-53 106 FCV-67-50 91 FCV-67-51 109 **RESET** EECW supplies to Control Air Compressors and RBCCW, at Unit 1 Panel 1-LPNL-925-0032 and Unit 2, 3 Panels 2(3)-25-32. **REFER TO** the EECW to the RCW Crossties for Control Air & RBCCW section of 0-OI-67, Emergency Equipment Cooling Water System.
	BOP	[7] **START** Control Air Compressors A, D and G as required and **MONITOR** system pressure. **REFER TO** 0-AOI-32-1, Loss of Control and Service Air Compressors. [7.1] **IF** an Air Compressor trips on high temperature, **THEN** (Otherwise N/A) **NOTIFY** Unit Supervisor for instructions. [8] **REFER TO** 3-AOI-32-2, Loss of Control Air, as necessary. [9] **PLACE** RPS MG Sets A and B in service. **REFER TO** 3-OI-99, Reactor Protection System. [10] **START** the Diesel Driven Fire Pump. **REFER TO** 0-OI-26, Fire Protection System.
	Driver	**If directed as the Outside/Work Control SRO to reset EECW, restore Control Air, restore RPS, or start the Diesel Driven Fire Pump, insert the following events as required:** • **Event 19 – Reset EECW** • **Event 20 – Restore Control Air** • **Event 21 – Restore RPS** • **Event 25 – Start Diesel Driven Fire Pump**
	NRC	**When the crew has restored power to 2A 480V Shutdown Board, has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using RCIC or HPCI, and has started EECW Pumps, end of scenario.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 9 Page 1 of 1

Event Description: DG Fails to Auto Start

Symptoms/Cues: Event is initiated during Simulator setup. 'C' DG will not automatically start upon the Loss of Offsite Power.

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9 is automatically entered on Simulator setup – no action is required by the Driver to insert Event 9.**
	NRC	**EXAMINER NOTE: Crew action is required to manually start 'C' DG – if no action is taken, Unit 2 will remain in Station Blackout.**
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) / Station Blackout 4.1 Immediate Actions [1] **ENSURE** Diesel Generators have started and tied to respective 4kV Shutdown Boards, **THEN DISPATCH** personnel to Diesel Generators. Determines that 'C' DG did not automatically start. Starts 'C' DG and informs the NUSO.
	NRC	**End of Event 9.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 10 Page 1 of 1

Event Description: Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start

Symptoms/Cues: Event is initiated during Simulator setup. EECW Pumps will not automatically start when the Emergency Diesels start on the Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 10 is automatically entered on Simulator setup – no action is required by the Driver to insert Event 10.**
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) / Station Blackout 4.1 Immediate Actions [2] **ENSURE** two EECW Pumps (not using the same EECW strainer) are in service supplying Diesel Generators. [3] IF two EECW Pumps (not using the same EECW strainer) are not in service supplying Diesel Generators, THEN PERFORM Attachment 9 (Cooling water is required to be established within 8 minutes)
	NRC	**Previously the crew should have aligned either the A1 or the C1 RHRSW Pump to the North EECW header. If C1 was placed in service, that pump will no longer be available due to subsequent failure of the 'B' DG to tie to the 'B' 4kV Shutdown Board. A1 RHRSW cannot be aligned to EECW within 8 minutes to meet 0-AOI-57-1A requirements. Consequently, the crew should place in service EECW South header pump(s) (B3/D3).**
	CREW	**Critical Task** **The crew restores EECW Flow with at least 1 EECW Pump within 8 minutes from the time the EDGs start to prevent damage to and a loss of the Emergency Diesel Generators.** **Critical Task Failure Criteria** **The operating crew fails to restore EECW Flow with at least one EECW Pump to the EDGs within 8 minutes following EDG start upon a Loss of Offsite Power.**
	BOP	Starts EECW Pumps to cool the DGs.
	NRC	**End of Event 10.**

Scenario Setup
UNIT 2

IC	28
Exam IC	286

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-2	114	EOI-1	21		
OI-67	126	APPX-5D	10		
OI-76	78	APPX-5C	8		
OI-85	156	APPX-11A	5		
GOI-100-12A	125				
ARP-9-6A	36				
ARP-9-7A	35				
ARP-9-7C	31				
ARP-9-8B	19				
ARP-9-8C	22				
AOI-57-1A	117				
AOI-85-3	26				
AOI-99-1	31				
AOI-100-1	121				

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Danger Tag on A3 EECW Pump
Schedule Files(s):	ILT 2304 NRC Scenario 2 UNIT 2.sch
Event Files(s):	ILT 2304 NRC Scenario 2 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ON LINE	Drywell Leakage (GPM)	Protected Equipment
	230		
	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	1.89	
Rx Power 70%		Floor Drain (GPM) 0.31	
MWe 862		Equipment Drain (GPM) 1.58	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

INFO ONLY TS 3.7.2, Condition A, due to A3 EECW tagged for scheduled maintenance

LCOs OF 72 HOURS OR LESS

None

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

A3 EECW Pump tagged for scheduled maintenance

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

To support preparations for Reactor SCRAM, continue with power reduction and planned equipment shutdowns in accordance with 2-GOI-100-12A, Section 5.3.1 [5.3], and 2-OI-2, Condensate System, Section 7.1 [3.4].

Support maintenance on A3 EECW Pump

OPERATOR WORK AROUNDS **OWAs – U0-1/U3-0** **Burdens - U0 – 0/U3-0** **Operator Challenges – U0 – 12/U3 - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-2
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 70% Reactor Power, Reactor Shutdown in Progress. A3 EECW Pump is Tagged for maintenance.

Turnover: To support preparations for Reactor SCRAM, continue with power reduction and planned equipment shutdowns in accordance with 3-GOI-100-12A, Section 5.3.1 [5.3], and 3-OI-2, Condensate System, Section 7.1 [3.4]. Support maintenance on A3 EECW Pump.

Critical Tasks:
1. With a failure of an Emergency Diesel Generator (EDG) to automatically start and tie on to its respective 4KV Shutdown Board, the crew takes manual action to restore power to 3A 480V Shutdown Board to exit a Station Blackout, or commences a plant cooldown within 20 minutes of the Station Blackout.

2. The crew restores EECW Flow with at least 1 EECW Pump within 8 minutes from the time the EDGs start to prevent damage to and a loss of the Emergency Diesel Generators.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Shutdown 3C Condensate/Condensate Booster Pumps
2.	N/A	R-OATC	Lower Reactor Power Using Control Rods
3.	RD12	C-OATC C-NUSO	Control Rod Difficult to Insert
4.	SW03C	C-BOP TS-NUSO	Emergency Equipment Cooling Water (EECW) Pump Trip
5.	PMP-76-110	TS-NUSO	H2/O2 Analyzer Failure
6.	ED08C	C-OATC MC-OATC C-NUSO	3C 4KV Unit Board Trip
7.	XA-55-7A_32	C-BOP C-NUSO	Generator Bus Duct High Temperature
8.	ED01 ED01A 52_BKR DG A	M-ALL	3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie
9.	DG01B	C-BOP MC-BOP C-NUSO	DG Fails to Auto Start
10.	SW07A SW07B SW07C SW07D	C-BOP MC-BOP C-NUSO	Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will shut down 3C Condensate/Condensate Booster Pumps in accordance with 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, and 3-OI-2, Condensate System.
2. The crew will continue the Reactor Shutdown by inserting Control Rods in accordance with 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, and 3-OI-85, Control Rod Drive System.
3. A Control Rod will be difficult to insert during the Reactor Shutdown. The crew will take action in accordance with 3-OI-85, Control Rod Drive System, to insert the Control Rod.
4. C3 Emergency Equipment Cooling Water (EECW) Pump will trip. The crew will start another pump to support the North EECW Header in accordance with 0-OI-67, Emergency Equipment Cooling Water. The Nuclear Unit Senior Operator (NUSO) will reference Technical Specifications.
5. The H2/O2 Analyzer will fail, requiring the NUSO to address Technical Specifications.
6. 3C 4KV Unit Board will trip and lock out due to an electrical fault. The crew will respond in accordance with Alarm Response Procedures and Abnormal Operating Instructions.
7. The crew will receive an annunciator that Generator Bus Duct Temperature is high, requiring action in accordance with Alarm Response Procedures.
8. 3B 480V Shutdown Board Normal Feeder Breaker will trip, followed by a loss of all Offsite Power. Coupled with the loss of 3B 480V Shutdown Board and the failure of 3A Emergency Diesel Generator (DG) to tie onto the 3EA 4KV Shutdown Board, the Loss of Offsite Power will result in a Station Blackout. The crew will respond in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) / Station Blackout, until the crew transfers 3A 480V Shutdown Board to its alternate power supply.
9. 3B DG will fail to automatically start on the Loss of Offsite Power, requiring the crew to manually start 3B DG.
10. Emergency Equipment Cooling Water (EECW) Pumps will fail to auto start when the DGs restore power to the 4160V Shutdown Boards. The crew will restart the EECW pumps within 8 minutes as required by 0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV)/Station Blackout.

The Scenario ends when the crew has restored power to 3A 480V Shutdown Board, has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) with high pressure makeup sources, and has restored EECW Flow to the Emergency Diesel Generators.

Critical Tasks 2

1. **With a failure of an Emergency Diesel Generator (EDG) to automatically start and tie on to its respective 4KV Shutdown Board, the crew takes manual action to restore power to 3A 480V Shutdown Board to exit a Station Blackout, or commences a plant cooldown within 20 minutes of the Station Blackout.**

   a. **Safety Significance**
   Mitigate Station Blackout.

   b. **Cues**
   Procedural compliance.
   Loss of power to both 480V Shutdown Boards.
   Loss of power to Low Pressure ECCS system valves.

   c. **Measured by:**
   Observation – Power restored to 3A 480V Shutdown Board or a plant cooldown is commenced.

   d. **Feedback**
   Power is restored to 3A 480 Shutdown Board.
   Power available to Low Pressure ECCS system valves.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to restore power to 3A Shutdown Board or commence a plant cooldown within 20 minutes of the Station Blackout on Unit 3.

2. **The crew restores EECW Flow with at least 1 EECW Pump within 8 minutes from the time the EDGs start to prevent damage to and a loss of the Emergency Diesel Generators.**

   a. **Safety Significance**
   Diesel Generator operation without EECW flow will cause EDG failure.

   b. **Cues**
   Procedural compliance.
   EECW Header Flow.
   EECW Pump Run Indications.

   c. **Measured by:**
   Observation – The crew verifies at least one EECW Pump is restarted as required when power is restored to the 4KV Shutdown Busses.

   d. **Feedback**
   EECW Header Flow.
   EECW Pump Run indications.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to restore EECW Flow with at least one EECW Pump to the EDGs within 8 minutes following EDG start upon a Loss of Offsite Power.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 1 Page 1 of 2

Event Description: Shutdown 3C Condensate/Condensate Booster Pumps

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**NOTE: The crew has been pre-briefed, but may elect to hold a re-focus brief.**
	NUSO	3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, Section 5.3.1, Reducing Reactor Power to 40% [5] **WHEN** Reactor is at less than 76% power (12 M lbm/hr) **THEN** [5.3] **REMOVE** from service one of the three Condensate Booster Pumps, as directed by the NUSO, per 3-OI-2, Condensate System. [5.2] **REMOVE** from service one of the three Condensate Pumps, as directed by the NUSO, per 3-OI-2, Condensate System.
	BOP	3-OI-2. Condensate System Section 3.1, General Precautions D. To remove a Condensate pump along with any other pump, then additional restrictions apply: 1. Pumps must be removed from high pressure to low 2. Condensate Flow should be reduced to 76% (12.0 M lbm/hr) prior to removing the Condensate Pump. 3. Condensate Flow can be restored to 87.5% (14.1 M lbm/hr). 4. The Condensate/Feedwater System is susceptible to perturbations from other evolutions. Any evolutions that have the potential to affect Condensate/Feedwater System flow/NPSH should be considered High Risk. Section 7.1, Condensate System Shutdown [1] **CHECK** Condensate system in operation in accordance with this instruction. [2] **REVIEW** the Precautions and Limitations in Section 3.0.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 1 Page 2 of 2

Event Description: Shutdown 3A Condensate/Condensate Booster Pumps

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **IF** removing the first two Condensate Booster Pumps/Condensate Pumps for power reductions or Condensate System shutdown, **THEN PERFORM** the following: [3.1] **ENSURE** Reactor Power is less than or equal to 76% (12.0 M lbm/hr) prior to removing a Condensate Pump. (Ref. P&L 3.1D.) [3.2] **ENSURE** Hydrogen Injection is secured to the Condensate Booster Pump(s) to be stopped in Section 7.1. **REFER TO** 3-OI-4, Hydrogen Water Chemistry System.
	Driver	**If contacted as the Turbine Building Assistant Unit Operator to shut down Hydrogen Injection, report that Hydrogen Injection has been stopped to Condensate Booster Pump 3C.**
	BOP	[3.3] **ENSURE** the number of Condensate/Condensate Booster Pumps to be left in service will supply adequate suction pressure to running RFP(s). [3.4] **WHEN** directed by 3-GOI-100-12, 3-GOI-100-12A, or the Unit 3 NUSO, **THEN STOP** CONDENSATE BOOSTER PUMP using one of the following: • 3-HS-2-68A, CONDENSATE BOOSTER PUMP 3C [3.5] **REMOVE** Condensate Demins from service as required, and PLACE in standby. **REFER TO** 3-OI-2A, Condensate Demineralizer System.
	Driver	**When contacted as the Turbine Building AUO to remove Demins from service, acknowledge any direction given.**
	BOP	[3.6] **NOTIFY** Chemistry which Condensate Pump(s) will be stopped and to realign oxygen injection to Condensate Pump(s).
	Driver	**When contacted as Chemistry to realign oxygen injection, report that realignment has been completed.**
	BOP	[3.7] **ENSURE** the number of Condensate/Condensate Booster Pumps to be left in service will supply adequate suction pressure to running RFP(s). [3.8] **WHEN** directed by 3-GOI-100-12, 3-GOI-100-12A, or the Unit 3 NUSO, **THEN STOP** CONDENSATE PUMP using one of the following: • 3-HS-2-15A, CONDENSATE PUMP 3C
	NRC	**End of Event 1. Continue to Event 2, Lower Reactor Power Using Control Rods.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 2 Page 1 of 2

Event Description: Lower Reactor Power Using Control Rods

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: During Event 2, Event 3 (Control Rod Difficult to Insert) is automatically entered on Simulator setup. No Action is required by the Driver to insert Event 3.**
	NUSO	3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, Section 5.3.1, Reducing Reactor Power to 40% [6] **CONTINUE** to **REDUCE** Reactor Power by a combination of Control Rod insertions and Core Flow changes, as recommended by Reactor Engineer.
	NUSO	Directs the Operator at the Controls (OATC) to continue the Reactor Shutdown by inserting Control Rods in accordance with 3-OI-85, Control Rod Drive System, and Reactivity Control Plan (RCP) RCP-NRC2304-U3S2.
	OATC	3-OI-85, Control Rod Drive System, Section 6.7, Control Rod Insertion Section 6.7.1, Initial Requirements [1] **REVIEW** Precautions and Limitations in Sections 3.7 and 3.8. [2] **ENSURE** the following prior to Control Rod movement: • CRD POWER, 3-HS-85-46 in ON • ROD WORTH MINIMIZER is operable and LATCHED in to the correct ROD GROUP, when Rod Worth Minimizer is enforcing [3] **OBSERVE** the following during Control Rod repositioning: • Control Rod reed switch position indicators (four rod display) agree with the indication on the Full Core Display. • Nuclear Instrumentation responds as Control Rods move through the core (This ensures Control Rod is following drive during Control Rod movement.) [4] **PERFORM** the following to insert the Control Rod as appropriate. • Control Rod Notch Insertion per Section 6.7.2 • Control Rod Continuous Insertion per Section 6.7.3

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 2 Page 2 of 2

Event Description: Lower Reactor Power Using Control Rods

Time	Position	Applicant's Actions or Behavior
	OATC	Section 6.7.3, Continuous Insertion of Control Rod [1] **CHECK** Section 6.7.1 has been performed. [2] **SELECT** the desired Control Rod by depressing the appropriate 3-XS-85-40, CRD ROD SELECT pushbutton.
	NRC	**EXAMINER NOTE: Control Rod insertion order per RCP-NRC2304-U3S2.**
	OATC	[3] **OBSERVE** the following for the selected Control Rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED • White light on the Full Core Display ILLUMINATED [4] **PLACE** and HOLD 3-HS-85-48, CRD CONTROL SWITCH, in ROD IN. [5] **WHEN** Control Rod notch reaches the even rod notch position prior to the desired final Control Rod notch position, **THEN RELEASE** 3-HS-85-48, CRD CONTROL SWITCH. [6] **OBSERVE** the Control Rod settles into desired position and the ROD SETTLE light extinguishes.
	NRC	**When Event 3 is complete and satisfied with Control Rod insertion, end of Events 2 and 3.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 3

Event Description: Control Rod Difficult to Insert

Symptoms/Cues: Event is initiated by Control Rod position. While driving in Control Rod 06-39, it will slow and stick at Position 38.

Time	Position	Applicant's Actions or Behavior
	NRC	**Event is inserted on simulator startup and is initiated by Control Rod position. No action is required by the Driver to insert Event 3.**
	OATC	While inserting Control Rods in accordance with 3-OI-85, Control Rod Drive System, determines that Control Rod 06-39 is difficult to insert. Informs the Nuclear Unit Senior Operator (NUSO).
	NUSO	Directs the OATC to take action in accordance with 3-OI-85.
	OATC	3-OI-85, Control Rod Drive System Section 8.17, Control Rod Difficult to Insert [1] **ENSURE** the Control Rod will NOT notch in, in accordance with Section 6.7 or Section 8.20. [2] **REVIEW** all Precautions and Limitations in Section 3.0. [3] N/A [4] **CHECK** CRD SYSTEM FLOW is between 40 gpm and 65 gpm, indicated by 3-FIC-85-11. [5] **ENSURE** 3-PDI-85-17A, CRD DRIVE WATER HEADER DP is between 250 -270 psid on by throttling 3-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE, as necessary. [6] N/A [7] **IF** Control Rod motion is observed, but the CRD fails to notch-in (example: settles back to the original position) with normal operating drive water pressure, **THEN**: [7.1] **NOTIFY** Reactor Engineer to determine what parameters should be recorded for further evaluation.
	Driver	**When contacted as the Reactor Engineer, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 3 Page 2 of 2

Event Description: Control Rod Difficult to Insert

Time	Position	Applicant's Actions or Behavior
	OATC	[7.2] **RAISE** 3-PDI-85-17A, CRD DRIVE WATER HEADER DP, NOT to exceed 300 psid, using 3-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE.
	NRC	**EXAMINER NOTE: Control Rod 06-39 will insert when CRD Drive Water Pressure is raised about 285 psig.**
	OATC	[7.3] **INSERT** Control Rod as directed in Section 6.7. [7.4] **LOWER** 3-PDI-85-17A, CRD DRIVE WATER HEADER DP, to between 250 psid and 270 psid using 3-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE.
	NRC	**End of Event 3. When satisfied with Control Rod insertion, end of Events 2 and 3. Request that the Driver insert Event 4.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 1 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. C3 EECW Pump will trip.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause C3 Emergency Equipment Cooling Water (EECW) Pump to trip.**
	NRC	**EXAMINER NOTE: A3 EECW Pump is tagged out as part of the initial conditions – therefore, when C3 EECW Pump Trips, there will be no EECW Pumps running on the North Header.**
	BOP	Acknowledges and reports the following alarms to the NUSO: • MOTOR TRIPOUT, 3-9-8C, Window 33
	BOP	Determines that C3 EECW Pump has tripped and informs the NUSO.
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure and 0-OI-67, Emergency Equipment Cooling Water.
	BOP	3-ARP-9-8C, Alarm Response Procedure MOTOR TRIPOUT, Window 33 A. **CHECK** Control Room for white disagreement light illuminated for affected equipment. B. **CLEAR** disagreement light. C. **DISPATCH** personnel to CHECK: 1. Relays at associated electrical board. 2. Equipment for abnormal conditions. 3. Safe-stop locally reset, if necessary. D. **REFER TO** 0-GOI-300-2 if relay targets are present or for motor starting limits. E. **REFER TO** appropriate OI for recovery or realignment of equipment.
	NRC	**The crew may elect to start either A1 or C1 RHRSW Pump and align to EECW in accordance with 0-OI-67, Emergency Equipment Cooling Water. Both procedures are shown below, but only one of the two will be used. (Procedure for C1 starts on page 10 of 38)**

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 2 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**EXAMINER NOTE: The crew may discuss filling and venting the EECW North Header – if so, direct the driver to inform the NUSO that time compression has been used, and that the North Header has been filled and vented.**
	BOP	0-OI-67, Emergency Equipment Cooling Water **IF THE CREW ELECTS TO USE A1 EECW PUMP:** Section 8.1, Operation of RHRSW Pump A1 (for EECW in place of A3) **NOTES** 1) RHRSW Pump A1 may be aligned for service by this section when: • It is used to meet the minimum number of Tech. Spec. operable pumps; or • At the discretion of the NUSO, it is needed to replace another pump's operation; or • At the discretion of the NUSO, it is needed to assist in supplying header flow/pressure demand 2) If used to meet EECW requirements, RHRSW Pump A1 must be aligned to EECW, the pump started, and should remain running. RHRSW Pump A1 does NOT have the same auto start signals as RHRSW Pump A3. 3) The RHRSW pump control switches and amp meters are located at Control Room Panel 9-3, Unit 1, 2, and 3. 4) When RHRSW Pump A1 is aligned for EECW, its RHRSW function required by the Safe Shutdown Program (NFPA 805) is inoperable. NFPA 805 FPR or FPRM requirement shall be addressed. 5) Minimum flow for A1 RHRSW/EECW Pump is 1700 gpm. **CAUTION** Only one RHRSW Pump in a given RHRSW Pump Room may be counted toward meeting Technical Specification 3.7.2 requirements for EECW Pump operability.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 3 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[1] To line up RHRSW Pump A1 for EECW System operation, **PERFORM** the following: [1.1] **ENSURE** EECW System is in prestartup/standby readiness alignment in accordance with Section 4.0. [1.2] **REVIEW** all precautions and limitations in Section 3.0. [1.3] **ENSURE** RHRSW Pump A1 is in standby readiness in accordance with 0-OI-23, Residual Heat Removal Service Water System. [1.4] through [1.10] – OUTSIDE ACTIONS
	Driver	**When contacted as the Outside AUO to perform Steps [1.4] – [1.9], acknowledge the direction and insert Event 14. Report that Steps [1.4] – [1.10] are complete, and A1 RHRSW Pump has been aligned for EECW.**
	BOP	[1.11] **REQUEST** a caution order be issued to tag RHRSW Pump A1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the A1 Pump should remain running to be operable for EECW.
	Driver	**When contacted as Work Control / Outside NUSO for a clearance on RHRSW Pump A1 and its associated crosstie valves, acknowledge the direction.**
	BOP	[2] To start RHRSW (EECW) Pump A1, PERFORM the following: [2.1] **START** RHRSW Pump A1 using one of the following: • 0-HS-23-1A/3, RHRSW PUMP A1, on Unit 3 [2.2] **CHECK** RHRSW Pump A1 running current is less than 61 amps using one of the following: • 0-EI-23-1/3, RHRSW PUMP A1 AMPS, on Unit 3 [2.3] AND [2.4] – OUTSIDE ACTIONS
	Driver	**When contacted as the Outside AUO to complete Steps [2.3] and [2.4], report that the steps are complete.**

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 4 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[2.5] **NOTIFY** Chemistry of running RHRSW (EECW) Pump(s). [2.6] **ENSURE** a caution order has been issued to tag RHRSW Pump A1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the A1 pump should remain running to be operable for EECW.
	Driver	**When contacted as Chemistry acknowledge any information or direction given.** **When contacted as Work Control / Outside NUSO concerning the caution order, report that the caution order is in progress.**
	BOP	**IF THE CREW ELECTS TO USE C1 EECW PUMP:** Section 8.3, Operation of RHRSW Pump C1 (for EECW in place of C3) **NOTES** 1) RHRSW Pump C1 may be aligned for service by this section when: • It is used to meet the minimum number of Tech. Spec. operable pumps; or • At the discretion of the NUSO, it is needed to replace another pump's operation; or • At the discretion of the NUSO, it is needed to assist in supplying header flow/pressure demand 2) If used to meet EECW requirements, RHRSW pump C1 must be aligned to EECW, the pump started, and should remain running. RHRSW Pump C1 does NOT have the same auto start signals as RHRSW Pump C3. 3) The RHRSW pump control switches and amp meters are located on Panel 9-3, Unit 1, 2, and 3. 4) When RHRSW Pump C1 is aligned for EECW, its RHRSW function required by the Safe Shutdown Program (NFPA 805) is inoperable. NFPA 805 FPR or FPRM requirement shall be addressed. 5) Minimum flow for C1 RHRSW/EECW pump is 1700 gpm.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 5 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTION** Only one RHRSW Pump in a given RHRSW Pump Room may be counted toward meeting Technical Specification 3.7.2 requirements for EECW Pump operability. [1] To line up RHRSW Pump C1 for EECW System operation, **PERFORM** the following: [1.1] **ENSURE** EECW System is in prestartup/standby readiness alignment in accordance with Section 4.0. [1.2] **REVIEW** all precautions and limitations in Section 3.0. [1.3] **ENSURE** RHRSW Pump C1 is in standby readiness in accordance with 0-OI-23, Residual Heat Removal Service Water System. [1.4] through [1.10] – OUTSIDE ACTIONS
	Driver	**When contacted as the Outside AUO to perform Steps [1.4] – [1.9], acknowledge the direction and insert Event 30. Report that Steps [1.4] – [1.10] are complete, and C1 RHRSW Pump has been aligned for EECW.**
	BOP	[1.11] **REQUEST** a caution order be issued to tag RHRSW Pump C1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the C1 Pump should remain running to be operable for EECW.
	Driver	**When contacted as Work Control / Outside NUSO for a clearance on RHRSW Pump C1 and its associated crosstie valves, acknowledge the direction.**
	BOP	[2] To start RHRSW (EECW) Pump C1, PERFORM the following: [2.1] **START** RHRSW Pump C1 using one of the following: • 0-HS-23-8A/3, RHRSW PUMP C1, on Unit 3 [2.2] **CHECK** RHRSW Pump C1 running current is less than 61 amps using one of the following: • 0-EI-23-8/3, RHRSW PUMP C1 AMPS, on Unit 3 [2.3] AND [2.4] – OUTSIDE ACTIONS

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 6 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as the Outside AUO to complete Steps [2.3] and [2.4], report that the steps are complete.**
	BOP	[2.5] **NOTIFY** Chemistry of running RHRSW (EECW) Pump(s). [2.6] **ENSURE** a caution order has been issued to tag RHRSW Pump C1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the C1 pump should remain running to be operable for EECW.
	Driver	**When contacted as Chemistry acknowledge any information or direction given.** **When contacted as Work Control / Outside NUSO concerning the caution order, report that the caution order is in progress.**
	NUSO	Technical Specification 3.7.1, Residual Heat Removal Service Water (RHRSW) System LCO 3.7.1 ---------- NOTE ---------- The number of required RHRSW Pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for ≥ 24 hours. ---------- Four RHRSW subsystems shall be OPERABLE with the number of OPERABLE Pumps as listed below: 1. 1 unit fueled - four OPERABLE RHRSW Pumps. 2. 2 units fueled - six OPERABLE RHRSW Pumps. 3. 3 units fueled - eight OPERABLE RHRSW Pumps. **CONDITION:** One required RHRSW Pump inoperable.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 4 Page 7 of 7

Event Description: Emergency Equipment Cooling Water (EECW) Pump Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** A.1 ----------NOTES---------- 1. Only applicable for the 2 units fueled condition. 2. Only four RHRSW Pumps powered from a separate 4 kV Shutdown Board are required to be OPERABLE if the other fueled unit has been in MODE 4 or 5 for > 24 hours. ---------- Verify five RHRSW Pumps powered from separate 4 kV Shutdown Boards are OPERABLE. <u>OR</u> A.2 Restore required RHRSW Pump to OPERABLE status.	**COMPLETION TIME:** A.1 – Immediately A.2 – 30 days
	NRC	**EXAMINER NOTE: SRO may elect to momentarily review Tech Spec 3.7.2 before moving to Tech Spec 3.7.1.**	
	NRC	**End of Event 4. Request that the Driver insert Event 5, H2O2 Analyzer Failure.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 1 of 4

Event Description: H2O2 Analyzer Failure

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The H2O2 Analyzer will lose control power.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a loss of the H2O2 Analyzer.**
	NRC	**3-9-7C, Window 22 will alarm about 1 minute after Event 5 is entered by the Driver.**
	BOP	Acknowledges and reports the following alarms: • DRYWELL/SUPPR CHAMBER H2O2 ANALYZER FAILURE, 3-9-7C, Window 22
	NUSO	Directs the Balance of Plant Operator (BOP) to respond in accordance with the Alarm Response Procedure.
	BOP	3-ARP-9-7C, Alarm Response Procedure DRYWELL/SUPPR CHAMBER H2O2 ANALYZER FAILURE, Window 22 Operator Action: A. **CHECK** Panel 3-9-54 and 3-9-55 for abnormal indicating lights such as low flow, H2 or O2 downscale, pump off, etc. B. **IF** sample pump is NOT running, **THEN ATTEMPT** to start pump. Reference 3-OI-76, Placing H2/O2 Analyzer in Service. C. **IF** sample pump will NOT start OR H2/O2 analyzer malfunction, **THEN ENSURE** CR initiated to address issue. D. **REFER** TO Technical Specification 3.6.3.2, TRM 3.3.11 and TRM 3.6.2.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 2 of 4

Event Description:

Time	Position	Applicant's Actions or Behavior
	BOP	3-OI-76, Containment Inerting System Section 5.5, Placing the H2/O2 Analyzer in Service **NOTES** 1) In order to ensure accurate readings, H2/O2 ANALYZER is required to have been energized (power supply breakers closed) for at least 1 hour to allow temperatures to stabilize following cold start. 2) Performance of this section can result in DRYWELL/SUPPR CHAMBER H2/O2 ANALYZER FAILURE, Panel 3-9-7C window 22, to alarm when sampling suction paths are alternated. 3) This section is performed at Panels 3-9-54 & 3-9-55. [1] **DEPRESS** 3-HS-76-91, H2/O2 ANALYZER ISOLATION RESET. [2] **PLACE** H2/O2 ANALYZER DRYWELL/ SUPPRESSION CHAMBER SELECT, 3-HS-76-110, in either SUPPRESSION CHAMBER or DRYWELL. [3] **IF** 3-HS-76-110, H2/O2 ANALYZER DRYWELL/ SUPPRESSION CHAMBER SELECT, is in SUPPRESSION CHAMBER, **THEN CHECK** the following red indicating lights illuminated: • 3-IL-76-49-1, SUPPRESSION CHAMBER SAMPLE VALVES 3-FSV-76-55/56 OPEN • 3-IL-76-49-3, SAMPLE RETURN VALVES 3-FSV-76-57/58 OPEN [4] **IF** 3-HS-76-110, H2/O2 ANALYZER DRYWELL/ SUPPRESSION CHAMBER SELECT, is in DRYWELL, **THEN CHECK** the following red indicating lights illuminated: • 3-IL-76-49-2, DRYWELL SAMPLE VALVES 3-FSV-76-49/50 OPEN • 3-IL-76-49-3, SAMPLE RETURN VALVES 3-FSV-76-57/58 OPEN [5] AT 3-MON-76-110 A. **TOUCH** 3-MON-76-110 display screen B. **DEPRESS** Go to Panel PROCESS VALUES soft key C. **DEPRESS** Go to Panel MAINT MENU soft key D. **DEPRESS** LOG ON soft key

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 3 of 4

Event Description: H2O2 Analyzer Failure

Time	Position	Applicant's Actions or Behavior	
	BOP	E. **ENTER** password 1915 on soft keypad F. **DEPRESS** ENT soft key on keypad G. **DEPRESS** STANDBY MODE ON soft key to enable sample pump operation H. **ENSURE** soft key reads STANDBY MODE OFF I. **DEPRESS** Go To Panel PROCESS VALUES soft key J. **DEPRESS** Go To Panel MAIN soft key K. **ENSURE** STANDBY MODE not displayed	
	BOP	Determines that the sample pump will not restart and informs the NUSO.	
	NUSO	Technical Requirement 3.3.11 – Hydrogen Monitoring Instrumentation LCO 3.3.11 The Primary Containment Hydrogen Analyzer shall be OPERABLE Applicability: MODE 1 during the time period a. From 24 hours after THERMAL POWER is > 15% RTP following startup, to b. 24 hours prior to reducing THERMAL POWER to < 15% RTP prior to the next scheduled Reactor shutdown. **CONDITION:** Primary Containment Hydrogen Analyzer inoperable.	
	NUSO	**REQUIRED ACTION:** A. Restore Primary Containment Hydrogen Analyzer to OPERABLE status.	**COMPLETION TIME:** A. – 7 days

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 5 Page 4 of 4

Event Description: H2O2 Analyzer Failure

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Requirement 3.6.2 – Oxygen Concentration Monitor LCO 3.6.2 The Primary Containment Oxygen Concentration Monitor shall be OPERABLE Applicability: MODES 1 and 2. **CONDITION:** Primary Containment Oxygen Concentration Monitor inoperable.	
	NUSO	**REQUIRED ACTION:** A. Begin alternate sampling and analyze results.	**COMPLETION TIME:** A. Immediately AND Once per 7 days thereafter
	NRC	**End of Event 5. Request that the Driver insert Event 6, 3C 4kV Unit Board Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 6 Page 1 of 2

Event Description: 3C 4kV Unit Board Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 4kV Unit Board 3C will trip, causing a loss of the running CRD Pump.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to trip 3C 4kV Unit Board.**
	Crew	Acknowledges and reports the following alarms: • 4KV UNIT BOARD 3C UNDERVOLTAGE, 3-9-8B, Window 14 • CONDENSATE BOOSTER PUMP C AUX OIL PRESSURE LOW, 3-9-6A, Window 14 • MOTOR TRIPOUT, 3-9-8C, Window 33
	OATC	Reports a loss of CRD System Flow due to 3A CRD Pump being de-energized.
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures and direct the OATC to respond in accordance with 3-AOI-85-3, CRD System Failure.
	OATC	3-AOI-85-3, CRD System Failure Immediate Actions [1] **IF** operating CRD PUMP has failed **AND** the standby CRD Pump is available, **THEN PERFORM** the following at Panel 3-9-5: (Otherwise N/A) [1.1] **PLACE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL in MAN at minimum setting. [1.2] **START** associated standby CRD Pump using the following: 3-HS-85-2A, CRD PUMP 3B [1.3] **ADJUST** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, to establish the following conditions: • 3-PDI-85-18A, CRD COOLING WATER HEADER DP, between 10 psid and 20 psid • 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, between 40 and 65 gpm [1.4] **BALANCE** CRD SYSTEM FLOW CONTROL, 3-FIC-85-11, and **PLACE** in AUTO or BALANCE.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 6 Page 2 of 2

Event Description: 3C 4kV Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	3-ARP-9-8B, Alarm Response Procedure 4KV UNIT BOARD 3C UNDERVOLTAGE, Window 14 Operator Action: A. **CHECK** Unit in stable condition by checking: • Condensate Pump 3C • Condensate Booster Pump 3C • RCW Pump 3C • CCW Pump 3C • CRD Pump 3A B. **IF** undervoltage has occurred, **THEN** 1. **CLEAR** disagreement lights on breakers. 2. **REDUCE** load as necessary to maintain stable operating conditions. 3. Condenser discharge may need to be throttled for two CCW pump operation. **REFER TO** 3-OI-27, Condenser Circulating Water System. C. **CHECK** Unit Bd 3C for abnormal conditions: relay targets, smoke, burned paint, etc. D. **REFER TO** 0-OI-57A, Switchyard and 4160V AC Electrical System, to re-energize board. E. **REFER TO** appropriate OI for recovery or realignment of equipment.
	NRC	**End of Event 6.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 7 Page 1 of 2

Event Description: Generator Bus Duct High Temperature

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Generator Bus Duct High Temperature alarm, Panel 3-9-7A, Window 32, will annunciate.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause Generator Bus Duct High Temperature annunciation.**
	BOP	Acknowledges and reports the following alarm: • GENERATOR BUS DUCT TEMPERATURE HIGH, 3-9-7A, Window 32
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	3-ARP-9-7A, Alarm Response Procedure GENERATOR BUS DUCT TEMPERATURE HIGH, Window 32 Operator Action: A. **IF** both fans are not operating, **THEN PERFORM** the following: • **START** all available fans GEN BUS DUCT HX FAN A(B) using 3-HS-262-1A1A(2A), on panel 3-9-8 • **ENSURE** system operation in accordance with 3-OI-47, Turbine-Generator System
	BOP	Starts 3B Bus Duct Cooling Fan.
	BOP	B. N/A C. N/A D. **CHECK** the status of alarm GEN BUS DUCT FAN FAILURE, 3-EA-262-161 (3-XA-55-7A, Window 31). E. **DISPATCH** personnel to the main Bus Cooling Fans on elevation 586' to check fans, cooler condition, and temperatures. F. **CHECK** temperature alarm is valid by available means including using a portable temperature monitor device locally at the 14 in-service thermostats. **REFER** to Figure 1.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 7 Page 2 of 2

Event Description: Generator Bus Duct High Temperature

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Turbine Building AUO to monitor Generator Temperatures locally, acknowledge the direction. Once the second Bus Duct Fan is started, wait 2 minutes and report that Main Bus Temperature is 73 C and steady.**
	BOP	**NOTE** With two fans operating at the design airflow of 45,000 scfm, the maximum allowable Bus Duct Cooler discharge temperature is 114ºF. G. N/A H. N/A I. **ENSURE** the system is operating in accordance with 3-OI-47, Turbine-Generator System.
	NRC	**End of Event 7. Request that the Driver insert Event 8, 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 1 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to cause a trip of 3B 480V Shutdown Board and a Loss of Offsite Power.**
	NRC	**EXAMINER NOTES:** **3B 480V Shutdown Board will trip when Event 8 is entered by the driver; the Loss of Offsite Power will occur 3 minutes after Event 8 is entered.** **3A Diesel Generator (EDG) will not tie onto 3EA 4kV Shutdown Board due to an electrical fault. This event is entered upon Simulator setup – no action is required by the driver for this malfunction.** **Event 9, DG Fails to Start, and Event 10, Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start, will occur during Event 8. No action is required by the Driver to insert Events 9 or 10.**
	NRC	**Numerous alarms will be received when 3B 480V Shutdown Board is lost. Only some are covered in this Scenario Guide.**
	BOP	Acknowledges and reports the following alarms: • 480V SHUTDOWN BOARD 3B UV OR XFR, 3-9-8B, Window 30 • 480V REACTOR MOV BOARD 3B OR 3E UV, 3-9-8C, Window 24
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures and 3-AOI-99-1, Loss of Power to one RPS Bus.
	NRC	**Only the immediate action of 3-AOI-99-1, Loss of Power to one RPS Bus is covered below.**
	CREW	3-AOI-99-1, Loss of Power to one RPS Bus 4.1 Immediate Action [1] **STOP** all testing with potential RPS half-scrams or PCIS logic isolation signals.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 2 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 3-ARP-9-8B 480V SHUTDOWN BOARD 3B UV OR XFR, Window 30 Operator Action: A. **CHECK** for indication of 480V Shutdown Board 3B loss: • RWCU Pump 3B shutdown • Fuel Pool Cooling Pump 3B shutdown • 480V Shutdown Board 3B voltage (3-EI-57-30) B. N/A C. N/A D. **IF** manual transfer is **NOT** accomplished, **THEN REFER TO** Tech Spec 3.8.1.
	Driver	**When contacted as the Reactor Building AUO to investigate the reason for the loss of 3B 480V Shutdown Board, acknowledge the direction. Wait 2 minutes and report to the crew that 3B 480V Shutdown Board overcurrent relay flags indicate an overcurrent condition has occurred.**
	NRC	**Due to time constraints, it is not expected that the NUSO reference Technical Specifications for this Event. Technical Specification evaluation should not be used to evaluate the applicant's Technical Specification competency.**
	BOP	Alarm Response Procedure, 3-ARP-9-8C 480V REACTOR MOV BD 3B OR 3E UV, Window 24 A. **CHECK** light indications for loss of any 480V equipment. B. **CHECK** 480V Rx MOV Bd 3B & 3E for abnormal conditions: relay targets, smoke, burned paint, breaker position, etc. **NOTE** If the mechanism resets (hear click and feel resistance when pushing in), this indicates that an overcurrent condition tripped the breaker.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 3 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	C. **IF** Normal or Alternate feeder breaker tripped, **THEN MANUALLY DEPRESS** mechanical trip/reset mechanism on breaker face to reset Bell Alarm lockout device. [NER/C II-B-92-069] D. **IF** undervoltage or transfer has occurred: 1. **REFER TO** Technical Specification Section 3.8.7. 2. **RESET** possible half-scram. **REFER TO** 3-OI-99, Reactor Protection System, E. **REFER TO** 0-OI-57B to re-energize or transfer board. F. **REFER TO** appropriate OI for recovery or realignment of equipment.
	BOP	Reports to the NUSO that Loop II Valves of RHR and Core Spray have lost power. Reports to the NUSO that a half SCRAM has occurred.
	Driver	**If contacted as the Control Bay AUO or Work Control / Outside NUSO to transfer the 3B 480V RMOV Board to Alternate, acknowledge the direction.**
	Crew	Determines that a Loss of Offsite Power has occurred.
	NRC	**Station Blackout conditions start 3 minutes and 30 seconds after Event 8 is inserted.**
	NUSO	Directs the crew to respond in accordance with 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout and 3-AOI-100-1, Reactor SCRAM.
	NRC	**The subsequent actions contained in 3-AOI-100-1, Reactor SCRAM, are too numerous to list here, and only immediate actions are listed below.**
	OATC	3-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 3-HS-99-5A/S3A, REACTOR SCRAM A and 3-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 3-9-5. [2] **PLACE** REACTOR MODE SWITCH, 3-HS-99-5A/S1, in SHUTDOWN. [3] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 4 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the NUSO: • Reactor SCRAM • MODE SWITCH is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Reactor Power level
	OATC	**NOTES** 1) Perform steps 4.1[5.3] and 4.1[5.4] in parallel. 2) Step 4.1[5.9] may be performed before step 4.1[5.8] if Reactor Water Level is slowly lowering. [5] N/A
	OATC	Provides a Reactor SCRAM report to the NUSO. Responds in accordance with 3-AOI-100-1, Reactor SCRAM.
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout **NOTE** Performing this instruction, in conjunction with an earthquake, may require resetting the individual Diesel Generator's 86G Lockout Relay and the Field Breaker (both locally at the Diesel Generator electrical cabinet). [1] **ENSURE** Diesel Generators have started and tied to respective 4kV Shutdown Boards, **THEN DISPATCH** personnel to Diesel Generators.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 5 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	Determines that 3A DG will not tie onto 3EA 4kV Shutdown Board. Determines that 3B DG failed to automatically start.
	NRC	**See Event 9, DG Fails to Auto Start, on page 35 of 38.**
	BOP	[4] **PERFORM** the following to ensure at least one train of Diesel Generator Room Fans are energized: • **CHECK** 480V DSL Aux Board 3EA or 3EB energized
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout 4.2 Subsequent actions **NOTES** 1) The SBO Unit does only Attachment 12 while the other two units perform only the subsequent actions of the AOI. 2) The following is the preferred pump combinations of Unit 1 and Unit 2 RHR pumps used in SDC: 1B/2D, 1A/2C, 1A/2D, 1B/2C. By using these pumps, it ensures that a loss of a Diesel during a station blackout would not result in a loss of Shutdown cooling for both Unit 1 and Unit 2. [1] **IF** ANY EOI entry condition is met, **THEN REFER TO** the appropriate EOI(s).
	NUSO	Enters 3-EOI-1, RPV Control Directs the crew to maintain Reactor Water Level using any available Table L-1 or Table L-2 Injection Subsystem (Normally HPCI and/or RCIC).

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 6 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	NUSO	

Table L-1
Preferred Injection Systems

SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available ◇2 ◇3 ◇5	5C, 20M	1200 psig
HPCI with CST suction if available ◇2 ◇5 ◇7	5D, 20N	1200 psig
CNDS	6A	470 psig
CS ◇2	6D, 6E	330 psig
LPCI ◇2	6B, 6C	320 psig

Table L-2
Alternate Injection Subsystems

SOURCE	APPX	INJ PRESS
EHPM Pump	7L	1210 psig
SLC (test tank)	7B	1450 psig
SLC (boron tank)	7B	1450 psig
CNDS transfer pumps to RHR and CS	7A	110 psig
RHR crosstie to other units	7C	320 psig
Stby coolant	7D	160 psig
RHR drain pumps	7E, 7F	50 psig
PSC head tank pumps	7G	30 psig
RCIC (aux boiler steam) with CST suction if available ◇2 ◇3 ◇5	7H	1200 psig
RCIC manual start	20A	1200 psig
HPCI (aux boiler steam) with CST suction if available ◇2 ◇5 ◇7	7J	780 psig
Fire Protection system	7K	150 psig
FLEX Pump Sys (CILRT/CS)	20D	150 psig
FLEX Pump Sys (Standby Coolant)	20B	150 psig
FLEX Pump Sys (CILRT/CRD)	20C	150 psig

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 7 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	3-EOI-Appendix-5D, Injection System Lineup HPCI [1] N/A [2] N/A [3] N/A [4] **VERIFY** at least one SGT train in operation.
	Driver	**Standby Gas Trains (SGT) should already be running, but if not and if contacted to start an SGT, insert Event 18 to start 'A' SGT.**
	OATC / BOP	**CAUTIONS** 1) Operating HPCI Turbine below 2400 RPM may result in unstable system operation and equipment damage. 2) Operating HPCI Turbine with suction temperatures above 140º F may result in equipment damage. [5] **VERIFY** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection **NOTE** HPCI Auxiliary Oil Pump will NOT start UNTIL 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open. [6] N/A [7] **PLACE** HPCI AUXILIARY OIL PUMP handswitch in START. [8] **PLACE** HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE. [10] **OPEN** 3-FCV-73-44, HPCI PUMP INJECTION VALVE. [11] **OPEN** 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 8 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 3-CKV-73-45, HPCI SYSTEM CHECK VLV, opens by observing 3-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 GPM. **CAUTION** 3-FCV-073-0030, HPCI PUMP MIN FLOW VALVE, automatically opens when system flow is at or below 900 GPM (lowering) only if a system initiation signal is present. Manually opening the Minimum Flow Valve may be required for pump min flow protection. [13] **ADJUST** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO. [16] N/A [17] N/A
	OATC / BOP	3-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 3-XS-71-51A(B) RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and **OPEN** 3-FCV-71-9, RCIC TURB TRIP/THROT VALVE. [4] **ENSURE** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 GPM.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 9 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[5] **OPEN** 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE. [6] **OPEN** 3-FCV-71-39, RCIC PUMP INJECTION VALVE. [7] **OPEN** 3-FCV-71-25, RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 3-HS-71-31A, RCIC VACUUM PUMP, in START. **CAUTIONS** 1) Operating RCIC Turbine below 2100 RPM may result in unstable system operation and equipment damage. 2) High Suppression Chamber pressure may trip RCIC. 3) Operating RCIC Turbine with suction temperatures above 240º F may result in equipment damage. [9] **OPEN** 3-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine. [10] **CHECK** proper RCIC operation by observing the following: A. RCIC turbine speed accelerates above 2100 RPM. B. Flow to RPV controlled automatically at 620 GPM. C. 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 GPM. [11] **ADJUST** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] **IF** BOTH of the following exist: • RCIC Initiation signal is NOT present, **AND** • RCIC flow is below 60 GPM, **THEN ENSURE OPEN** 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 10 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>OATC / BOP</td><td>[13] N/A
[14] N/A</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to maintain Reactor Pressure below 1073 psig using 3-EOI-Appendix-11A, Alternate RPV Pressure Control Systems MSRVs.</td></tr>
<tr><td></td><td></td><td>3-EOI-APPENDIX-11A, Alternate RPV Pressure Control Systems MSRVs

[1] N/A
[2] N/A
[3] OPEN MSRVs using the following sequence to control RPV Pressure as directed by SRO:

<table>
<tr><td>1</td><td>3-PCV-1-179</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>2</td><td>3-PCV-1-180</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>3</td><td>3-PCV-1-4</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>4</td><td>3-PCV-1-31</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>5</td><td>3-PCV-1-23</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>6</td><td>3-PCV-1-42</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>7</td><td>3-PCV-1-30</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>8</td><td>3-PCV-1-19</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>9</td><td>3-PCV-1-5</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>10</td><td>3-PCV-1-41</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>11</td><td>3-PCV-1-22</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>12</td><td>3-PCV-1-18</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>13</td><td>3-PCV-1-34</td><td>MN STM LINE C RELIEF VALVE</td></tr>
</table>

[4] N/A
[5] N/A
[6] N/A</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 11 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	Continuing 0-AOI-57-1A, Loss of Offsite Power (161 and 500KV)/Station Blackout 4.2 Subsequent actions [2] **IF** any Unit is under a Station Blackout, **THEN PERFORM** the following only for the Unit in Station Blackout • Perform Attachment 12 for that Unit
	NRC	**The Station Blackout time starts when power is lost to all 480V Shutdown Boards.**
	CREW	**Critical Task: START:_____ STOP:_____** **With a failure of an Emergency Diesel Generator (EDG) to automatically start and tie on to its respective 4KV Shutdown Board, the crew takes manual action to restore power to 3A 480V Shutdown Board to exit a Station Blackout, or commences a plant cooldown within 20 minutes of the Station Blackout.** **Critical Task Failure Criteria:** **The operating crew fails to restore power to 3A Shutdown Board or commence a plant cooldown within 20 minutes of the Station Blackout on Unit 3.**
	BOP	Determines that Unit 3 is in a Station Blackout condition due to the following plant conditions: • 3B 480V Shutdown Board has tripped and locked out • 3A DG will not tie onto 3EA 4 kV Shutdown Board • 4KV Shutdown Board 3EB is de-energized until 3B DG is manually started
	SRO	Acknowledges that Unit 3 is in a Station Blackout condition. Once the 3B DG is started and tied to 3EB 4 kV Shutdown Board, directs the BOP to transfer 3A 480V Shutdown Board to the alternate power supply to exit Station Blackout.

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 12 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	[3] N/A [4] **CHECK** automatic actions and **PERFORM** any that failed to occur. **NOTE** The following subsequent actions may be performed out of order, depending on plant conditions. [5] **REFER TO** 3-AOI-78-1, Fuel Pool Cooling System Failure for a complete Loss of AC POWER, as necessary. **NOTES** 1) EECW supply valves to the Control Air Compressors and Reactor Building Closed Cooling Water (RBCCW) are air operated. If initial air pressure is low, air compressors may trip on high temperature, until cooling water flow is established. 2) At US discretion, 0-FCV-67-53, EECW NORTH HDR SUPPLY VLV TO AIR COMP, can be placed in the open position with hand switch. The valve will automatically come open once EECW pressure is above setpoint. **REFER TO** 0-OI-67, Emergency Equipment Cooling Water System, for valve operation. 3) The North header supply to Unit 1 RBCCW, the North header supply to Unit 2 RBCCW and the South header supply to Unit 3 RBCCW are normally isolated with a manual valve; therefore, no flow will occur when either 1-FCV-67-50, EECW NORTH HEADER SUPPLY TO RBCCW, 2-FCV-67-50 EECW NORTH HEADER SUPPLY TO RBCCW, or 3-FCV-67-51, EECW SOUTH HEADER SUPPLY TO RBCCW, opens.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 8 Page 13 of 13

Event Description: 3B 480V Shutdown Board Trip / Loss of Offsite Power / Emergency Diesel Generator (DG) Fails to Tie

Time	Position	Applicant's Actions or Behavior
	BOP	6] **WHEN** EECW Header Pressure is restored above the reset pressure setpoint (psig) for the valves listed below, **THEN** Common / Unit 3 0-FCV-67-53: 106 FCV-67-50: 92 FCV-67-51: 113 **RESET** EECW supplies to Control Air Compressors and RBCCW, at Unit 1 Panel 1-LPNL-925-0032 and Unit 2, 3 Panels 2(3)-25-32. **REFER TO** the EECW to the RCW Crossties for Control Air & RBCCW section of 0-OI-67, Emergency Equipment Cooling Water System.
	BOP	[7] **START** Control Air Compressors A, D and G as required and **MONITOR** system pressure. **REFER TO** 0-AOI-32-1, Loss of Control and Service Air Compressors. [7.1] **IF** an Air Compressor trips on high temperature, **THEN** (Otherwise N/A) **NOTIFY** Unit Supervisor for instructions. [8] **REFER TO** 3-AOI-32-2, Loss of Control Air, as necessary. [9] **PLACE** RPS MG Sets A and B in service. **REFER TO** 3-OI-99, Reactor Protection System. [10] **START** the Diesel Driven Fire Pump. **REFER TO** 0-OI-26, Fire Protection System.
	Driver	**If directed as the Outside/Work Control SRO to reset EECW, restore Control Air, restore RPS, or start the Diesel Driven Fire Pump, insert the following events as required:** • **Event 19 – Reset EECW** • **Event 20 – Restore Control Air** • **Event 21 – Restore RPS** • **Event 25 – Start Diesel Driven Fire Pump**
	NRC	**When the crew has restored power to 3A 480V Shutdown Board, has control of Reactor Water Level above the Top of Active Fuel (TAF, (-) 162 inches) using RCIC or HPCI, and has started EECW Pumps, end of scenario.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 9 Page 1 of 1

Event Description: DG Fails to Auto Start

Symptoms/Cues: Event is initiated during Simulator setup. 3EB DG will not automatically start upon the Loss of Offsite Power.

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9 is automatically entered on Simulator setup – no action is required by the Driver to insert Event 9.**
	NRC	**EXAMINER NOTE: Crew action is required to manually start 3B DG – if no action is taken, Unit 3 will remain in Station Blackout.**
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) / Station Blackout 4.1 Immediate Actions [1] **ENSURE** Diesel Generators have started and tied to respective 4kV Shutdown Boards, **THEN DISPATCH** personnel to Diesel Generators. Determines that 3EB DG did not automatically start. Starts 3EB DG and informs the NUSO.
	NRC	**End of Event 9.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-2 Event No.: 10 Page 1 of 1

Event Description: Emergency Equipment Cooling Water (EECW) Pumps Fail to Auto Start

Symptoms/Cues: Event is initiated during Simulator setup. EECW Pumps will not automatically start when the Emergency Diesels start on the Loss of Offsite Power

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 10 is automatically entered on Simulator setup – no action is required by the Driver to insert Event 10.**
	BOP	0-AOI-57-1A, Loss of Offsite Power (161 and 500 KV) / Station Blackout 4.1 Immediate Actions [2] **ENSURE** two EECW Pumps (not using the same EECW strainer) are in service supplying Diesel Generators. [3] **IF** two EECW Pumps (not using the same EECW strainer) are not in service supplying Diesel Generators, **THEN PERFORM** Attachment 9 (Cooling water is required to be established within 8 minutes).
	NRC	**Previously the crew should have aligned either the A1 or the C1 RHRSW Pump to the North EECW header. If the crew elects to place a North EECW Header pump in service, it should be the pump previously aligned.**
	CREW	**Critical Task** **The crew restores EECW Flow with at least 1 EECW Pump within 8 minutes from the time the EDGs start to prevent damage to and a loss of the Emergency Diesel Generators.** **Critical Task Failure Criteria** **The operating crew fails to restore EECW Flow with at least one EECW Pump to the EDGs within 8 minutes following EDG start upon a Loss of Offsite Power.**
	BOP	Starts EECW Pumps to cool the DGs.
	NRC	**End of Event 10.**

Scenario Setup
UNIT 3

IC	28
Exam IC	264

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-2	79	EOI-1	16		
GOI-100-12A	77	APPX-5D	8		
OI-85	99	APPX-5C	6		
ARP-9-8C	21	APPX-11A	2		
OI-67	126				
ARP-9-7C	41				
OI-76	46				
ARP-9-8B	24				
ARP-9-6A	31				
AOI-85-3	14				
ARP-9-7A	30				
AOI-99-1	20				
AOI-57-1A	117				
AOI-100-1	80				

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Danger Tag A3 EECW Pump
Schedule Files(s):	ILT 2304 NRC Scenario 2 UNIT 3.sch
Event Files(s):	ILT 2304 NRC Scenario 2 UNIT 3.evt

UNIT 3 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ON LINE**	**Drywell Leakage (GPM)**	**Protected Equipment**
	230		
	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	**1.89**	
Rx Power		**Floor Drain (GPM)**	
70%		**0.31**	
MWe		**Equipment Drain (GPM)**	
862		**1.58**	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS ☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

INFO ONLY TS 3.7.2, Condition A, due to A3 EECW tagged for scheduled maintenance

LCOs OF 72 HOURS OR LESS

None

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

A3 EECW Pump tagged for scheduled maintenance

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

To support preparations for Reactor SCRAM, continue with power reduction and planned equipment shutdowns in accordance with 3-GOI-100-12A, Section 5.3.1 [5.3], and 3-OI-2, Condensate System, Section 7.1 [3.4].

Support maintenance on A3 EECW Pump

OPERATOR WORK AROUNDS **OWAs – U0-1/U3-0** **Burdens - U0 – 0/U3-0** **Operator Challenges – U0 – 12/U3 - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-3
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 95% Reactor Power, EHPM tagged for motor repair on 2-FCV-007-008, EHPM PUMP INJECTION VALVE, Loop I RHR is tagged for a scheduled work-week outage.

Turnover: Start 'C' SGT on Unit 2 Reactor Zone. Raise Reactor Power to 100%, Support EHPM Injection Valve Maintenance and RHR Loop I outage.

Critical Tasks:

1. When plant conditions indicate that two plant areas are above their Maximum Safe values, without delay, implement Emergency RPV Depressurization.

2. With a loss of Table L-3 Injection Systems, stop Emergency Depressurization before HPCI and RCIC isolate on low Steam Pressure.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 2 Reactor Zone
2.	N/A	R-OATC	Raise Reactor Power Using Core Flow
3.	OG04A	C-BOP C-NUSO	2A Steam Jet Air Ejector Trip
4.	RF RD01A ANN 5A_W2	C-OATC MC-OATC C-NUSO	Control Rod Drive (CRD) Suction Pressure Low
5.	RM08A	TS-NUSO	2-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale
6.	RD04R2015	C-OATC TS-NUSO	Control Rod Drift Out
7.	HS-35-35A PMP-35-36	C-BOP MC-BOP C-NUSO	Stator Cooling Water Pump (SCW) Trip
8.	RD04R3023 RD04R3039 TH23 CU04	M-ALL	Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak
9.	ED10B ED12A CS01A CS01C	C-BOP C-NUSO	Loss of Low Pressure Injection Systems / Limited Emergency Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The Balance of Plant (BOP) Operator will start 'C' SGT in accordance with 0-OI-65, Standby Gas Treatment System, and align it to the Unit 2 Reactor Zone.
2. The crew will raise Reactor Power to 100% using Core Flow in accordance with 2-OI-68, Reactor Recirculation System.
3. 2A Steam Jet Air Ejector (SJAE) will isolate due to a faulty relay, requiring that the crew place the Standby SJAE in service in accordance with 2-OI-66, Off-Gas System, Attachment 6, Standby SJAE System Lineup Hard Card.
4. The crew will observe indications of low Control Rod Drive System Pressure, requiring them to take action in accordance with Alarm Response Procedures and 2-AOI-85-3, CRD System Failure.
5. 2-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor, will fail downscale, requiring the crew to reference the Alarm Response Procedures. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
6. A Control Rod will drift out of the core, requiring the crew to take action in accordance with 2-AOI-85-6, Rod Drift Out, and 2-AOI-85-7, Mispositioned Control Rod. The Control Rod will not settle to position 00 until individually SCRAMMED. The NUSO will address Technical Specifications 3.1.3 and 3.1.5.
7. 2A Stator Cooling Water (SCW) Pump will trip, and 2B SCW Pump will not automatically start, requiring the crew to manually start 2B SCW to prevent a Main Turbine Trip and Reactor SCRAM.
8. Two additional Control Rods will drift out of the Core, requiring the crew to insert a Manual Reactor SCRAM. As a result of the Control Rod Drift transient and the subsequent SCRAM, fuel damage will occur, causing rising radiation levels. An un-isolable Reactor Water Cleanup (RWCU) leak will occur, eventually causing maximum safe radiation levels in various areas of the plant; and as a result, the crew will be required to Emergency Depressurize the Reactor.
9. Various electrical failures will cause a loss of Emergency Operating Instruction (EOI) Table L-3 low pressure injection systems, requiring the NUSO to enter 2-EOI-C1, Alternate Level/Pressure Control, and limit the Emergency Depressurization to ensure Adequate Core Cooling with High Pressure Injection Systems.

The Scenario ends when the crew has Emergency Depressurized the Reactor to 150-250 psig, and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available high-pressure injection system.

Critical Tasks 2

1. **When plant conditions indicate that two plant areas are above their Maximum Safe values, without delay, implement Emergency RPV Depressurization.**

   a. **Safety Significance**
   Places the primary system in the lowest possible energy state, rejects heat to the Suppression Pool in preference to outside the Primary Containment, and reduces driving head and flow of system discharging into the Secondary Containment.

   b. **Cues**
   Procedural Compliance.
   Secondary Containment Area Radiation indications.

   c. **Measured by:**
   Observation - NUSO transitions to Emergency RPV Depressurization, and directs opening at least 6 MSRVs when two or more areas are greater than their Maximum Safe operating values for the same parameter.

   d. **Feedback**
   Reactor Pressure Trend.
   MSRV status indications.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to proceed without delay to initiate Emergency Depressurization from the time it is announced that two Area Radiation Levels exceed Maximum Safe value.

2. **With a loss of Table L-3 Injection Systems, stop Emergency Depressurization before HPCI and RCIC isolate on low Steam Pressure.**

   a. **Safety Significance**
   Full Emergency Depressurization in the case of a loss of Low-Pressure injection subsystems will result in a loss of steam-driven injection systems – therefore Emergency Depressurization must be coordinated with core cooling strategies to prevent a loss of Adequate Core Cooling. A Loss of Adequate Core Cooling would compound radioactivity release.

   b. **Cues**
   Procedural Compliance.
   Loss of Low-Pressure injection systems.

   c. **Measured by:**
   Observation – NUSO transitions to 2-EOI-C-1 and directs limited Emergency Depressurization when Table L-3 injection systems become unavailable.

   d. **Feedback:**
   Reactor Water Level.
   Reactor Pressure.

   e. **Critical Task Failure Criteria:**
   The operating crew reduces Reactor Pressure below the HPCI and RCIC isolation setpoints, causing a loss of Reactor Injection sources.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 1 Page 1 of 3

Event Description: Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 2 Reactor Zone

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**EXAMINER NOTE: The crew has been pre-briefed on Event 1, but may elect to hold a re-focus brief before proceeding.**
	NUSO	Directs the Balance of Plant Operator (BOP) to start 'C' Standby Gas Treatment Train (SGT) and align it to the Unit 2 Reactor Zone.
	BOP	0-OI-65, Standby Gas Treatment System Section 5.2, Standby Gas Treatment System Manual Initiation [1] **ENSURE** the following requirements are satisfied: • SGT Train A(B)(C) in standby readiness. **REFER TO** Section 4.0. • Main Stack Radiation Monitoring in Service. **REFER TO** 2-OI-90, Radiation Monitoring System • As plant conditions permit, Normal ventilation has been in operation for a minimum of 2 hours following painting, fire, or chemical release for the affected zones from which SGT will take suction
	Driver	**If contacted as another Unit, acknowledge any information or direction given.**
	BOP	[2] **REVIEW** the Precautions and Limitations in Section 3.0. [3] **ENSURE** suction path is aligned to SGT System as follows: [3.1] N/A [3.2] **IF** alignment to Reactor Zone Ventilation suction path is desired, **THEN ENSURE OPEN** the following dampers for the desired unit(s) to be aligned. • 2-HS-64-40 and 2-HS-64-41, REACTOR ZONE EXHAUST TO SGTS Dampers, on Panel 2-9-25 [3.3] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 1 Page 2 of 3

Event Description: Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 2 Reactor Zone

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **START** SGT FAN A(B)(C) as follows: [4.1] N/A [4.2] N/A [4.3] **IF** starting SGT FAN C from Panel 2-9-25, **THEN DEPRESS** 0-HS-65-40A/3, SGTS TRAIN C FAN push-button. [5] **CHECK** SGT TRAIN C INLET DAMPER as follows: [5.1] N/A [5.2] N/A [5.3] **IF** SGT FAN C was started, **THEN CHECK OPEN** SGTS TRAIN C INLET DAMPER, 0-HS-65-51A(0-IL-65-51) indicates OPEN on Panel 2-9-25. [6] **CHECK** SGT TRAIN A(B)(C) RH CONTROL HTR as follows: [6.1] N/A [6.2] **IF** SGT FAN C was started, **THEN CHECK** ENERGIZED 0-HS-65-60, SGTS TRAIN C RH CONTROL HTR, on Panel 2-9-25.
	Driver	**If requested, report that SGTS TRAIN C RH CONTROL HEATER is energized as required.**
	BOP	[7] **RECORD** start time and filter bank differential pressure for SGT Train C as follows: [7.1] **N/A** [7.2] **IF** SGT FAN C was started, **THEN RECORD** start time and FILTER BANK DIFFERENTIALPRESSURE, 0-PDI-65-53 on Panel 2-9-25, in the Narrative Log.
	Driver	**If contacted as Unit 2 for Narrative Log readings, report that Unit 2 will make all required entries for SGT C.**
	BOP	[8] **DISPATCH** Operator to the Standby Gas Treatment building as soon as time allows to check for abnormal conditions (i.e. belt tightness, rubbing or vibration noises).

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 1 Page 3 of 3

Event Description: Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 2 Reactor Zone

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Outside Assistant Unit Operator (AUO), acknowledge any information or direction given.**
	BOP	[9] **MONITOR** Standby Gas Treatment Train operation. **REFER TO** Section 6.0.
	NRC	**End of Event 1. Proceed to Event 2, Raise Reactor Power Using Core Flow.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 2 Page 1 of 1

Event Description: Raise Reactor Power Using Core Flow

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew has pre-briefed Event 2, but may elect to conduct a re-focus brief.**
	NUSO	Directs the Operator at the Controls (OATC) to raise Reactor Power to 100% in accordance with 2-OI-68, Reactor Recirculation System and Reactivity Control Plan (RCP) RCP-NRC2304-U2S3.
	NRC	**The crew may elect to raise Reactor Power using any method below.**
	OATC	2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown on 0-TI-248, Unit 2 Core Operating Limits Report (COLR) and 2-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor pump speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pump 2A and/or 2B speed with Recirc Individual Control, **THEN PERFORM** the following: • Raise Recirc Pump 2A using, 2-HS-96-15A(15B), RAISE SLOW (MEDIUM) AND/OR • Raise Recirc Pump 2B using, 2-HS-96-16A(16B), RAISE SLOW (MEDIUM). [2] **WHEN** desired to control Recirc Pump 2A and/or 2B speed with RECIRC MASTER CONTROL, **THEN** **ADJUST** Recirc Pump 2A and/or 2B speed using the following pushbuttons as required: 2-HS-96-31(32), RAISE SLOW (MEDIUM)
	NRC	**End of Event 2. Request that the Driver insert Event 3, 2A Steam Jet Air Ejector Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 3 Page 1 of 4

Event Description: 2A Steam Jet Air Ejector Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Steam Jet Air Ejector will trip, causing lowering vacuum

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to trip 2A Steam Jet Air Ejector.**
	BOP	Acknowledges and reports the following alarms to the SRO: • OG HOLDUP LINE INLET FLOW LOW, 2-9-53, Window 4 • H2 WATER CHEMISTRY ABNORMAL, 2-9-53, Window 10 • H2 WATER CHEMISTRY SHUTDOWN, 2-9-53, Window 20
	BOP	2-ARP-9-53, Alarm Response Procedure OG HOLDUP LINE INLET FLOW LOW, Window 4 A. **CHECK** holdup line flow on: 1. 2-XR-66-103, OFFGAS H2 CONT/OFFGAS TO 6 HR FLOW/ANALYZER (CH 3-Narrow Range), Panel 2-9-53 2. 2-FR-66-20, OFFGAS FLOW TO HOLDUP VOLUME, Panel 2-9-8 3. Integrated Computer System, computer point 66-111A (D419) B. **CHECK OPEN** in-service Recombiner Inlet And Outlet Valves at Panel 2-9-53. C. **CHECK OPEN**, 2-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE. D. **CHECK** that SJAE auto isolation has **NOT** occurred. E. **DISPATCH** PERSONNEL to 2-LPNL-925-0095 to verify the alarm, 2-FS-066-0111A. F. N/A
	BOP	Determines the cause for the above alarms to be a trip of 2A Steam Jet Air Ejector (SJAE), informs the Nuclear Unit Senior Operator (NUSO).
	NUSO	Directs the BOP to respond in accordance with Alarm Response Procedures and to place 2B SJAE in service in accordance with 2-OI-66, Standby SJAE System Lineup Hard Card.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 3 Page 2 of 4

Event Description: 2A Steam Jet Air Ejector Trip

Time	Position	Applicant's Actions or Behavior
	BOP	2-OI-66, Off-Gas System Standby SJAE System Lineup Hard Card Section 3.0, OPERATOR ACTIONS FOR SJAE 2B **NOTES** Radiation Protection should be notified prior to placing a SJAE in service. If time does not permit this due to plant conditions, then notification should be made when possible. 2-HS-001-0375, SJAE TRAIN PERMISSIVE (located on 2-LPNL-925-0105, U2 TB, el 586') should normally be in the position of the standby SJAE. If problems are encountered while placing a SJAE in service and time permits, operate this switch as required during the performance of this section.
	Driver	**If contacted as Radiation Protection, acknowledge any information or direction given.**
	BOP	[1] **ENSURE RESET** Off-Gas isolation using 2-HS-90-155, OFFGAS OUTLET/DRAIN ISOLATION VALVES. [2] **ENSURE OPEN** the following valves: • 2-HS-66-15, SJAE 2B INLET VALVE • 2-HS-1-156A, STEAM TO SJAE 2B [3] **ENSURE** in **AUTO/OPEN 2**-HS-66-18, SJAE 2B OG OUTLET VALVE. [4] **PLACE 2**-HS-1-152, SJAE 2B PRESS CONTROLLER, in **CLOSE** and then in **OPEN**. [5] **ENSURE OPEN** the following valves (red light illuminated): • 2-PCV-1-153/167, STEAM TO SJAE 2B STAGES 1,2, AND 3 • 2-FCV-1-152, SJAE 2B INTMD CONDENSER DRAIN
	BOP	When the standby SJAE is placed in service, acknowledges and reports the following alarms: • OFFGAS HOLDUP LINE INLET FLOW HIGH, 2-9-53, Window 14 • HIGH OFFGAS % H2 TRAIN A, 2-9-53, Window 3 • HIGH OFFGAS % H2 TRAIN B, 2-9-53, Window 13

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 3 Page 3 of 4

Event Description: 2A Steam Jet Air Ejector Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew may enter 2-AOI-66-1, Off Gas H2 High.**
	BOP	**2-AOI-66-1, Off Gas H2 High** 4.1 Immediate Actions – None 4.2 Subsequent Actions [1] **PLACE** both OFFGAS TRAIN A(B) AUTO CHANNEL CHECK / BYPASS control switches, 2-HS-066-1007 and 1008, on OFFGAS SAMPLE PANEL, 2-LPNL-925-0588, in BYPASS to assure continuous availability of Hydrogen monitoring.
	Driver	**If contacted as the Turbine Building AUO to perform Step [1], acknowledge the direction and report that Step [1] is complete.**
	BOP	[2] N/A [3] **VERIFY** proper operation of in service SJAE. [3.1] N/A [4] **IF** Hydrogen concentration is ≥ 4%, **THEN REFER TO** TRM 3.7.2.
	NRC	**EXAMINER NOTE: TRM requirements are on page 8 of 39.**
	OATC	Monitors Main Condenser Vacuum.
	BOP	(2-OI-66, continued): [6] **MONITOR** Hotwell Pressure as indicated on recorder 2-XR-2-26 (Point 3), CONDENSATE, on Panel 2-9-6. [7] **FOR** the SJAE **NOT** being placed in service, **ENSURE CLOSED** the following valves: • 2-HS-66-14, SJAE 2A OG OUTLET VALVE • 2-HS-1-150, SJAE 2A PRESSURE CONTROLLER • 3-HS-1-155A, STEAM TO SJAE 2A [8] **WHEN** Condenser Vacuum has stabilized, **THEN PROCEED** to 2-OI-66, Off-Gas System, "Placing Standby SJAE in Operation" AND **PERFORM** the applicable steps to establish acceptable SJAE parameters to obtain stable and live pressure indications.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 3 Page 4 of 4

Event Description: 2A Steam Jet Air Ejector Trip

Time	Position	Applicant's Actions or Behavior	
	BOP	Informs the NUSO that 2B SJAE is in service. Monitors Main Condenser Vacuum.	
	NUSO	Technical Requirement 3.7.2 – Airborne Effluents LCO 3.7.2 Whenever the SJAE is in service, the concentration of Hydrogen in the Offgas downstream of the Recombiners shall be limited to ≤4% by volume. Applicability: During Main Condenser Offgas Treatment System operation. ----------NOTE---------- TRM LCO 3.0.3 is not applicable. ---------- Condition: A. With the concentration of Hydrogen >4% by volume.	
	NUSO	**REQUIRED ACTION:** A. Restore the concentration to within the limit.	**COMPLETION TIME:** A. 48 hours
	Driver	**If the crew directs the Turbine Building AUO perform the following steps, acknowledge the direction:** • **2-OI-4, Hydrogen Water Chemistry System, Section 8.9, Restore H_2 Water Chemistry. Report to the Control Room for a brief on restoring H_2 Water Chemistry, acknowledge the direction.** • **2-OI-66, Off-Gas System, Section 8.4, Placing Standby SJAE in Operation** **If the crew notifies Radiation Protection that an RPHP exists, acknowledge the report. If requested provide a name for the NOMS log.**	
	Driver	**If contacted as Work Control / Outside NUSO, or Maintenance, acknowledge any information or direction given.**	
	NRC	**End of Event 3. Request that the Driver insert Event 4, Control Rod Drive (CRD) Suction Pressure Low.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 4 Page 1 of 3

Event Description: Control Rod Drive (CRD) Suction Pressure Low

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2A CRD Pump parameters will indicate low Suction Pressure

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause low Suction Pressure on 2A CRD Pump.**
	OATC	Acknowledges and reports the following alarm: • CRD PUMP A SUCTION PRESSURE LOW, 2-9-5A, Window 2
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure and applicable Abnormal Operating Instruction.
	OATC	2-ARP-9-5A, Alarm Response Procedure CRD PUMP A SUCTION PRESSURE LOW, Window 2 A. **CHECK** alarm by: 1. Control switch lights (green, pump out of service and white, disagreement light on). 2. Checking 2-FI-85-11A, CRD SYSTEM FLOW and 2-PI-85-13A, CRD ACCUM CHARGING WATER HEADER PRESSURE lowering (Panel 2-9-5).
	NRC	**EXAMINER NOTE: The CRD Pump Low Suction Pressure trip is overridden. The OATC must manually trip 2A CRD Pump.**
	OATC	B. **IF** alarm is valid, **THEN REFER TO** 2-AOI-85-3, CRD System Failure. C. **DISPATCH** personnel to CHECK suction path.
	Driver	**If contacted as the Reactor Building AUO, acknowledge any information or direction given.**
	OATC	2-AOI-85-3, CRD System Failure 4.1 Immediate Actions [1] N/A [2] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 4 Page 2 of 3

Event Description: Control Rod Drive (CRD) Suction Pressure Low

Time	Position	Applicant's Actions or Behavior
	OATC	4.1 Subsequent Actions **NOTES** 1) Operation with loss of CRD system should be minimized since CRD Mechanism Cooling Water, Reactor Recirc Pump Seal Purge Flow, 2B Reactor Water Cleanup Recirc Pumps Purge Water Flow, and CRD HCU Accumulator Charging Water will be lost. Technical Specification Section 3.1.3 and 3.1.5 should be consulted. 2) Do NOT attempt to start a CRD Pump which has tripped for unknown reasons before verifying locally that the pump, motor, and breaker are in operating condition. 3) Steps 4.2[1] and 4.2[2] should be performed concurrently. [1] N/A [2] N/A [3] **IF** operating CRD Pump has NOT tripped, **THEN PERFORM** the following:(**REFER TO** 2-OI-85, Control Rod Drive System): [3.1] **ENSURE** FULL OPEN 2-FCV-85-56, CRD PUMPS 2A & 2B UNIT 2 SUCTION. [3.2] **IF** running CRD Pump Suction Pressure is approaching 18-inches HG absolute, **THEN ALTERNATE** CRD Pumps. [3.3] **IF** in-service Filter Differential Pressure is greater than 20.5 psid, **THEN ALTERNATE** CRD Drive Water Filters. [3.4] N/A [3.5] N/A [3.6] N/A [4] N/A [5] **NOTIFY** Reactor Engineer of CRD System failure.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 4 Page 3 of 3

Event Description: Control Rod Drive (CRD) Suction Pressure Low

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Reactor Engineer, Work Control / Outside NUSO, or Electrical / Mechanical Maintenance, acknowledge any information or direction given.**
	NRC	**End of Event 4. Request that the Driver insert Event 5, 2-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 5 Page 1 of 3

Event Description: 2-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to fail 2-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor downscale.**
	BOP	Acknowledges and reports the following alarm: • REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE, 2-9-3A, Window 28
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	2-ARP-9-3A, Alarm Response Procedure Operator Action: A. **CHECK** alarm condition on the following: 1. 2-RR-90-144, REACTOR & REFUEL ZONE EXHAUST RADIATION recorder, on Panel 3-9-2. 2. 2-RM-90-140/142, RX & REFUEL ZONE EXH CH A RAD MON RTMR, on Panel 2-9-10. 3. 2-RM-90-141/143, RX & REFUEL ZONE EXH CH B RAD MON RTMR, on Panel 2-9-10. B. **CHECK** power supply to the Radiation Monitors. C. **CHECK** other Radiation Monitors for Radiation Levels below release limits. D. **NOTIFY** Unit SRO, Unit 1 and Unit 3 E. **REFER** TO Tech Spec 3.3.6.2 and 3.3.7.1.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 5 Page 2 of 3

Event Description: 2-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.3.6.2 – Secondary Containment Isolation Instrumentation LCO 3.3.6.2 The Secondary Containment isolation instrumentation for each Function in Table 3.3.6.2-1 shall be OPERABLE. Applicability: According to Table 3.3.6.2-1. ----------NOTE---------- Separate Condition entry is allowed for each channel. ---------- Condition: A. One or more channels inoperable.	
	NUSO	**REQUIRED ACTION:** A. Place channel in trip.	**COMPLETION TIME:** A. 12 hours for Functions 1 and 2 AND 24 hours for Functions other than Functions 1 and 2
	NUSO	Technical Specification 3.3.7.1 – Control Room Emergency Ventilation (CREV) System Instrumentation LCO 3.3.7.1 The CREV System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE. Applicability: According to Table 3.3.7.1-1. ----------NOTE---------- Separate Condition entry is allowed for each channel. ---------- Condition: A. One or more channels inoperable.	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 5 Page 3 of 3

Event Description: 2-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** A. Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	**COMPLETION TIME:** A. Immediately
	NUSO	**REQUIRED ACTION:** C.1 Declare associated CREV subsystem inoperable. AND C.2 Place channel in trip.	**COMPLETION TIME:** C.1 – 1 hour from discovery of loss of CREV initiation capability C.2 – 24 hours
	NRC	**End of Event 5. Request that the Driver insert Event 6, Control Rod Drift Out.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 1 of 6

Event Description: Control Rod Drift Out

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Control Rod 22-15 will drift out

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause Control Rod 22-15 to drift out.**
	OATC	Acknowledges and reports the following alarms: • CONTROL ROD DRIFT, 2-9-5A, Window 28
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure and applicable Abnormal Operating Instructions.
	OATC	2-ARP-9-5A, Alarm Response Procedure CONTROL ROD DRIFT, Window 28 Operator Action: A. **DETERMINE** which rod is drifting from Full Core Display. B. **IF** no Control Rod motion is observed, **THEN RESET** Rod Drift as follows: 1. **PLACE** ROD DRIFT ALARM TEST switch, 2-HS-85-3A-S7, in RESET and **RELEASE**. 2. **RESET** the annunciator. C. N/A D. **IF** rod drifting out, **THEN REFER TO** 2-AOI-85-6, Rod Drift Out and 2-AOI-85-7, Mispositioned Control Rod. E. **REFER TO** Tech Spec Section 3.1.3, 3.10.8.
	OATC	2-AOI-85-6, Rod Drift Out 4.1 Immediate Actions [1] N/A 4.2 Subsequent Actions [1] N/A [2] **IF** a Control Rod is moving from its intended position without operator actions, **THEN SELECT** the drifting Control Rod and **INSERT** to the FULL IN (00) position.

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 2 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	[3] N/A [4] N/A [5] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. [6] N/A **NOTES** 1) A Control Rod that will NOT latch could be indicative of either of the following: • A failed open withdraw Directional Control Valve (2-FCV-85-40C) • Foreign material from the reactor entering and lodging in the CRD Collet area wedging the collet open • Collet retainer tube cracking failure 2) If Control Rod does NOT latch at position 00, Step 4.2[8] may be skipped to avoid Control Rod drift to full out position when removed from service. [7] IF the Control Rod will not latch into position "00" and continues to demonstrate occurrences of inadvertent withdrawal, **THEN PERFORM** the following: [7.1] **MAINTAIN** an insert signal on the Control Rod to keep the Control Rod at position 00 or full in overtravel. [7.2] **PLACE** the appropriate Rod Scram Switch, on 2-PNLA-009-0016 in the Auxiliary Instrument Room, in the DOWN position.
	Driver	**If contacted as Work Control / Outside NUSO to SCRAM Control Rod 22-15, acknowledge the direction and any information given. When directed to go to the Aux Instrument Room, wait 1 minute and report that you are standing by to SCRAM Control Rod 22-15. When directed to SCRAM the Control Rod, insert Event 16 and report that Control Rod 22-15 is SCRAMMED.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 3 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	[7.3] **RELEASE** the control switch placing an insert signal on the Control Rod. [7.4] **PLACE** the appropriate Rod Scram Switch, on 2-PNLA-009-0016 in the Auxiliary Instrument Room, in the UP position.
	Driver	**When directed, insert Event 26 and report that the Individual Control Rod SCRAM switch is in the UP position.**
	OATC	[7.5] N/A [8] **IF** the Control Rod is latched into position "00", **THEN REMOVE** associated HCU from service per 2-OI-85, Control Rod Drive System. [9] **DECLARE** Control Rod INOP per Technical Specification 3.1.3. [10] **REFER** TO 2-AOI-85-7 Mispositioned Control Rod. [11] **INITIATE** Condition Report/Work Order. [12] **NOTIFY** Reactor Engineer to perform the following for current condition: • **EVALUATE** condition of Core to assure no resultant Fuel Damage has occurred • **EVALUATION** of impact on Thermal Limits and PCIOMOR restraints • **DETERMINE** if other Control Rods need to be repositioned in order to safely restore Core symmetry to prevent local Fuel Damage
	Driver	**If contacted as the Reactor Engineer or System Engineer, acknowledge any information or direction given.**
	OATC	[13] **NOTIFY** System Engineering to PERFORM 0-TI-20, Control Rod Drive System Testing and Troubleshooting to determine problem with faulty Control Rod. [14] **IF** a manual SCRAM was not inserted and Reactor Startup or Shutdown is not in progress, **THEN ENSURE** 2-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. [15] N/A [16] N/A [17] **NOTIFY** Reactor Engineer to EVALUATE impact on preconditioning envelope, prior to returning to normal power operation.
	Driver	**If contacted as the Reactor Engineer, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 4 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	2-AOI-85-7, Mispositioned Control Rod 4.1 Immediate Actions – None 4.2 Subsequent Actions [1] **STOP** all intentional Control Rod Movement. **NOTE** Notifications per step 4.2[2] should be performed as soon as practical, but should NOT delay performance of steps 4.2[3] or 4.2[4]. Step 4.2[2] may be completed after performance of steps 4.2[3] and 4.2[4]. [2] IF Control Rod is determined to be mispositioned, THEN NOTIFY the following: • Reactor Engineer (RE) • Shift Technical Advisor (STA) • Unit SRO • Shift Manager (SM) • Operations Superintendent
	Driver	**If contacted by the Reactor Engineer, Shift Technical Advisor, Shift Manager, or Operations Superintendent, acknowledge any information given.**
	OATC	[3] **IF** the wrong Control Rod is moved OR a Control Rod that was not being moved is found to be mispositioned, **THEN PERFORM** the following: [3.1] **INSERT** the mispositioned rod to "00". [3.2] **IF** a Reactor Startup or Shutdown is not in progress, **THEN ENSURE** 2-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. [3.3] N/A [4] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 5 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	[5] **CHECK** the following radiation recorders for a rise in activity to determine if any fuel damage occurred: • 2-RR-90-135, MAIN STEAM LINE RADIATION (Panel 2-9-2) • 2-RR-90-266, OFFGAS RADIATION (Panel 2-9-2) [6] N/A [7] [8] N/A [9] **NOTIFY** Reactor Engineer to perform the following when time permits: • **EVALUATE** the possible consequences • **DOCUMENT** in Reactor Engineer log [10] N/A
	Driver	**If contacted as the Reactor Engineer, acknowledge any information or direction given.**
	NUSO	Technical Specification 3.1.3, Control Rod OPERABILITY LCO 3.1.3 Each Control Rod shall be OPERABLE Applicability: Modes 1 and 2 ---------- NOTE: Separate Condition entry is allowed for each Control Rod. ---------- **CONDITION:** C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 6 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** C.1 ----------NOTE---------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation. --- Fully insert INOPERABLE Control Rod AND C.2 Disarm the associated CRD.	**COMPLETION TIME:** C.1 – 3 hours C.2 – 4 hours
	NUSO	Technical Specification 3.1.5, Control Rod OPERABILITY LCO 3.1.5 Each Control Rod SCRAM Accumulator shall be OPERABLE Applicability: Modes 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod SCRAM Accumulator. **CONDITION:** A. One or more Control Rod SCRAM Accum INOPERABLE with Reactor Steam Dome Pressure ≥ 900 psig.	
	NUSO	**REQUIRED ACTION:** A.2 Declare the associated Control Rod INOPERABLE	**COMPLETION TIME:** 8 hours
	NRC	**Tech Spec 3.10.8, Shutdown Margin (SDM) Test – Refueling is not applicable to current plant conditions.**	
	NRC	**End of Event 6. Request that the Driver insert Event 7, Stator Cooling Water Pump Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 7 Page 1 of 2

Event Description: Stator Cooling Water Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2A Stator Cooling Water Pump will trip, and 2B Stator Cooling Water Pump will not automatically start.

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: The Unit 2 Main Turbine will trip and the Reactor will SCRAM in approximately 1 minute from the loss of 2A Stator Cooling Water Pump if the crew does not manually start the standby pump (it will fail to automatically start).**
	BOP	Acknowledges and reports the following alarms: • GEN STATOR COOLANT SYS ABNORMAL, 2-9-7A, Window 22 TURBINE TRIP TIMER INITIATED, 2-9-8A, Window 1
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures.
	BOP	2-ARP-9-7A, Alarm Response Procedure GEN STATOR COOLANT SYS ABNORMAL, Window 22 **NOTE** The control room alarm typer can be used to confirm this alarm. A. **IF** while performing the action of this ARP 2-XA-55-9-8A window 1 alarms, **THEN** 1. **ENSURE** all available Stator Cooling Water Pumps running. 2. N/A 3. N/A B. **ENSURE** a Stator Cooling Water Pump is running and **CHECK** stator temperature recorder, 2-TR-57-59, Panel 2-9-8.
	BOP	Starts 2B SCW Pump. Verifies SCW has been restored and that TURBINE TRIP TIMER INITIATED, 2-9-8A, Window 1, can be reset.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 7 Page 2 of 2

Event Description: Stator Cooling Water Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	C. **CHECK** alarm and MONITOR Stator Coolant System parameters using ICS "STATCWA" or 'MAINGEN". D. **REQUEST** personnel to **REFER TO** Local Panel ARP for correct alarm response actions to be taken. E. N/A
	Driver	**If contacted as the Turbine Building AUO, Work Control NUSO, or Electrical Maintenance, acknowledge any report or direction given.** **After 2 minutes, report as the Turbine Building AUO that 2A Stator Cooling Water Pump is hot to the touch.**
	BOP	2-ARP-9-8A, Alarm Response Procedure TURBINE TRIP TIMER INITIATED, Window 1 **NOTE** The control room alarm typer can be used to confirm this alarm. Operator Action: A. **CHECK** Stator Cooling Water Flow and Temperature and Generator Stator temperatures using ICS. B. **ENSURE** all available Stator Cooling Water Pumps running. **NOTE** The full capacity of the Turbine Bypass valves with all nine valves open is 21.3% reactor power. To determine the capacity of the bypass valves, subtract 2.37% for each out of service bypass valve from the 21.3%. (Example, one bypass valve out of service, [21.3% - 2.37% = 18.93%], therefore, the capacity of the bypass valves with one bypass valve out of service is 18.93%.)
	BOP	C. N/A D. **DISPATCH** personnel to Stator Coolant Unit to investigate.
	NRC	**End of Event 7. Request that that driver insert Event 8, Multiple Control Rod Drift/Fuel Damage/Reactor Water Cleanup (RWCU) Leak**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Two Control Rods will drift in – when a Reactor SCRAM is inserted, A RWCU leak and Fuel Damage will occur

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to cause Control Rods 30-23 and 30-29 to drift out.**
	NRC	**EXAMINER NOTES:** • **Control Rods 30-23 and 30-29 will drift out** • **No action is required by the Driver to insert the RWCU Leak or the Fuel Damage – those malfunctions are inserted when the Reactor MODE SWITCH is taken to the SHUTDOWN position**
	OATC	Acknowledges and reports the following alarm: • CONTROL ROD DRIFT, 2-9-5A, Window 28
	NRC	**Only the Immediate Actions of 2-AOI-85-6, Rod Drift Out, are shown below, as the Subsequent Actions are not applicable once a Reactor SCRAM is inserted.**
	OATC	2-AOI-85-6, Rod Drift Out 4.1 Immediate Actions [1] **IF** multiple Control Rod drifts are identified, **THEN MANUALLY SCRAM** the Reactor and enter 2-AOI-100-1, Reactor SCRAM.
	NUSO	Directs the OATC to enter 2-AOI-100-1, Reactor SCRAM.
	NRC	**When the MODE SWITCH is placed in SHUTDOWN, all Condensate Pumps will trip. See Attachment 1 on page 31 of 39 for crew actions.** **Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 2 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	OATC	2-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A, and 2-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 2-9-5. [2] **PLACE** 2-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] N/A [4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the UNIT SRO: • Reactor SCRAM • MODE SWITCH is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power Level [5] N/A
	BOP	Acknowledges and reports the following alarms: • RWCU LEAK DETECTION TEMP HIGH, 2-9-3D, Window 17 • RWCU ISOL LOGIC CHANNEL A TEMP HIGH, 2-9-5B, Window 32 • RWCU ISOL LOGIC CHANNEL B TEMP HIGH, 2-9-5B, Window 33
	NUSO	Directs the BOP to respond in accordance with the applicable Alarm Response Procedures.
	BOP	2-ARP-9-3D, Alarm Response Procedure RWCU LEAK DETECTION TEMP HIGH, 2-9-3D, Window 17 Operator Action: [A] **IF** this alarm is received in conjunction with RWCU ISOL LOGIC CHANNEL A TEMP HIGH [2-XA-55-5B, Window 32] and RWCU ISOL LOGIC CHANNEL B TEMP HIGH [2-XA-55-5B, Window 33], **THEN EXIT** this procedure and **GO TO** 2-ARP-9-5B. Otherwise, **CONTINUE** in this procedure.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 3 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NRC	**RWCU ISOL LOGIC CHANNEL B TEMP HIGH, 2-9-5B, Window 33 is effectively the same actions as those for Window 32. Therefore, only Window 32 actions are shown below.**
	BOP	2-ARP-9-5B, Alarm Response Procedure RWCU ISOL LOGIC CHANNEL A TEMP HIGH, 2-9-5B, Window 32 Operator Action: A. **CHECK** alarm by checking: 1. ATUs on Panel 2-9-83 and 2-9-85. 2. RWCU LEAK DETECTION TEMP HIGH annunciator in alarm (2-XA-55-3D, Window 17). 3. Area temperature indication on LEAK DETECTION SYSTEM TEMPERATURE, 2-TR-69-29, on Panel 2-9-22. 4. ARMs 2-RR-90-1, 2-RR-90-50B on Panel 2-9-2 and 0-MON-90-361 on Panel 1-9-2. 5. ICS 'HPTURB' & 'RWCU' mimics for the 834 and 835 temperature loops. B. **IF** leak is suspected, **THEN MANUALLY ISOLATE** RWCU C. **IF** RWCU automatically isolates, **REFER TO** 2-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation.
	Driver	**If contacted as Unit 1 Operator to check Area Radiation Monitors or Radiation Recorders, acknowledge the request.**
	BOP	Determines that RWCU failed to automatically isolate, and attempts to manually isolate RWCU. Informs the NUSO that RWCU will not isolate.
		D. **IF** TIS-69-835A(C) indicates greater than 131 ºF, **THEN ENTER** 2-EOI-3, Secondary Containment Control. E. **REFER TO** Tech. Spec. Table 3.3.6.1-1. F. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	Driver	**If contacted as Work Control / Outside NUSO, acknowledge any information or direction given.**

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 4 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

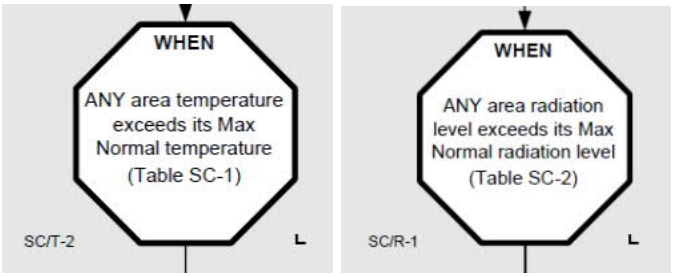
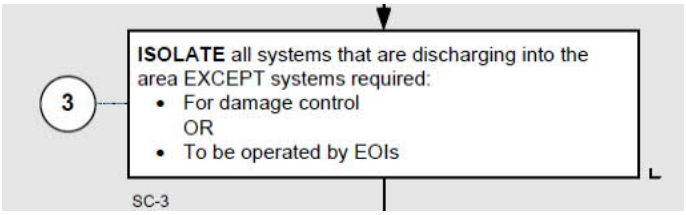
<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Driver</td><td>If contacted as the Control Bay AUO to investigate or check temperatures, acknowledge any information or direction given.</td></tr>
<tr><td></td><td>NRC</td><td>It is not intended that the SRO reference Technical Specifications for this event.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-EOI-3, Secondary Containment Control

SC-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Reactor Zone Ventilation Exhaust Radiation Level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation Exhaust Radiation Level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Reactor Zone Ventilation is isolated
AND
Reactor Zone Ventilation Exhaust Radiation level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation is isolated
AND
Refuel Zone Ventilation Exhaust Radiation Level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 5 of 11

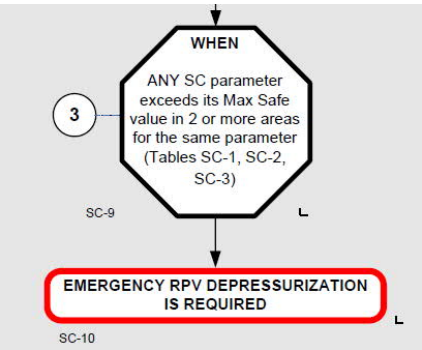
Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-2 **IF** Reactor Zone or Refuel Zone ventilation exhaust radiation level is below 72 mR/hr **THEN** operate available Reactor Zone or Refuel Zone ventilation
	NUSO	
	NUSO	
	NRC	**EXAMINER NOTE: The crew will NOT be able to anticipate Emergency Depressurization, as the Turbine Bypass valves are over-ridden during Simulator Setup.**
	NUSO	Directs the BOP to monitor Secondary Containment parameters (Temperature and Radiation) for approach to Maximum Safe Values.
	CREW	**Critical Task: When plant conditions indicate that two plant areas are above their Maximum Safe values, without delay, implement Emergency RPV Depressurization.** **Critical Task Failure Criteria: The operating crew fails to proceed without delay to initiate Emergency Depressurization from the time it is announced that two Area Radiation Levels exceed Maximum Safe value.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 6 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	Table SC-2 Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 7 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>Monitors Secondary Containment parameters and informs the NUSO when two (2) Radiation Levels are at Maximum Safe Value.</td></tr>
<tr><td></td><td>NUSO</td><td>Updates the crew that Emergency Depressurization is required. Enters 2-EOI-1, Emergency RPV Depressurization.</td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-1, Emergency Depressurization

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 feet</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-7
<table>
<tr><td>IF Drywell Pressure is above 2.45 psig THEN PREVENT injection from ONLY those Core Spray and LPCI Pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr>
</table>
RC/P-8
<table>
<tr><td>EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Level is above 5.5 ft THEN OPEN 6 MSRVs (ADS Valves preferred)
➢ OK to exceed 100°F/hr cooldown rate</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to open 6 ADS Valves.</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 8 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization, is automatically entered when the first SRV is opened for Emergency Depressurization. See Event 9 on page 34 of 39.**
	BOP	Opens 6 ADS Valves.
	BOP	Determines that 2B 480V Shutdown Board has tripped. See Event 9 on page 34 of 39.
	NRC	**End of Event 8. See Attachment 1 below for action for Reactor Water Level Control following the loss of Condensate Pumps. Otherwise, continue to Event 9 on page 34 of 39.**

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 9 of 11

Attachment 1
Page 1

Time	Position	Applicant's Actions or Behavior
	NUSO	When Condensate Pumps trip, directs the BOP to maintain Reactor Water Level using HPCI or RCIC.
	BOP	2-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 2-XS-71-51A(B), RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and OPEN 2-FCV-71-9, RCIC TURB TRIP/THROT VLV. [4] **ENSURE** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 gpm. [5] **OPEN** 2-FCV-71-34 RCIC PUMP MIN FLOW VALVE. [6] **OPEN** 2-FCV-71-39 RCIC PUMP INJECTION VALVE. [7] **OPEN** 2-FCV-71-25 RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 2-HS-71-31A, RCIC VACUUM PUMP, in START. [9] **OPEN** 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine. [10] **CHECK** proper RCIC operation by observing the following: A. RCIC turbine speed accelerates above 2100 rpm. B. Flow to RPV controlled automatically at 620 gpm. C. 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 gpm. [11] **ADJUST** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] N/A [13] N/A [14] N/A

Time	Position	Applicant's Actions or Behavior
	BOP	2-AOI-Appendix-5D, Injection System Lineup HPCI [1] N/A [2] N/A [3] N/A [4] **VERIFY** at least one SGTS Train in operation.
	Driver	**Standby Gas should be running following the Reactor SCRAM (based on the Reactor Water Level transient), but if not and if contacted to start a train of Standby Gas, insert Event 29.**
	BOP	[5] **VERIFY** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL Controller is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection [6] N/A [7] **PLACE** HPCI AUXILIARY OIL PUMP handswitch in START. [8] **PLACE** HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE. [10] **OPEN** 2-FCV-73-44, HPCI PUMP INJECTION VALVE. [11] **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine. [12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 2-CKV-73-45, HPCI SYSTEM CHECK VALVE, opens by observing 2-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 gpm.

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 11 of 11

Attachment 1
Page 3

Time	Position	Applicant's Actions or Behavior
	BOP	[13] **ADJUST** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL Controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO. [16] N/A [17] N/A
	NRC	**End of Event 8, Attachment 1.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 9 Page 1 of 4

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

Symptoms/Cues: Event is initiated automatically entered based on MODE SWITCH and ADS Valve Position. 2B 480V Shutdown Board and both Loop I Core Spray Pumps will trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Event 9 malfunctions are inserted as follows – no action is required by the Driver to insert these malfunctions:
All Condensate Pumps trip when the MODE SWITCH is placed in SHUTDOWN
2B 480V Shutdown Board Trips when the first ADS Valve is opened
2A and 2C Core Spray Pump Trips are inserted on Simulator setup</td></tr>
<tr><td></td><td>NRC</td><td>Numerous alarms will be received during this event, and not all are covered in this guide.</td></tr>
<tr><td></td><td>BOP</td><td>When Emergency Depressurization commences, determines that 2B 480V Shutdown Board has tripped.
Reports a loss of Loop I LPCI Injection Valve Power to the NUSO.</td></tr>
<tr><td></td><td>BOP</td><td>When an attempt to is made to start Loop I Core Spray Pumps, they will trip.</td></tr>
<tr><td></td><td>NUSO</td><td>In accordance with 2-AOI-1, Emergency Depressurization override RC/P-6, determines that Table L-3 subsystems alone cannot assure Adequate Core Cooling.

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure Adequate Core Cooling</td><td>TERMINATE the RPV Depressurization
AND
C1 Alternate Level/Pressure Control</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 feet</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 9 Page 2 of 4

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to terminate Emergency Depressurization.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-C-1, Alternate Level/Pressure Control

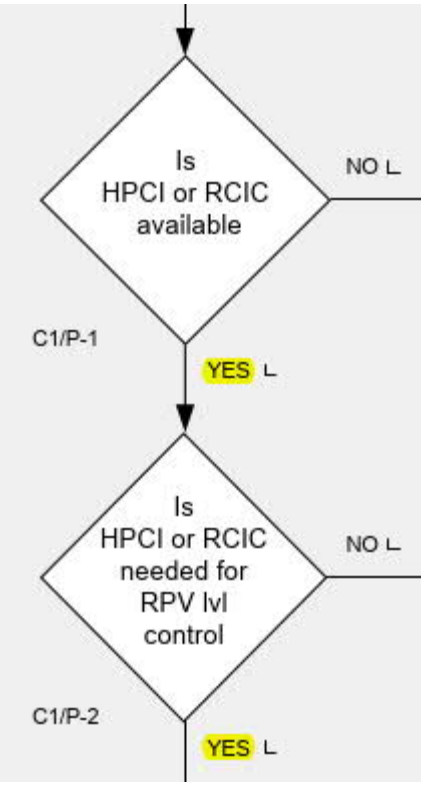
C1-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>It is determined that the Reactor will NOT remain subcritical without boron under all conditions</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>One or more available Injection Subsystems (Table L-3) can assure Adequate Core Cooling (Table M-3)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is determined that Core Damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>

</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 9 Page 3 of 4

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	
	CREW	**Critical Task: With a loss of Table L-3 Injection Systems, stop Emergency Depressurization before HPCI and RCIC isolate on low Steam Pressure.** **Critical Task Failure Criteria: The operating crew reduces Reactor Pressure below the HPCI and RCIC isolation setpoints, causing a loss of Reactor Injection sources.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 9

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
C1/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>➢ HPCI and RCIC are NOT available for injection
OR
➢ HPCI and RCIC are no longer needed for injection</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>All criteria that will assure Adequate Core Cooling is met (Table L-4)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

AND

RPV depressurization to 150-250 psig will NOT result in loss of injection required for Adequate Core Cooling</td><td>RAPIDLY DEPRESSURIZE the RPV to 150-250 psig using ANY RPV Depressurization Systems (Table P-2)
➢ Ok to exceed 100ºF/hr cooldown rate
➢ OK to defeat isolations and interlocks and exceed rad release rate limits if necessary</td></tr>
</table>
C1/P-4
<table>
<tr><td>DEPRESSURIZE to 150-250 psig AND CONTROL pressure within that band using ANY RPV Depressurization Systems (Table P-2)
➢ Ok to exceed 100ºF/hr cooldown rate
➢ OK to defeat isolations and interlocks and exceed rad release rate limits if necessary
➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary</td></tr>
</table>
</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 9. When the crew has depressurized the Reactor to 150-250 psig and has control of Reactor Water Level using HPCI and/or RCIC, end of scenario.</td></tr>
</table>

Scenario Setup
UNIT 2

IC	28
Exam IC	287

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-65	56	ARP-9-5A	63	EOI-3	19
OI-66	124	ARP-9-5B	33	APPX-5C	8
OI-68	168	ARP-9-7A	35	APPX-5D	10
AOI-66-1	21	ARP-9-8A	49		
AOI-85-3	26	ARP-9-53	42		
AOI-85-6	23	TRM 3.7.2	0		
AOI-85-7	25	TS 3.1.3	253		
AOI-100-1	121	TS 3.3.6.2	253		
ARP-9-3A	58	TS 3.3.7.1	253		
ARP-9-3D	38	EOI-1	21		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Danger Tags on the following: • Loop I of RHR • EHPM Pump Place Protected Equipment tags on the following components: • Loop II RHR • Loop II Core Spray • DG C, D • HPCI • 4KV Shutdown Boards C, D
Schedule Files(s):	ILT 2304 NRC Scenario 3 UNIT 2.sch
Event Files(s):	ILT 2304 NRC Scenario 3 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ON LINE 230	Drywell Leakage (GPM) 1.89	Protected Equipment
			LOOP II RHR, LOOP II CS
			DG C, D
			HPCI
Rx Power 95%	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	Floor Drain (GPM) 0.31	4KV SHUTDOWN BOARDS C, D
MWe 1232		Equipment Drain (GPM) 1.58	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

On Day 2 of planned 4-day RHR Loop I Outage - LCO 3.5.1.A – 7 Days

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

EHPM tagged for 2-FCV-007-008, EHPM INJECTION VALVE motor repair

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Start ‘C’ Standby Gas from Panel 2-9-25 and align to Unit 2 Reactor Zone for Engineering Data gathering

Return Reactor Power to 100% using Core Flow (RCP provided). RE is on call and not required to be in Control Room

Support Loop I RHR Outage

Support motor repair on EHPM Injection Valve

OPERATOR WORK AROUNDS OWAs – U0-1/U3-0 Burdens - U0 – 0/U3-0 Operator Challenges – U0 – 12/U3 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-3
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 95% Reactor Power, EHPM tagged for motor repair on 3-FCV-007-008, EHPM PUMP INJECTION VALVE, Loop I RHR is tagged for a scheduled work-week outage.

Turnover: Align 'C' Standby Gas Train to the Unit 3 Reactor Zone. Raise Reactor Power to 100%, support EHPM Injection Valve Maintenance and RHR Loop I outage.

Critical Tasks:

1. When plant conditions indicate that two plant areas are above their Maximum Safe values, without delay, implement Emergency RPV Depressurization.

2. With a loss of Table L-3 Injection Systems, stop Emergency Depressurization before HPCI and RCIC isolate on low Steam Pressure.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 3 Reactor Zone
2.	N/A	R-OATC	Raise Reactor Power Using Core Flow
3.	OG04A	C-BOP C-NUSO	3A Steam Jet Air Ejector Trip
4.	RF RD01A ANN 5A_W2	C-OATC MC-OATC C-NUSO	Control Rod Drive (CRD) Suction Pressure Low
5.	RM08A	TS-NUSO	3-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale
6.	RD04R2015	C-OATC TS-NUSO	Control Rod Drift Out
7.	HS-35-35A PMP-35-36	C-BOP MC-BOP C-NUSO	Stator Cooling Water Pump (SCW) Trip
8.	RD04R3023 RD04R3039 TH23 CU04	M-ALL	Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak
9.	ED10B ED12A CS01A CS01C	C-BOP C-NUSO	Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The Balance of Plant (BOP) Operator will start 'C' SGT in accordance with 0-OI-65, Standby Gas Treatment System, and align it to the Unit 3 Reactor Zone.
2. The crew will raise Reactor Power to 100% using Core Flow in accordance with 3-OI-68, Reactor Recirculation System.
3. 3A Steam Jet Air Ejector (SJAE) will isolate due to a faulty relay, requiring that the crew place the Standby SJAE in service in accordance with 3-OI-66, Off-Gas System, Attachment 6, Standby SJAE System Lineup Hard Card.
4. The crew will observe indications of low Control Rod Drive System Pressure, requiring them to take action in accordance with Alarm Response Procedures and 3-AOI-85-3, CRD System Failure.
5. 3-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor, will fail downscale, requiring the crew to reference the Alarm Response Procedures. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
6. A Control Rod will drift out of the core, requiring the crew to take action in accordance with 3-AOI-85-6, Rod Drift Out, and 3-AOI-85-7, Mispositioned Control Rod. The Control Rod will not settle to position 00 until individually SCRAMMED. The NUSO will address Technical Specifications.
7. 3A Stator Cooling Water (SCW) Pump will trip, and 3B SCW Pump will not automatically start, requiring the crew to manually start 3B SCW to prevent a Main Turbine Trip and Reactor SCRAM.
8. Two additional Control Rods will drift out of the Core, requiring the crew to insert a Manual Reactor SCRAM. As a result of the Control Rod Drift transient and the subsequent SCRAM, fuel damage will occur, causing rising radiation levels. An un-isolable Reactor Water Cleanup (RWCU) leak will occur, eventually causing maximum safe radiation levels in various areas of the plant; and as a result, the crew will be required to Emergency Depressurize the Reactor.
9. Various electrical failures will cause a loss of Emergency Operating Instruction (EOI) Table L-3 low pressure injection systems, requiring the NUSO to enter 3-EOI-C1, Alternate Level/Pressure Control, and limit the Emergency Depressurization to ensure Adequate Core Cooling with High Pressure Injection Systems.

The Scenario ends when the crew has Emergency Depressurized the Reactor to 150-250 psig, and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available high-pressure injection system.

Critical Tasks 2

1. **When plant conditions indicate that two plant areas are above their Maximum Safe values, without delay, implement Emergency RPV Depressurization.**

   a. **Safety Significance**
   Places the primary system in the lowest possible energy state, rejects heat to the Suppression Pool in preference to outside the Primary Containment, and reduces driving head and flow of system discharging into the Secondary Containment.

   b. **Cues**
   Procedural Compliance.
   Secondary Containment Area Radiation indications.

   c. **Measured by:**
   Observation - NUSO transitions to Emergency RPV Depressurization, and directs opening at least 6 MSRVs when two or more areas are greater than their Maximum Safe operating values for the same parameter.

   d. **Feedback**
   Reactor Pressure Trend.
   MSRV status indications.

   e. **Critical Task Failure Criteria:**
   The operating crew fails to proceed without delay to initiate Emergency Depressurization from the time it is announced that two Area Radiation Levels exceed Maximum Safe value.

2. **With a loss of Table L-3 Injection Systems, stop Emergency Depressurization before HPCI and RCIC isolate on low Steam Pressure.**

   a. **Safety Significance**
   Full Emergency Depressurization in the case of a loss of Low-Pressure injection subsystems will result in a loss of steam-driven injection systems – therefore Emergency Depressurization must be coordinated with core cooling strategies to prevent a loss of Adequate Core Cooling. A Loss of Adequate Core Cooling would compound radioactivity release.

   b. **Cues**
   Procedural Compliance.
   Loss of Low-Pressure injection systems.

   c. **Measured by:**
   Observation – NUSO transitions to 3-EOI-C-1 and directs limited Emergency Depressurization when Table L-3 injection systems become unavailable.

   d. **Feedback:**
   Reactor Water Level.
   Reactor Pressure.

   e. **Critical Task Failure Criteria:**
   The operating crew reduces Reactor Pressure below the HPCI and RCIC isolation setpoints, causing a loss of Reactor Injection sources.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 1 Page 1 of 3

Event Description: Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 3 Reactor Zone

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**EXAMINER NOTE: The crew has been pre-briefed on Event 1, but may elect to hold a re-focus brief before proceeding.**
	NUSO	Directs the Balance of Plant Operator (BOP) to start 'C' Standby Gas Treatment Train (SGT) and align it to the Unit 3 Reactor Zone.
	BOP	0-OI-65, Standby Gas Treatment System Section 5.2, Standby Gas Treatment System Manual Initiation [1] **ENSURE** the following requirements are satisfied: • SGT Train A(B)(C) in standby readiness. **REFER TO** Section 4.0. • Main Stack Radiation Monitoring in Service. **REFER TO** 2-OI-90, Radiation Monitoring System • As plant conditions permit, Normal ventilation has been in operation for a minimum of 2 hours following painting, fire, or chemical release for the affected zones from which SGT will take suction
	Driver	**If contacted as another Unit, acknowledge any information or direction given.**
	BOP	[2] **REVIEW** the Precautions and Limitations in Section 3.0. [3] **ENSURE** suction path is aligned to SGT System as follows: [3.1] N/A [3.2] **IF** alignment to Reactor Zone Ventilation suction path is desired, **THEN ENSURE OPEN** the following dampers for the desired unit(s) to be aligned. • 3-HS-64-40 and 3-HS-64-41, REACTOR ZONE EXHAUST TO SGTS Dampers, on Panel 3-9-25 [3.3] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 1 Page 2 of 3

Event Description: Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 3 Reactor Zone

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **START** SGT FAN A(B)(C) as follows: [4.1] N/A [4.2] N/A [4.3] **IF** starting SGT FAN C from Panel 3-9-25, **THEN DEPRESS** 0-HS-65-69A/3, SGTS TRAIN C FAN push-button. [5] **CHECK** SGT TRAIN C INLET DAMPER as follows: [5.1] N/A [5.2] N/A [5.3] **IF** SGT FAN C was started, **THEN CHECK OPEN** SGTS TRAIN C INLET DAMPER, 0-HS-65-51A(0-IL-65-51) indicates OPEN on Panel 3-9-25. [6] **CHECK** SGT TRAIN A(B)(C) RH CONTROL HTR as follows: [6.1] N/A [6.2] **IF** SGT FAN C was started, **THEN CHECK** ENERGIZED 0-HS-65-60, SGTS TRAIN C RH CONTROL HTR, on Panel 2-9-25.
	Driver	**If contacted as Unit 2, when requested report that SGTS TRAIN C RH CONTROL HEATER is energized as required.**
	BOP	[7] **RECORD** start time and filter bank differential pressure for SGT Train C as follows: [7.1] **N/A** [7.2] **IF** SGT FAN C was started, **THEN RECORD** start time and FILTER BANK DIFFERENTIALPRESSURE, 0-PDI-65-53 on Panel 2-9-25, in the Narrative Log.
	Driver	**If contacted as Unit 2 for Narrative Log readings, report that Unit 2 will make all required entries for SGT C.**
	BOP	[8] **DISPATCH** Operator to the Standby Gas Treatment building as soon as time allows to check for abnormal conditions (i.e. belt tightness, rubbing or vibration noises).

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 1 Page 3 of 3

Event Description: Start 'C' Standby Gas Treatment Train (SGT) and Align to Unit 3 Reactor Zone

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Outside Assistant Unit Operator (AUO), acknowledge any information or direction given.**
	BOP	[9] **MONITOR** Standby Gas Treatment Train operation. **REFER TO** Section 6.0.
	NRC	**End of Event 1. Proceed to Event 2, Raise Reactor Power Using Core Flow.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 2 Page 1 of 2

Event Description: Raise Reactor Power Using Core Flow

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew has pre-briefed Event 2, but may elect to conduct a re-focus brief.**
	NUSO	Directs the Operator at the Controls (OATC) to raise Reactor Power to 100% in accordance with 3-OI-68, Reactor Recirculation System and Reactivity Control Plan (RCP) RCP-NRC2304-U3S3.
	NRC	**The crew may elect to raise Reactor Power using any method below.**
	OATC	3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown on 0-TI-248, Unit 3 Core Operating Limits Report (COLR) and 3-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor pump speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pump 3A speed with Recirc Individual Control, **THEN PERFORM** the following: [1.1] Raise Recirc Pump 3A using, 3-HS-96-15A(15B), RAISE SLOW (MEDIUM). [1.2] N/A [2] **IF** desired to control Recirc Pump 3B speed with Recirc Individual Control, **THEN PERFORM** the following: [2.1] Raise Recirc Pump 3B using 3-HS-96-16A(16B), RAISE SLOW (MEDIUM), [2.2] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 2 Page 2 of 2

Event Description: Raise Reactor Power Using Core Flow

Time	Position	Applicant's Actions or Behavior
	OATC	[3] **WHEN** desired to control Recirc Pumps 3A and/or 3B Speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 3A & 3B using the following push buttons as required: 3-HS-96-31, **RAISE** SLOW 3-HS-96-32, **RAISE** MEDIUM
	NRC	**End of Event 2. Request that the Driver insert Event 3, 3A Steam Jet Air Ejector Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 3 Page 1 of 4

Event Description: 3A Steam Jet Air Ejector Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Steam Jet Air Ejector will trip, causing lowering vacuum

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to trip 3A Steam Jet Air Ejector.**
	BOP	Acknowledges and reports the following alarms to the SRO: • OG HOLDUP LINE INLET FLOW LOW, 3-9-53, Window 4 • H2 WATER CHEMISTRY ABNORMAL, 3-9-53, Window 10 • H2 WATER CHEMISTRY SHUTDOWN, 3-9-53, Window 20
	BOP	3-ARP-9-53, Alarm Response Procedure OG HOLDUP LINE INLET FLOW LOW, Window 4 A. **CHECK** holdup line flow on: 1. Integrated Computer System, computer point 66-111A (A419) 2. 3-FR-66-20, OFFGAS FLOW TO HOLDUP VOLUME, Panel 3-9-8 3. 3-XR-66-103/3, OFFGAS H2 CONT/OFFGAS TO 6 HR FLOW/ANALYZER, Panel 3-9-53 B. **CHECK OPEN** in-service Recombiner Inlet And Outlet Valves at Panel 3-9-53. C. **CHECK OPEN**, 3-FCV-66-28, OFF-GAS SYSTEM ISOLATION VALVE. D. **CHECK** that SJAE auto isolation has **NOT** occurred. E. **DISPATCH** PERSONNEL to 3-LPNL-925-0095 to verify the alarm, 3-FS-066-0111A. F. N/A
	BOP	Determines the cause for the above alarms to be a trip of 3A Steam Jet Air Ejector (SJAE), informs the Nuclear Unit Senior Operator (NUSO).
	NUSO	Directs the BOP to respond in accordance with Alarm Response Procedures and to place 3B SJAE in service in accordance with 3-OI-66, Standby SJAE System Lineup Hard Card.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: x Page 2 of 4

Event Description: 3A Steam Jet Air Ejector Trip

Time	Position	Applicant's Actions or Behavior
	BOP	3-OI-66, Off-Gas System Standby SJAE System Lineup Hard Card Section 3.0, OPERATOR ACTIONS FOR SJAE 3B **NOTES** Radiation Protection should be notified prior to placing a SJAE in service. If time does not permit this due to plant conditions, then notification should be made when possible. 3-HS-001-0375, SJAE TRAIN PERMISSIVE (located on 3-LPNL-925-0105, U3 TB, el 586') should normally be in the position of the standby SJAE. If problems are encountered while placing a SJAE in service and time permits, operate this switch as required during the performance of this section.
	Driver	**If contacted as Radiation Protection, acknowledge any information or direction given.**
	BOP	[1] **ENSURE RESET** Off-Gas isolation using 3-HS-90-155, OFFGAS OUTLET/DRAIN ISOLATION VALVES. [2] **ENSURE OPEN** the following valves: • 3-HS-66-15, SJAE 3B INLET VALVE • 3-HS-1-156A, STEAM TO SJAE 3B [3] **ENSURE** in **AUTO/OPEN** 3-HS-66-18, SJAE 3B OG OUTLET VALVE. [4] **PLACE** 3-HS-1-152, SJAE 3B PRESS CONTROLLER, in **CLOSE** and then in **OPEN**. [5] **ENSURE OPEN** the following valves (red light illuminated): • 3-PCV-1-153/167, STEAM TO SJAE 3B STAGES 1,2, AND 3 • 3-FCV-1-152, SJAE 3B INTMD CONDENSER DRAIN
	BOP	When the standby SJAE is placed in service, acknowledges and reports the following alarms: • OFFGAS HOLDUP LINE INLET FLOW HIGH, 3-9-53, Window 14 • HIGH OFFGAS % H2 TRAIN A, 3-9-53, Window 3 • HIGH OFFGAS % H2 TRAIN B, 3-9-53, Window 13

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 3 Page 3 of 4

Event Description: 3A Steam Jet Air Ejector Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**The crew may enter 3-AOI-66-1, Off Gas H2 High.**
	BOP	**3-AOI-66-1, Off Gas H2 High** 4.1 Immediate Actions – None 4.2 Subsequent Actions [1] **PLACE** both OFFGAS TRAIN A(B) AUTO CHANNEL CHECK / BYPASS control switches, 3-HS-066-1007 and 1008, on OFFGAS SAMPLE PANEL, 3-LPNL-925-0588, in BYPASS to assure continuous availability of Hydrogen monitoring.
	Driver	**If contacted as the Turbine Building AUO to perform Step [1], acknowledge the direction and report that Step [1] is complete.**
	BOP	[2] N/A [3] **VERIFY** proper operation of in service SJAE. [3.1] N/A [4] **IF** Hydrogen concentration is ≥ 4%, **THEN REFER TO** TRM 3.7.2.
	NRC	**EXAMINER NOTE: TRM requirements are on page 9 of 40.**
	OATC	Monitors Main Condenser Vacuum.
	BOP	[6] **MONITOR** Hotwell Pressure as indicated on recorder 3-XR-2-26, CONDENSATE, on Panel 3-9-6. [7] **FOR** the SJAE **NOT** being placed in service, **ENSURE CLOSED** the following valves: • 3-HS-66-14, SJAE 3A OG OUTLET VALVE • 3-HS-1-150, SJAE 3A PRESSURE CONTROLLER • 3-HS-1-155A, STEAM TO SJAE 3A [8] **WHEN** Condenser Vacuum has stabilized, **THEN PROCEED** to 3-OI-66, Off-Gas System, "Placing Standby SJAE in Operation" AND **PERFORM** the applicable steps to establish acceptable SJAE parameters to obtain stable and live pressure indications.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 3 Page 4 of 4

Event Description: 3A Steam Jet Air Ejector Trip

Time	Position	Applicant's Actions or Behavior	
	BOP	Informs the NUSO that 3B SJAE is in service. Monitors Main Condenser Vacuum.	
	NUSO	Technical Requirement 3.7.2 – Airborne Effluents LCO 3.7.2 Whenever the SJAE is in service, the concentration of Hydrogen in the Offgas downstream of the Recombiners shall be limited to ≤4% by volume. Applicability: During Main Condenser Offgas Treatment System operation. ----------NOTE---------- TRM LCO 3.0.3 is not applicable. ---------- Condition: A. With the concentration of Hydrogen >4% by volume.	
	NUSO	**REQUIRED ACTION:** A. Restore the concentration to within the limit.	**COMPLETION TIME:** A. 48 hours
	Driver	**If the crew directs the Turbine Building AUO perform the following steps, acknowledge the direction:** • **3-OI-4, Hydrogen Water Chemistry System, Section 8.9, Restore H_2 Water Chemistry. Report to the Control Room for a brief on restoring H_2 Water Chemistry, acknowledge the direction.** • **3-OI-66, Off-Gas System, Section 8.4, Placing Standby SJAE in Operation** **If the crew notifies Radiation Protection that an RPHP exists, acknowledge the report. If requested provide a name for the NOMS log.**	
	Driver	**If contacted as Work Control / Outside NUSO, or Maintenance, acknowledge any information or direction given.**	
	NRC	**End of Event 3. Request that the Driver insert Event 4, Control Rod Drive (CRD) Suction Pressure Low.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 4 Page 1 of 3

Event Description: Control Rod Drive (CRD) Suction Pressure Low

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3A CRD Pump parameters will indicate low Suction Pressure

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause low Suction Pressure on 3A CRD Pump.**
	OATC	Acknowledges and reports the following alarm: • CRD PUMP A SUCTION PRESSURE LOW, 3-9-5A, Window 2
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure and applicable Abnormal Operating Instruction.
	OATC	3-ARP-9-5A, Alarm Response Procedure CRD PUMP A SUCTION PRESSURE LOW, Window 2 A. **CHECK** alarm by CRD pump disagreement light and CRD flow and pressure.
	NRC	**EXAMINER NOTE: The CRD Pump Low Suction Pressure trip is overridden. The OATC must manually trip 3A CRD Pump.**
	OATC	B. **IF** alarm is valid, **THEN REFER TO** 3-AOI-85-3, CRD System Failure. C. **DISPATCH** personnel to CHECK suction path.
	Driver	**If contacted as the Reactor Building AUO, acknowledge any information or direction given.**
	OATC	3-AOI-85-3, CRD System Failure 4.1 Immediate Actions [1] N/A [2] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 4 Page 2 of 3

Event Description: Control Rod Drive (CRD) Suction Pressure Low

Time	Position	Applicant's Actions or Behavior
	OATC	4.1 Subsequent Actions **NOTES** 1) Operation with loss of CRD system should be minimized since CRD Mechanism Cooling Water, Reactor Recirc Pump Seal Purge Flow, Reactor Water Cleanup Recirc Pumps Purge Water Flow, and CRD HCU Accumulator Charging Water will be lost. Technical Specification Section 3.1.3 and 3.1.5 should be consulted. 2) Do NOT attempt to start a CRD Pump which has tripped for unknown reasons before verifying locally that the pump, motor, and breaker are in operating condition. 3) Steps 4.2[1] and 4.2[2] should be performed concurrently. [1] N/A [2] N/A [3] **IF** operating CRD Pump has NOT tripped, **THEN PERFORM** the following:(**REFER TO** 3-OI-85, Control Rod Drive System): [3.1] **ENSURE** FULL OPEN 3-FCV-85-56, CRD PUMPS 3A & 3B UNIT 3 SUCTION. [3.2] **IF** running CRD Pump Suction Pressure is approaching 18-inches HG absolute, **THEN ALTERNATE** CRD Pumps. [3.3] **IF** in-service Filter Differential Pressure is greater than 20.5 psid, **THEN ALTERNATE** CRD Drive Water Filters. [3.4] N/A [3.5] N/A [3.6] N/A [4] N/A [5] **NOTIFY** Reactor Engineer of CRD System failure.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 4 Page 3 of 3

Event Description: Control Rod Drive (CRD) Suction Pressure Low

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Reactor Engineer, Work Control / Outside NUSO, or Electrical / Mechanical Maintenance, acknowledge any information or direction given.**
	NRC	**End of Event 4. Request that the Driver insert Event 5, 3-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 5 Page 1 of 3

Event Description: 3-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to fail 3-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor downscale.**
	BOP	Acknowledges and reports the following alarm: • REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE, 3-9-3A, Window 28
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	3-ARP-9-3A, Alarm Response Procedure Operator Action: A. **CHECK** alarm condition on the following: 1. 3-RR-90-144, REACTOR & REFUEL ZONE EXHAUST RADIATION recorder, points 3 and 4 on Panel 3-9-2. 2. 3-RM-90-140/142, RX & REFUEL ZONE EXH CH A RAD MON RTMR, on Panel 3-9-10. 3. 3-RM-90-141/143, RX & REFUEL ZONE EXH CH B RAD MON RTMR, on Panel 3-9-10. B. **CHECK** power supply to the Radiation Monitors. C. **CHECK** other Radiation Monitors for Radiation Levels below release limits. D. **REFER** TO Tech Spec 3.3.6.2 and 3.3.7.1. E. N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 5 Page 2 of 3

Event Description: 3-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.3.6.2 – Secondary Containment Isolation Instrumentation LCO 3.3.6.2 The Secondary Containment isolation instrumentation for each Function in Table 3.3.6.2-1 shall be OPERABLE. Applicability: According to Table 3.3.6.2-1. ----------NOTE---------- Separate Condition entry is allowed for each channel. ---------- Condition: A. One or more channels inoperable.	
	NUSO	**REQUIRED ACTION:** A. Place channel in trip.	**COMPLETION TIME:** A. 12 hours for Functions 1 and 2 AND 24 hours for Functions other than Functions 1 and 2
	NUSO	Technical Specification 3.3.7.1 – Control Room Emergency Ventilation (CREV) System Instrumentation LCO 3.3.7.1 The CREV System instrumentation for each Function in Table 3.3.7.1-1 shall be OPERABLE. Applicability: According to Table 3.3.7.1-1. ----------NOTE---------- Separate Condition entry is allowed for each channel. ---------- Condition: A. One or more channels inoperable.	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 5 Page 3 of 3

Event Description: 3-RE-90-140A, Refuel Zone Channel A Detector A Radiation Monitor Fails Downscale

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** A. Enter the Condition referenced in Table 3.3.7.1-1 for the channel.	**COMPLETION TIME:** A. Immediately
	NUSO	**REQUIRED ACTION:** C.1 Declare associated CREV subsystem inoperable. AND C.2 Place channel in trip.	**COMPLETION TIME:** C.1 – 1 hour from discovery of loss of CREV initiation capability C.2 – 24 hours
	NRC	**End of Event 5. Request that the Driver insert Event 6, Control Rod Drift Out.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 1 of 6

Event Description: Control Rod Drift Out

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Control Rod 22-15 will drift out

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause Control Rod 22-15 to drift out.**
	OATC	Acknowledges and reports the following alarms: • CONTROL ROD DRIFT, 3-9-5A, Window 28
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure and applicable Abnormal Operating Instructions.
	OATC	3-ARP-9-5A, Alarm Response Procedure CONTROL ROD DRIFT, Window 28 Operator Action: A. **DETERMINE** which rod is drifting from Full Core Display. B. **IF** no Control Rod motion is observed, **THEN RESET** Rod Drift as follows: 1. **PLACE** ROD DRIFT ALARM TEST switch, 3-HS-85-3A-S7, in RESET and **RELEASE**. 2. **RESET** the annunciator. C. N/A D. **IF** rod drifting out, **THEN REFER TO** 3-AOI-85-6, Rod Drift Out and 3-AOI-85-7, Mispositioned Control Rod. E. **REFER TO** Tech Spec Section 3.1.3, 3.10.8.
	OATC	3-AOI-85-6, Rod Drift Out 4.1 Immediate Actions [1] N/A 4.2 Subsequent Actions [1] N/A [2] **IF** a Control Rod is moving from its intended position without operator actions, **THEN SELECT** the drifting Control Rod and **INSERT** to the FULL IN (00) position.

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 2 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	[3] N/A [4] N/A [5] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. [6] N/A **NOTES** 1) A Control Rod that will NOT latch could be indicative of either of the following: • A failed open withdraw Directional Control Valve (3-FCV-85-40C) • Foreign material from the reactor entering and lodging in the CRD Collet area wedging the collet open • Collet retainer tube cracking failure 2) If Control Rod does NOT latch at position 00, Step 4.2[8] may be skipped to avoid Control Rod drift to full out position when removed from service. [7] IF the Control Rod will not latch into position "00" and continues to demonstrate occurrences of inadvertent withdrawal, **THEN PERFORM** the following: [7.1] **MAINTAIN** an insert signal on the Control Rod to keep the Control Rod at position 00 or full in overtravel. [7.2] **PLACE** the appropriate Rod Scram Switch, on 3-PNLA-009-0016 in the Auxiliary Instrument Room, in the DOWN position.
	Driver	**If contacted as Work Control / Outside NUSO to SCRAM Control Rod 22-15, acknowledge the direction and any information given. When directed to go to the Aux Instrument Room, wait 1 minute and report that you are standing by to SCRAM Control Rod 22-15. When directed to SCRAM the Control Rod, insert Event 16 and report that Control Rod 22-15 is SCRAMMED.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 3 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	[7.3] **RELEASE** the control switch placing an insert signal on the Control Rod. [7.4] **PLACE** the appropriate Rod Scram Switch, on 3-PNLA-009-0016 in the Auxiliary Instrument Room, in the UP position.
	Driver	**When directed, insert Event 26 and report that the Individual Control Rod SCRAM switch is in the UP position.**
	OATC	[7.5] N/A [8] **IF** the Control Rod is latched into position "00", **THEN REMOVE** associated HCU from service per 3-OI-85, Control Rod Drive System. [9] **DECLARE** Control Rod INOP per Technical Specification 3.1.3. [10] **REFER** TO 3-AOI-85-7 Mispositioned Control Rod. [11] **INITIATE** Condition Report/Work Order. [12] **NOTIFY** Reactor Engineer to perform the following for current condition: • **EVALUATE** condition of Core to assure no resultant Fuel Damage has occurred • **EVALUATION** of impact on Thermal Limits and PCIOMOR restraints • **DETERMINE** if other Control Rods need to be repositioned in order to safely restore Core symmetry to prevent local Fuel Damage
	Driver	**If contacted as the Reactor Engineer or System Engineer, acknowledge any information or direction given.**
	OATC	[13] **NOTIFY** System Engineering to PERFORM 0-TI-20, Control Rod Drive System Testing and Troubleshooting to determine problem with faulty Control Rod. [14] **IF** a manual SCRAM was not inserted and Reactor Startup or Shutdown is not in progress, **THEN ENSURE** 3-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. [15] N/A [16] N/A [17] NOTIFY Reactor Engineer to EVALUATE impact on preconditioning envelope, prior to returning to normal power operation.
	Driver	**If contacted as the Reactor Engineer, acknowledge any information or direction given.**

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 4 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	3-AOI-85-7, Mispositioned Control Rod 4.1 Immediate Actions – None 4.2 Subsequent Actions [1] **STOP** all intentional Control Rod Movement. **NOTE** Notifications per step 4.2[2] should be performed as soon as practical, but should NOT delay performance of steps 4.2[3] or 4.2[4]. Step 4.2[2] may be completed after performance of steps 4.2[3] and 4.2[4]. [2] IF Control Rod is determined to be mispositioned, THEN NOTIFY the following: • Reactor Engineer (RE) • Shift Technical Advisor (STA) • Unit SRO • Shift Manager (SM) • Operations Superintendent
	Driver	**If contacted by the Reactor Engineer, Shift Technical Advisor, Shift Manager, or Operations Superintendent, acknowledge any information given.**
	OATC	[3] **IF** the wrong Control Rod is moved OR a Control Rod that was not being moved is found to be mispositioned, **THEN PERFORM** the following: [3.1] **INSERT** the mispositioned rod to "00". [3.2] **IF** a Reactor Startup or Shutdown is not in progress, **THEN ENSURE** 3-GOI-100-12, Power Maneuvering, has been entered if a power change occurred. [3.3] N/A [4] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 5 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior
	OATC	[5] **CHECK** the following radiation recorders for a rise in activity to determine if any fuel damage occurred: • 3-RR-90-135, MAIN STEAM LINE RADIATION (Panel 3-9-2) • 3-RR-90-266, OFFGAS RADIATION (Panel 3-9-2) [6] N/A [7] [8] N/A [9] **NOTIFY** Reactor Engineer to perform the following when time permits: • **EVALUATE** the possible consequences • **DOCUMENT** in Reactor Engineer log [10] N/A
	Driver	**If contacted as the Reactor Engineer, acknowledge any information or direction given.**
	NUSO	Technical Specification 3.1.3, Control Rod OPERABILITY LCO 3.1.3 Each Control Rod shall be OPERABLE Applicability: Modes 1 and 2 --- NOTE: Separate Condition entry is allowed for each Control Rod. --- **CONDITION:** C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 6 Page 6 of 6

Event Description: Control Rod Drift Out

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** C.1 ----------NOTE---------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation. --- Fully insert INOPERABLE Control Rod AND C.2 Disarm the associated CRD.	**COMPLETION TIME:** C.1 – 3 hours C.2 – 4 hours
	NUSO	Technical Specification 3.1.5, Control Rod OPERABILITY LCO 3.1.5 Each Control Rod SCRAM Accumulator shall be OPERABLE Applicability: Modes 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod SCRAM Accumulator. **CONDITION:** A. One or more Control Rod SCRAM Accum INOPERABLE with Reactor Steam Dome Pressure ≥ 900 psig.	
	NUSO	**REQUIRED ACTION:** A.2 Declare the associated Control Rod INOPERABLE	**COMPLETION TIME:** 8 hours
	NRC	**Tech Spec 3.10.8, Shutdown Margin (SDM) Test – Refueling is not applicable to current plant conditions.**	
	NRC	**End of Event 6. Request that the Driver insert Event 7, Stator Cooling Water Pump Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 7 Page 1 of 2

Event Description: Stator Cooling Water Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3A Stator Cooling Water Pump will trip, and 3B Stator Cooling Water Pump will not automatically start.

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: The Unit 3 Main Turbine will trip and the Reactor will SCRAM in approximately 1 minute from the loss of 3A Stator Cooling Water Pump if the crew does not manually start the standby pump (it will fail to automatically start).**
	BOP	Acknowledges and reports the following alarms: • GEN STATOR COOLANT SYS ABNORMAL, 3-9-7A, Window 22 TURBINE TRIP TIMER INITIATED, 3-9-8A, Window 1
	NUSO	Directs the BOP to respond in accordance with applicable Alarm Response Procedures.
	BOP	3-ARP-9-7A, Alarm Response Procedure GEN STATOR COOLANT SYS ABNORMAL, Window 22 **NOTE** The control room alarm typer can be used to confirm this alarm. A. **IF** while performing the action of this ARP 3-XA-55-9-8A window 1 alarms, **THEN** 1. **ENSURE** all available Stator Cooling Water Pumps running. 2. N/A 3. N/A B. **ENSURE** a Stator Cooling Water Pump is running and **CHECK** stator temperature recorder, 3-TR-57-59, Panel 3-9-8.
	BOP	Starts 3B SCW Pump. Verifies SCW has been restored and that TURBINE TRIP TIMER INITIATED, 3-9-8A, Window 1, can be reset.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 7 Page 2 of 2

Event Description: Stator Cooling Water Pump Trip

Time	Position	Applicant's Actions or Behavior
	BOP	C. **CHECK** alarm by dispatching personnel to check the Stator Coolant Control Cabinet. D. **REQUEST** personnel to **REFER TO** Local Panel ARP for correct alarm response actions to be taken. E. N/A
	Driver	**If contacted as the Turbine Building AUO, Work Control NUSO, or Electrical Maintenance, acknowledge any report or direction given.** **After 2 minutes, report as the Turbine Building AUO that 3A Stator Cooling Water Pump is hot to the touch.**
	BOP	3-ARP-9-8A, Alarm Response Procedure TURBINE TRIP TIMER INITIATED, Window 1 **NOTE** The control room alarm typer can be used to confirm this alarm. Operator Action: A. **CHECK** Stator Cooling Water Flow and Temperature and Generator Stator temperatures using ICS. B. **ENSURE** all available Stator Cooling Water Pumps running. **NOTE** The full capacity of the Turbine Bypass valves with all nine valves open is 21.3% reactor power. To determine the capacity of the bypass valves, subtract 2.37% for each out of service bypass valve from the 21.3%. (Example, one bypass valve out of service, [21.3% - 2.37% = 18.93%], therefore, the capacity of the bypass valves with one bypass valve out of service is 18.93%.)
	BOP	C. N/A D. **DISPATCH** personnel to Stator Coolant Unit to investigate.
	NRC	**End of Event 7. Request that that driver insert Event 8, Multiple Control Rod Drift/Fuel Damage/Reactor Water Cleanup (RWCU) Leak**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Two Control Rods will drift in – when a Reactor SCRAM is inserted, A RWCU leak and Fuel Damage will occur

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to cause Control Rods 30-23 and 30-29 to drift out.**
	NRC	**EXAMINER NOTES:** • **Control Rods 30-23 and 30-29 will drift out** • **No action is required by the Driver to insert the RWCU Leak or the Fuel Damage – those malfunctions are inserted when the Reactor MODE SWITCH is taken to the SHUTDOWN position**
	OATC	Acknowledges and reports the following alarm: • CONTROL ROD DRIFT, 3-9-5A, Window 28
	NRC	**Only the Immediate Actions of 3-AOI-85-6, Rod Drift Out, are shown below, as the Subsequent Actions are not applicable once a Reactor SCRAM is inserted.**
	OATC	3-AOI-85-6, Rod Drift Out 4.1 Immediate Actions [1] **IF** multiple Control Rod drifts are identified, **THEN MANUALLY SCRAM** the Reactor and enter 3-AOI-100-1, Reactor SCRAM.
	NUSO	Directs the OATC to enter 3-AOI-100-1, Reactor SCRAM.
	NRC	**When the MODE SWITCH is placed in SHUTDOWN, all Condensate Pumps will trip. See Attachment 1 on page 32 of 40 for crew actions.** **Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 2 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	OATC	3-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 3-HS-99-5A/S3A, REACTOR SCRAM A, and 3-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 3-9-5. [2] **PLACE** 3-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] N/A [4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the UNIT SRO: • Reactor SCRAM • MODE SWITCH is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power Level [5] N/A
	BOP	Acknowledges and reports the following alarms: • RWCU LEAK DETECTION TEMP HIGH, 3-9-3D, Window 17 • RWCU ISOL LOGIC CHANNEL A TEMP HIGH, 3-9-5B, Window 32 • RWCU ISOL LOGIC CHANNEL B TEMP HIGH, 3-9-5B, Window 33
	NUSO	Directs the BOP to respond in accordance with the applicable Alarm Response Procedures.
	BOP	3-ARP-9-3D, Alarm Response Procedure RWCU LEAK DETECTION TEMP HIGH, 3-9-3D, Window 17 Operator Action: [A] **IF** this alarm is received in conjunction with RWCU ISOL LOGIC CHANNEL A TEMP HIGH [3-XA-55-5B, Window 32] and RWCU ISOL LOGIC CHANNEL B TEMP HIGH [3-XA-55-5B, Window 33], **THEN EXIT** this procedure and **GO TO** 3-ARP-9-5B. Otherwise, **CONTINUE** in this procedure.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 3 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NRC	**RWCU ISOL LOGIC CHANNEL B TEMP HIGH, 3-9-5B, Window 33 is effectively the same actions as those for Window 32. Therefore, only Window 32 actions are shown below.**
	BOP	3-ARP-9-5B, Alarm Response Procedure RWCU ISOL LOGIC CHANNEL A TEMP HIGH, 3-9-5B, Window 32 Operator Action: A. **CHECK** alarm by checking: 1. ATUs on Panel 3-9-83 and 3-9-85. 2. RWCU LEAK DETECTION TEMP HIGH annunciator in alarm (3-XA-55-3D, Window 17). 3. Area temperature indication on LEAK DETECTION SYSTEM TEMPERATURE, 3-TR-69-29, on Panel 3-9-22. 4. ARMs 3-RR-90-1, 3-CONS-90-50A on Panel 3-9-2 and 0-CONS-90-361 on Panel 1-9-2. 5. ICS 'HPTURB' & 'RWCU' mimics for the 834 and 835 temperature loops. B. **IF** leak is suspected, **THEN MANUALLY ISOLATE** RWCU C. **IF** RWCU automatically isolates, **REFER TO** 3-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation.
	Driver	**If contacted as Unit 1 Operator to check Area Radiation Monitors or Radiation Recorders, acknowledge the request.**
	BOP	Determines that RWCU failed to automatically isolate, and attempts to manually isolate RWCU. Informs the NUSO that RWCU will not isolate.
		D. **IF** TIS-69-835A(C) indicates greater than 131 ºF, **THEN ENTER** 3-EOI-3, Secondary Containment Control. E. **REFER TO** Tech. Spec. Table 3.3.6.1-1. F. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	Driver	**If contacted as Work Control / Outside NUSO, acknowledge any information or direction given.**

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 4 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

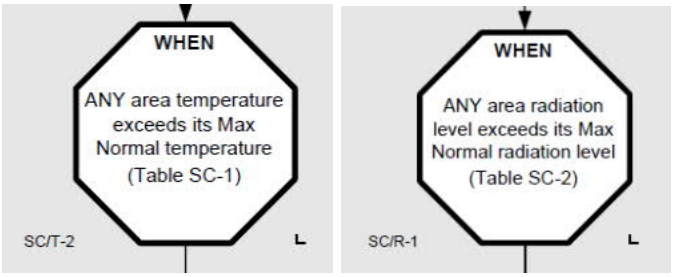
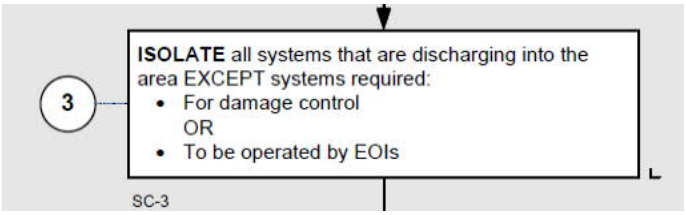
<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>Driver</td><td>If contacted as the Control Bay AUO to investigate or check temperatures, acknowledge any information or direction given.</td></tr>
<tr><td></td><td>NRC</td><td>It is not intended that the SRO reference Technical Specifications for this event.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 3-EOI-3, Secondary Containment Control

SC-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Reactor Zone Ventilation Exhaust Radiation Level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation Exhaust Radiation Level is above 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Reactor Zone Ventilation is isolated
AND
Reactor Zone Ventilation Exhaust Radiation level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Refuel Zone Ventilation is isolated
AND
Refuel Zone Ventilation Exhaust Radiation Level is below 72 mR/hr</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 5 of 11

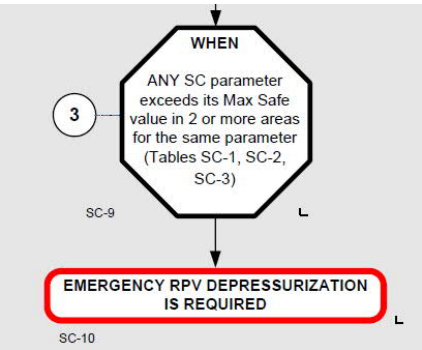
Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	SC-2 **IF** Reactor Zone or Refuel Zone ventilation exhaust radiation level is below 72 mR/hr **THEN** operate available Reactor Zone or Refuel Zone ventilation
	NUSO	
	NUSO	
	NRC	**EXAMINER NOTE: The crew will NOT be able to anticipate Emergency Depressurization, as the Turbine Bypass valves are over-ridden during Simulator Setup.**
	NUSO	Directs the BOP to monitor Secondary Containment parameters (Temperature and Radiation) for approach to Maximum Safe Values.
	CREW	**Critical Task: When plant conditions indicate that two plant areas are above their Maximum Safe values, without delay, implement Emergency RPV Depressurization.** **Critical Task Failure Criteria: The operating crew fails to proceed without delay to initiate Emergency Depressurization from the time it is announced that two Area Radiation Levels exceed Maximum Safe value.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 6 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	Table SC-2 Secondary Cntmt Area Radiation

Area	Applicable Radiation Indicators	Max Normal Value mR/hr	Max Safe Value mR/hr	Potential Isolation Sources
RHR sys I pumps	90-25A	Alarmed	1000	FCV-74-47, 48
RHR sys II pumps	90-28A	Alarmed	1000	FCV-74-47, 48
HPCI room	90-24A	Alarmed	1000	FCV-73-2, 3, 44, 81
CS sys I pumps RCIC room	90-26A	Alarmed	1000	FCV-71-2, 3, 39
CS sys II pumps	90-27A	Alarmed	1000	None
Top of torus General area	90-29A	Alarmed	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB el 565 W	90-20A	Alarmed	1000	FCV-69-1, 2, 12 SDV vents & drains
RB el 565 E	90-21A	Alarmed	1000	SDV vents & drains
RB el 565 NE	90-23A	Alarmed	1000	None
TIP room	90-22A	Alarmed	100,000	TIP ball valve
RB el 593	90-13A, 14A	Alarmed	1000	FCV-74-47, 48
RB el 621	90-9A	Alarmed	1000	FCV-43-13, 14
Recirc MG sets	90-4A	Alarmed	1000	None
Refuel floor	90-1A, 2A 3A	Alarmed	1000	None

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 7 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>BOP</td><td>Monitors Secondary Containment parameters and informs the NUSO when two (2) Radiation Levels are at Maximum Safe Value.</td></tr>
<tr><td></td><td>NUSO</td><td>Updates the crew that Emergency Depressurization is required. Enters 3-EOI-1, Emergency RPV Depressurization.</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-1, Emergency Depressurization

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 feet</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-7
<table>
<tr><td>IF Drywell Pressure is above 2.45 psig THEN PREVENT injection from ONLY those Core Spray and LPCI Pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr>
</table>
RC/P-8
<table>
<tr><td>EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Level is above 5.5 ft THEN OPEN 6 MSRVs (ADS Valves preferred)
➢ OK to exceed 100°F/hr cooldown rate</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to open 6 ADS Valves.</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 8 of 11

Event Description: Multiple Control Rod Drift / Fuel Damage / Reactor Water Cleanup (RWCU) Leak

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9, Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization, is automatically entered when the first SRV is opened for Emergency Depressurization. See Event 9 on page 35 of 40.**
	BOP	Opens 6 ADS Valves.
	BOP	Determines that 3B 480V Shutdown Board has tripped. See Event 9 on page 35 of 40.
	NRC	**End of Event 8. See Attachment 1 below for action for Reactor Water Level Control following the loss of Condensate Pumps. Otherwise, continue to Event 9 on page 35 of 40.**

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 9 of 11

Attachment 1
Page 1

Time	Position	Applicant's Actions or Behavior
	NUSO	When Condensate Pumps trip, directs the BOP to maintain Reactor Water Level using HPCI or RCIC.
	BOP	3-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 3-XS-71-51A(B), RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and OPEN 3-FCV-71-9, RCIC TURB TRIP/THROT VLV. [4] **ENSURE** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 gpm. [5] **OPEN** 3-FCV-71-34 RCIC PUMP MIN FLOW VALVE. [6] **OPEN** 3-FCV-71-39 RCIC PUMP INJECTION VALVE. [7] **OPEN** 3-FCV-71-25 RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 3-HS-71-31A, RCIC VACUUM PUMP, in START. [9] **OPEN** 3-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine. [10] **CHECK** proper RCIC operation by observing the following: A. RCIC turbine speed accelerates above 2100 rpm. B. Flow to RPV controlled automatically at 620 gpm. C. 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 gpm. [11] **ADJUST** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] N/A [13] N/A [14] N/A

Time	Position	Applicant's Actions or Behavior
	BOP	3-AOI-Appendix-5D, Injection System Lineup HPCI [1] N/A [2] N/A [3] N/A [4] **VERIFY** at least one SGTS Train in operation.
	Driver	**Standby Gas should be running following the Reactor SCRAM (based on the Reactor Water Level transient), but if not and if contacted to start a train of Standby Gas, insert Event 29.**
	BOP	[5] **VERIFY** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL Controller is in one of the following configurations, as desired: • in AUTO and set for 5300 gpm for rapid injection • in AUTO and set for approximately 2500 gpm for slower injection • in MANUAL with output at approximately 50% for slower injection [6] N/A [7] **PLACE** HPCI AUXILIARY OIL PUMP handswitch in START. [8] **PLACE** HPCI STEAM PACKING EXHAUSTER handswitch in START. [9] **OPEN** 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE. [10] **OPEN** 3-FCV-73-44, HPCI PUMP INJECTION VALVE. [11] **OPEN** 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine. [12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine speed accelerates. B. 3-CKV-73-45, HPCI SYSTEM CHECK VALVE, opens by observing 3-ZI-73-45A, DISC POSITION, red light illuminated. C. HPCI flow to RPV stabilizes and is controlled automatically at the setpoint. (N/A if controller in manual). D. 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds approximately 1200 gpm.

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8

Attachment 1
Page 3

Time	Position	Applicant's Actions or Behavior
	BOP	[13] **ADJUST** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL Controller as necessary to control injection. [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO. [16] N/A [17] N/A
	NRC	**End of Event 8, Attachment 1.**

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 9 Page 1 of 4

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

Symptoms/Cues: Event is initiated automatically entered based on MODE SWITCH and ADS Valve Position. 3B 480V Shutdown Board and both Loop I Core Spray Pumps will trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Event 9 malfunctions are inserted as follows – no action is required by the Driver to insert these malfunctions:
All Condensate Pumps trip when the MODE SWITCH is placed in SHUTDOWN
3B 480V Shutdown Board Trips when the first ADS Valve is opened
3A and 3C Core Spray Pump Trips are inserted on Simulator setup</td></tr>
<tr><td></td><td>NRC</td><td>Numerous alarms will be received during this event, and not all are covered in this guide.</td></tr>
<tr><td></td><td>BOP</td><td>When Emergency Depressurization commences, determines that 3B 480V Shutdown Board has tripped.
Reports a loss of Loop I LPCI Injection Valve Power to the NUSO.</td></tr>
<tr><td></td><td>BOP</td><td>When an attempt to is made to start Loop I Core Spray Pumps, they will trip.</td></tr>
<tr><td></td><td>NUSO</td><td>In accordance with 3-AOI-1, Emergency Depressurization override RC/P-6, determines that Table L-3 subsystems alone cannot assure Adequate Core Cooling.

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure Adequate Core Cooling</td><td>TERMINATE the RPV Depressurization
AND
C1 Alternate Level/Pressure Control</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 feet</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 9 Page 2 of 4

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to terminate Emergency Depressurization.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 3-C-1, Alternate Level/Pressure Control

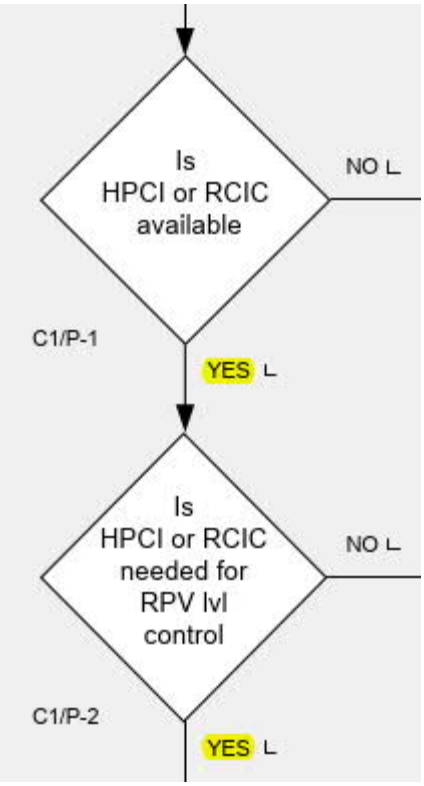
C1-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>It is determined that the Reactor will NOT remain subcritical without boron under all conditions</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>One or more available Injection Subsystems (Table L-3) can assure Adequate Core Cooling (Table M-3)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is determined that Core Damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>

</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 9 Page 3 of 4

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NUSO	
	CREW	**Critical Task: With a loss of Table L-3 Injection Systems, stop Emergency Depressurization before HPCI and RCIC isolate on low Steam Pressure.** **Critical Task Failure Criteria: The operating crew reduces Reactor Pressure below the HPCI and RCIC isolation setpoints, causing a loss of Reactor Injection sources.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-3 Event No.: 8 Page 4 of 4

Event Description: Loss of Low-Pressure Injection Systems / Limited Emergency Depressurization

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
C1/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>➢ HPCI and RCIC are NOT available for injection
OR
➢ HPCI and RCIC are no longer needed for injection</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>All criteria that will assure Adequate Core Cooling is met (Table L-4)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

AND

RPV depressurization to 150-250 psig will NOT result in loss of injection required for Adequate Core Cooling</td><td>RAPIDLY DEPRESSURIZE the RPV to 150-250 psig using ANY RPV Depressurization Systems (Table P-2)
➢ Ok to exceed 100ºF/hr cooldown rate
➢ OK to defeat isolations and interlocks and exceed rad release rate limits if necessary</td></tr>
</table>
C1/P-4
<table>
<tr><td>DEPRESSURIZE to 150-250 psig AND CONTROL pressure within that band using ANY RPV Depressurization Systems (Table P-2)
➢ Ok to exceed 100ºF/hr cooldown rate
➢ OK to defeat isolations and interlocks and exceed rad release rate limits if necessary
➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary</td></tr>
</table>
</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 9. When the crew has depressurized the Reactor to 150-250 psig and has control of Reactor Water Level using HPCI and/or RCIC, end of scenario.</td></tr>
</table>

Scenario Setup
UNIT 3

IC	28
Exam IC	265

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-65	56	ARP-9-5A	56	EOI-3	17
OI-66	87	ARP-9-5B	33	APPX-5C	6
OI-68	106	ARP-9-7A	30	APPX-5D	8
AOI-66-1	9	ARP-9-8A	42		
AOI-85-3	14	ARP-9-53	30		
AOI-85-6	13	TRM 3.7.2	0		
AOI-85-7	11	TS 3.1.3	298		
AOI-100-1	80	TS 3.3.6.2	213		
ARP-9-3A	60	TS 3.3.7.1	213		
ARP-9-3D	36	EOI-1	16		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Danger Tags on the following: • Loop I of RHR • EHPM Pump Place Protected Equipment tags on the following components: • Loop II RHR • Loop II Core Spray • DG 3C, 3D • HPCI • 4KV Shutdown Boards 3EC, 3ED
Schedule Files(s):	ILT 2304 NRC Scenario 3 UNIT 3.sch
Event Files(s):	ILT 2304 NRC Scenario 3 UNIT 3.evt

UNIT 3 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ON LINE** **230**	**Drywell Leakage (GPM)** **1.89**	**Protected Equipment** LOOP II RHR, LOOP II CS DG 3C, 3D HPCI
Rx Power **95%**	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (GPM)** **0.31**	4KV SHUTDOWN BOARDS 3EC, 3ED
MWe **1232**		**Equipment Drain (GPM)** **1.58**	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

On Day 2 of planned 4-day RHR Loop I Outage - LCO 3.5.1.A – 7 Days

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

EHPM tagged for 3-FCV-007-008, EHPM INJECTION VALVE motor repair

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Start ‘C’ Standby Gas from Panel 3-9-25 and align to Unit 3 Reactor Zone for Engineering Data gathering

Return Reactor Power to 100% using Core Flow (RCP provided). RE is on call and not required to be in Control Room

Support Loop I RHR Outage

Support motor repair on EHPM Injection Valve

OPERATOR WORK AROUNDS OWAs – U0-1/U3-0 Burdens - U0 – 0/U3-0 Operator Challenges – U0 – 12/U3 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-4
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 100% Reactor Power, HPCI is tagged for a scheduled outage, PSC Head Tank Pump 2B is tagged for coupling repair.

Turnover: Alternate EHC Pumps due to high filter D/P. 2B Bus Duct fan is running for PMT. Support HPCI outage.

Critical Tasks:
1. When Reactor Water Level cannot be restored and maintained above (-) 162 inches, and before Reactor Water Level reaches (-) 180 inches, Emergency Depressurize the Reactor.
2. With a primary system discharging into the Primary Containment, initiate Drywell Sprays before Drywell Temperature exceeds 280ºF to prevent reaching component qualification and structural design limits.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Alternate EHC Pumps
2.	SW02A	C-OATC C-NUSO MC-OATC	Reactor Building Closed Cooling Water (RBCCW) Pump Trip
3.	DG01B	TS-NUSO	'B' Diesel Generator Logic Breaker Trip
4.	TU02A-M	R-OATC	Main Turbine Vibration
5.	EG13B	C-BOP C-NUSO	Bus Duct Fan Trip
6.	RD08R3043	TS-NUSO	Control Rod Accumulator INOPERABLE
7.	TH07A	C-OATC C-NUSO	2A Recirculation Pump Cell Bypass
8.	ED07B	C-BOP C-NUSO MC-BOP	2B 480V Unit Board Trip
9.	TH21	M-ALL	Unisolable Recirc Piping Leak
10.	FW14A-C	C-BOP C-NUSO	Reactor Feedwater Pumps (RFPs) Trip

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will alternate EHC Pumps in accordance with 2-OI-47A, EHC System.
2. 2A Reactor Building Closed Cooling Water (RBCCW) Pump will trip, and 2-FCV-70-48, RBCCW SECTIONALIZING VALVE, will fail to automatically close, requiring the crew to take action in accordance with Alarm Response Procedures and 2-AOI-70-1, Loss of Reactor Building Closed Cooling Water. The crew must take manual action to close 2-FCV-70-48.
3. The 'B' Emergency Diesel Generator (EDG) Logic Breaker will trip, causing an annunciation in the Control Room and disabling 'B' EDG. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
4. The Main Turbine will experience a high vibration condition. The crew will respond in accordance with Alarm Response Procedures, and reduce Reactor Power. Following the Reactor Power reduction, Main Turbine vibrations levels will lower, precluding further action by the crew.
5. 2B Bus Duct Fan will trip and the crew will respond in accordance with Alarm Response Procedure 2-ARP-9-7A, Window 31, Gen Bus Duct Fan Failure, and start the standby Bus Duct Fan.
6. Control Rod 30-43 Accumulator Pressure will lower below 940 psig, requiring the NUSO to address Technical Specifications.
7. 2A Recirculation Pump Variable Frequency Drive (VFD) Cell C5 will bypass, requiring the crew to take action in accordance with 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease.
8. 2B 480V Unit Board will trip, and as a result 2B EHC Pump will trip, requiring the crew to respond in accordance with Alarm Response Procedures. 2A EHC Pump will fail to automatically start, requiring the crew to manually start 2A EHC Pump.
9. An unisolable Recirculation System piping leak will occur in the Drywell. The crew may respond in accordance with 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, and will be forced to insert a manual Reactor SCRAM. The NUSO will respond in accordance with 2-EOI-2, Primary Containment Control. Reactor Water Level will not be able to be maintained above (-) 180 inches, requiring the crew to Emergency Depressurize the Reactor.
10. Following the Reactor SCRAM, the Reactor Feedwater Pumps (RFPs) will trip, requiring the crew to take action to align a high-pressure makeup source for Reactor Vessel Injection.

The Scenario ends when the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available low-pressure injection system.

Critical Tasks 2

1. **When Reactor Water Level cannot be restored and maintained above (-) 162 inches, and before Reactor Water Level reaches (-) 180 inches, Emergency Depressurize the Reactor.**

   a. **Safety Significance**
   Maintain Adequate Core Cooling and prevent degradation of fission product barrier.

   b. **Cues**
   Procedural Compliance.
   Reactor Water Level trend.

   c. **Measured by:**
   Observation – MRSVs are opened when Reactor Water Level cannot be restored and maintained above (-) 180 inches.

   d. **Feedback**
   RPV Pressure trend.
   MSRV open status indicators.

   e. **Critical Task Failure Criteria:**
   Emergency Depressurization is not directed when Reactor Water Level cannot be restored and maintained above (-) 180 inches.

2. **With a primary system discharging into the Primary Containment, initiate Drywell Sprays before Drywell Temperature exceeds 280ºF to prevent reaching component qualification and structural design limits.**

   a. **Safety Significance**
   Precludes failure of Primary Containment.

   b. **Cues**
   Procedural Compliance.
   Drywell Temperature trend.

   c. **Measured by:**
   Observation – Drywell Sprays are initiated prior to exceeding 280ºF Drywell Temperature.

   d. **Feedback**
   Lowering Drywell Temperature.
   RHR System Flow to containment cooling/sprays.

   e. **Critical Task Failure Criteria:**
   Drywell sprays are not directed before Drywell Temperature reaches 280ºF.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 1

Event Description: Alternate EHC Pumps

Symptoms/Cues: Crew is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**The crew has briefed on the procedure to alternate EHC Pumps, but may hold a re-focus brief prior to commencing the procedure.**
	NUSO	Directs the Balance of Plant Operator (BOP) to alternate EHC Pumps in accordance with 2-OI-47A, EHC System, Section 6.3.
	Driver	**If contacted as the Turbine Building Assistant Unit Operator (AUO), acknowledge any information or direction given.**
	BOP	2-OI-47A, EHC System Section 6.3, EHC Pump Alternation [1] **ENSURE** the EHC System is in service. **REFER TO** Section 5.1. [2] **REVIEW** the Precautions and Limitations listed in Section 3.0. [3] **START** the non-operating EHC Pump 2B using 2-HS-47-2A, EHC HYDRAULIC FLUID PUMP 2B. **NOTE** **ALLOW** both EHC Pumps to operate for at least 30 seconds to allow the standby pump to expel any air which may have accumulated in the pump casing. [4] **CHECK** the started EHC Pump 2-PI-047-0002, EHC HYDRAULIC PUMP 2B DISCHARGE PRESSURE, indicates between 1550 psig and 1750 psig, locally at the EHC skid.
	Driver	**If contacted as the Turbine Building AUO to report EHC 2B Discharge Pressure, report that pressure is 1600 psig.**
	BOP	[5] N/A [6] **STOP** EHC PUMP 2A using 2-HS-47-1A, EHC HYDRAULIC FLUID PUMP 2A.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 1 Page 2 of 2

Event Description: Alternate EHC Pumps

Time	Position	Applicant's Actions or Behavior
	BOP	[7] **CHECK** EHC HEADER PRESSURE, 2-PI-47-7, indicates between 1550 psig and 1650 psig [8] N/A [9] N/A [10] **CHECK** the started EHC PUMP 2-EI-47-2, 2B MOTOR AMPS, indicates less than 140 amps. [11] **CHECK** EHC Pump 2B, EHC PUMP SUCTION FILTER and EHC AUX PUMP SUCTION FILTERS, indicate CLEAN, locally at the EHC Skid. [12] **ENSURE** 2-047-0588, MANUAL FLOW CONTROL VALVE, is set at 2.5 turns clockwise.
	Driver	**If contacted as the Turbine Building AUO to perform steps [11] and [12], report that the EHC Pump Suction Filters indicate clean at the EHC Skid and the Manual Flow Control Valve is set at 2.5 turns clockwise.**
	NRC	**End of Event 1. Request that the Driver insert Event 2, Reactor Building Closed Cooling Water (RBCCW) Pump Trip.** **EXAMINER NOTE: For Event 2, the BALANCE OF PLANT OPERATOR (BOP) MUST BE REMOVED FROM THE CONTROL ROOM TO ALLOW THE OPERATOR AT THE CONTROLS (OATC) TO RESPOND TO THE NEXT EVENT.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 2 Page 1 of 3

Event Description: Reactor Building Closed Cooling Water (RBCCW) Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2A RBCCW Pump will trip, and 2-FCV-70-48, RBCCW SECTIONALIZING VALVE, will not automatically close.

Time	Position	Applicant's Actions or Behavior
	Driver	**CONTACT THE BOP AS REQUESTED BY THE CHIEF EXAMINER WITH A REASON TO LEAVE THE CONTROL ROOM.** **VERIFY THAT THE BOP HAS BEEN REMOVED FROM THE CONTROL ROOM PRIOR TO INSERTING EVENT 2.** **When requested by the Chief Examiner and the BOP is not in the Control Room, insert Event 2 to trip 2A Reactor Building Closed Cooling Water (RBCCW) Pump Trip.**
	OATC	Acknowledges and reports the following alarms: • RBCCW PUMP DISCHARGE HEADER PRESSURE LOW, 2-9-4C, Window 12 • RWCU DEMINERALIZER ABNORMAL, 2-9-4B, Window 24
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response Procedures and 2-AOI-70-1, Loss of Reactor Building Closed Cooling Water.
	OATC	2-ARP-9-4C, Alarm Response Procedure RBCCW PUMP DISCHARGE HEADER PRESSURE LOW, Window 12 Operator Action: A. **ENSURE** 2-FCV-70-48 CLOSING/CLOSED.
	OATC	Determines that 2-FCV-70-48 did not automatically close, takes action to manually close it. Notifies the Nuclear Unit Senior Operator.
	OATC	B. **CHECK** RBCCW Pumps A and B in service. C. **CHECK** RBCCW Surge Tank Low Level Alarm is reset. D. **DISPATCH** personnel to check the following: • RBCCW Surge Tank Level locally • RBCCW Pumps for proper operation
	Driver	**If contacted as the Reactor Building AUO, acknowledge any direction or information given.**

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 2 Page 2 of 3

Event Description: Reactor Building Closed Cooling Water (RBCCW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	OATC	E. **REFER TO** 2-AOI-70-1, Loss of Reactor Building Closed Cooling Water, for RBCCW System failure and 2-OI-70, Reactor Building Closed Cooling Water System, for starting spare pump.
	NRC	**EXAMINER NOTE: Once the OATC has manually closed 2-FCV-70-48, the BOP can be allowed to re-enter the Control Room.**
	OATC	2-AOI-70-1, Loss of Reactor Building Closed Cooling Water 4.1 Immediate Actions [1] **IF** RBCCW Pump(s) has tripped or 2-FCV-70-48, RBCCW SECTIONALIZING VALVE, is closed/closing due to low pressure, **THEN PERFORM** the following (Otherwise N/A): • SECURE RWCU Pumps • ENSURE RBCCW SECTIONALIZING VALVE, 2-FCV-70-48 CLOSED 4.2 Subsequent Actions [1] N/A [2] N/A [3] N/A [4] **IF** unable to restart a tripped pump, **THEN PLACE** Spare RBCCW Pump in service in accordance with 2-OI-70.
	Driver	**If contacted as Unit 1 to place the spare RBCCW Pump in service to Unit 2, wait 1 minute, insert Event 12 to align the spare RBCCW Pump to Unit 2, and inform the crew that the spare RBCCW Pump is aligned to Unit 2.**
	OATC	[5] **IF** RBCCW flow was restored to two pump operation, **THEN PERFORM** the following (Otherwise N/A): [5.1] **REOPEN** 2-HS-70-48A. RBCCW SECTIONALIZING VALVE. [5.2] **RESTORE** the Reactor Water Cleanup (RWCU) System to operation. (**REFER TO** 2-OI-69, Reactor Water Cleanup System)
	NRC	**EXAMINER NOTE: It is not expected that the crew return RWCU to service.**
	NUSO	When RWCU is isolated, addresses Technical Requirements Manual 3.4.1, Coolant Chemistry Surveillance Requirements.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 2 Page 3 of 3

Event Description: Reactor Building Closed Cooling Water (RBCCW) Pump Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Requirement 3.4.1 – Coolant Chemistry TSR 3.4.1.1	
	NUSO	**SURVEILLANCE:** ----------NOTE---------- Not required when there is no fuel in the Reactor vessel. ---------- Monitor Reactor Coolant Conductivity	**FREQUENCY:** Continuously OR 4 hours when the continuous conductivity monitor is inoperable and the reactor is not in MODE 4 or 5 OR 8 hours when the continuous conductivity monitor is inoperable and the reactor is in MODE 4 or 5
	NRC	**End of Event 2. Request that the driver insert Event 3, 'B' Diesel Generator Logic Breaker Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 3 Page 1 of 3

Event Description: 'B' Diesel Generator Logic Breaker Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 'B' Diesel Generator Logic Breaker will open, resulting in the DG being inoperable.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to open the 'B' Diesel Generator (DG) Logic Breaker.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • DIESEL GENERATOR B CONTROL POWER OFF, 0-9-23B, Window 14
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	1/2-ARP-9-23B, Alarm Response Procedure DIESEL GENERATOR 'B' CONTROL POWER OFF, Window 14 Operator Action: A. **OBSERVE** any other alarms on panels 9-8 or 0-9-23-7 which may indicate problem area. B. **DISPATCH** personnel to check the Probable Causes listed above. C. **IF** loss of normal power has occurred, **THEN TRANSFER** to alternate power source per 0-OI-57D, DC Electrical System. D. **IF** unable to restore power or to correct problem, **REFER TO** Tech Spec Sections 3.8.1, 3.3.8.1, 3.8.4, & 3.8.7. E. **LOG** valid events and actions taken in narrative log.
	Driver	**If contacted as an AUO, acknowledge any information or direction given.** **If contacted as the Work Control / Outside NUSO or Electrical Maintenance, acknowledge any information or direction given.**

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 3 Page 2 of 3

Event Description: 'B' Diesel Generator Logic Breaker Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.8.1 – AC Sources – Operating LCO 3.8.1 - The following AC electrical power sources shall be OPERABLE: a. Two qualified circuits between the Offsite transmission network and the Onsite Class 1E AC Electrical Power Distribution System; b. Unit 1 and 2 Diesel Generators (DGs) with two divisions of 480 V Load Shed Logic and Common Accident Signal Logic OPERABLE; and c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating." Applicability: Modes 1, 2, and 3. ACTIONS ---------- NOTE ---------- LCO 3.0.4.b is not applicable to DGs ----------	
		CONDITION B. One required Unit 1 and 2 DG INOPERABLE	
	NUSO	**REQUIRED ACTION:** B.1 Verify power availability from the offsite transmission network. AND B.2 Evaluate the availability of both temporary Diesel Generators B.3 Declare required feature(s), supported by the inoperable Unit 1 and 2 DG, inoperable when the redundant required feature(s) are inoperable. AND (Continued below)	**COMPLETION TIME:** B.1 – 1 hour AND Once every 8 hours thereafter. B.2 – 1 hour AND Once per 12 hours thereafter. B.3 – 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s).

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 3 Page 3 of 3

Event Description: 'B' Diesel Generator Logic Breaker Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** B.4.1 Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure. OR B.4.2 Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s). AND B.5 Restore Unit 1 and 2 DG to OPERABLE status.	**COMPLETION TIME:** B.4.1 – 24 hours B.4.2 – 24 hours B.5 – 7 days from discovery of unavailability of TDG(s) AND 24 hours from discovery of Condition B entry > 6 days concurrent with unavailability of TDG(s) AND 14 days AND 21 days from discovery of failure to meet LCO
	NRC	**In the interest of time management, if the NUSO directs the crew to perform Surveillance Requirement 3.8.7.1, Monthly Check of Power Availability, inform NUSO that another operator will perform the required power availability check.**	
	NRC	**End of Event 3. Request that the Driver insert Event 4, Main Turbine Vibration.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 4 Page 1 of 2

Event Description: Main Turbine Vibration

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Main Turbine vibration will rise to the alarm setpoint.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause Main Turbine vibrations to rise.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • MAIN TURBINE VIBRATION HIGH, 2-9-7B, Window 32
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	2-ARP-9-7B, Alarm Response Procedure Operator Action: A. **VALIDATE** alarm by checking the following: • On EHC WORKSTATION, Turbine Vibration screen • On ICS, MAIN TURBINE BEARINGS (TURBBRG) screen • TURBINE GENERATOR VIBRATION on, 2-MON-47-94 (Panel 2-9-7) • Computer points 47-1501A&B thru 47-1512A&B B. **IF** alarm is valid, **THEN PERFORM** the following: 1. **DETERMINE** cause by checking PROBABLE CAUSE section above. 2. **REDUCE** load and **OBSERVE** vibration. 3. N/A 4. N/A
	NUSO	Directs the OATC to reduce Reactor Power in an attempt to reduce Main Turbine vibration.
	NRC	**The crew may reduce Reactor Power by any method in 2-OI-68, Reactor Recirculation System.** **When the crew reduces Reactor Power less than 99.5%, Main Turbine vibrations will lower below the alarm setpoint.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 4 Page 2 of 2

Event Description: Main Turbine Vibration

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow [1] N/A [2] **WHEN** desired to control Recirc Pumps 2A and/or 2B Speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 2A & 2B using the following push buttons as required: • 2-HS-96-33, **LOWER** SLOW • 2-HS-96-34, **LOWER** MEDIUM • 2-HS-96-35, **LOWER** FAST
	BOP	When Reactor Power is lowered, reports that Main Turbine Vibration levels have returned to normal values.
	NRC	**End of Event 4. Request that the Driver insert Event 5, Bus Duct Fan Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 5 Page 1 of 2

Event Description: Bust Duct Fan Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2B Bus Duct Fan will trip.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause 2B Bus Duct Fan to trip.**
	BOP	Acknowledges and reports the following alarm: • GENERATOR BUS DUCT FAN FAILURE, 2-9-7A, Window 31
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	2-ARP-9-7A, Alarm Response Procedure GENERATOR BUS DUCT FAN FAILURE, Window 31 Operator Action: A. **CHECK** Main Bus Cooling Fans, 2-HS-262-1A or 3-HS-262-2A, indicates running on Panel 2-9-8. B. **START** 2-HS-262-1A, GEN BUS DUCT HX FAN A using on Panel 2-9-8 to start the standby fan.
	BOP	Starts 2A Bus Duct Fan and informs the NUSO that the Standby Bus Duct Fan is running.
	BOP	C. N/A D. N/A E. **DISPATCH** personnel as necessary to check the following: 1. Main Bus Cooling Fan on elevation 586' to check fan condition. 2. Monitor Bus Duct Temperature by available means including using a portable temperature monitor device locally at the 15 in-service thermostats. **REFER** to Window 32, Figure 1. 3. 480V Unit Board 2A on elevation 586' to check breaker 5C closed. 4. 480V Unit Board 2B on elevation 604' to check breaker 5C closed. F. **ENSURE** the system is operating in accordance with 2-OI-47.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 5 Page 2 of 2

Event Description: Bus Duct Fan Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Turbine Building AUO, acknowledge any information or direction given.** **If contacted as the Work Control / Outside NUSO or Electrical Maintenance, acknowledge any information or direction given.**
	NRC	**End of Event 5. Request that the Driver insert Event 6, Control Rod Accumulator INOPERABLE.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6 Page 1 of 4

Event Description: Control Rod Accumulator INOPERABLE

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The CRD Accumulator Low Pressure Alarm will be received for Control Rod Accumulator 30-43.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the NRC, insert Event 6 to cause Control Rod 30-43 Accumulator low pressure.**
	OATC	Acknowledges and reports the following alarm: • CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, 2-9-5A, Window 29
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	OATC	3-9-5A, Alarm Response Procedure CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, Window 29 Operator Action: A. **CHECK** alarm by amber background light illuminated on Full Core Display. B. **LOG** in the narrative log the Control Rod number and time alarm was received. **NOTE** If any of the following fuses is/are cleared, the local indications at Panel 25-4 and 25-22 will NOT illuminate. C. **IF** multiple accumulator lights are lit on Panel 2-9-5, **THEN CHECK** for cleared fuses 2-FU1-085-25-004G, -004H in Panel 25-4 and 2-FU1-085-25-022G, -022H in Panel 25-22. D. **DISPATCH** personnel to Panel 25-4 (east side), Panel 25-22 (west side) El 565', to determine if level high or pressure low. E. **DEPRESS** push-button for associated HCU to determine if alarm is caused by level high or pressure low as follows: 1. If alarm is due to high level, the red light will extinguish. 2. If light stays illuminated, alarm is due to low N2 pressure.

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6 Page 2 of 4

Event Description: Control Rod Accumulator INOPERABLE

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as the Reactor Building AUO to respond to the Accumulator Alarm, acknowledge any direction or information given. Wait two minutes and report to the crew that Control Rod 30-43 Accumulator Pressure is 930 psig and slowly lowering. The N2 charging bottle is empty and you are working to get another N2 bottle to recharge Accumulator 30-43.**
	OATC	F. **IF** alarm is valid, **THEN REFER TO** 2-OI-85, Control Rod Drive System and 2-AOI-85-3, CRD System Failure.
	OATC	**NOTE** If accumulator pressure is greater than 940 psig the accumulator is not required to be declared Inoperable when the "star" valve is CLOSED, unless accumulator is unattended. G. **IF** Accumulator Pressure is less than or equal to 940 psig, **THEN DECLARE** Control Rod HCU "INOPERABLE". H. **IF** the associated HCU's nitrogen pressure is found less than 940 psi, **THEN INITIATE** a Condition Report (CR) to calibrate the pressure switch. The HCU will NOT be declared operable until the switch has been calibrated. I. **IF** alarm is due to low pressure with pressure greater than 940 psig and accumulator CANNOT be recharged within one hour, **THEN EVALUATE** per Tech Spec 3.1.5. 1. **IF** the Control Rod is declared SLOW **REFER TO** TECH SPEC 3.1.4. Currently no more than 13 OPERABLE Control Rods shall be slow and no more than 2 OPERABLE Control Rods that are slow shall occupy adjacent locations. 2. **IF** the Control Rod is declared INOPERABLE, **THEN REFER** TO TECH SPEC 3.1.3. J. **RECORD** this evaluation in narrative log. K. N/A L. N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6 Page 3 of 4

Event Description: Control Rod Accumulator INOPERABLE

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.1.5, Control Rod SCRAM Accumulators LCO 3.1.5: Each Control Rod SCRAM Accumulator shall be OPERABLE Applicability: Modes 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod SCRAM Accumulator. Condition: A.1 One Control Rod SCRAM Accumulator inoperable with Reactor Steam Dome Pressure ≥ 900 psig.	
	NUSO	**REQUIRED ACTION:** A.1 NOTE: Only applicable if the associated Control Rod SCRAM time was within the limits of Table 3.1.4-1 during the last SCRAM time Surveillance. Declare the associated Control Rod SCRAM Time "slow". OR A.2- Declare the associated Control Rod INOPERABLE.	**COMPLETION TIME:** A.1 – 8 hours A.2 – 8 hours
	NRC	**If Tech Spec 3.1.5 A.2 is Declared, THEN Tech Spec 3.1.3 is applicable.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6 Page 4 of 4

Event Description: Control Rod Accumulator INOPERABLE

Time	Position	Applicant's Actions or Behavior	
	NUSO	If the Control Rod is declared INOPERABLE from Tech Spec 3.1.5.A.2, references Tech Spec 3.1.3, Control Rod OPERABILITY. LCO 3.1.3 Each Control Rod shall be OPERABLE APPLICABILITY: MODES 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod. **CONDITION:** C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.	
	NUSO	**REQUIRED ACTION:** NOTE: RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation. C.1 – Fully insert INOPERABLE Control Rod. AND C.2 – Disarm the associated CRD.	**COMPLETION TIME:** C.1 – 3 hours C.2 – 4 hours
	NUSO	If the Control Rod is declared SLOW, references Tech Spec 3.1.4, Control Rod SCRAM Times. Currently, there are less than 13 OPERABLE Control Rods that are slow, and there are no OPERABLE Control Rods that are slow that occupy adjacent locations. Therefore, there are no actions required for Tech Spec 3.1.4, Control Rod SCRAM Times.	
	NRC	**End of Event 6. Request that the driver insert Event 7, 2A Recirculation Pump Cell Bypass.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 7 Page 1 of 3

Event Description: 2A Recirculation Pump Cell Bypass

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2A recirculation Pump VFD Cell C5 will bypass, causing 2A Recirculation Pump Speed to lower by about 200 rpm

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause 2A Recirculation Pump Cell C5 to bypass.**
	OATC	Acknowledges and reports the following alarm: • RECIRC DRIVE 2A DRIVE ALARM, 2-9-4A, Window 32
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure and 2-AOI-68-1, Recirc Pump Trip / Core Flow Decrease.
	OATC	2-ARP-9-4A, Alarm Response Procedure RECIRC DRIVE 2A DRIVE ALARM, Window 32 Operator Action: A. **REFER** TO ICS Group Display "GD @VFDADA" and **DETERMINE** cause of alarm. B. N/A C. N/A D. N/A E. N/A F. For all other alarms, or any problems encountered. **NOTIFY** System Engineering. G. At discretion of Unit SRO, **ATTEMPT** to reset alarm. **REFER TO** 2-OI-68 for guidance.
	Driver	**If contacted as the Work Control / Outside NUSO, an AUO, or System Engineering acknowledge any direction or information given.**
	OATC	2-AOI-68-1, Recirc Pump Trip / Core Flow Decrease. 4.1 Immediate Actions [1] N/A 4.2 Subsequent Actions [1] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 7 Page 2 of 3

Event Description: 2A Recirculation Pump Cell Bypass

Time	Position	Applicant's Actions or Behavior
	OATC	[2] N/A [3] **IF** a VFD Cell(s) has bypassed, **THEN** with Shift Manager AND Reactor Engineering concurrence, **RAISE** speed of the affected VFD to approximately the original speed. (REFER to 2-OI-68) (otherwise N/A) [3.1] **IF** the affected VFD cannot be restored to approximately the original speed, **THEN PERFORM** 2-SR-3.4.2.1, Jet Pump Mismatch and Operability. (otherwise N/A)
	Driver	**If contacted as the Reactor Engineer or Shift Manager, concur with raising 2A Recirc Pump speed to match 2B Recirc Pump Speed. If asked as the Reactor Engineer, no RCP is required, there are no ramp rate limits, and Reactor Engineering will monitor Thermal Limits.**
	OATC	[4] **REFERENCE** the following ICS screens to help determine the cause of Recirc Pump trip/Core Flow lowering. • VFDPMPA(VFDPMPB) • VFDAAL(VFDBAL) [5] **CHECK** parameters associated with Recirc Drive and Recirc Pump/Motor 2A on ICS and 2-TR-68-71 to determine cause of trip/core flow lowering event. [6] **PERFORM** visual inspection of tripped/affected Reactor Recirc Drive. [7] **PERFORM** visual inspection of Reactor Recirc Pump Drive relay boards for relay targets.
	Driver	**If contacted as the Reactor Building AUO, acknowledge any information or direction given.**
	OATC	[8] N/A [9] **INITIATE** actions required to make the necessary repairs. [10] N/A
	NUSO	Directs the OATC to raise 2A Recirc Pump Speed in accordance with 2-OI-68, Reactor Recirculation System.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 7 Page 3 of 3

Event Description: 3A Recirculation Pump Cell Bypass

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-68, Reactor Recirculation System, Section 6.2 Adjusting Recirc Flow [1] **IF** desired to control Recirc Pump 2A speed with Recirc Individual Control, **THEN PERFORM** the following; • Raise Recirc Pump 2A using, 2-HS-96-15A(15B), RAISE SLOW (MEDIUM). • N/A [2] N/A
	NRC	**End of Event 7. Request that the Driver insert Event 8, 2B 480V Unit Board Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 8 Page 1 of 2

Event Description: 2B 480V Unit Board Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2B Unit Board will Trip, causing a loss of electrical loads

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to trip 2B 480V Unit Board.**
	BOP	Acknowledges and reports the following alarms: • 480V UNIT BD 2B UV OR XFR, 2-9-8B, Window 17 • 480V TB VENT BD 2B UV OR XFR, 2-9-8C, Window 22 • 480V TURB MOV BD 2B UV OR XFR, 2-9-8C, Window 23 • 480V TURB MOV BD 2C UV OR XFR, 2-9-8C, Window 30 • EHC HYD FLUID HDR PRESS LOW, 2-9-7B, Window 1 • STANDBY EHC PUMP FAILED, 2-9-7B, Window 15
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedures.
	NRC	**Only the 2B 480V Unit Board alarm is covered below.**
	BOP	2-9-8B, Alarm Response Procedure 480V UNIT BD 2B UV OR XFR, Window 17 A. **ENSURE** automatic action has occurred. B. **INSPECT** 480V Unit Bd 2B for abnormal conditions: relay targets, smoke, burned paint, breaker position, etc. C. **REFER TO** 0-OI-57B, 480V/240V AC Electrical System to re-energize board. D. **REFER TO** appropriate OI for recovery or realignment of equipment.
	Driver	**If contacted as an AUO, Electrical Maintenance, or Work Control / Outside NUSO, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 8 Page 2 of 2

Event Description: 2B 480V Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	2-ARP-9-7B, Alarm Response Procedure • EHC HYD FLUID HDR PRESS LOW, Window 1 Operator Action: A. **ENSURE** Standby EHC PUMP 2A, 2-HS-47-1A running.
	BOP	Determines that the standby EHC Pump did not start and starts 2A EHC Pump.
	BOP	B. **CHECK** EHC HEADER PRESSURE, pressure indicator, 2-PI-47-7 between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC Pump unit. D. N/A
	Driver	If contacted as the Turbine Building AUO, acknowledge any information or direction given.
	BOP	2-9-7B, Alarm Response Procedure STANDBY EHC PUMP FAILED, Window 15 Operator Action: A. **PERFORM** the following on Panel 2-9-7: 1. **CHECK** alarm by checking EHC HEADER PRESSURE, 2-PI-47-7. 2. **CHECK** 2-HS-47-2A, EHC PUMP 2B and/or 2-HS-47-1A, EHC PUMP 3A running. 3. **CHECK** EHC PUMP 2B PUMP MTR CURRENT 2-EI-47-2 and/or EHC PUMP 2A PUMP MTR CURRENT 2-EI-47-1. 4. **CHECK** lights above EHC PUMP 2A TEST pushbutton 2-HS-47-4A and EHC PUMP 2B TEST pushbutton 2-HS-47-5A. B. **DISPATCH** personnel to pumping unit to check for abnormal conditions. C. N/A
	NRC	**End of Event 8. Request that the Driver insert Event 9, Unisolable Recirc Piping Leak.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 1 of 12

Event Description: Unisolable Recirc Piping Leak

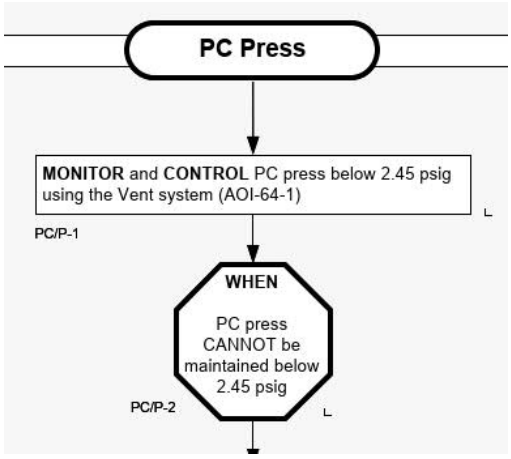
Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. A Recirculation System leak will occur in the Drywell, causing Drywell Pressure and Temperature to rise

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 9 to cause an unisolable Recirc Piping leak in the Drywell.**
	NRC	**Drywell Pressure rises quickly enough that the crew may not have time to address all alarms prior to having to insert a manual Reactor SCRAM.**
	CREW	Observes Drywell Pressure rising, acknowledges and reports Drywell Pressure alarms as they are received.
	NRC	**The NUSO may direct the OATC to insert a Core Flow Runback if there is time before Drywell Pressure reaches 2.2 psig, which is when a Reactor SCRAM will be ordered.**
	NUSO	When Drywell Pressure reaches 2.2 psig, directs the OATC to insert a Reactor SCRAM and to enter 2-AOI-100-1, Reactor SCRAM.
	NRC	**Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**
	OATC	Inserts a manual Reactor SCRAM, and enters 2-AOI-100-1. 2-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A and 2-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 2-9-5. [2] **PLACE** 2-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] N/A
	NRC	**When the OATC places the Reactor MODE SWITCH in SHUTDOWN, Event 10, Reactor Feedwater Pumps Trip, is automatically entered. See Event 10 on page 34 of 45 for Event 10 actions.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 2 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the UNIT SRO: • Reactor SCRAM • MODE SWITCH is in SHUTDOWN • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power level [5] N/A
	NRC	**The NUSO will enter 2-EOI-1, RPV Control Modes 1-3 if Reactor Water Level reaches +2 inches following the SCRAM.**
	BOP	When Drywell Pressure reaches 2.45 psig, informs the NUSO.
	NUSO	When informed that Drywell Pressure has reached 2.45 psig, enters 2-EOI-2, Primary Containment Control and re-enters 2-EOI-1 (if previously entered).
	NUSO	2-EOI-2, Primary Containment Control  

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 3 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	PC/P-3 **IF** / **THEN** Primary Containment Pressure reduction is required to restore and maintain Adequate Core cooling or reduce total offsite radiation dose — **NO ACTION REQUIRED** Primary Containment Pressure CANNOT be maintained above (-) 2 psig — **NO ACTION REQUIRED** Suppression Chamber Pressure approaches 0 psig — **NO ACTION REQUIRED** Drywell Pressure approaches 0 psig — **NO ACTION REQUIRED** PC/P-4 **BEFORE** Suppression Chamber Pressure rises to 12 psig PC/P-5 **INITIATE** Suppression Chamber Sprays ➢ Use only sources NOT required for continuous RPV injection (APPX 17C) **IF** / **THEN** Needed to augment Suppression Chamber Sprays — **NO ACTION REQUIRED** PC/P-6 **WHEN** Suppression Chamber Pressure exceeds 12 psig

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 4 of 12

Event Description: Unisolable Recirc Piping Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>PC/P-7
IF Suppression Pool Level is below 19 ft AND Drywell Temperature is in safe area of Curve 5 THEN
1. SHUT DOWN Recirc Pumps
2. SHUT DOWN Drywell Blowers
3. INITIATE Drywell Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B, 20L)
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Needed to augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to spray the Suppression Chamber in accordance with 2-EOI-Appendix-17C, RHR System Operations Suppression Chamber Sprays.</td></tr>
<tr><td></td><td>BOP</td><td>2-EOI-Appendix-17C, RHR System Operations Suppression Chamber Sprays

[1] BEFORE Suppression Chamber pressure drops below 0 psig CONTINUE In this procedure at Step 1.0[6].
[2] IF Adequate Core Cooling is assured OR directed to spray the Suppression Chamber irrespective of Adequate Core Cooling, THEN BYPASS LPCI Injection Valve auto open signal as necessary:
• PLACE 2-HS-74-155A, LPCI SYS I OUTBD INJECTION VALVE BYPASS SELECT IN BYPASS
• PLACE 2-HS-74-155B, LPCI SYS II OUTBD INJECTION VALVE BYPASS SELECT IN BYPASS
[3] N/A
[4] N/A
[5] INITIATE Suppression Chamber Sprays as follows:
[5.1] ENSURE at least one RHRSW Pump supplying each EECW header.</td></tr>
</table>

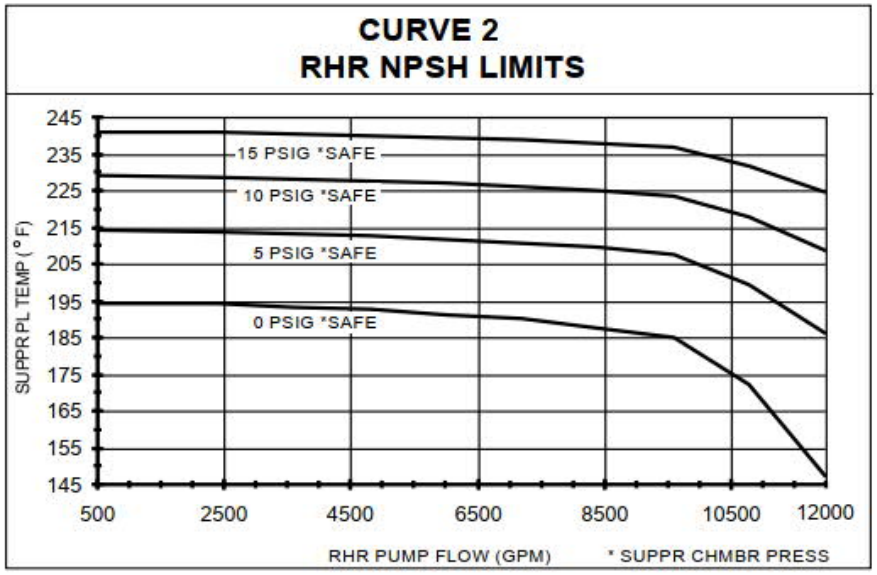
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 5 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[5.2] **IF** EITHER of the following exists: • LPCI Initiation signal is NOT present OR • Directed by SRO, **THEN** **PLACE** keylock switch 2-XS-74-122(130), RHR SYSTEM I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE. [5.3] **MOMENTARILY PLACE** 2-XS-74-121 (129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT. [5.4] **IF** 2-FCV-74-53 (67), RHR SYSTEM I (II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 2-FCV-74-52 (66), RHR SYSTEM I (II) LPCI OUTBOARD INJECTION VALVE. [5.5] **ENSURE** OPERATING the desired System I (II) RHR Pump(s) for Suppression Chamber Spray. [5.6] **ENSURE** OPEN 2-FCV-74-57(71), RHR SYS I(II) SUPPRESSION CHAMBER/POOL ISOLATION VALVE [5.7] **OPEN** 2-FCV-74-58 (72), RHR SYS I (II) SUPPRESSION CHAMBER SPRAY VALVE. [5.8] **IF** RHR System I(II) is operating ONLY in Suppression Chamber Spray mode, **THEN CONTINUE** in this procedure at step 1.0[5.11]. [5.9] **ENSURE** CLOSED 2-FCV-74-7 (30), RHR SYSTEM I (II) MIN FLOW VALVE. [5.10] **RAISE** system flow by placing the second RHR System I(II) Pump in service as necessary. [5.11] **MONITOR** RHR Pump NPSH using Attachment 2. [5.12] **ENSURE** RHRSW pump supplying desired RHR Heat Exchanger(s). [5.13] **THROTTLE** the following in-service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW Flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 6 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[5.14] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any information or direction given.**
	BOP	[6] N/A [7] N/A [8] N/A [9] N/A [10] N/A [11] N/A Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. 

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 7 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	CREW	**Critical Task: With a primary system discharging into the Primary Containment, initiate Drywell Sprays before Drywell Temperature exceeds 280ºF to prevent reaching component qualification and structural design limits.** **Critical Task Failure Criteria: Drywell sprays are not directed before Drywell Temperature reaches 280ºF.**
	BOP	2-EOI-Appendix-17B, RHR System Operation Drywell Sprays [1] **BEFORE** Drywell Pressure drops below 0 psig **CONTINUE** in this procedure at Step 1.0[7]. [2] **IF** Adequate Core Cooling is assured OR directed to spray the Drywell irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI Injection Valve auto open signal as necessary: • PLACE 2-HS-74-155A, LPCI SYS I OUTBOARD INJECTION VALVE BYPASS SELECT IN BYPASS • PLACE 2-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT IN BYPASS [3] **ENSURE** Recirc Pumps and Drywell Blowers shutdown. [4] N/A [5] N/A [6] **INITIATE** Drywell Sprays as follows: [6.1] **ENSURE** at least one RHRSW pump supplying each EECW header. [6.2] **IF** EITHER of the following exists: • LPCI Initiation signal is NOT present, OR • Directed by SRO, **THEN** **PLACE** keylock switch 2-XS-74-122(130), RHR SYS I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE. [6.3] **MOMENTARILY PLACE** 2-XS-74-121 (129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT.

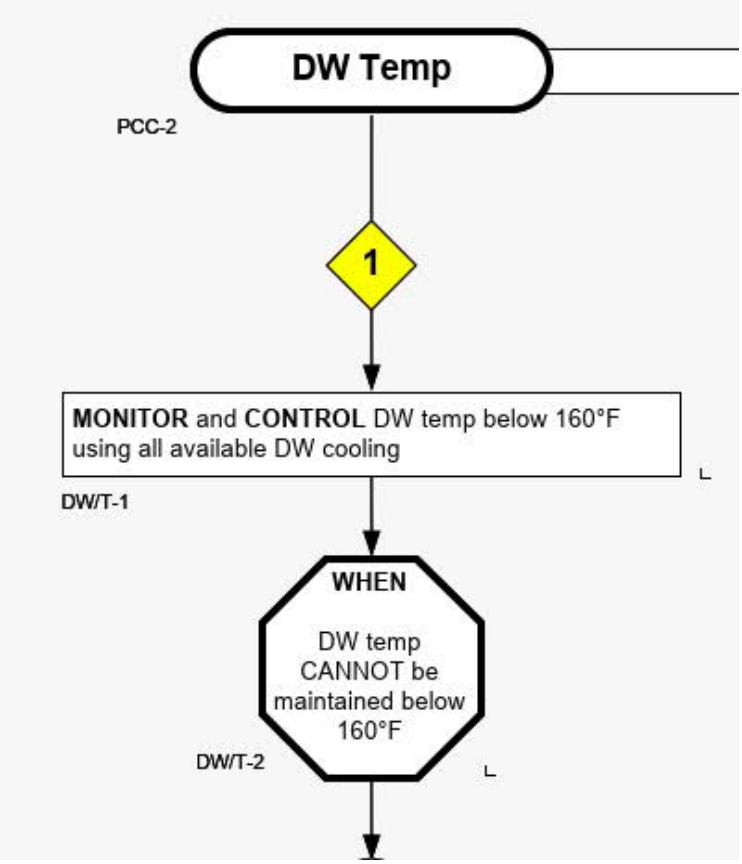
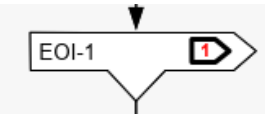
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 8 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[6.4] **IF** 2-FCV-74-53 (67), RHR SYS I (II) LPCI INBD INJECT VALVE, is OPEN, **THEN ENSURE CLOSED** 2-FCV-74-52 (66), RHR SYS I (II) LPCI OUTBD INJECT VALVE. [6.5] **ENSURE OPERATING** the desired System I (II) RHR Pump(s) for Drywell Spray. [6.6] **OPEN** the following valves: • 2-FCV-74-60(74), RHR SYSTEM I(II) DRYWELL SPRAY OUTBOARD VALVE • 2-FCV-74-61 (75), RHR SYSTEM I (II) DRYWELL SPRAY INBOARD VALVE [6.7] **ENSURE CLOSED** 2-FCV-74-7 (30), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [6.8] **IF** Additional Drywell Spray Flow is necessary, **THEN PLACE** the second System I (II) RHR Pump in service. [6.9] **MONITOR** RHR Pump NPSH using Attachment 2. [6.10] ENSURE RHRSW Pump supplying desired RHR Heat Exchanger(s). [6.11] **THROTTLE** the following in-service RHRSW outlet valves to obtain 1700 to 4500 gpm RHRSW Flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE [6.12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any information or direction given.**
	BOP	[7] N/A [8] N/A [9] N/A [10] N/A [11] N/A [12] N/A

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 9 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	 DW/T-3 **BEFORE** Drywell Temperature rises to 280 ºF 

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 10 of 12

Event Description: Unisolable Recirc Piping Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>DW/T-4
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
DW/T-5
<table>
<tr><td colspan="2">IF Suppression Pool Level is below 19 ft AND Drywell Temperature is in safe area of Curve 5 THEN
1. SHUT DOWN Recirc Pumps
2. SHUT DOWN Drywell Blowers
3. INITIATE Drywell Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B, 20L)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-1, RPV Control MODES 1-3
 
</td></tr>
<tr><td></td><td>NRC</td><td>See Event 10 on page 34 of 45 for 2-EOI-1 Reactor Water Level control actions.</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 11 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	RC/P-1

IF	THEN
A high Drywell Pressure ECCS signal exists (2.45 psig)	**PREVENT** inj from ONLY those CS and LPCI pumps NOT required for RPV injection before depressurizing below 330 psig (APPX 4)
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED or has been required	B
Emergency Depressurization is anticipated	**RAPIDLY DEPRESSURIZE** the RPV with Main Turbine Bypass Valves ➢ OK to exceed 100ºF/hr cooldown rate
Suppression Pool Temperature and Level CANNOT be maintained in a safe area of Curve 3 at the existing RPV Pressure OR Suppression Pool Level CANNOT be maintained in the safe are of Curve 4	**NO ACTION REQUIRED**

RC/P-2

STABILIZE RPV PRESSURE below 1073 psig using the Main Turbine Bypass Valves (APPX 8B)
- ➢ OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
- ➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary

IF	THEN
DW Control Air is or becomes unavailable	**NO ACTION REQUIRED**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 12 of 12

Event Description: Unisolable Recirc Piping Leak

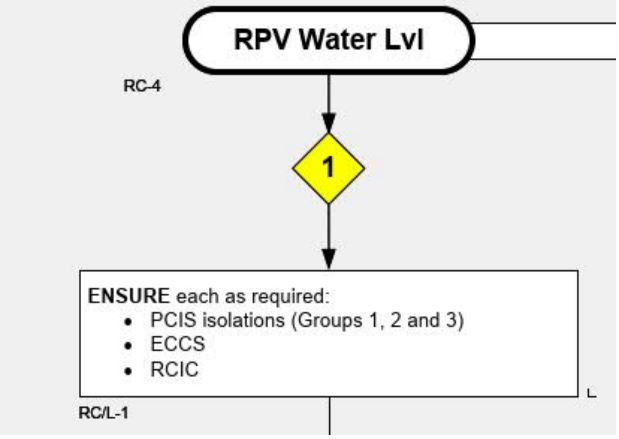
Time	Position	Applicant's Actions or Behavior
	NRC	**When it is apparent that available high-pressure injection systems cannot maintain Reactor Water Level above (-) 162 inches due to the trend in Reactor Water Level, the crew may anticipate Emergency Depressurization.**
	NRC	**End of Event 9. When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF – (-) 162 inches), end of Scenario.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 1 of 10

Event Description: Reactor Feedwater Pumps Trip

Symptoms/Cues: Event is initiated when the Reactor MODE SWITCH is placed in SHUTDOWN. All Reactor Feedwater Pumps will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 10 is initiated when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required to insert Event 10.**
	NRC	**Reactor Feedwater Pumps (RFPTs) will trip 30 seconds after the Reactor MODE SWITCH is placed in SHUTDOWN.**
	OATC	Informs the NUSO that all RFPTs have tripped.
	NRC	**EXAMINER NOTE: The EHPM will trip when it is started – no action is required by the Driver is insert this malfunction.**
	NUSO	 RC/L-2

IF	THEN
Raising RPV Water Level above +51 in. will facilitate use of Shutdown Cooling, steam-driven injection systems, or Alternate Injection subsystems (Table L-2)	**NO ACTION REQUIRED**
Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7	**NO ACTION REQUIRED**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 2 of 10

Event Description: Reactor Feedwater Pumps Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>In accordance with 2-EOI-1, RPV Control MODES 1-3, directs the BOP to maintain Reactor Water Level using any or all of the following EOI Appendices:
CRD – 2-EOI-Appendix-5B
RCIC – 2-EOI-Appendix-5C
EHPM Pump – 2-EOI-Appendix-7L</td></tr>
<tr><td></td><td>NUSO</td><td>RC/L-3
RESTORE and **MAINTAIN** RPV Water Level between +2 in. and +51 in. with ANY Preferred Injection Systems (Table L-1)
➢ OK to use ANY Alternate Injection Subsystems (Table L-2)

RC/L-4
WHEN RPV Water Level CANNOT be restored and maintained between +2 in. and +51 in.

RC/L-5<table><tr><th>IF</th><th>THEN</th></tr><tr><td>RPV Water Level can be restored and maintained above +2 in.</td><td>**NO ACTION REQUIRED**</td></tr></table>
RC/L-6
IF RPV Water Level CANNOT be maintained above -122 in. **THEN INHIBIT** ADS</td></tr>
<tr><td></td><td>**NRC**</td><td>**Due to the size of the leak, available high-pressure systems will not maintain Reactor Water Level above (-) 180 inches. Reactor Water Level will reach (-) 180 inches in about 14 minutes from the time Event 9 is entered.**</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 3 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 2-XS-71-51A(B) RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and OPEN 2-FCV-71-9, RCIC TURBINE TRIP/THROT VALVE. [4] **ENSURE** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 gpm. [5] **OPEN** 2-FCV-71-34 RCIC PUMP MINIMUM FLOW VALVE. [6] **OPEN** 2-FCV-71-39 RCIC PUMP INJECTION VALVE. [7] **OPEN** 2-FCV-71-25 RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 2-HS-71-31A, RCIC VACUUM PUMP, in START. **CAUTIONS** 1) Operating RCIC Turbine below 2100 rpm may result in unstable system operation and equipment damage. 2) High Suppression Chamber Pressure may trip RCIC. 3) Operating RCIC Turbine with Suction Temperatures above 240 ºF may result in equipment damage. [9] **OPEN** 2-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine. [10] **CHECK** proper RCIC operation by observing the following: A. RCIC Turbine Speed accelerates above 2100 rpm. B. Flow to RPV controlled automatically at 620 gpm. C. 2-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE, closes as flow rises above 120 gpm. [11] **ADJUST** 2-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] N/A [13] N/A [14] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 4 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	OATC	2-EOI-Appendix-5B, Injection System Lineup CRD [1] N/A [2] IF **BOTH** of the following exists • CRD is NOT required for rod insertion, **AND** • Maximum injection flow is required **THEN LINE UP** ALL available CRD pumps to the RPV as follows: [2.1] **IF** CRD Pump 2A is available, **THEN VERIFY** RUNNING CRD Pump 2A. [2.2] **IF** CRD Pump 1B is available, **THEN VERIFY** RUNNING CRD Pump 1B. A. **NOTIFY** Unit 1 Operator to verify closed 1-FCV-85-8, CRD PUMP B DISCH TO U2, (Panel 1-9-5) B. **START** CRD Pump 1B C. **OPEN** 2-FCV-85-8, CRD PUMP 1B DISCH TO U2 [2.3] OPEN the following valves to increase CRD Flow to the RPV: • 2-PCV-85-23, CRD DRIVE WATER PRESSURE CONTROL VALVE • 2-PCV-85-27, CRD COOLING WATER PRESSURE CONTROL VALVE • 2-FCV-85-50, CRD EXHAUST RETURN LINE SHUTOFF VALVE [2.4] **ADJUST** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL to control injection WHILE maintaining 2-PI-85-13A, CRD ACCUMULATOR CHARGING WATER HEADER PRESSURE, above 1450 psig, if possible. [2.5] IF additional flow is necessary to prevent or mitigate Core Damage, THEN DISPATCH personnel to fully open the following valves as required: • 2-THV-085-0527, CRD PUMP DISCH THROTTLING (RB NE, el 565') • 2-BYV-085-0551, CRD PUMP TEST BYPASS (RB NE, el 565')

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 5 of 10

Event Description: Reactor Feedwater Pumps Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>CREW</td><td>Determines that Reactor Water Level cannot be maintained above (-) 162 inches.</td></tr>
<tr><td></td><td>CREW</td><td>Critical Task: When Reactor Water Level cannot be restored and maintained above (-) 162 inches, and before Reactor Water Level reaches (-) 180 inches, Emergency Depressurize the Reactor.

Critical Task Failure Criteria: Emergency Depressurization is not directed when Reactor Water Level cannot be restored and maintained above (-) 180 inches.</td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-1, RPV Control MODES 1-3
EMERGENCY RPV DEPRESSURIZATION

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 ft</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-7
<table>
<tr><td>IF DW Pressure is above 2.45 psig THEN PREVENT injection from ONLY those Core Spray and LPCI Pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 6 of 10

Event Description: Reactor Feedwater Pumps Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>RC/P-8
<table>
<tr><td colspan="2">EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Level is above 5.5 ft THEN OPEN 6 MSRVs (ADS Valves preferred)
➢ OK to exceed 100°F/hr cooldown rate</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Less than 4 MSRVs can be opened
AND
RPV Pressure is 60 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>Directs the crew to inject to the Reactor using low-pressure systems:
• Condensate System – 2-EOI-Appendix-6A
• Core Spray System (Loop I, II, or both) - 2-EOI-Appendix-6D/6E
• RHR System (Loop I, II, or both) - 2-EOI-Appendix-6B/6C</td></tr>
<tr><td></td><td>BOP</td><td>2-EOI-Appendix-6A, injection Subsystems Lineup Condensate

[1] ENSURE CLOSED the following Feedwater Heater Return Valves:
• 2-FCV-3-71, HP HTR 2A1 LONG CYCLE TO CONDENSER
• 2-FCV-3-72, HP HTR 2B1 LONG CYCLE TO CONDENSER
• 2-FCV-3-73, HP HTR 2C1 LONG CYCLE TO CONDENSER
[2] ENSURE CLOSED the following RFP Discharge Valves (N/A if Condensate Flood up is anticipated per the FSSs):
• 2-FCV-3-19, RFP 2A DISCHARGE VALVE
• 2-FCV-3-12, RFP 2B DISCHARGE VALVE
• 2-FCV-3-5, RFP 2C DISCHARGE VALVE</td></tr>
</table>

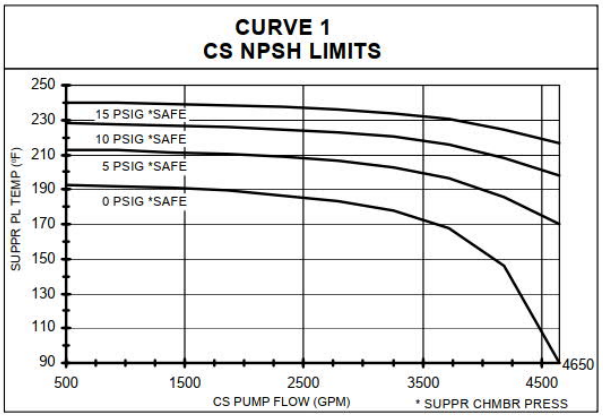
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 7 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE OPEN** the following Drain Cooler Inlet Valves: • 2-FCV-2-72, DRAIN COOLER 2A5 CONDENSER INLET ISOLATION VALVE • 2-FCV-2-84, DRAIN COOLER 2B5 CONDENSER INLET ISOLATION VALVE • 2-FCV-2-96, DRAIN COOLER 2C5 CONDENSER INLET ISOLATION VALVE [4] **ENSURE OPEN** the following Heater Outlet Valves: • 2-FCV-2-124, LP HEATER 2A3 CONDENSER OUTLET ISOLATION VALVE • 2-FCV-2-125, LP HEATER 2B3 CONDENSER OUTLET ISOLATION VALVE • 2-FCV-2-126, LP HEATER 2C3 CONDENSER OUTLET ISOLATION VALVE [5] **ENSURE OPEN** the following Heater Isolation Valves: • 2-FCV-3-38, HP HTR 2A2 FW INLET ISOLATION VALVE • 2-FCV-3-31, HP HTR 2B2 FW INLET ISOLATION VALVE • 2-FCV-3-24, HP HTR 2C2 FW INLET ISOLATION VALVE • 2-FCV-3-75, HP HTR 2A1 FW OUTLET ISOLATION VALVE • 2-FCV-3-76, HP HTR 2B1 FW OUTLET ISOLATION VALVE • 2-FCV-3-77, HP HTR 2C1 FW OUTLET ISOLATION VALVE [6] **ENSURE OPEN** the following RFP Suction Valves: • 2-FCV-2-83, RFP 2A SUCTION VALVE • 2-FCV-2-95, RFP 2B SUCTION VALVE • 2-FCV-2-108, RFP 2C SUCTION VALVE [7] **ENSURE** all available Condensate Pumps running. [8] **ENSURE** all available Condensate Booster Pumps running. [9] **ADJUST** 2-LIC-3-53, RFW START-UP LEVEL CONTROL, on Panel 2-9-5, to control injection (N/A if Condensate Flood up is anticipated per the FSSs). [10] **CHECK** RFW Flow to RPV (N/A if Condensate Flood up is anticipated per the FSSs).

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 8 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Only one EOI Appendix for Core Spray Injection (2-EOI-Appendix-6D) and one for RHR Injection (2-EOI-Appendix-6B) is covered below, as System I and System II procedures are the same (with different valve numbers).**
	BOP	2-EOI-Appendix-6D, Injection System Lineup Core Spray System I [1] **VERIFY OPEN** the following valves: • 2-FCV-75-2, CORE SPRAY PUMP 2A SUPPRESSION POOL SUCTION VALVE • 2-FCV-75-11, CORE SPRAY PUMP 2C SUPPRESSION POOL SUCTION VALVE • 2-FCV-75-23, CORE SPRAY SYS I OUTBOARD INJECT VALVE [2] **VERIFY CLOSED** 2-FCV-75-22, CORE SPRAY SYS I TEST VALVE. [3] **VERIFY** CS Pump 2A and/or 2C running. [4] **WHEN** RPV Pressure is below 450 psig, **THEN THROTTLE** 2-FCV-75-25, CORE SPRAY SYS I INBD INJECT VALVE, as necessary to control injection at or below 4000 gpm per pump [5] **MONITOR** Core Spray Pump NPSH using Attachment 1. **NPSH MONITORING** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. 

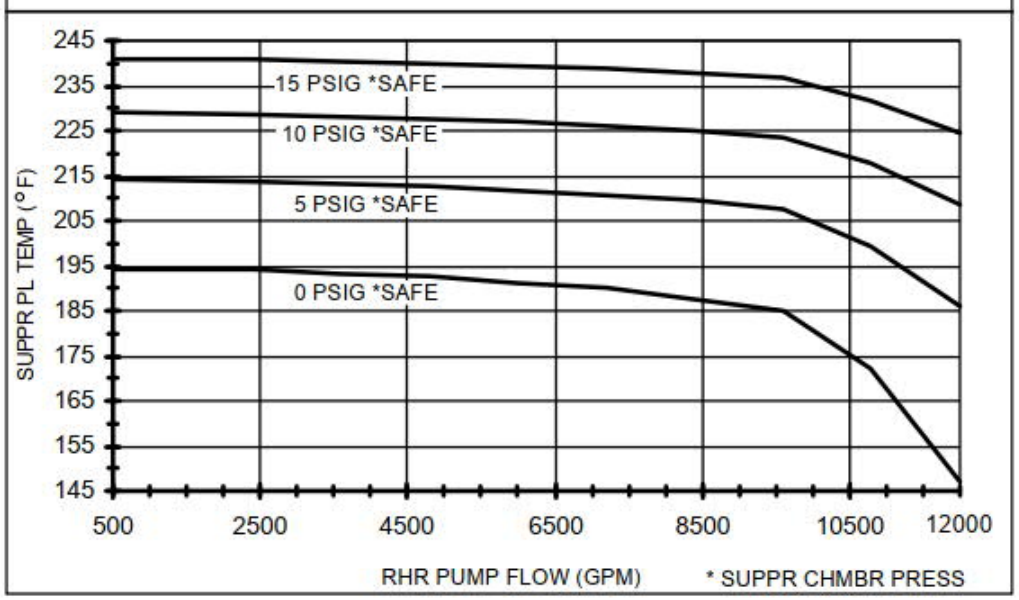
Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 9 of 10
Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI-Appendix-6B, Injection Subsystems Lineup RHR System I LPCI Mode [1] **IF** Adequate Core Cooling is assured AND it becomes necessary to bypass LPCI Injection Valve auto open signal to control injection, **THEN PLACE** 2-HS-74-155A, LPCI SYSTEM-I OUTBOARD INJECTION VALVE BYPASS SELECT, in BYPASS. [2] **ENSURE OPEN** 2-FCV-74-7, RHR SYSTEM I MIN FLOW VALVE. [3] **ENSURE OPEN** the following valves: • 2-FCV-74-1, RHR PUMP 2A SUPPRESSION POOL SUCT VALVE • 2-FCV-74-12, RHR PUMP 2C SUPPRESSION POOL SUCT VALVE [4] **ENSURE CLOSED** the following valves: • 2-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE • 2-FCV-74-60, RHR SYSTEM I DRYWELL SPRAY OUTBOARD VALVE • 2-FCV-74-57, RHR SYSTEM I SUPPRESSION CHAMBER/POOL ISOLATION VALVE • 2-FCV-74-58, RHR SYSTEM I SUPPRESSION CHAMBER SPRAY VALVE • 2-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE [5] **ENSURE** RHR Pump 2A and / or 2C running. [6] **WHEN** RPV Pressure is below 450 psig, **THEN ENSURE OPEN** 2-FCV-74-53, RHR SYS I LPCI INBOARD INJECT ION VALVE. [7] **IF** RPV Pressure is below 230 psig, **THEN ENSURE CLOSED** 2-FCV-68-79, RECIRC PUMP 2B DISCHARGE VALVE. [8] **THROTTLE** 2-FCV-74-52, RHR SYS I LPCI OUTBOARD INJECTION VALVE, as necessary to control injection. [9] **MONITOR** RHR Pump NPSH using Attachment 1. [10] **PLACE** RHRSW pumps in service as soon as possible on ANY RHR Heat Exchangers discharging to the RPV.

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 10 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[11] **THROTTLE** the following in-service RHRSW Outlet Valves to maintain 1700 to 4500 gpm flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE [12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers. **NPSH MONITORING** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. **CURVE 2** **RHR NPSH LIMITS** 
	NRC	**When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF – (-) 162 inches), end of Scenario.**

Scenario Setup
UNIT 2

IC	28
Exam IC	288

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-47A	53	ARP-9-4B	51	TRM-3.4.1	21
AOI-70-1	34	ARP-9-4C	12	TS-3.8.1	307
OI-68	168	ARP-9-23B	35	TS-3.1.3	253
AOI-68-1	39	ARP-9-7B	39	TS-3.1.5	253
AOI-100-1	121	ARP-9-7A	35		
EOI-1	21	ARP-9-5A	63		
EOI-2	18	ARP-9-4A	52		
EOI-APP-17C	19	ARP-9-8B	19		
EOI-APP-17B	18	EOI-APP-6A	6		
EOI-APP-5C	8	EOI-APP-6B	12		
EOI-APP-5B	4	EOI-APP-6D	8		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Danger Tags on HPCI Hang Danger Tags on 2B PSC Head Tank Pump Hang Protected Equipment tags on the following: • RCIC • Loop I and II of Core Spray • Loop I and II of RHR • 4KV Shutdown Boards A, B, C, and D
Schedule Files(s):	ILT 2304 NRC Scenario 4 UNIT 2.sch
Event Files(s):	ILT 2304 NRC Scenario 4 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

MODE 1	DAYS ON LINE	Drywell Leakage (GPM)	Today
	230	1.89	Protected Equipment
Rx Power	PRA (Phoenix) -Green	Floor Drain (GPM)	RCIC
100%	500Kv GRID - Qualified	0.31	Loop I and II of Core Spray
MWe	161Kv Grid -Qualified	Equipment Drain (GPM)	Loop I and II of RHR
1307		1.58	4KV Shutdown Boards A, B, C, and D

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

On Day 5 of a planned 10-day HPCI outage – LCO 3.5.1.C – 14 days

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2B PSC Head Tank Pump coupling failure

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Alternate EHC Pumps due to high filter D/P
Support HPCI Outage
Support repair of 2B PSC Head Tank Pump
Continue PMT on 2B Bus Duct Fan

OPERATOR WORK AROUNDS **OWAs – U0-1/U3-0** **Burdens - U0 – 0/U3-0** **Operator Challenges – U0 – 12/U3 - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-4
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 100% Reactor Power, HPCI is tagged for a scheduled outage, PSC Head Tank Pump 3B is tagged for coupling repair.

Turnover: Alternate EHC Pumps due to high filter D/P. 3B Bus Duct fan is running for PMT. Support HPCI outage.

Critical Tasks:
1. When Reactor Water Level cannot be restored and maintained above (-) 162 inches, and before Reactor Water Level reaches (-) 180 inches, Emergency Depressurize the Reactor.
2. With a primary system discharging into the Primary Containment, initiate Drywell Sprays before Drywell Temperature exceeds 280ºF to prevent reaching component qualification and structural design limits.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Alternate EHC Pumps
2.	SW02A	C-OATC C-NUSO MC-OATC	Reactor Building Closed Cooling Water (RBCCW) Pump Trip
3.	DG01B	TS-NUSO	3B Diesel Generator Logic Breaker Trip
4.	TU02A-M	R-OATC	Main Turbine Vibration
5.	EG13B	C-BOP C-NUSO	Bus Duct Fan Trip
6.	RD08R3043	TS-NUSO	Control Rod Accumulator INOPERABLE
7.	TH07A	C-OATC C-NUSO	3A Recirculation Pump Cell Bypass
8.	ED07B	C-BOP C-NUSO MC-BOP	3B 480V Unit Board Trip
9.	TH21	M-ALL	Unisolable Recirc Piping Leak
10.	RP11	C-BOP C-NUSO	Reactor Feedwater Pumps Trip

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will alternate EHC Pumps in accordance with 3-OI-47A, EHC System.
2. 3A Reactor Building Closed Cooling Water (RBCCW) Pump will trip, and 3-FCV-70-48, RBCCW SECTIONALIZING VALVE, will fail to automatically close, requiring the crew to take action in accordance with Alarm Response Procedures and 3-AOI-70-1, Loss of Reactor Building Closed Cooling Water. The crew must take manual action to close 3-FCV-70-48.
3. The 3B Emergency Diesel Generator (EDG) Logic Breaker will trip, causing an annunciation in the Control Room and disabling 3B EDG. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
4. The Main Turbine will experience a high vibration condition. The crew will respond in accordance with Alarm Response Procedures, and reduce Reactor Power. Following the Reactor Power reduction, Main Turbine vibrations levels will lower, precluding further action by the crew.
5. 3B Bus Duct Fan will trip and the crew will respond in accordance with Alarm Response Procedure 3-ARP-9-7A, Window 31, Gen Bus Duct Fan Failure, and start the standby Bus Duct Fan.
6. Control Rod 30-43 Accumulator Pressure will lower below 940 psig, requiring the NUSO to address Technical Specifications.
7. 3A Recirculation Pump Variable Frequency Drive (VFD) Cell C5 will bypass, requiring the crew to take action in accordance with 3-AOI-68-1, Recirc Pump Trip/Core Flow Decrease.
8. 3B 480V Unit Board will trip, and as a result 3B EHC Pump will trip, requiring the crew to respond in accordance with Alarm Response Procedures. 3A EHC Pump will fail to automatically start, requiring the crew to manually start 3A EHC Pump.
9. An unisolable Recirculation System piping leak will occur in the Drywell. The crew may respond in accordance with 3-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell, and will be forced to insert a manual Reactor SCRAM. The NUSO will respond in accordance with 3-EOI-2, Primary Containment Control. Reactor Water Level will not be able to be maintained above (-) 180 inches, requiring the crew to Emergency Depressurize the Reactor.
10. Following the Reactor SCRAM, the Reactor Feedwater Pumps will trip, requiring the crew to take action to provide high-pressure injection to the Reactor.

The Scenario ends when the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available low-pressure injection system.

Critical Tasks 2

1. **When Reactor Water Level cannot be restored and maintained above (-) 162 inches, and before Reactor Water Level reaches (-) 180 inches, Emergency Depressurize the Reactor.**

   a. **Safety Significance**
   Maintain Adequate Core Cooling and prevent degradation of fission product barrier.

   b. **Cues**
   Procedural Compliance.
   Reactor Water Level trend.

   c. **Measured by:**
   Observation – MRSVs are opened when Reactor Water Level cannot be restored and maintained above (-) 180 inches.

   d. **Feedback**
   RPV Pressure trend.
   MSRV open status indicators.

   e. **Critical Task Failure Criteria:**
   Emergency Depressurization is not directed when Reactor Water Level cannot be restored and maintained above (-) 180 inches.

2. **With a primary system discharging into the Primary Containment, initiate Drywell Sprays before Drywell Temperature exceeds 280ºF to prevent reaching component qualification and structural design limits.**

   a. **Safety Significance**
   Precludes failure of Primary Containment.

   b. **Cues**
   Procedural Compliance.
   Drywell Temperature trend.

   c. **Measured by:**
   Observation – Drywell Sprays are initiated prior to exceeding 280ºF Drywell Temperature.

   d. **Feedback**
   Lowering Drywell Temperature.
   RHR System Flow to containment cooling/sprays.

   e. **Critical Task Failure Criteria:**
   Drywell sprays are not directed before Drywell Temperature reaches 280ºF.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 1 Page 1 of 2

Event Description: Alternate EHC Pumps

Symptoms/Cues: Crew is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**The crew has briefed on the procedure to alternate EHC Pumps, but may hold a re-focus brief prior to commencing the procedure.**
	NUSO	Directs the Balance of Plant Operator (BOP) to alternate EHC Pumps in accordance with 3-OI-47A, EHC System, Section 6.3.
	Driver	**If contacted as the Turbine Building Assistant Unit Operator (AUO), acknowledge any information or direction given.**
	BOP	3-OI-47A, EHC System Section 6.3, EHC Pump Alternation [1] **ENSURE** the EHC System is in service. **REFER TO** Section 5.1. [2] **REVIEW** the Precautions and Limitations listed in Section 3.0. [3] **START** the non-operating EHC Pump 3B using 3-HS-47-2A, EHC HYDRAULIC FLUID PUMP 3B. **NOTE** **ALLOW** both EHC Pumps to operate for at least 30 seconds to allow the standby pump to expel any air which may have accumulated in the pump casing. [4] **CHECK** the started EHC Pump 3-PI-047-0002, EHC HYDRAULIC PUMP B DISCHARGE PRESSURE, indicates between 1550 psig and 1750 psig, locally at the EHC skid.
	Driver	**If contacted as the Turbine Building AUO to report EHC B Discharge Pressure, report that pressure is 1600 psig.**
	BOP	[5] N/A [6] **STOP** EHC PUMP 3A using 3-HS-47-1A, EHC HYDRAULIC FLUID PUMP 3A.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 1 Page 2 of 2

Event Description: Alternate EHC Pumps

Time	Position	Applicant's Actions or Behavior
	BOP	[7] **CHECK** EHC HEADER PRESSURE, 3-PI-47-7, indicates between 1550 psig and 1650 psig [8] N/A [9] N/A [10] **CHECK** the started EHC PUMP 3-EI-47-2, 3B MOTOR AMPS, indicates less than 140 amps. [11] **CHECK** EHC Pump 3B, EHC PUMP SUCTION FILTER and EHC AUX PUMP SUCTION FILTERS, indicate CLEAN, locally at the EHC Skid. [12] **ENSURE** 3-047-0588, MANUAL FLOW CONTROL VALVE, is set at 2.0 turns clockwise.
	Driver	**If contacted as the Turbine Building AUO to perform steps [11] and [12], report that the EHC Pump Suction Filters indicate clean at the EHC Skid and the Manual Flow Control Valve is set at 2 turns clockwise.**
	NRC	**End of Event 1. Request that the Driver insert Event 2, Reactor Building Closed Cooling Water (RBCCW) Pump Trip.** **EXAMINER NOTE: For Event 2, the BALANCE OF PLANT OPERATOR (BOP) MUST BE REMOVED FROM THE CONTROL ROOM TO ALLOW THE OPERATOR AT THE CONTROLS (OATC) TO RESPOND TO THE NEXT EVENT.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 2 Page 1 of 3

Event Description: Reactor Building Closed Cooling Water (RBCCW) Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3A RBCCW Pump will trip, and 3-FCV-70-48, RBCCW SECTIONALIZING VALVE, will not automatically close.

Time	Position	Applicant's Actions or Behavior
	Driver	**CONTACT THE BOP AS REQUESTED BY THE CHIEF EXAMINER WITH A REASON TO LEAVE THE CONTROL ROOM.** **VERIFY THAT THE BOP HAS BEEN REMOVED FROM THE CONTROL ROOM PRIOR TO INSERTING EVENT 2.** **When requested by the Chief Examiner and the BOP is not in the Control Room, insert Event 2 to trip 3A Reactor Building Closed Cooling Water (RBCCW) Pump Trip.**
	OATC	Acknowledges and reports the following alarms: • RBCCW PUMP DISCHARGE HEADER PRESSURE LOW, 3-9-4C, Window 12 • RWCU DEMINERALIZER ABNORMAL, 3-9-4B, Window 24
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response Procedures and 3-AOI-70-1, Loss of Reactor Building Closed Cooling Water.
	OATC	3-ARP-9-4C, Alarm Response Procedure RBCCW PUMP DISCHARGE HEADER PRESSURE LOW, Window 12 Operator Action: A. **CHECK** 3-FCV-70-48 CLOSING/CLOSED.
	OATC	Determines that 3-FCV-70-48 did not automatically close, takes action to manually close it. Notifies the Nuclear Unit Senior Operator.
	OATC	B. **CHECK** RBCCW Pumps A and B in service. C. **CHECK** RBCCW Surge Tank Low Level Alarm is reset. D. **DISPATCH** personnel to check the following: • RBCCW Surge Tank Level locally • RBCCW Pumps for proper operation
	Driver	**If contacted as the Reactor Building AUO, acknowledge any direction or information given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 2 Page 2 of 3

Event Description: Reactor Building Closed Cooling Water (RBCCW) Pump Trip

Time	Position	Applicant's Actions or Behavior
	OATC	E. **REFER TO** 3-AOI-70-1, Loss of Reactor Building Closed Cooling Water, for RBCCW System failure and 3-OI-70, Reactor Building Closed Cooling Water System, for starting spare pump.
	NRC	**EXAMINER NOTE: Once the OATC has manually closed 3-FCV-70-48, the BOP can be allowed to re-enter the Control Room.**
	OATC	3-AOI-70-1, Loss of Reactor Building Closed Cooling Water 4.1 Immediate Actions [1] **IF** RBCCW Pump(s) has tripped or 3-FCV-70-48, RBCCW SECTIONALIZING VALVE, is closed/closing due to low pressure, **THEN PERFORM** the following (Otherwise N/A): • SECURE RWCU Pumps • ENSURE RBCCW SECTIONALIZING VALVE, 3-FCV-70-48 CLOSED [2] N/A [3] N/A [4] **IF** unable to restart a tripped pump, **THEN PLACE** Spare RBCCW Pump in service in accordance with 3-OI-70.
	Driver	**If contacted as Unit 1 to place the spare RBCCW Pump in service to Unit 3, wait 1 minute, insert Event 12 to align the spare RBCCW Pump to Unit 3, and inform the crew that the spare RBCCW Pump is aligned to Unit 3.**
	OATC	[5] **IF** RBCCW flow was restored to two pump operation, **THEN PERFORM** the following (Otherwise N/A): [5.1] **REOPEN** 3-HS-70-48A. RBCCW SECTIONALIZING VALVE. [5.2] **RESTORE** the Reactor Water Cleanup (RWCU) System to operation. (**REFER TO** 3-OI-69, Reactor Water Cleanup System)
	NRC	**EXAMINER NOTE: It is not expected that the crew return RWCU to service.**
	NUSO	When RWCU is isolated, addresses Technical Requirements Manual 3.4.1, Coolant Chemistry Surveillance Requirements.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 2 Page 3 of 3

Event Description: Reactor Building Closed Cooling Water (RBCCW) Pump Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Requirement 3.4.1 – Coolant Chemistry TSR 3.4.1.1	
	NUSO	**SURVEILLANCE:** ----------NOTE---------- Not required when there is no fuel in the Reactor vessel. ---------- Monitor Reactor Coolant Conductivity	**FREQUENCY:** Continuously OR 4 hours when the continuous conductivity monitor is inoperable and the reactor is not in MODE 4 or 5 OR 8 hours when the continuous conductivity monitor is inoperable and the reactor is in MODE 4 or 5
	NRC	**End of Event 2. Request that the driver insert Event 3, 3B Diesel Generator Logic Breaker Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 3 Page 1 of 3

Event Description: 3B Diesel Generator Logic Breaker Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3B Diesel Generator Logic Breaker will open, resulting in the DG being inoperable.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to open the 3B Diesel Generator (DG) Logic Breaker.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • DIESEL GENERATOR 3B CONTROL POWER OFF, 3-9-23B, Window 14
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	3-ARP-9-23B, Alarm Response Procedure DIESEL GENERATOR 3B CONTROL POWER OFF, Window 14 Operator Action: A. **OBSERVE** any other alarms on panels 9-8 or 9-23 which may indicate problem area. B. **CHECK** panels, breakers and batteries. **IF** necessary, **THEN CHECK** fuses and relays. C. **IF** loss of normal power has occurred, **THEN TRANSFER** to alternate power source per 0-OI-57D, DC Electrical System. D. **REFER TO** Tech Spec Sections 3.8.1 and 3.8.2.
	Driver	**If contacted as an AUO, acknowledge any information or direction given.** **If contacted as the Work Control / Outside NUSO or Electrical Maintenance, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 3 Page 2 of 3

Event Description: 3B Diesel Generator Logic Breaker Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.8.1 – AC Sources – Operating LCO 3.8.1 - The following AC electrical power sources shall be OPERABLE: a. Two qualified circuits between the Offsite transmission network and the Onsite Class 1E AC Electrical Power Distribution System; b. Unit 3 Diesel Generators (DGs) with two divisions of 480 V Load Shed Logic and Common Accident Signal Logic OPERABLE; and c. Unit 1 and 2 DG(s) capable of supplying the Unit 1 and 2 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating." Applicability: Modes 1, 2, and 3.	
		CONDITION B. One required unit 3 DG INOPERABLE.	
	NUSO	**REQUIRED ACTION:** B.1 Verify power availability from the offsite transmission network. AND B.2 Evaluate the availability of both temporary Diesel Generators B.3 Declare required feature(s), supported by the inoperable Unit 3 DG, inoperable when the redundant required feature(s) are inoperable. AND (Continued below)	**COMPLETION TIME:** B.1 – 1 hour AND Once every 8 hours thereafter. B.2 – 1 hour AND Once per 12 hours thereafter. B.3 – 4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s).

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 3 Page 3 of 3

Event Description: 3B Diesel Generator Logic Breaker Trip

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION:** B.4.1 Determine OPERABLE Unit 3 DG(s) are not inoperable due to common cause failure. OR B.4.2 Perform SR 3.8.1.1 for OPERABLE Unit 3 DG(s). AND B.5 Restore Unit 3 DG to OPERABLE status.	**COMPLETION TIME:** B.4.1 – 24 hours B.4.2 – 24 hours B.5 – 7 days from discovery of unavailability of TDG(s) AND 24 hours from discovery of Condition B entry > 6 days concurrent with unavailability of TDG(s) AND 14 days AND 21 days from discovery of failure to meet LCO
	NRC	**In the interest of time management, if the NUSO directs the crew to perform Surveillance Requirement 3.8.7.1, Monthly Check of Power Availability, inform NUSO that another operator will perform the required power availability check.**	
	NRC	**End of Event 3. Request that the Driver insert Event 4, Main Turbine Vibration.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 4 Page 1 of 2

Event Description: Main Turbine Vibration

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Main Turbine vibration will rise to the alarm setpoint.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to cause Main Turbine vibrations to rise.**
	BOP	Acknowledges and reports the following alarm to the NUSO: • MAIN TURBINE VIBRATION HIGH, 3-9-7B, Window 32
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	3-ARP-9-7B, Alarm Response Procedure Operator Action: A. **CHECK** the following: • On EHC WORKSTATION, Turbine Vibration screen • On ICS, MAIN TURBINE BEARINGS (TURBBRG) screen • TURBINE GENERATOR VIBRATION recorder, 3-XR-47-15 (Panel 3-9-7) • Computer points 47-15A thru 47-15M B. **IF** alarm is valid, **THEN** 1. **DETERMINE** cause by checking PROBABLE CAUSE section above. 2. **REDUCE** load and OBSERVE vibration. 3. N/A 4. N/A
	NUSO	Directs the OATC to reduce Reactor Power in an attempt to reduce Main Turbine vibration.
	NRC	**The crew may reduce Reactor Power by any method in 3-OI-68, Reactor Recirculation System.** **When the crew reduces Reactor Power less than 99.5%, Main Turbine vibrations will lower below the alarm setpoint.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 4 Page 2 of 2

Event Description: Main Turbine Vibration

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow [1] N/A [2] N/A [3] **WHEN** desired to control Recirc Pumps 3A and/or 3B Speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump Speed 3A & 3B using the following push buttons as required: • 3-HS-96-33, **LOWER** SLOW • 3-HS-96-34, **LOWER** MEDIUM • 3-HS-96-35, **LOWER** FAST
	BOP	When Reactor Power is lowered, reports that Main Turbine Vibration levels have returned to normal values.
	NRC	**End of Event 4. Request that the Driver insert Event 5, Bus Duct Fan Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 5 Page 1 of 2

Event Description: Bust Duct Fan Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3B Bus Duct Fan will trip.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause 3B Bus Duct Fan to trip.**
	BOP	Acknowledges and reports the following alarm: • GENERATOR BUS DUCT FAN FAILURE, 3-9-7A, Window 31
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	BOP	3-ARP-9-7A, Alarm Response Procedure GENERATOR BUS DUCT FAN FAILURE, Window 31 Operator Action: A. **CHECK** Main Bus Cooling Fans, 3-HS-262-1A or 3-HS-262-2A, indicates running on Panel 3-9-8. B. **START** 3-HS-262-1A, GEN BUS DUCT HX FAN A using on Panel 3-9-8 to start the standby fan.
	BOP	Starts 3A Bus Duct Fan and informs the NUSO that the Standby Bus Duct Fan is running.
	BOP	C. N/A D. N/A E. **DISPATCH** personnel as necessary to check the following: 1. Main Bus Cooling Fan on elevation 586' to check fan condition. 2. Monitor Bus Duct Temperature by available means including using a portable temperature monitor device locally at the 14 in-service thermostats. **REFER** to Window 32, Figure 1. 3. 480V Unit Board 3A on elevation 586' to check breaker 5C closed. 4. 480V Unit Board 3B on elevation 604' to check breaker 5C closed. F. **ENSURE** the system is operating in accordance with 3-OI-47.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 5 Page 2 of 2

Event Description: Bus Duct Fan Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as the Turbine Building AUO, acknowledge any information or direction given.** **If contacted as the Work Control / Outside NUSO or Electrical Maintenance, acknowledge any information or direction given.**
	NRC	**End of Event 5. Request that the Driver insert Event 6, Control Rod Accumulator INOPERABLE.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6 Page 1 of 4

Event Description: Control Rod Accumulator INOPERABLE

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. The CRD Accumulator Low Pressure Alarm will be received for Control Rod Accumulator 30-43.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the NRC, insert Event 6 to cause Control Rod 30-43 Accumulator low pressure.**
	OATC	Acknowledges and reports the following alarm: • CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, 3-9-5A, Window 29
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure.
	OATC	3-9-5A, Alarm Response Procedure CRD ACCUMULATOR PRESSURE LOW / LEVEL HIGH, Window 29 Operator Action: A. **CHECK** alarm by amber background light illuminated on Full Core Display. B. **LOG** in the narrative log the Control Rod number and time alarm was received. **NOTE** If any of the following fuses is/are cleared, the local indications at Panel 25-4 and 25-22 will NOT illuminate. C. **IF** multiple accumulator lights are lit on Panel 3-9-5, **THEN CHECK** for cleared fuses 3-FU1-085-25-004G, -004H in Panel 25-4 and 3-FU1-085-25-022G, -022H in Panel 25-22. D. **DISPATCH** personnel to Panel 25-4 (east side), Panel 25-22 (west side) El 565', to determine if level high or pressure low. E. **DEPRESS** push-button for associated HCU to determine if alarm is caused by level high or pressure low as follows: 1. If alarm is due to high level, the red light will extinguish. 2. If light stays illuminated, alarm is due to low N2 pressure.

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6 Page 2 of 4

Event Description: Control Rod Accumulator INOPERABLE

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as the Reactor Building AUO to respond to the Accumulator Alarm, acknowledge any direction or information given. Wait two minutes and report to the crew that Control Rod 30-43 Accumulator Pressure is 930 psig and slowly lowering. The N2 charging bottle is empty and you are working to get another N2 bottle to recharge Accumulator 30-43.**
	OATC	F. **IF** alarm is valid, **THEN REFER TO** 3-OI-85, Control Rod Drive System and 3-AOI-85-3, CRD System Failure.
	OATC	**NOTE** If accumulator pressure is greater than 940 psig the accumulator is not required to be declared Inoperable when the "star" valve is CLOSED, unless accumulator is unattended. G. **IF** Accumulator Pressure is less than or equal to 940 psig, **THEN DECLARE** Control Rod HCU "INOPERABLE". H. **IF** the associated HCU's nitrogen pressure is found less than 940 psi, **THEN INITIATE** a Condition Report (CR) to calibrate the pressure switch. The HCU will NOT be declared operable until the switch has been calibrated. I. **IF** alarm is due to low pressure with pressure greater than 940 psig and accumulator CANNOT be recharged within one hour, **THEN EVALUATE** per Tech Spec 3.1.5. 1. **IF** the Control Rod is declared SLOW **REFER TO** TECH SPEC 3.1.4. Currently no more than 13 OPERABLE Control Rods shall be slow and no more than 2 OPERABLE Control Rods that are slow shall occupy adjacent locations. 2. **IF** the Control Rod is declared INOPERABLE, **THEN REFER** TO TECH SPEC 3.1.3. J. **RECORD** this evaluation in narrative log. K. N/A L. N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6 Page 3 of 4

Event Description: Control Rod Accumulator INOPERABLE

Time	Position	Applicant's Actions or Behavior	
	NUSO	Technical Specification 3.1.5, Control Rod SCRAM Accumulators LCO 3.1.5: Each Control Rod SCRAM Accumulator shall be OPERABLE Applicability: Modes 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod SCRAM Accumulator. Condition: A.1 One Control Rod SCRAM Accumulator inoperable with Reactor Steam Dome Pressure ≥ 900 psig.	
	NUSO	**REQUIRED ACTION:** A.1 NOTE: Only applicable if the associated Control Rod SCRAM time was within the limits of Table 3.1.4-1 during the last SCRAM time Surveillance. Declare the associated Control Rod SCRAM Time "slow". OR A.2- Declare the associated Control Rod INOPERABLE.	**COMPLETION TIME:** A.1 – 8 hours A.2 – 8 hours
	NRC	**If Tech Spec 3.1.5 A.2 is Declared, THEN Tech Spec 3.1.3 is applicable.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 6

Event Description: Control Rod Accumulator INOPERABLE

Time	Position	Applicant's Actions or Behavior	
	NUSO	If the Control Rod is declared INOPERABLE from Tech Spec 3.1.5.A.2, references Tech Spec 3.1.3, Control Rod OPERABILITY. LCO 3.1.3 Each Control Rod shall be OPERABLE APPLICABILITY: MODES 1 and 2 NOTE: Separate Condition entry is allowed for each Control Rod. **CONDITION:** C. One or more Control Rods INOPERABLE for reasons other than Condition A or B.	
	NUSO	**REQUIRED ACTION:** NOTE: RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of INOPERABLE Control Rod and continued operation. C.1 – Fully insert INOPERABLE Control Rod. AND C.2 – Disarm the associated CRD.	**COMPLETION TIME:** C.1 – 3 hours C.2 – 4 hours
	NUSO	If the Control Rod is declared SLOW, references Tech Spec 3.1.4, Control Rod SCRAM Times. Currently, there are less than 13 OPERABLE Control Rods that are slow, and there are no OPERABLE Control Rods that are slow that occupy adjacent locations. Therefore, there are no actions required for Tech Spec 3.1.4, Control Rod SCRAM Times.	
	NRC	**End of Event 6. Request that the driver insert Event 7, 3A Recirculation Pump Cell Bypass.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 7 Page 1 of 3

Event Description: 3A Recirculation Pump Cell Bypass

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3A recirculation Pump VFD Cell C5 will bypass, causing 3A Recirculation Pump Speed to lower by about 200 rpm

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to cause 3A Recirculation Pump Cell C5 to bypass.**
	OATC	Acknowledges and reports the following alarm: • RECIRC DRIVE 3A DRIVE ALARM, 3-9-4A, Window 32
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure and 3-AOI-68-1, Recirc Pump Trip / Core Flow Decrease.
	OATC	3-ARP-9-4A, Alarm Response Procedure RECIRC DRIVE 3A DRIVE ALARM, Window 32 Operator Action: A. **REFER** TO ICS Group Display "GD @VFDADA" B. **DETERMINE** cause of alarm. C. N/A D. N/A E. N/A F. N/A G. N/A H. **NOTIFY** System Engineering for all other alarms or any problems encountered.
	Driver	**If contacted as the Work Control / Outside NUSO, an AUO, or System Engineering acknowledge any direction or information given.**
	OATC	3-AOI-68-1, Recirc Pump Trip / Core Flow Decrease. 4.1 Immediate Actions [1] N/A 4.2 Subsequent Actions [1] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 7 Page 2 of 3

Event Description: 3A Recirculation Pump Cell Bypass

Time	Position	Applicant's Actions or Behavior
	OATC	[2] N/A [3] **IF** a VFD Cell(s) has bypassed, **THEN** with Shift Manager AND Reactor Engineering concurrence, **RAISE** speed of the affected VFD to approximately the original speed. (REFER to 3-OI-68) (otherwise N/A) [3.1] **IF** the affected VFD cannot be restored to approximately the original speed, **THEN PERFORM** 3-SR-3.4.2.1, Jet Pump Mismatch and Operability. (otherwise N/A)
	Driver	**If contacted as the Reactor Engineer or Shift Manager, concur with raising 3A Recirc Pump speed to match 3B Recirc Pump Speed. If asked as the Reactor Engineer, no RCP is required, there are no ramp rate limits, and Reactor Engineering will monitor Thermal Limits.**
	OATC	[4] **REFERENCE** the following ICS screens to help determine the cause of Recirc Pump trip/Core Flow lowering. • VFDPMPA(VFDPMPB) • VFDAAL(VFDBAL) [5] **CHECK** parameters associated with Recirc Drive and Recirc Pump/Motor 3A on ICS and 3-TR-68-71 to determine cause of trip/core flow lowering event. [6] **PERFORM** visual inspection of tripped/affected Reactor Recirc Drive. [7] **PERFORM** visual inspection of Reactor Recirc Pump Drive relay boards for relay targets.
	Driver	**If contacted as the Reactor Building AUO, acknowledge any information or direction given.**
	OATC	[8] N/A [9] **INITIATE** actions required to make the necessary repairs. [10] N/A
	NUSO	Directs the OATC to raise 3A Recirc Pump Speed in accordance with 3-OI-68, Reactor Recirculation System.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 7 Page 3 of 3

Event Description: 3A Recirculation Pump Cell Bypass

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-68, Reactor Recirculation System, Section 6.2 Adjusting Recirc Flow [1] **IF** desired to control Recirc Pump 3A speed with Recirc Individual Control, **THEN PERFORM** the following; [1.1] Raise Recirc Pump 3A using, 3-HS-96-15A(15B), RAISE SLOW (MEDIUM). [1.2] N/A [2] N/A [3] N/A
	NRC	**End of Event 7. Request that the Driver insert Event 8, 3B 480V Unit Board Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 8 Page 1 of 2

Event Description: 3B 480V Unit Board Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3B Unit Board will Trip, causing a loss of electrical loads

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8 to trip 3B 480V Unit Board.**
	BOP	Acknowledges and reports the following alarms: • 480V UNIT BD 3B UV OR XFR, 3-9-8B, Window 17 • 480V TB VENT BD 3B UV OR XFR, 3-9-8C, Window 22 • 480V TURB MOV BD 3B UV OR XFR, 3-9-8C, Window 23 • 480V TURB MOV BD 3C UV OR XFR, 3-9-8C, Window 30 • EHC HYD FLUID HDR PRESS LOW, 3-9-7B, Window 1 • STANDBY EHC PUMP FAILED, 3-9-7B, Window 15
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedures.
	NRC	**Only the 3B 480V Unit Board alarm is covered below.**
	BOP	3-9-8B, Alarm Response Procedure 480V UNIT BD 3B UV OR XFR, Window 17 A. **ENSURE** automatic action has occurred. B. **INSPECT** 480V Unit Bd 3B for abnormal conditions: relay targets, smoke, burned paint, breaker position, etc. C. **REFER TO** 0-OI-57B, 480V/240V AC Electrical System to re-energize board. D. **REFER TO** appropriate OI for recovery or realignment of equipment.
	Driver	**If contacted as an AUO, Electrical Maintenance, or Work Control / Outside NUSO, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 8 Page 2 of 2

Event Description: 3B 480V Unit Board Trip

Time	Position	Applicant's Actions or Behavior
	BOP	3-ARP-9-7B, Alarm Response Procedure • EHC HYD FLUID HDR PRESS LOW, Window 1 Operator Action: A. **ENSURE** Standby EHC PUMP 3A, 3-HS-47-1A running.
	BOP	Determines that the standby EHC Pump did not start and starts 3A EHC Pump.
	BOP	B. **CHECK** EHC HEADER PRESSURE, pressure indicator, 3-PI-47-7 between 1550 and 1650 psig. C. **DISPATCH** personnel to inspect EHC Pump unit. D. N/A
	Driver	If contacted as the Turbine Building AUO, acknowledge any information or direction given.
	BOP	3-9-7B, Alarm Response Procedure STANDBY EHC PUMP FAILED, Window 15 Operator Action: A. On Panel 3-9-7: 1. **CHECK** EHC HEADER PRESSURE, 3-PI-47-7. 2. **ENSURE** 3-HS-47-2A, EHC PUMP 3B and/or 3-HS-47-1A, EHC PUMP 3A running. 3. **CHECK** EHC PUMP 3B PUMP MTR CURRENT 3-EI-47-2 and/or EHC PUMP 3A PUMP MTR CURRENT 3-EI-47-1. 4. **CHECK** lights above EHC PUMP 3A TEST pushbutton 3-HS-47-4A and EHC PUMP 3B TEST pushbutton 3-HS-47-5A. B. **DISPATCH** personnel to pumping unit to check for abnormal conditions. C. N/A
	NRC	**End of Event 8. Request that the Driver insert Event 9, Unisolable Recirc Piping Leak.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 1 of 12

Event Description: Unisolable Recirc Piping Leak

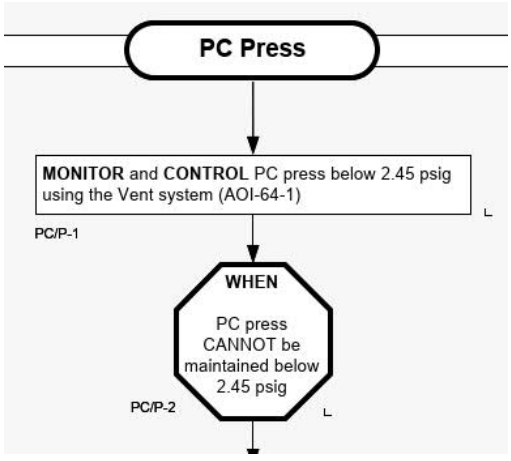
Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. A Recirculation System leak will occur in the Drywell, causing Drywell Pressure and Temperature to rise

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 9 to cause an unisolable Recirc Piping leak in the Drywell.**
	NRC	**Drywell Pressure rises quickly enough that the crew may not have time to address all alarms prior to having to insert a manual Reactor SCRAM.**
	CREW	Observes Drywell Pressure rising, acknowledges and reports Drywell Pressure alarms as they are received.
	NRC	**The NUSO may direct the OATC to insert a Core Flow Runback if there is time before Drywell Pressure reaches 2.2 psig, which is when a Reactor SCRAM will be ordered.**
	NUSO	When Drywell Pressure reaches 2.2 psig, directs the OATC to insert a Reactor SCRAM and to enter 3-AOI-100-1, Reactor SCRAM.
	NRC	**Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**
	OATC	Inserts a manual Reactor SCRAM, and enters 3-AOI-100-1. 3-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 3-HS-99-5A/S3A, REACTOR SCRAM A and 3-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 3-9-5. [2] **PLACE** 3-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] N/A
	NRC	**When the OATC places the Reactor MODE SWITCH in SHUTDOWN, Event 10, Reactor Feedwater Pumps Trip, is automatically entered. See Event 10 on page 34 of 45 for Event 10 actions.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 2 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	OATC	[4] **IF** Reactor Power is 5% or BELOW, **THEN REPORT** the following to the UNIT SRO: • Reactor SCRAM • MODE SWITCH is in SHUTDOWN • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Power level [5] N/A
	NRC	**The NUSO will enter 3-EOI-1, RPV Control Modes 1-3 if Reactor Water Level reaches +2 inches following the SCRAM.**
	BOP	When Drywell Pressure reaches 2.45 psig, informs the NUSO.
	NUSO	When informed that Drywell Pressure has reached 2.45 psig, enters 3-EOI-2, Primary Containment Control and re-enters 3-EOI-1 (if previously entered).
	NUSO	3-EOI-2, Primary Containment Control  

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 3 of 12

Event Description: Unisolable Recirc Piping Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
PC/P-3
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Primary Containment Pressure reduction is required to restore and maintain Adequate Core cooling or reduce total offsite radiation dose</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Primary Containment Pressure CANNOT be maintained above (-) 2 psig</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression Chamber Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
PC/P-4
<table>
<tr><td>BEFORE Suppression Chamber Pressure rises to 12 psig</td></tr>
</table>
PC/P-5
<table>
<tr><td>INITIATE Suppression Chamber Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17C)</td></tr>
</table>
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to augment Suppression Chamber Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table>
PC/P-6
<table>
<tr><td>WHEN Suppression Chamber Pressure exceeds 12 psig</td></tr>
</table>
</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 4 of 12

Event Description: Unisolable Recirc Piping Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>PC/P-7
IF Suppression Pool Level is below 19 ft AND Drywell Temperature is in safe area of Curve 5 THEN
1. SHUT DOWN Recirc Pumps
2. SHUT DOWN Drywell Blowers
3. INITIATE Drywell Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B, 20L)
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Needed to augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to spray the Suppression Chamber in accordance with 3-EOI-Appendix-17C, RHR System Operations Suppression Chamber Sprays.</td></tr>
<tr><td></td><td>BOP</td><td>3-EOI-Appendix-17C, RHR System Operations Suppression Chamber Sprays

[1] BEFORE Suppression Chamber pressure drops below 0 psig CONTINUE In this procedure at Step 1.0[6].
[2] IF Adequate Core Cooling is assured OR directed to spray the Suppression Chamber irrespective of Adequate Core Cooling, THEN BYPASS LPCI Injection Valve auto open signal as necessary:
• PLACE 3-HS-74-155A, LPCI SYS I OUTBD INJECTION VALVE BYPASS SELECT IN BYPASS
• PLACE 3-HS-74-155B, LPCI SYS II OUTBD INJECTION VALVE BYPASS SELECT IN BYPASS
[3] N/A
[4] N/A
[5] INITIATE Suppression Chamber Sprays as follows:
[5.1] ENSURE at least one RHRSW Pump supplying each EECW header.</td></tr>
</table>

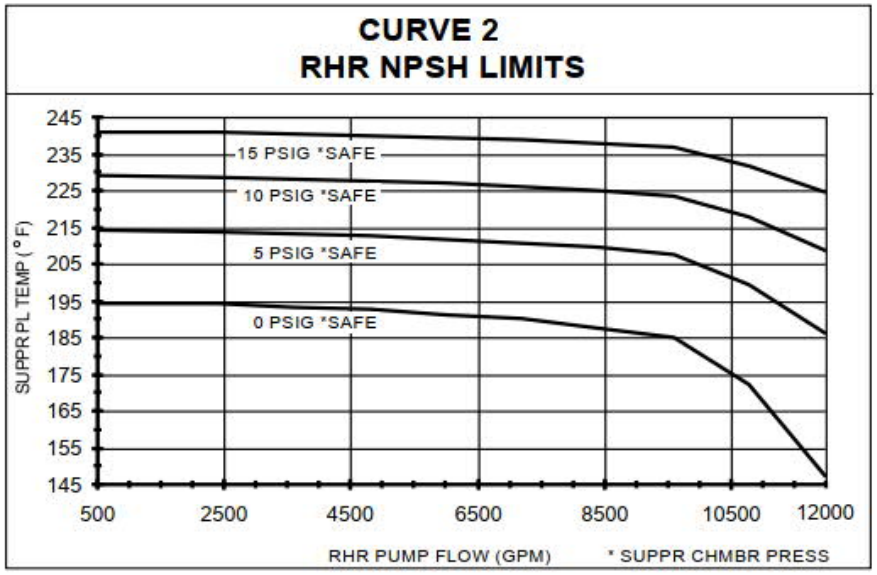
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 5 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[5.2] **IF** EITHER of the following exists: • LPCI Initiation signal is NOT present OR • Directed by SRO, **THEN** **PLACE** keylock switch 3-XS-74-122(130), RHR SYSTEM I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE. [5.3] **MOMENTARILY PLACE** 3-XS-74-121 (129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT. [5.4] **IF** 3-FCV-74-53 (67), RHR SYSTEM I (II) LPCI INBOARD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 3-FCV-74-52 (66), RHR SYSTEM I (II) LPCI OUTBOARD INJECTION VALVE. [5.5] **ENSURE** OPERATING the desired System I (II) RHR Pump(s) for Suppression Chamber Spray. [5.6] **ENSURE** OPEN 3-FCV-74-57(71), RHR SYS I(II) SUPPRESSION CHAMBER/POOL ISOLATION VALVE [5.7] **OPEN** 3-FCV-74-58 (72), RHR SYS I (II) SUPPRESSION CHAMBER SPRAY VALVE. [5.8] **IF** RHR System I(II) is operating ONLY in Suppression Chamber Spray mode, **THEN CONTINUE** in this procedure at step 1.0[5.11]. [5.9] **ENSURE** CLOSED 3-FCV-74-7 (30), RHR SYSTEM I (II) MIN FLOW VALVE. [5.10] **RAISE** system flow by placing the second RHR System I(II) Pump in service as necessary. [5.11] **MONITOR** RHR Pump NPSH using Attachment 2. [5.12] **ENSURE** RHRSW pump supplying desired RHR Heat Exchanger(s). [5.13] **THROTTLE** the following in-service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW Flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VALVE • 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VALVE • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VALVE • 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VALVE

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 6 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[5.14] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any information or direction given.**
	BOP	[6] N/A [7] N/A [8] N/A [9] N/A [10] N/A [11] N/A Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. 

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 7 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	CREW	**Critical Task: With a primary system discharging into the Primary Containment, initiate Drywell Sprays before Drywell Temperature exceeds 280ºF to prevent reaching component qualification and structural design limits.** **Critical Task Failure Criteria: Drywell sprays are not directed before Drywell Temperature reaches 280ºF.**
	BOP	3-EOI-Appendix-17B, RHR System Operation Drywell Sprays [1] **BEFORE** Drywell Pressure drops below 0 psig **CONTINUE** in this procedure at Step 1.0[7]. [2] **IF** Adequate Core Cooling is assured OR directed to spray the Drywell irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI Injection Valve auto open signal as necessary: • PLACE 3-HS-74-155A, LPCI SYS I OUTBOARD INJECTION VALVE BYPASS SELECT IN BYPASS • PLACE 3-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT IN BYPASS [3] **ENSURE** Recirc Pumps and Drywell Blowers shutdown. [4] N/A [5] N/A [6] **INITIATE** Drywell Sprays as follows: [6.1] **ENSURE** at least one RHRSW pump supplying each EECW header. [6.2] **IF** EITHER of the following exists: • LPCI Initiation signal is NOT present, OR • Directed by SRO, **THEN** **PLACE** keylock switch 3-XS-74-122(130), RHR SYS I (II) LPCI 2/3 CORE HEIGHT OVERRIDE, in MANUAL OVERRIDE. [6.3] **MOMENTARILY PLACE** 3-XS-74-121 (129), RHR SYSTEM I (II) CONTAINMENT SPRAY/COOLING VALVE SELECT, switch in SELECT.

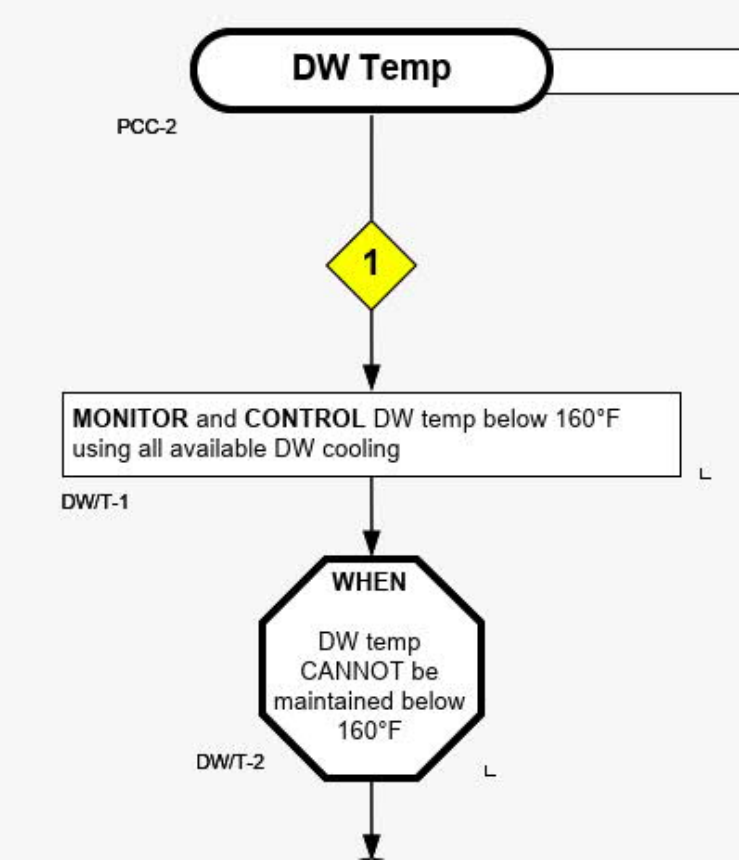
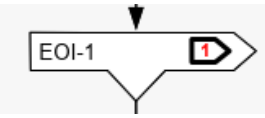
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 8 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[6.4] **IF** 3-FCV-74-53 (67), RHR SYS I (II) LPCI INBD INJECT VALVE, is OPEN, **THEN ENSURE CLOSED** 3-FCV-74-52 (66), RHR SYS I (II) LPCI OUTBD INJECT VALVE. [6.5] **ENSURE OPERATING** the desired System I (II) RHR Pump(s) for Drywell Spray. [6.6] **OPEN** the following valves: • 3-FCV-74-60(74), RHR SYSTEM I(II) DRYWELL SPRAY OUTBOARD VALVE • 3-FCV-74-61 (75), RHR SYSTEM I (II) DRYWELL SPRAY INBOARD VALVE [6.7] **ENSURE CLOSED** 3-FCV-74-7 (30), RHR SYSTEM I (II) MINIMUM FLOW VALVE. [6.8] **IF** Additional Drywell Spray Flow is necessary, **THEN PLACE** the second System I (II) RHR Pump in service. [6.9] **MONITOR** RHR Pump NPSH using Attachment 2. [6.10] ENSURE RHRSW Pump supplying desired RHR Heat Exchanger(s). [6.11] **THROTTLE** the following in-service RHRSW outlet valves to obtain 1700 to 4500 gpm RHRSW Flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VALVE • 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VALVE • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VALVE • 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VALVE [6.12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	Driver	**If contacted as Chemistry, acknowledge any information or direction given.**
	BOP	[7] N/A [8] N/A [9] N/A [10] N/A [11] N/A [12] N/A

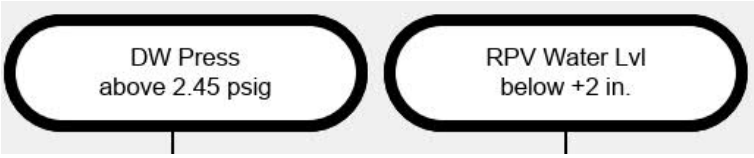
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 9 of 12

Event Description: Unisolable Recirc Piping Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	 DW/T-3 **BEFORE** Drywell Temperature rises to 280 ºF 

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 10 of 12

Event Description: Unisolable Recirc Piping Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
DW/T-4
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
DW/T-5
<table>
<tr><td colspan="2">IF Suppression Pool Level is below 19 ft AND Drywell Temperature is in safe area of Curve 5 THEN
1. SHUT DOWN Recirc Pumps
2. SHUT DOWN Drywell Blowers
3. INITIATE Drywell Sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B, 20L)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-1, RPV Control MODES 1-3
 
</td></tr>
<tr><td></td><td>NRC</td><td>See Event 10 on page 34 of 45 for 3-EOI-1 Reactor Water Level control actions.</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 11 of 12

Event Description: Unisolable Recirc Piping Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
RC/P-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>A high Drywell Pressure ECCS signal exists (2.45 psig)</td><td>PREVENT inj from ONLY those CS and LPCI pumps NOT required for RPV injection before depressurizing below 330 psig (APPX 4)</td></tr>
<tr><td>EMERGENCY RPV DEPRESSURIZATION IS REQUIRED or has been required</td><td>B</td></tr>
<tr><td>Emergency Depressurization is anticipated</td><td>RAPIDLY DEPRESSURIZE the RPV with Main Turbine Bypass Valves
➢ OK to exceed 100ºF/hr cooldown rate</td></tr>
<tr><td>Suppression Pool Temperature and Level CANNOT be maintained in a safe area of Curve 3 at the existing RPV Pressure
OR
Suppression Pool Level CANNOT be maintained in the safe are of Curve 4</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-2
<table>
<tr><td colspan="2">STABILIZE RPV PRESSURE below 1073 psig using the Main Turbine Bypass Valves (APPX 8B)
➢ OK to use ANY Alternate RPV Pressure Control Systems (Table P-1)
➢ Crosstie CAD or MSRV carts to DW Control Air (APPX 8G, 20H) if necessary</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>DW Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 9 Page 12 of 12

Event Description: Unisolable Recirc Piping Leak

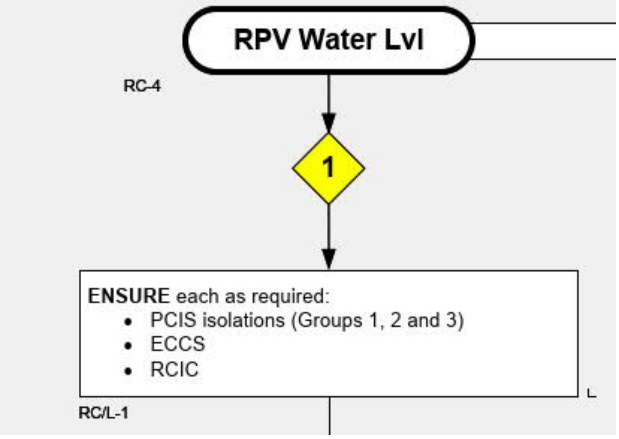
Time	Position	Applicant's Actions or Behavior
	NRC	**When it is apparent that available high-pressure injection systems cannot maintain Reactor Water Level above (-) 162 inches due to the trend in Reactor Water Level, the crew may anticipate Emergency Depressurization.**
	NRC	**End of Event 9. When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF – (-) 162 inches), end of Scenario.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 1 of 10

Event Description: Reactor Feedwater Pumps Trip

Symptoms/Cues: Event is initiated when the Reactor MODE SWITCH is placed in SHUTDOWN. All Reactor Feedwater Pumps will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 10 is initiated when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required to insert Event 10.**
	NRC	**Reactor Feedwater Pumps (RFPTs) will trip 30 seconds after the Reactor MODE SWITCH is placed in SHUTDOWN.**
	OATC	Informs the NUSO that all RFPTs have tripped.
	NRC	**EXAMINER NOTE: The EHPM will trip when it is started – no action is required by the Driver is insert this malfunction.**
	NUSO	 RC/L-2

IF	THEN
Raising RPV Water Level above +51 in. will facilitate use of Shutdown Cooling, steam-driven injection systems, or Alternate Injection subsystems (Table L-2)	**NO ACTION REQUIRED**
Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7	**NO ACTION REQUIRED**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 2 of 10

Event Description: Reactor Feedwater Pumps Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>In accordance with 3-EOI-1, RPV Control MODES 1-3, directs the BOP to maintain Reactor Water Level using any or all of the following EOI Appendices:
CRD – 3-EOI-Appendix-5B
RCIC – 3-EOI-Appendix-5C
EHPM Pump – 3-EOI-Appendix-7L</td></tr>
<tr><td></td><td>NUSO</td><td>RC/L-3
RESTORE and MAINTAIN RPV Water Level between +2 in. and +51 in. with ANY Preferred Injection Systems (Table L-1)
➢ OK to use ANY Alternate Injection Subsystems (Table L-2)

RC/L-4
WHEN RPV Water Level CANNOT be restored and maintained between +2 in. and +51 in.

RC/L-5<table><tr><th>IF</th><th>THEN</th></tr><tr><td>RPV Water Level can be restored and maintained above +2 in.</td><td>NO ACTION REQUIRED</td></tr></table>
RC/L-6
IF RPV Water Level CANNOT be maintained above -122 in. THEN INHIBIT ADS</td></tr>
<tr><td></td><td>NRC</td><td>Due to the size of the leak, available high-pressure systems will not maintain Reactor Water Level above (-) 180 inches. Reactor Water Level will reach (-) 180 inches in about 14 minutes from the time Event 9 is entered.</td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 3 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	3-EOI-Appendix-5C, Injection System Lineup RCIC [1] N/A [2] **ENSURE RESET** auto isolation logic using 3-XS-71-51A(B) RCIC AUTO-ISOL LOGIC A (B) RESET pushbuttons. [3] **ENSURE RESET** and OPEN 3-FCV-71-9, RCIC TURBINE TRIP/THROT VALVE. [4] **ENSURE** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, in AUTO with setpoint at 620 gpm. [5] **OPEN** 3-FCV-71-34 RCIC PUMP MINIMUM FLOW VALVE. [6] **OPEN** 3-FCV-71-39 RCIC PUMP INJECTION VALVE. [7] **OPEN** 3-FCV-71-25 RCIC LUBE OIL COOLING WATER VALVE. [8] **PLACE** 3-HS-71-31A, RCIC VACUUM PUMP, in START. **CAUTIONS** 1) Operating RCIC Turbine below 2100 rpm may result in unstable system operation and equipment damage. 2) High Suppression Chamber Pressure may trip RCIC. 3) Operating RCIC Turbine with Suction Temperatures above 240 ºF may result in equipment damage. [9] **OPEN** 3-FCV-71-8, RCIC TURBINE STEAM SUPPLY VALVE, to start RCIC turbine. [10] **CHECK** proper RCIC operation by observing the following: A. RCIC Turbine Speed accelerates above 2100 rpm. B. Flow to RPV controlled automatically at 620 gpm. C. 3-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE, closes as flow rises above 120 gpm. [11] **ADJUST** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, as necessary to control injection. [12] N/A [13] N/A [14] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 4 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	OATC	3-EOI-Appendix-5B, Injection System Lineup CRD [1] N/A [2] IF **BOTH** of the following exists • CRD is NOT required for rod insertion, **AND** • Maximum injection flow is required **THEN LINE UP** ALL available CRD pumps to the RPV as follows: [2.1] **IF** CRD Pump 3A is available, **THEN VERIFY** RUNNING CRD Pump 3A. [2.2] **IF** CRD Pump 3B is available, **THEN VERIFY** RUNNING CRD Pump 3B. [2.3] OPEN the following valves to increase CRD Flow to the RPV: • 3-PCV-85-23, CRD DRIVE WATER PRESSURE CONTROL VALVE • 3-PCV-85-27, CRD COOLING WATER PRESSURE CONTROL VALVE • 3-FCV-85-50, CRD EXHAUST RETURN LINE SHUTOFF VALVE [2.4] ADJUST 3-FIC-85-11, CRD SYSTEM FLOW CONTROL to control injection WHILE maintaining 3-PI-85-13A, CRD ACCUMULATOR CHARGING WATER HEADER PRESSURE, above 1450 psig, if possible. [2.5] IF additional flow is necessary to prevent or mitigate Core Damage, THEN DISPATCH personnel to fully open the following valves as required: • 3-THV-085-0527, CRD PUMP DISCH THROTTLING (RB NE, el 565') • 3-BYV-085-0551, CRD PUMP TEST BYPASS (RB NE, el 565')

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 5 of 10

Event Description: Reactor Feedwater Pumps Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>CREW</td><td>Determines that Reactor Water Level cannot be maintained above (-) 162 inches.</td></tr>
<tr><td></td><td>CREW</td><td>Critical Task: When Reactor Water Level cannot be restored and maintained above (-) 162 inches, and before Reactor Water Level reaches (-) 180 inches, Emergency Depressurize the Reactor.

Critical Task Failure Criteria: Emergency Depressurization is not directed when Reactor Water Level cannot be restored and maintained above (-) 180 inches.</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-1, RPV Control MODES 1-3
EMERGENCY RPV DEPRESSURIZATION

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that available Injection Subsytems (Table L-3) alone CANNOT assure Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 ft</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-7
<table>
<tr><td>IF DW Pressure is above 2.45 psig THEN PREVENT injection from ONLY those Core Spray and LPCI Pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 6 of 10

Event Description: Reactor Feedwater Pumps Trip

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>RC/P-8
<table>
<tr><td colspan="2">EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Level is above 5.5 ft THEN OPEN 6 MSRVs (ADS Valves preferred)
➢ OK to exceed 100°F/hr cooldown rate</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Control Air is or becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Less than 4 MSRVs can be opened
AND
RPV Pressure is 60 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
<tr><td></td><td>NUSO</td><td>Directs the crew to inject to the Reactor using low-pressure systems:
• Condensate System – 3-EOI-Appendix-6A
• Core Spray System (Loop I, II, or both) - 3-EOI-Appendix-6D/6E
• RHR System (Loop I, II, or both) - 3-EOI-Appendix-6B/6C</td></tr>
<tr><td></td><td>BOP</td><td>3-EOI-Appendix-6A, injection Subsystems Lineup Condensate

[1] ENSURE CLOSED the following Feedwater Heater Return Valves:
• 3-FCV-3-71, HP HTR 3A1 LONG CYCLE TO CONDENSER
• 3-FCV-3-72, HP HTR 3B1 LONG CYCLE TO CONDENSER
• 3-FCV-3-73, HP HTR 3C1 LONG CYCLE TO CONDENSER
[2] ENSURE CLOSED the following RFP Discharge Valves (N/A if Condensate Flood up is anticipated per the FSSs):
• 3-FCV-3-19, RFP 3A DISCHARGE VALVE
• 3-FCV-3-12, RFP 3B DISCHARGE VALVE
• 3-FCV-3-5, RFP 3C DISCHARGE VALVE</td></tr>
</table>

Form 3.3-2 Required Operator Actions

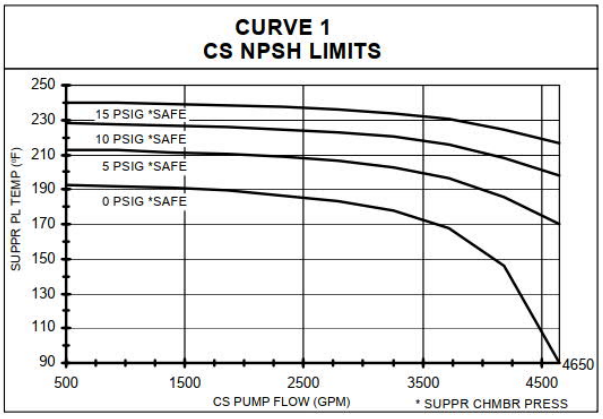
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 7 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE OPEN** the following Drain Cooler Inlet Valves: • 3-FCV-2-72, DRAIN COOLER 3A5 CONDENSER INLET ISOLATION VALVE • 3-FCV-2-84, DRAIN COOLER 3B5 CONDENSER INLET ISOLATION VALVE • 3-FCV-2-96, DRAIN COOLER 3C5 CONDENSER INLET ISOLATION VALVE [4] **ENSURE OPEN** the following Heater Outlet Valves: • 3-FCV-2-124, LP HEATER 3A3 CONDENSER OUTLET ISOLATION VALVE • 3-FCV-2-125, LP HEATER 3B3 CONDENSER OUTLET ISOLATION VALVE • 3-FCV-2-126, LP HEATER 3C3 CONDENSER OUTLET ISOLATION VALVE [5] **ENSURE OPEN** the following Heater Isolation Valves: • 3-FCV-3-38, HP HTR 3A2 FW INLET ISOLATION VALVE • 3-FCV-3-31, HP HTR 3B2 FW INLET ISOLATION VALVE • 3-FCV-3-24, HP HTR 3C2 FW INLET ISOLATION VALVE • 3-FCV-3-75, HP HTR 3A1 FW OUTLET ISOLATION VALVE • 3-FCV-3-76, HP HTR 3B1 FW OUTLET ISOLATION VALVE • 3-FCV-3-77, HP HTR 3C1 FW OUTLET ISOLATION VALVE [6] **ENSURE OPEN** the following RFP Suction Valves: • 3-FCV-2-83, RFP 3A SUCTION VALVE • 3-FCV-2-95, RFP 3B SUCTION VALVE • 3-FCV-2-108, RFP 3C SUCTION VALVE [7] **ENSURE** all available Condensate Pumps running. [8] **ENSURE** all available Condensate Booster Pumps running. [9] **ADJUST** 3-LIC-3-53, RFW START-UP LEVEL CONTROL, on Panel 3-9-5, to control injection (N/A if Condensate Flood up is anticipated per the FSSs). [10] **CHECK** RFW Flow to RPV (N/A if Condensate Flood up is anticipated per the FSSs).

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 8 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**Only one EOI Appendix for Core Spray Injection (3-EOI-Appendix-6D) and one for RHR Injection (3-EOI-Appendix-6B) is covered below, as System I and System II procedures are the same (with different valve numbers).**
	BOP	3-EOI-Appendix-6D, Injection System Lineup Core Spray System I [1] **VERIFY OPEN** the following valves: • 3-FCV-75-2, CORE SPRAY PUMP 3A SUPPRESSION POOL SUCTION VALVE • 3-FCV-75-11, CORE SPRAY PUMP 3C SUPPRESSION POOL SUCTION VALVE • 3-FCV-75-23, CORE SPRAY SYS I OUTBOARD INJECT VALVE [2] **VERIFY CLOSED** 3-FCV-75-22, CORE SPRAY SYS I TEST VALVE. [3] **VERIFY** CS Pump 3A and/or 3C running. [4] **WHEN** RPV Pressure is below 450 psig, **THEN THROTTLE** 3-FCV-75-25, CORE SPRAY SYS I INBD INJECT VALVE, as necessary to control injection at or below 4000 gpm per pump [5] **MONITOR** Core Spray Pump NPSH using Attachment 1. **NPSH MONITORING** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. 

Form 3.3-2 Required Operator Actions

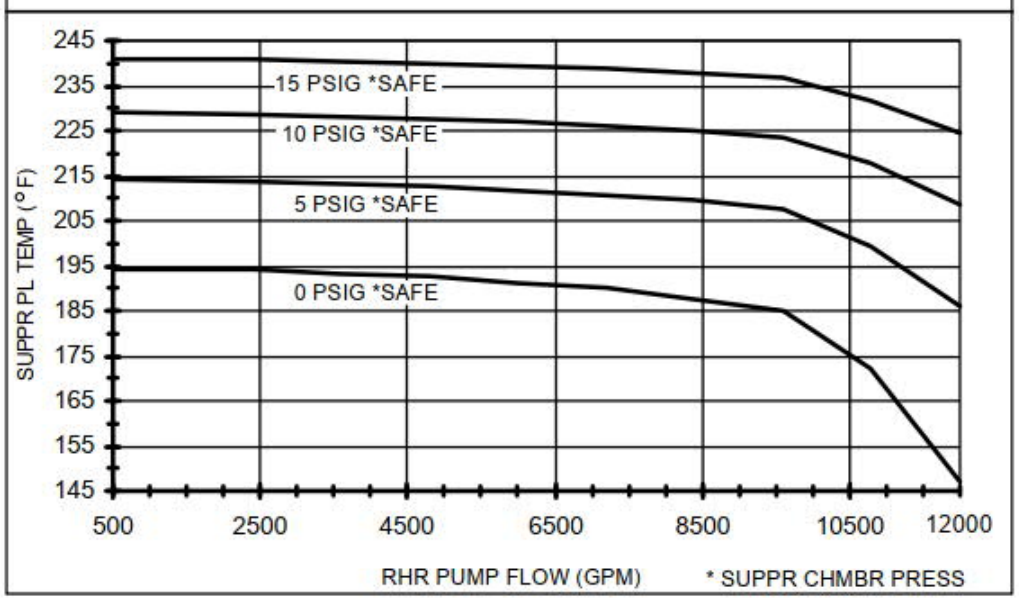
Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 9 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	3-EOI-Appendix-6B, Injection Subsystems Lineup RHR System I LPCI Mode [1] **IF** Adequate Core Cooling is assured AND it becomes necessary to bypass LPCI Injection Valve auto open signal to control injection, **THEN PLACE** 3-HS-74-155A, LPCI SYSTEM-I OUTBOARD INJECTION VALVE BYPASS SELECT, in BYPASS. [2] **ENSURE OPEN** 3-FCV-74-7, RHR SYSTEM I MIN FLOW VALVE. [3] **ENSURE OPEN** the following valves: • 3-FCV-74-1, RHR PUMP 3A SUPPRESSION POOL SUCT VALVE • 3-FCV-74-12, RHR PUMP 3C SUPPRESSION POOL SUCT VALVE [4] **ENSURE CLOSED** the following valves: • 3-FCV-74-61, RHR SYSTEM I DRYWELL SPRAY INBOARD VALVE • 3-FCV-74-60, RHR SYSTEM I DRYWELL SPRAY OUTBOARD VALVE • 3-FCV-74-57, RHR SYSTEM I SUPPRESSION CHAMBER/POOL ISOLATION VALVE • 3-FCV-74-58, RHR SYSTEM I SUPPRESSION CHAMBER SPRAY VALVE • 3-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE [5] **ENSURE** RHR Pump 3A and / or 3C running. [6] **WHEN** RPV Pressure is below 450 psig, **THEN ENSURE OPEN** 3-FCV-74-53, RHR SYS I LPCI INBOARD INJECT ION VALVE. [7] **IF** RPV Pressure is below 230 psig, **THEN ENSURE CLOSED** 3-FCV-68-79, RECIRC PUMP 3B DISCHARGE VALVE. [8] **THROTTLE** 3-FCV-74-52, RHR SYS I LPCI OUTBOARD INJECTION VALVE, as necessary to control injection. [9] **MONITOR** RHR Pump NPSH using Attachment 1. [10] **PLACE** RHRSW pumps in service as soon as possible on ANY RHR Heat Exchangers discharging to the RPV.

Op Test No.: 23-04 Scenario No. NRC-4 Event No.: 10 Page 10 of 10

Event Description: Reactor Feedwater Pumps Trip

Time	Position	Applicant's Actions or Behavior
	BOP	[11] **THROTTLE** the following in-service RHRSW Outlet Valves to maintain 1700 to 4500 gpm flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VALVE • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VALVE [12] **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers. **NPSH MONITORING** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. **CURVE 2** **RHR NPSH LIMITS** 
	NRC	**When the crew has Emergency Depressurized the Reactor and has control of Reactor Water Level above the Top of Active Fuel (TAF – (-) 162 inches), end of Scenario.**

Form 3.3-2 Required Operator Actions

Scenario Setup
UNIT 3

IC	28
Exam IC	266

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-47A	40	AOI-68-1	18	EOI-APP-6D	4
OI-68	106	AOI-70-1	21	EOI-APP-17B	12
ARP-9-4A	51	AOI-85-3	14	EOI-APP-17C	13
ARP-9-4B	54	AOI-100-1	80	TRM 3.4.1	21
ARP-9-4C	46	EOI-1	16	TS 3.1.3	212
ARP-9-5A	56	EOI-2	15	TS 3.1.5	212
ARP-9-7A	30	EOI-APP-5B	2	TS 3.8.1	244
ARP-9-7B	31	EOI-APP-5C	6		
ARP-9-8B	24	EOI-APP-6A	4		
ARP-9-23B	31	EOI-APP-6B	7		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Danger Tags on HPCI Hang Danger Tags on 3B PSC Head Tank Pump Hang Protected Equipment tags on the following: • RCIC • Loop I and II of Core Spray • Loop I and II of RHR • 4KV Shutdown Boards 3EA, 3EB, 3EC, and 3ED
Schedule Files(s):	ILT 2304 NRC Scenario 4 UNIT 3.sch
Event Files(s):	ILT 2304 NRC Scenario 4 UNIT 3.evt

UNIT 3 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ON LINE 230	Drywell Leakage (GPM) 1.89	Protected Equipment RCIC Loop I and II of Core Spray Loop I and II of RHR
Rx Power 100%	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	Floor Drain (GPM) 0.31	4KV SHUTDOWN BOARDS 3EA, 3EB, 3EC, and 3ED
MWe 1307		Equipment Drain (GPM) 1.58	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

On Day 5 of a planned 10-day HPCI outage – LCO 3.5.1.C – 14 days

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

3B PSC Head Tank Pump coupling failure

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Alternate EHC Pumps due to high filter D/P
Support HPCI Outage
Support repair of 3B PSC Head Tank Pump
Continue PMT on 3B Bus Duct Fan

OPERATOR WORK AROUNDS OWAs – U0-1/U3-0 Burdens - U0 – 0/U3-0 Operator Challenges – U0 – 12/U3 - 6

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-5
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 26% Reactor Power, Single Loop on 2B Recirc Pump, 1B CRD Pump is tagged for circuit breaker inspection, IRM 'G' is bypassed due to amplifier failure.

Turnover: Restore 2A Recirc Pump to service, match speed of 2B Recirc Pump, then await instructions from the Reactor Engineer. Support maintenance on 1B CRD Pump.

Critical Tasks:
1. When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the Recirc Pumps reaching zero speed to prevent possible fuel damage.
2. With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Start 2A Recirculation Pump
2.	N/A	R-OATC	Match Recirculation Pump Speeds
3.	HS-75-42A	C-BOP MC-BOP TS-NUSO	Inadvertent Core Spray Pump Start
4.	RD01A	C-OATC MC-OATC C-NUSO	Control Rod Drive (CRD) Pump Trip
5.	ED18A	C-BOP C-NUSO	Loss of I&C 'A'
6.	PC01C HS-65-69A/2	TS-NUSO	Standby Gas Train (SGT) Fails to Start
7.	TH03A TH03B	C-OATC C-NUSO	Dual Recirculation Pump Trip
8.	RD09A RD09B	M-ALL	Hydraulic Anticipated Transient Without SCRAM (ATWS)
9.	RP11	C-BOP C-NUSO	Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will restart 2A Recirculation Pump in accordance with 2-OI-68, Recirculation System.
2. The crew will raise speed of 2A Recirculation Pump to match 2B Recirculation Pump in accordance with 2-OI-68, Reactor Recirculation System.
3. 2B Core Spray Pump will inadvertently start, requiring the crew to shut down 2B Core Spray Pump. When contacted, the Assistant Unit Operator (AUO) will report there is no visible oil level on 2B Core Spray Pump. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
4. Control Rod Drive (CRD) Pump 2A will trip, requiring the crew to enter 2-AOI-85-3, CRD System Failure. The standby CRD Pump is tagged out and not available; therefore, the crew will dispatch an AUO to verify the status of the tripped pump and restart the tripped pump.
5. A Loss of I&C Bus A will occur, requiring the crew to respond in accordance with 2-AOI-57-5A, Loss of I&C Bus A.
6. The Immediate Actions of 2-AOI-57-5A, Loss of I&C Bus A, direct that Standby Gas Train(s) (SGT) be started; one SGT will not start when the crew attempts to start it, requiring the NUSO to reference Technical Specifications.
7. 2A Recirculation Pump will trip, requiring the crew to take action in accordance with 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease. During the immediate actions for the single Recirculation Pump trip, 2B Recirculation Pump will trip, requiring the crew to insert a manual Reactor SCRAM.
8. When Operator at the Controls enters a manual Reactor SCRAM due to the dual Recirc Pump trip, a hydraulic Anticipated Transient Without SCRAM (ATWS) will occur, requiring the crew to respond in accordance with 2-AOI-100-1, Reactor SCRAM, and 2-EOI-1A, ATWS RPV Control.
9. Following the Reactor SCRAM, the Main Steam Isolation Valves (MSIVs) will close due to an inadvertent isolation signal, requiring the crew to take action to restore Reactor Pressure and Reactor Water Level.

The Scenario ends when the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available high-pressure injection system.

Critical Tasks 2

1. **When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the Recirc Pumps reaching zero speed to prevent possible fuel damage.**

   a. **Safety Significance**

   With thermal power being produced in the Reactor and actions to lower Reactor Power have not brought power out of the heating range, power oscillations and subsequent fuel damage may be generated.

   b. **Cues**

   The Reactor is SCRAMMED and ATWS actions are taken by the OATC and Reactor Power is greater than 5% or unknown.

   c. **Measured by**

   Reactor Power on APRM displays.

   d. **Feedback**

   Lowering Reactor Water Level.
   Lowering Reactor Power.

   e. **Critical Task Failure Criteria:**

   The operating crew fails to STOP and PREVENT injection and take action to lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds after Recirculation Flow is stopped.

2. **With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.**

   a. **Safety Significance**

   Precludes core damage due to an uncontrolled reactivity addition.

   b. **Cues**

   Procedural Compliance.

   c. **Measured by**

   ADS Logic inhibited prior to an automatic initiation.

   d. **Feedback**

   Reactor Pressure trend.
   Reactor Water Level trend.
   ADS LOGIC BUS A/B INHIBITED (2-9-3C, Window 18/31) annunciator status.

   e. **Critical Task Failure Criteria**

   ADS automatic initiation with Control Rods out.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 1 of 6

Event Description: Start 2A Recirculation Pump

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**The crew has been briefed that another operator is responsible for 2-SR-3.4.9.3&4, Reactor Recirculation Pump Start Limitations.** **The crew has been briefed to start in 2-OI-68, Section 5.3 on Step [12].** **The crew has been pre-briefed on this procedure, but may elect to hold a re-focus brief.**
	NUSO	Directs the Balance of Plant Operator (BOP) to start 2A Recirculation Pump Speed.
	BOP	2-OI-68, Reactor Recirculation System Section 5.3, Recirc Pump Startup [12] **CHECK** RECIRC LOOP A(B) DIFFERENTIAL PRESS LOW annunciation, 2-XA-55-4A(4B), window 31 in ALARM. **NOTE** All faults and alarms should be reset prior to closing a Recirc Pump Discharge Valve to ensure the DRIVE READY light is obtained thus preventing the loop from cooling down causing the start limitations of the Recirc Pump to possibly be exceeded thus delaying the startup. [13] **DEPRESS** 2-HS-96-13, FAULT RESET, **AND CHECK** all Faults or Alarms are reset on ICS Screen VFDAAL [13.1] **IF** all faults and alarms are clear, **THEN CONTINUE** Recirc Pump Startup at Step 5.3[14]. [13.2] N/A

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 2 of 6

Event Description: Start 2A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTION** The closure circuit for 2-HS-68-3A, RECIRC PUMP 2A DISCHARGE VALVE, is a seal-in and is NOT to be held in the CLOSE position unless specifically directed. If the handswitch is held to the close position (approx. 2.5 seconds) after the valve has reached its full close valve position this will allow the reclosure circuit to engage. This reclosure causes the motor to re-engage at a rapid rate (approx. 10 times a second), which Can cause the motor to heat and also to exceed the recommended valve seat loading. [14] **ENSURE** 2-FCV-68-3, RECIRC PUMP 2A DISCHARGE VALVE, CLOSED. **NOTES** 1) The DRIVE READY light will only light after the FAULT RESET push button is depressed, if all the active faults are reset. ICS screen VFDAAL(VFDBAL) can be referred to verify the Number of Active Faults is zero after depressing the fault reset push buttons. Any faults listed after the FAULT RESET push button has been depressed, are keeping the drive from being ready to run. 2) Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161 source and large pumps are started. Unit SROs are to evaluate the need to place the Capacitor Banks in Manual prior to starting a Recirc pump from the Alternate feeder. The evaluation should consider placing a Caution order on the Recirc pump's start switches stating, "evaluate the need to place CAP Banks in Manual prior to starting Pumps from the ALTERNATE FEED." 3) The actions which occur after the START push-button is depressed are listed in Step [20]. 4) When depressing the switches which control the Recirc Drives these switches are to be firmly depressed to ensure all the contacts are made-up.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 3 of 6

Event Description: Start 2A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[15] IF the Recirc Drive startup sequence incomplete alarm is received and the Recirc Discharge Valve is jogging OPEN, THEN (N/A if alarm is not received) [15.1] **CLOSE** and **HOLD**, 2-FCV-68-3, RECIRC PUMP 2A DISCHARGE VALVE. [15.2] **OPEN** then **CLOSE** RECIRC PUMP 'A' INSTR BKR 519(520) Panel 9-9 cabinet 5. [15.3] **RELEASE** 2-FCV-68-3, RECIRC PUMP 2A DISCHARGE VALVE. NOTES 1) Tech Spec LCO 3.4.1 should be reviewed prior to starting Recirc Pump 2A in Step 5.3[16] The applicability for the Tech Spec is Modes 1 & 2. 2) Step 5.3[16] is used to start Recirc Pump 2A under normal conditions. 3) Step 5.3[17] can be used to start Recirc Pump 2A during cold shutdown (Mode 4) when it is desired to start the pump and open the discharge valve without the jogging circuit. [16] **IF** starting Recirc Pump 2A, **THEN PERFORM** the following: [16.1] **CHECK** Tech Spec LCO 3.4.1 requirements are satisfied. [16.2] **CHECK** DRIVE READY, 2-IL-96-36 is LIT. [16.3] **FIRMLY** DEPRESS DRIVE START, 2-HS-96-12. [16.4] **GO TO** Step 5.3[20]. [17] N/A [18] N/A [19] N/A

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 4 of 6

Event Description: Start 2A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[20] **CHECK** the following for Recirc Pump being started: • DRIVE RUNNING, 2-IL-96-41 is LIT • Pump DP rises to above 5 psid as indicated on 2-PDI-68-65, PUMP DP • Recirc Pump 2A(2B) Speed rises to ~345 RPM, as indicated on 2-SI-68-59, RECIRC PUMP 2A SPEED • RECIRC LOOP A(B) DIFF PRESS LOW 2-PDS-68-65, 2-XA-55-4A, Window 31, is reset. • 2-FCV-68-3, RECIRC PUMP 2A DISCHARGE VALVE, begins to open by the automatic jogging circuit **NOTES** 1) In order to achieve balanced jet pump flows, the Recirc Pumps Speed may require a mismatch. 2) Recirc Pump Speed cannot be raised above 480 rpm (28%) until total Feedwater Flow is greater than 16 percent. Recirc Pump Speed can be controlled between ~345 RPM and ~480 RPM using the Recirc Drive/Recirc Pump Speed Controllers. 3) Recirc Pump A will trip 85 seconds after initiation of the automatic jogging sequence if 2-FCV-68-3, RECIRC PUMP A DISCHARGE VALVE, is less than 90 percent open. 4) Performance of 2-SR-3.4.2.1 is required 24 hours after reaching > 23% RTP and/or 4 hours after returning a Recirc Pump to service. 5) After Recirc loop(s) are in service, Technical Specifications Surveillance Requirements SR 3.4.1.1 and SR 3.4.2.1 have performance requirements which are to be referred to while performing Step 5.3[22]. [21] **CHECK OPEN** 2-FCV-68-3, RECIRC PUMP 2A DISCHARGE VALVE.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 5 of 6

Event Description: Start 2A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[22] **NOTIFY** Reactor Engineer to **CHECK** that the following has been performed, as applicable, depending on the number of operating Recirc loops, and **RECORD** the applicable requirements and their completion status in the narrative log. [22.1] 2-SR-3.4.1(DLO), Reactor Recirculation System Dual Loop Operation. OR [22.2] 2-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation.
	Driver	**If contacted as the Reactor Engineer, acknowledge any information or direction given. Report that all Surveillance Requirements (including 2-SR-3.4.2.1) are in progress or completed as required.**
	BOP	[23] **PERFORM** 2-SR-3.4.2.1 within the next 4 hours following placing a Recirc loop in service. (REFER TO Tech Specs 3.4.1 and 3.4.2.) [24] **DEPRESS** push-button 2-HS-68-32, RECIRC PUMP 2A RUNBACK RESET. [25] **MONITOR** Recirc Pump Seal Pressures during pressurization for proper operation. (Number 2 Seal Pressure should be approximately one half that of Number 1 Seal Pressure). [26] THROTTLE 2-THV-024-0976, VFD 2A HX OUTLET RCW THROTTLE VALVE, to maintain VFD inlet and outlet temperatures. **CAUTION** The Recirc System should be operated with balanced Jet Pump Flows to reduce hydraulic forces and vibration stresses on Jet Pumps and Retainers. [27] **ADJUST** Recirc Pump Speed 2A using 2-HS-96-15A(15B), RAISE SLOW (MEDIUM), or 2-HS-96-17A(17B), LOWER SLOW(MEDIUM), push buttons, to 480 rpm. [28] **WHEN** total Feedwater flow is greater than 16%, **THEN CHECK** reset, RECIRC LOOP A(B) FLOW LIMITER ENFORCING, 2-XA-55-4A, Window 35.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 6 of 6

Event Description: Start 2A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[29] N/A [30] **IF** this is the second Recirc Pump started, **THEN PERFORM** 2-SR-3.4.2.1 for the Jet Pump Mismatch requirements, within the next 24 hours (**REFER TO** Tech Specs 3.4.1.1). [31] **RAISE** Recirc Pumps 2A and/or 2B Speed as directed by 2-GOI-100-1A or 2-GOI-100-12, or 2-GOI-100-12A **REFER TO** Section 6.2.
	Driver	**If contacted as the Reactor Building Assistant Unit Operator (AUO), acknowledge any information or direction given.**
	NRC	**End of Event 1. Proceed to Event 2, Match Recirculation Pump Speeds.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 2 Page 1 of 1

Event Description: Match Recirculation Pump Speeds

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**No action is required by the Driver to insert Event 2. Event 2 is cued by the turnover sheet.**
	NRC	**The crew will match Recirculation Pump Speeds by raising 2A Recirc Pump Speed in accordance with 2-OI-68 and the provided Reactivity Control Plan (RCP).**
	OATC	2-OI-68, Reactor Recirculation System [1] **IF** desired to control Recirc Pump 2A speed with Recirc Individual Control, **THEN PERFORM** the following; • Raise Recirc Pump 2A Speed using, 2-HS-96-15A(15B), RAISE SLOW (MEDIUM) (Otherwise N/A) • Lower Recirc Pump 2A Speed using 2-HS-96-17A(17B)(17C), SLOW (MEDIUM) (FAST) (Otherwise N/A) • N/A • N/A
	NRC	**When satisfied with the Reactor Power change, end of Event 2. Request that the Driver insert Event 3, Inadvertent Core Spray Pump Start.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 1 of 4

Event Description: Inadvertent Core Spray Pump Start

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Core Spray Pump 2B will start without an initiation signal

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to start 2B Core Spray Pump.**
	BOP	Acknowledges and reports the following alarms: • CORE SPRAY SYS II PUMP B START, 2-9-3F, Window 1 • RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 2-9-3C, Window 10 • CORE SPRAY PUMP 2B RUNNING, 1/2-9-23C, Window 37 • RHR SERVICE WATER PUMP D3 RUNNING, 1/2-9-23D, Window 34 **(only received if the 2B Core Spray Pump runs longer than 28 seconds)** Reports that 2B Core Spray Pump has started and that there are no valid signals that should result in Core Spray Pumps starting.
	NUSO	Directs the BOP to respond in accordance with Alarm Response Procedures.
	BOP	Alarm Response Procedure, 2-ARP-9-3F CORE SPRAY SYS II PUMP B START, Window 1 Operator Action: A. **CHECK** auto start signals by multiple indications. B. **CHECK** Core Spray Pump 2B operation by motor amps, discharge pressure and flow. C. **IF** pump is NOT needed, **THEN STOP** Core Spray Pump 2B. D. N/A
	BOP	Alarm Response Procedure, 1/2-9-23C CORE SPRAY PUMP 2B RUNNING, Window 37 Operator Action: A. **IF** accident signal is NOT present, **THEN** 1. CHECK panel 9-3 to verify pump is running.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 2 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior
	BOP	2. **INITIATE** actions to determine why pump is running. 3. **SECURE** pump.
	BOP	Alarm Response Procedure, 1/2-9-23D RHR SERVICE WATER PUMP D3 RUNNING, Window 34 Operator Action: A. **IF** accident signal is **NOT** present, **THEN PERFORM** the following: 1. **CHECK** Panel 2-9-3 to verify RHRSW Pump D3 is running. 2. **ENSURE** adequate Raw Cooling Water (RCW) Pressure PI-24-18, Panel 9-20. 3. **CHECK** Diesel Generator or Core Spray Pump running.
	Driver	**If informed as Unit 1, acknowledge that 2B Core Spray Pump inadvertently started, which caused D3 RHRSW Pump to start (if 2B Core Spray Pump is not secured in 28 seconds). If asked, Unit 1 doesn't require that D3 RHRSW Pump remain running.** **If contacted as the Control Bay AUO to check the ATUs in the Aux. Instrument Room, acknowledge the direction. Wait 2 minutes and report that there are no ATUs tripped that are common to Core Spray Initiation Logic.** **When contacted as Maintenance, the Outside/Work Control SRO, or an AUO to investigate, wait 3 minutes and report that the oil level on 2B Core Spray Pump is below the sight glass.**
	BOP	Determines that 2B Core Spray Pump inadvertently started, recommends to the Nuclear Unit Senior Operator (NUSO) that 2B Core Spray Pump be secured, and secures 2B Core Spray Pump and D3 RHRSW Pump when directed.
	BOP	0-OI-67, Emergency Equipment Cooling Water System Section 7.2, Return to Standby Readiness of the South Header [1] **ENSURE** Chemistry treatment to associated RHRSW (EECW) Pump discharge line secured.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 3 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **ENSURE** there are NO emergency Diesel Generators running on any unit. [3] **ENSURE** there are NO Core Spray Pumps running on any unit. [4] **ENSURE** EECW system is **NOT** being used to supply cooling water to RBCCW. [5] **ENSURE** EECW System is **NOT** being used to supply cooling water to the Control Air Compressors. **NOTE** The RHRSW Pump Control Switches and amp meters are located at Control Room Panel 9-3, Unit 1, 2, and 3. **CAUTION** It is normal practice to keep one RHRSW Pump running on each of the North and South EECW headers to ensure the headers remain solid with no trapped air, and to ensure operability requirements. [6] N/A [7] **IF** Pump D3 is to be **SHUT DOWN**, **THEN PLACE** one of the following switches momentarily to the STOP position: • 0-HS-23-94A/2, RHRSW PUMP D3 EECW SOUTH HDR, on Unit 2 [8] **IF** either pump remains in operation, **THEN CHECK** running current for the remaining pump is less than 61 amps using one of the following: • 0-EI-23-88/2, RHRSW PUMP B3 AMPS, on Unit 2
	NRC	**The candidate may reference Technical Specification 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation; however, given that the Driver reported as an AUO that there were no common ATUS for Core Spray Logic tripped in the Aux instrument Room, Technical Specification 3.3.5.1 is not required.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 4 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior	
	NUSO	References Technical Specification 3.5.1, ECCS-Operating LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE. Applicability: MODE 1, MODES 2 and 3, except High Pressure Coolant Injection (HPCI) and ADS Valves are not required to be OPERABLE with Reactor Steam Dome Pressure ≤ 150 PSIG. <u>Condition:</u> A. One Low Pressure ECCS Injection/Spray subsystem inoperable. <u>OR</u> One Low Pressure Coolant Injection (LPCI) Pump in both LPCI subsystems inoperable.	
		REQUIRED ACTION: A. Restore Low Pressure ECCS Injection/Spray subsystem(s) to OPERABLE status.	**COMPLETION TIME:** B.1 – 7 days
	Driver	**If contacted as the Outside / Work Control NUSO or as an AUO to hang a clearance on or open the circuit breaker for 2B Core Spray Pump, wait 2 minutes and insert Event 13. Contact the crew to inform them that the breaker for 2B Core Spray Pump is open.** **If contacted as Engineering to perform an evaluation, acknowledge the direction.** **If contacted as the Outside/Work Control SRO to address Protected Equipment acknowledge the direction.**	
	NRC	**End of Event 3. Request that the Driver insert Event 4, Control Rod Drive (CRD) Pump Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 4

Event Description: Control Rod Drive (CRD) Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2A CRD Pump will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to trip 2A CRD Pump.**
	NRC	**CRD Temperature alarms will be received in approximately 4 and a half minutes after the CRD Pump trip.**
	OATC	Informs the NUSO that 2A Control Rod Drive (CRD) Pump has tripped.
	NUSO	Directs the Operator at the Controls (OATC) to respond in accordance with 2-AOI-85-3, CRD System Failure.
	OATC	2-AOI-85-3, CRD System Failure 4.1 Immediate Actions: [1] N/A [2] N/A 4.2 Subsequent Actions **NOTES** 1) Operation with loss of CRD system should be minimized since CRD Mechanism Cooling Water, Reactor Recirc Pump Seal Purge Flow, Reactor Water Cleanup Recirc Pump Seal Water Flow, and CRD HCU Accumulator Charging Water will be lost. Technical Specification Section 3.1.3 and 3.1.5 should be consulted. 2) Do NOT attempt to start a CRD Pump which has tripped for unknown reasons before verifying locally that the pump, motor and breaker are in operating condition. 3) Steps 4.2[1] and 4.2[2] should be performed concurrently. [1] **IF** the operating CRD Pump has tripped **AND** backup CRD Pump is **NOT** available, **THEN PERFORM** the following: (Otherwise N/A) [1.1] **SECURE** RWCU Pump 2B.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 4 Page 2 of 2

Event Description: Control Rod Drive (CRD) Pump Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as the Reactor Building and Control Bay AUOs to verify 2A CRD Pump, motor, and breaker are in operating condition, acknowledge the direction.** **Wait 2 minutes and report that 2A CRD Pump, motor, and breaker are in operating condition. As the Reactor Building AUO report that there were scaffold builders in the area of 2A CRD Pump – you have directed them to report to the Control Room.**
	OATC	1.2] **DISPATCH** personnel to locally verify the tripped pump, motor and breaker are in operating condition. [1.3] **IF** the tripped pump is determined to be in operating condition AND with Unit supervisor permission, **THEN PERFORM** the following at Panel 2-9-5: (Otherwise N/A) [1.3.1] **PLACE** CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, in MAN at minimum setting. [1.3.2] **ATTEMPT TO RESTART** tripped CRD Pump using one of the following: • CRD PUMP 1B, using 2-HS-85-2A • CRD Pump 2A, using 2-HS-85-1A [1.3.3] **ADJUST** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, to establish the following conditions: • 2-PDI-85-18A, CRD CLG WATER HEADER DP, between 10 psid and 20 psid • 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, between 40 and 65 gpm [1.3.4] **BALANCE** 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, and **PLACE** in AUTO or BALANCE.
	NRC	**End of Event 4. Request that the Driver insert Event 5, Loss of I&C 'A'.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 1 of 4

Event Description: Loss of I&C 'A'

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. I&C 'A' will de-energize, causing a loss of loads and various alarms

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a loss of I&C 'A'.**
	NRC	**Numerous alarms are received during this event. Only some alarms are addressed in this guide.** **When Event 5 is inserted, a Reactor Building Ventilation Static Isolation will occur.**
	BOP	Acknowledges and reports the following alarm: • I&C BUS A VOLTAGE ABNORMAL, 2-9-8C, Window 21 • REACTOR BUILDING VENTILATION ABNORMAL, 2-9-3D, Window 3
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure and 2-AOI-57-5A, Loss of I&C Bus A.
	BOP	2-ARP-9-8C, Alarm Response Procedure I&C BUS A VOLTAGE ABNORMAL, Window 21 Operator Action: A. **CHECK** alarm by checking the following: • Loss of instrument power and remote position indication to Core Spray Div I and RHR Div I (Panel 9-3) • RWCU Filter Demin A isolation • Reactor Building/Refuel Zone Ventilation isolation B. **NOTIFY** Unit 3 Unit Supervisor. C. **REFER TO** 2-AOI-57-5A and 0-GOI-300-2.
	BOP	2-AOI-57-5A, Loss of I&C Bus A 4.1 Immediate Actions – None 4.2 – Subsequent Actions [1] **ENSURE** Automatic Actions have occurred.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 2 of 4

Event Description: Loss of I&C 'A'

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **IF** a Reactor Scram occurs, **THEN REFER** TO 2-AOI-100-1 and PERFORM it concurrently with this procedure. [3] **ENSURE** a flow path for Condensate System OR STOP the Condensate Pumps/Booster Pumps. **REFER TO** 2-OI-2, Condensate System. [4] **START** Standby Gas Train(s) and **CHECK** Reactor Building Pressure at or below 0.25" H2O vacuum (PDIC 64-1, Panel 25-215; PDIC 64-2, Panel 25-213). **REFER TO** Standby Gas Treatment System Manual Initiation section of 0-OI-65, Standby Gas Treatment System.
	NRC	**Based on their training in the Initial License Training Program, the crew may elect to either call Unit 1 to start a Standby Gas Train (SGT) or go to Panel 9-25 and start the SGT themselves. The first SGT the crew attempts to start will not start. See Event 6 on page 18 of 43 for SGT actions.**
	BOP	[5] **ENSURE** Steam Jet Air Ejector (SJAE) 2B in service to maintain Condenser Vacuum. **REFER TO** 2-AOI-47-3, Loss of Condenser Vacuum.
	BOP	Places 2B Steam Jet Air Ejector in service in accordance with 2-OI-66/HARDCARD, Standby SJAE System Lineup Hard Card. 3.0 Operator Actions for SJAE 2B **NOTES** Radiation Protection should be notified prior to placing a SJAE in service. If time does not permit this due to plant conditions, then notification should be made when possible. 2-HS-001-0375, SJAE TRAIN PERMISSIVE (located on 2-LPNL-925-0105, U2 TB, el 586') should normally be in the position of the standby SJAE. If problems are encountered while placing a SJAE in service and time permits, operate this switch as required during the performance of this section.
	Driver	**If contacted as Radiation Protection, acknowledge any information or direction given.**
	BOP	[1] **ENSURE RESET** Off-Gas isolation using 2-HS-90-155, OFFGAS OUTLET/DRAIN ISOLATION VALVES. [2] **ENSURE OPEN** the following valves: • 2-HS-66-15, SJAE 2B INLET VALVE • 2-HS-1-156A, STEAM TO SJAE 2B

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 3 of 4

Event Description: Loss of I&C 'A'

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE** in AUTO/OPEN 2-HS-66-18, SJAE 2B OFFGAS OUTLET VALVE. [4] **PLACE** 2-HS-1-152, SJAE 2B PRESS CONTROLLER, in **CLOSE** and then in **OPEN**. [5] **ENSURE OPEN** the following valves (red light illuminated): • 2-PCV-1-153/167, STEAM TO SJAE 2B STAGES 1,2, AND 3. • 2-FCV-1-152, SJAE 2B INTMD CONDENSER DRAIN. [6] **MONITOR** Hotwell Pressure as indicated on recorder 2-XR-2-26, CONDENSATE (Point 3), on Panel 2-9-6. [7] **FOR** the SJAE **NOT** being placed in service, **ENSURE CLOSED** the following valves: • 2-HS-66-14, SJAE 2A OFFGAS OUTLET VALVE • 2-HS-1-150, SJAE 2A PRESSURE CONTROLLER • 2-HS-1-155A STEAM TO SJAE 2A [8] **WHEN** Condenser Vacuum has stabilized, **THEN PROCEED** to 2-OI-66, Off-Gas System "Placing Standby SJAE in Operation" **AND PERFORM** the applicable steps to establish acceptable SJAE parameters to obtain stable and live pressure indications.
	BOP	(Continuing actions of 2-AOI-57-5A, Loss of I&C Bus A) [6] N/A [7] **WHEN** Reactor Water Level is normal, **THEN RESET** PCIS Group 6 Inboard Isolation and **RETURN** the affected systems to service or standby readiness. REFER to 2-AOI-100-1, if a Reactor SCRAM occurred, otherwise REFER TO 2-AOI-64-2D, Ventilation System Isolation. [8] **REQUEST** Electrical Maintenance investigate I&C Bus A distribution malfunction(s).
	Driver	**When contacted as Electrical Maintenance, Work Control / Outside NUSO, or an AUO to investigate, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 4 of 4

Event Description: Loss of I&C 'A'

Time	Position	Applicant's Actions or Behavior
	BOP	[9] **DETERMINE** affected components, using Attachment 1, I&C BUS 2A Load Listing, and NOTIFY the following sections as applicable of the associated inoperable components: • Fire Protection - Panel 2-9-9 Breakers 214, 232, Batt Bd 2, Panel 8, Breakers 807, 808, 810 • Chemistry - Panel 2-9-9 Breakers 208, 209, 216, 210 • RADCON - Panel 2-9-9 Breakers 202, 234, 231, 229 **[10] SEE NRC NOTE BELOW** [11] **REQUEST** Unit SRO evaluate Attachment 1 for Limiting Conditions of Operation and Actions required for Technical Specification compliance.
	BOP	2-ARP-9-3D, Alarm Response Procedure REACTOR BUILDING VENTILATION ABNORMAL, Window 3 A. N/A B. **NOTIFY** the Unit SROs, Unit 1 and Unit 3 C. N/A D. **DISPATCH** personnel to check Bldg DP (PDIC 64-2, El 639' Rx Bldg.). E. **DISPATCH** personnel to check 480V AC Rx Bldg Vent Bd 2A and 2B. F. N/A
	NRC	**In the interest of time management, it is not expected that the crew address Technical Specifications or Fire Protection Requirements for this event.**
	NRC	**End of Event 5. See Event 6 for Driver Actions.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 6 Page 1 of 3

Event Description: Standby Gas Train (SGT) Fails to Start

Symptoms/Cues: Event is initiated by either the crew contacting another Unit for SGT Train start or automatically by Simulator setup if the crew attempts to manually start a SGT Train. One SGT Train will not be able to be started

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 6 is inserted as follows:** • **Method 1: When the crew attempts to start any train of SGT, it will fail. Ensure that the Event File is updated (this will automatically occur) to allow any other train to be started** • **Method 2: If contacted as Unit 1 or Unit 2 to start a train of SGT, acknowledge the direction. Wait 2 minutes, and insert Event 16 to start SGT 'B'. Report to the crew that 'C' SGT will not automatically start**
	NRC	**EXAMINER NOTE: See page 20 of 43 for actions if the crew contacts another Unit to start a SGT Train.**
	BOP	Method 1: 0-OI-65, Standby Gas Treatment System Section 5.2, Standby Gas Treatment System Manual Initiation **NOTE** When manually starting SGT, verify that the train with the highest run time is used so that all train beds are not exhausted at the same time.
	Driver	**If contacted as Unit 1 and asked which SGT Train should be started, inform the crew that 'C' SGT Train should be started.**
	BOP	[1] **ENSURE** the following requirements are satisfied: • SGT Train A(B)(C) in standby readiness. **REFER TO** Section 4.0. • Main Stack Radiation Monitoring in Service. **REFER TO** 2-OI-90, Radiation Monitoring System. • As plant conditions permit, Normal ventilation has been in operation for a minimum of 2 hours following painting, fire, or chemical release for the affected zones from which SGT will take suction. [2] **REVIEW** the Precautions and Limitations in Section 3.0.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 6 Page 2 of 3

Event Description: Standby Gas Train (SGT) Fails to Start

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE** suction path is aligned to SGT System as follows: [3.1] **IF** alignment to Containment Venting or Purge suction path is desired, **THEN REFER TO** 1(2)(3)-OI-64, Primary Containment System. [3.2] **IF** alignment to Reactor Zone Ventilation suction path is desired, **THEN ENSURE OPEN** the following dampers for the desired unit(s) to be aligned. • 2-HS-64-40 and 2-HS-64-41, REACTOR ZONE EXH TO SGTS Dampers, on Panel 2-9-25 [3.3] N/A [4] **START** SGT FAN A(B)(C) as follows: [4.1] N/A [4.2] **IF** starting SGT FAN C from Panel 2-9-25, **THEN PLACE** 0-HS-65-69A/2, SGTS FAN C, in START. [4.3] N/A
	NRC	**SGT Train 'C' will fail to start**
	BOP	[5] **CHECK** SGT TRAIN A(B)(C) INLET DAMPER as follows: [5.1] **IF** SGT FAN A was started, **THEN CHECK** 0-HS-65-3A(0-IL-65-3), SGTS FILTER BANK A INLET DAMPER, indicates OPEN on Panel 1-9-25(3-9-25). [5.2] **IF** SGT FAN B was started, **THEN CHECK OPEN** 0-HS-65-25A(0-IL-65-25), SGTS FILTER BANK B INLET DAMPER, indicates OPEN on Panel 1-9-25(3-9-25). [5.3] **IF** SGT FAN C was started, **THEN CHECK OPEN** 0-HS-65-51A(0-IL-65-51), SGTS TRAIN C INLET DAMPER, indicates OPEN on Panel 3-9-25. [6] **CHECK** SGT TRAIN A(B)(C) RH CONTROL HTR as follows: [6.1] **IF** SGT FAN A(B) was started, **THEN CHECK** ENERGIZED 0-HS-65-12(34), SGTS TRAIN A(B) RH CONTROL HTR, on Panel 1-9-25. [6.2] **IF** SGT FAN C was started, **THEN CHECK** ENERGIZED 0-HS-65-60, SGTS TRAIN C RH CONTROL HTR, on Panel 2-9-25.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 6 Page 3 of 3

Event Description: Standby Gas Train (SGT) Fails to Start

Time	Position	Applicant's Actions or Behavior	
	Driver	**If contacted as Unit 1 to verify inlet damper position or heater status, report that conditions are normal for SGT Train start.**	
	BOP	[7] Record start time and SGT Filter Bank Differential Pressure. [8] **DISPATCH** Operator to the Standby Gas Treatment building as soon as time allows to check for abnormal conditions (i.e. belt tightness, rubbing or vibration noises). [9] **MONITOR** Standby Gas Treatment Train operation. **REFER TO** Section 6.0	
	Driver	**If contacted as an AUO to monitor SGT Trains, acknowledge any information or direction given.**	
	BOP	Method 2: Contacts Unit 1 to start a SGT Train.	
	Driver	**If contacted as Unit 1 to start a SGT Train, acknowledge any information or direction given. Wait 2 minutes, insert Event 6 to start 'B' SGT, and then contact the crew and inform them that 'C' SGT would not start from Unit 1(3). Inform the crew that 'B' SGT is running.**	
	NUSO	References Technical Specification 3.6.4.3, Standby Gas Treatment (SGT) System LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE Applicability: MODES 1, 2, and 3. <u>Condition:</u> A. One SGT subsystem inoperable.	
		REQUIRED ACTION: A.1 Restore SGT subsystem to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 days
	NRC	**End of Event 6. Request that the Driver insert Event 7, Dual Recirc Pump Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 7 Page 1 of 2

Event Description: Dual Recirc Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 2A Recirc Pump will trip, and 90 seconds later 2B Recirc Pump will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to trip both Recirc Pumps.**
	OATC	Acknowledges and reports the following alarms: • RECIRC DRIVE 2A PROCESS ALARM, 2-9-4A, Window 28 • RECIRC DRIVE 2A DRIVE ALARM, 2-9-4A, Window 32 Reports to the NUSO that 2A Recirc Pump has tripped.
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedures and 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease.
	NRC	**Because the 2B Recirc Pump trips 90 seconds after 2A, the crew may not have time to perform many ARP Actions.**
	OATC	2-ARP-9-4A, Alarm Response Procedure RECIRC DRIVE 2A PROCESS ALARM, Window 28 Operator Action: A. N/A B. **REFER TO** ICS Screen "VFDAAL" and determine cause of alarm. C. N/A D. N/A E. N/A F. N/A G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	OATC	2-ARP-9-4A, Alarm Response Procedure RECIRC DRIVE 2A DRIVE ALARM, Window 32 Operator Action: A. **REFER TO** ICS Group Display "GD @VFDADA" and DETERMINE cause of alarm.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 7 Page 2 of 2

Event Description: Dual Recirc Pump Trip

Time	Position	Applicant's Actions or Behavior
	OATC	B. **IF** the VFD has shutdown, **THEN NOTIFY** engineering to determine if a random reboot has occurred by reviewing the fault logs. • **IF** a random reboot is confirmed by engineering, **THEN START** the VFD per 2-OI-68, Reactor Recirculation System.
	Driver	**If contacted as Engineering, acknowledge any information or direction given.**
	OATC	C. N/A D. N/A E. N/A F. For all other alarms, or any problems encountered **NOTIFY** system engineering. G. At discretion of Unit SRO, **ATTEMPT** to reset alarm. **REFER TO** 2-OI-68, Reactor Recirculation System, for guidance.
	OATC	When 2B Recirc Pump Trips, informs the NUSO that both Recirc Pumps have tripped.
	OATC	2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease 4.1 Immediate Actions [1] **IF** both Recirc Pumps are tripped in MODES 1 or 2, **THEN**, SCRAM the Reactor.
	NUSO	Directs the OATC to insert a manual Reactor SCRAM.
	NRC	**End of Event 7. Proceed to Event 8, Hydraulic Anticipated Transient Without SCRAM (ATWS).**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 1 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Symptoms/Cues: Event is initiated automatically on Simulator setup. When a manual Reactor SCRAM is inserted for Event 7, all Control Rods will not go fully into the Core

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8 is entered on Simulator setup. No action is required by the Driver to insert Event 8.**
	NRC	**EXAMINER NOTE: Event 9, Inadvertent Main Steam Isolation Valves (MSIVs) Isolation, will automatically be inserted 2 minutes after the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the Driver to insert Event 9. See page 37 of 43 for Event 9 actions.**
	NRC	**Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**
	OATC	2-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 2-HS-99-5A/S3A, REACTOR SCRAM A and 2-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 2-9-5. [2] **PLACE** 2-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] **IF** all Control Rods can NOT be verified fully inserted, **THEN INITIATE** ARI. [4] N/A **NOTES** 1) Perform steps 4.1[5.3] and4.1[5.4] in parallel. 2) Step 4.1[5.9] may be performed before step 4.1[5.8] if Reactor Water Level is slowly lowering. [5] **IF** Reactor Power is ABOVE 5% or unknown, **THEN PERFORM** the following: [5.1] **REPORT** the following to the UNIT SRO: • REACTOR SCRAM, MODE SWITCH in SHUTDOWN • Control Rods out • Reactor Power • Continuing with ATWS Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 2 of 14

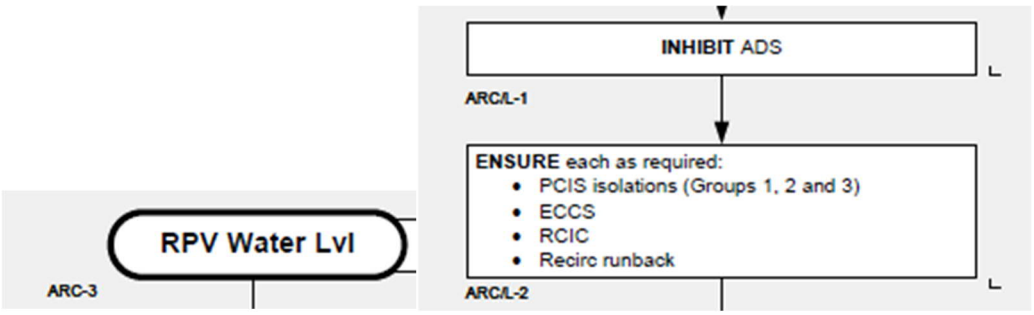
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[5.2] **ENSURE** shutdown both Recirc Pumps [5.3] **STOP** and **PREVENT** injection from CONDENSATE and FEEDWATER per APP 4. • Maintain Reactor Water Level -180 inches to -50 inches [5.4] **STOP** and **PREVENT** injection from HPCI, RHR, and Core Spray per EOI Appendix 4.
	CREW	**Critical Task: When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the Recirc Pumps reaching zero speed to prevent possible fuel damage.** **Critical Task Failure Criteria: The operating crew fails to STOP and PREVENT injection and take action to lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds after Recirculation Flow is stopped.**
	OATC / BOP	[5.5] **INITIATE** SLC and **ENSURE** injection. [5.6] **INHIBIT** ADS LOGIC.
	CREW	**Critical Task: With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.** **Critical Task Failure Criteria: ADS automatic initiation with Control Rods out.**
	OATC	[5.7] **BYPASS** Group 1 RPV Low-Low-Low Level isolation interlocks per EOI Appendix 8A. [5.8] **AFTER** Reactor Water Level reaches -50 inches, **THEN REPORT** "ATWS Actions Complete." • Reactor Water Level and trend • Reactor Power level • Reactor Pressure and trend • MSIV position • SLC injection status

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 3 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
		[5.9] **IF** Reactor Water Level is slowly lowering, **THEN** (Otherwise N/A) **REPORT** ATWS action complete. • Reactor Water Level and trend • Reactor Power level • Reactor Pressure and trend • MSIV position • SLC injection status
	OATC	[5.10] **IF** Any Main steam line is OPEN, **THEN CALL** for EOI Appendix 8E, Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation interlocks.
	NUSO	Enters 2-EOI-1A, ATWS RPV Control.
	Driver	**Insert the Events noted below as called for when contacted as the Work Control / Outside NUSO to perform the EOI Appendices.** • **2-EOI-Appendix-1F – Event 20** • **2-EOI-Appendix-2 – Event 21** • **2-EOI-Appendix-8E – Event 23** • **2-EOI-Appendix-8A – Event 22**
	NRC	**EXAMINER NOTE: Normally 2-EOI-Appendix-8A is performed in the Control Room at Panel 2-9-4 using keylock switches, but the crew may call for field performance.** **EOI Appendices are covered starting on page 33 of 43.**
	NRC	**EXAMINER NOTE: Due to the low Reactor Power Level for this ATWS and SLC Injection, the crew may not be able to lower Reactor Water Level to (-) 50 inches.**
	NUSO	

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 4 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	ARC/L-3

IF	THEN
Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7	**NO ACTION REQUIRED**

ARC/L-4

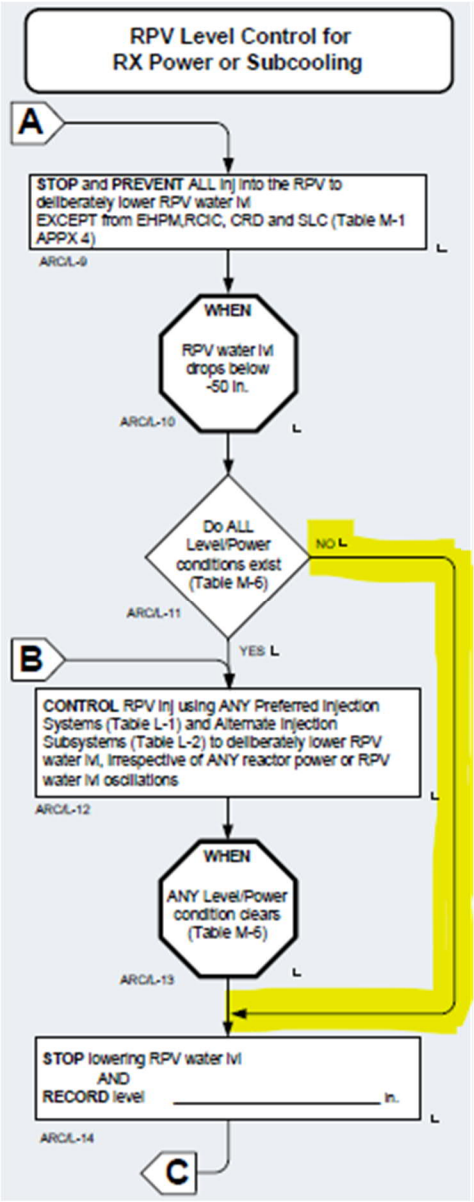
IF	THEN
Reactor Power is above 5% or unknown AND RPV Water Level is above (-) 50 in.	A
ALL Level/Power conditions exist (Table M-6)	**NO ACTION REQUIRED**
Reactor Power is above 10% or unknown AND RPV Water Level is below (-) 162 in. AND MSRV open/cycling OR Drywell Pressure above 2.4 psig	**NO ACTION REQUIRED**

Table M-6
Level/Power Conditions

- Suppression Pool Temperature is above 110°F ☐
- Reactor Power above 5% OR unknown ☐
- RPV Level above -162 in. ☐
- MSRV open/cycling OR DW pressure above 2.4 psig ☐

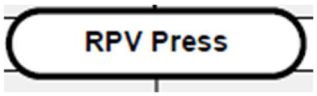
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 5 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	Directs the crew to maintain Reactor Water Level in any band between (-) 50 and (-180) inches to keep Reactor Power low.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 6 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	 ARC/P-1 **IF**: **EMERGENCY DEPRESSURIZATION IS REQUIRED** — **THEN**: **NO ACTION REQUIRED** ARC/P-2 **IF** ANY MSRV is cycling **THEN MANUALLY OPEN** MSRVs UNTIL RPV Pressure drops to the Pressure at which all Main Turbine Bypass Valves are fully open (PPX 11A) ARC/P-3 **IF**: **BORON INJECTION IS REQUIRED** AND The Main Condenser is available AND There has been no indication of a steam line break — **THEN**: **NO ACTION REQUIRED (MSIV ISOLATION)** **IF**: RPV Depressurization will not result in loss of injection required for Adequate Core Cooling AND: • Suppression Pool Temperature and Level CANNOT be maintained in the safe area of Curve 3 at the existing RPV Pressure OR • Suppression Pool Level CANNOT be maintained in the safe area of Curve 4 — **THEN**: **NO ACTION REQUIRED**

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 7 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	 ARC/P-4 **RESTORE** and **MAINTAIN** RPV Pressure between 1073 psig and 350 psig using the Main Turbine Bypass Valves (APPX 8B) and maintain cooldown rate below 100 ºF/hr ➢ Use Alternate RPV Pressure Control Systems (Table P-1), if necessary ➢ Crosstie CAD or MSRV carts to Drywell Control Air (APPX 8G, 20H), if necessary

IF	THEN
Drywell Control Air is or becomes unavailable	**NO ACTION REQUIRED**

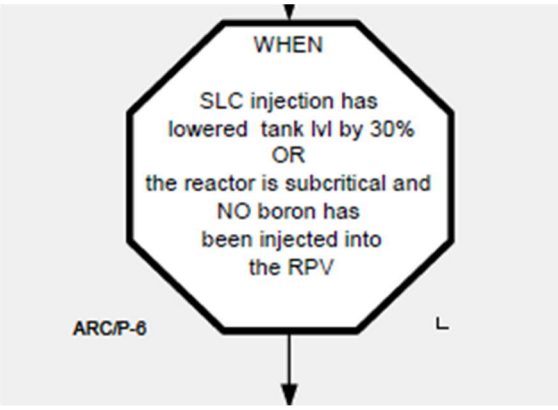
Table P-1
Alternate RPV Pressure Control Systems

Source	APPX
Main turb bypass vlvs	8B
MSRVs only if suppr lvl is above 5.5 ft IF "MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW" annunciator (XA-55-3D-18) is in alarm THEN **MINIMIZE** MSRV cycling by using sustained opening for depressurization	11A
HPCI with CST suction if available (2) (5) (7)	11C
RCIC with CST suction if available (2) (3) (5)	11B
RFPTs on min flow	11F
Main Steam system drains	11D
Steam seals	11G
SJAEs	11G
Off Gas preheater	11G
HPCI and RCIC drains	11J
RWCU if NO boron has been injected into the RPV	11E
MSRV Remote Operation only if suppr lvl is above 5.5 ft	EDMG-24 APPX. D FSI-3E

Form 3.3-2 Required Operator Actions

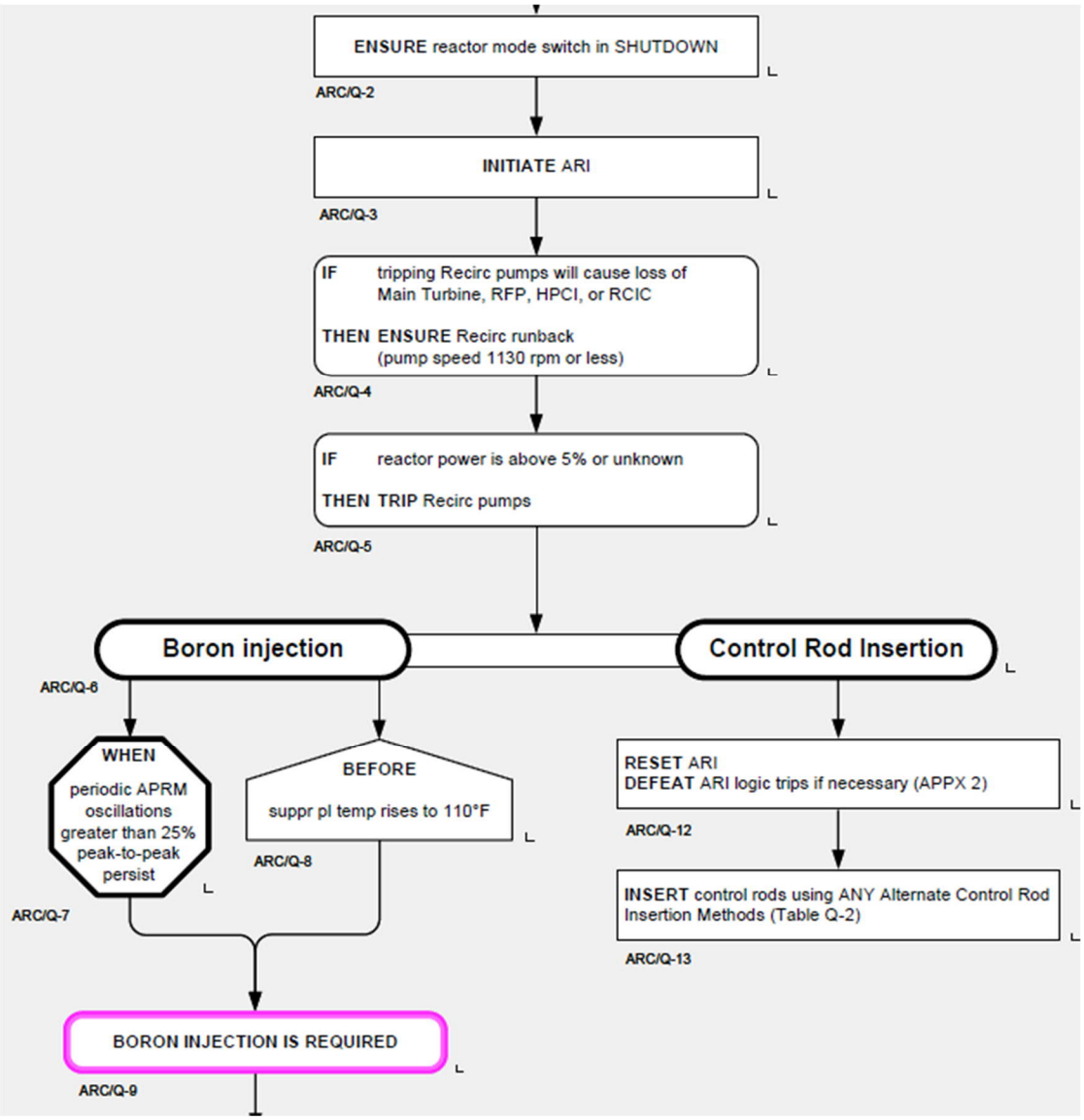
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 8 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>While MSIVs are OPEN, directs the crew to maintain Reactor Pressure using the Main Turbine (until it is required to be tripped at < 50MWE) and then using 2-EOI-Appendix-8B, Reopening MSIVs/Bypass Valve Operation.</td></tr>
<tr><td></td><td>NRC</td><td>When the MSIVs isolate, this method of Reactor Pressure Control will not be available. See Event 9 for Reactor Pressure Control actions following MSIV isolation.</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-5
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>The Reactor is NOT subcritical</td><td>E</td></tr>
<tr><td>It is anticipated that RPV Depressurization will result in a loss of injection for Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
</table>

</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>The Reactor is subcritical
AND
NO boron has been injected</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 9 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 10 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	Table Q-2 Alternate Control Rod Insertion Methods CONDITIONS / METHODS / APPX Scram valves failed to open: **DEENERGIZE** scram solenoids — 1A; **VENT** scram air header — 1B Scram valves opened but SDV is full: 1. **RESET** scram **DEFEAT** RPS logic if necessary 2. **DRAIN** SDV 3. **RECHARGE** accumulators 4. **INITIATE** scram — 1F Manual control rod insertion methods: **DRIVE** control rods **BYPASS** RWM and **RAISE** CRD drive water differential pressure if necessary — 1D; **RAISE** CRD cooling water header dp — 1G; **SCRAM** individual control rods — 1C; **VENT** control rod over piston volumes — 1E
	NUSO	1. **INITIATE** SLC 2. **INHIBIT** ADS 3. RECORD SLC tank level . IF / THEN Boron CANNOT be injected using SLC — **INJECT** boron into RPV using CRD (APPX 3B) SLC tank water lvl drops to 0% — **TRIP** the SLC pumps ARC/Q-10 **ENSURE** RWCU system isolation ARC/Q-11

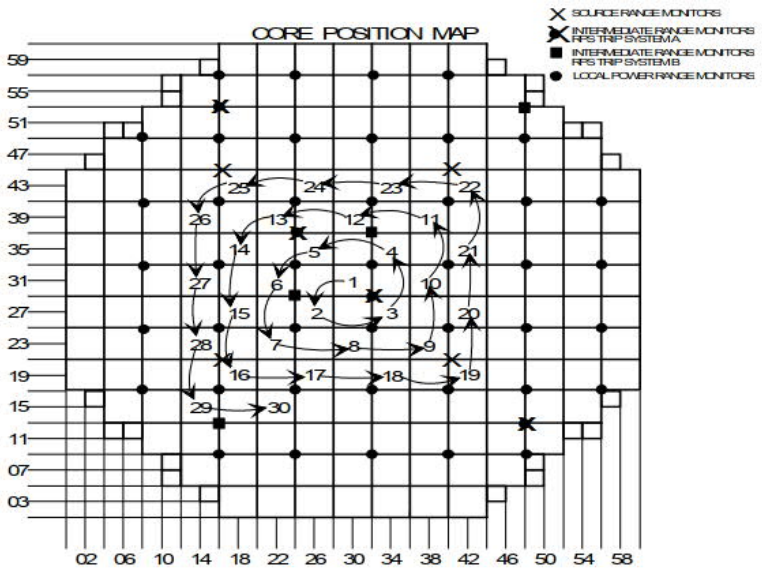
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 11 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI-APPENDIX-8A, Bypassing Group 1 Low Low Low Level Isolation Interlocks [1] **BYPASS** Group 1 RPV Low-Low-Low Level Isolation Interlocks as follows (Unit 3 Control Room, Panel 9-4): [1.1] **PLACE** keylock switch 2-HS-064-0056A, GRP 1 RPV LOW LEVEL BYPASS (SYS A1), in BYPASS. [1.2] **PLACE** keylock switch 2-HS-064-0056B, GRP 1 RPV LOW LEVEL BYPASS (SYS B1), in BYPASS. [1.3] **PLACE** keylock switch 2-HS-064-0056C, GRP 1 RPV LOW LEVEL BYPASS (SYS A2), in BYPASS. [1.4] **PLACE** keylock switch 2-HS-064-0056D, GRP 1 RPV LOW LEVEL BYPASS (SYS B2), in BYPASS. [1.5] **ENSURE** closed the following valves (Unit 3 Control Room, Panel 9-3): • 2-FCV-43-13, RX RECIRC SAMPLE INBOARD ISOLATION VALVE • 2-FCV-43-14, RX RECIRC SAMPLE OUTBOARD ISOLATION VALVE [2] N/A
	OATC	Drives Control Rods using 2-EOI-Appendix-1D, Insert Control Rods Using Reactor Manual Control System [1] **VERIFY** at least one CRD Pump in service. **NOTES** 1) Closing 2-85-586, CHARGING WATER ISOL valve may reduce the effectiveness of EOI Appendix 1A or 1B. 2) A ladder may be required to perform the following step. REFER TO Tools and Equipment, Attachment 1. 3) IF necessary, an alternate ladder is available at the HCU Modules, East and West banks. It is stored by the CRD Charging Cart.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 12 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	[2] **IF** Reactor SCRAM or ARI CANNOT be reset, **THEN DISPATCH** personnel to close 2-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft). [3] **VERIFY** REACTOR MODE SWITCH in SHUTDOWN. [4] **BYPASS** Rod Worth Minimizer. [5] **REFER TO** Attachment 2 and INSERT Control Rods in the area of highest power as follows: [5.1] **SELECT** Control Rod. [5.2] **PLACE** CRD NOTCH OVERRIDE switch in EMERG ROD IN position UNTIL Control Rod is NOT moving inward. [5.3] **REPEAT** Steps 1.0[5.1] and 1.0[5.2] for each Control Rod to be inserted. [6] **WHEN** NO further Control Rod movement is possible or desired, **THEN DISPATCH** personnel to verify open 2-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft).
	OATC	**Attachment 2** **(Page 1 of 1)** **Core Position Map**  (EXAMPLE ONLY)

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 13 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	2-EOI-APPENDIX-1F, Manual SCRAM [1] **VERIFY** Reactor Scram and ARI reset. [1.1] **IF** ARI CANNOT be reset, **THEN EXECUTE** EOI Appendix 2, Defeating ARI Logic Trips concurrently with Step 1.0[1.2] of this procedure. [1.2] N/A for the Control Room, this step is performed locally in the Auxiliary Instrument Room [2] **WHEN** RPS Logic has been defeated, **THEN RESET** Reactor Scram. [3] **VERIFY** OPEN Scram Discharge Volume Vent and Drain Valves. [4] **DRAIN** SDV UNTIL the following annunciators clear: • WEST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 2-9-4, 2-XA-55-4A, Window 1) • EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 2-9-4, 2-XA-55-4A, Window 29)
	NRC	**After the SCRAM has been reset, the accumulators will drain in approximately 7 minutes and the alarms at Panel 2-9-4, Windows 1 and 29, will clear. The OATC may then attempt a Reactor SCRAM.**
	OATC	[5] **DISPATCH** personnel to **VERIFY OPEN** 2-SHV-085-0586, CHARGING WATER ISOLATION. **NOTES** 1) If EOI Appendix 2 has been executed, ARI initiation or reset will NOT be possible or necessary in Step 1.0[6]. 2) If Reactor Pressure is greater than 600 psig, NUSO may direct performance of step 1.0[6] prior to accumulators being fully recharged. [6] **WHEN** CRD Accumulators are recharged, **THEN INITIATE** manual Reactor SCRAM and ARI.
	NRC	**Control Rods will insert the first time the OATC attempts a Reactor SCRAM after the ATWS.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	[7] **CONTINUE** to perform Steps 1.0[1] through 1.0[6] UNTIL ANY of the following exists: • ALL Control Rods are fully inserted, OR • NO inward movement of Control Rods is observed, OR • NUSO directs otherwise
	NRC	**End of Event 8. When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 1 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Symptoms/Cues: Event is initiated when the Reactor MODE SWITCH is placed in SHUTDOWN. The Main Steam Isolation Valves will close two (2) minutes after the Event is initiated

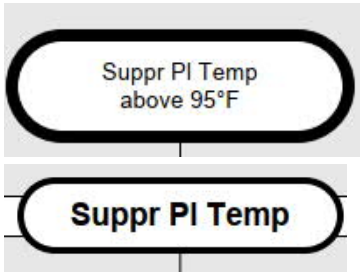
<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NRC</td><td>Event 9 is automatically entered on Simulator setup. No action is required by the Driver to insert Event 9.</td></tr>
<tr><td></td><td>BOP</td><td>Informs the NUSO that MSIVs have closed.</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the BOP to maintain Reactor Pressure using 2-EOI-Appendix-11A, Alternate Pressure Control Systems MSRVs.</td></tr>
<tr><td></td><td>BOP</td><td>2-EOI-APPENDIX-11A, Alternate RPV Pressure Control Systems MSRVs

[1] N/A
[2] N/A
[3] OPEN MSRVs using the following sequence to control RPV pressure as directed by the NUSO:

<table>
<tr><td>1</td><td>2-PCV-1-179</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>2</td><td>2-PCV-1-180</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>3</td><td>2-PCV-1-4</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>4</td><td>2-PCV-1-31</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>5</td><td>2-PCV-1-23</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>6</td><td>2-PCV-1-42</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>7</td><td>2-PCV-1-30</td><td>MN STM LINE C RELIEF VALVE</td></tr>
<tr><td>8</td><td>2-PCV-1-19</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>9</td><td>2-PCV-1-5</td><td>MN STM LINE A RELIEF VALVE</td></tr>
<tr><td>10</td><td>2-PCV-1-41</td><td>MN STM LINE D RELIEF VALVE</td></tr>
<tr><td>11</td><td>2-PCV-1-22</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>12</td><td>2-PCV-1-18</td><td>MN STM LINE B RELIEF VALVE</td></tr>
<tr><td>13</td><td>2-PCV-1-34</td><td>MN STM LINE C RELIEF VALVE</td></tr>
</table>
</td></tr>
</table>

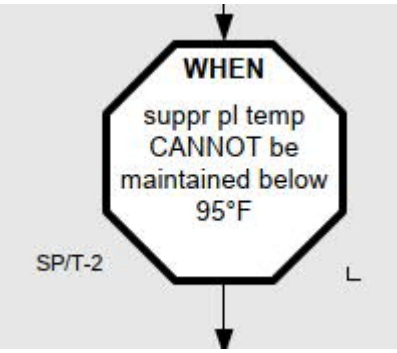
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 2 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	BOP	[4] N/A [5] N/A [6] N/A
	BOP	Acknowledges and reports the following alarm when received: • SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, 2-9-3E, Window 12
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	(If received) 2-ARP-9-3E, Alarm Response Procedure SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, Window 12 Operator Action: A. **IF** alarm is valid, **THEN ENTER** 2-EOI-2, Primary Containment Control.
	NUSO	Enters 2-EOI-2, Primary Containment Control.
	NUSO	 SP/T-1 **MONITOR** and **CONTROL** Suppr Pool Temperature below 95°F using available Suppression Pool Cooling (APPX 17A)

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 3 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	NUSO	 SP/T-3 **OPERATE** all available Suppression Pool Cooling using only RHR Pumps NOT required for continuous injection (APPX 17A)
	NUSO	Directs the BOP to place Suppression Pool Cooling in service in accordance with 2-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling.
	BOP	2-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling **NOTE** Placing a BYPASS SEL switch in BYPASS in Step 1.0[1] below prevents automatic opening of the affected RHR loop's Outboard Injection Valve. This makes LPCI Mode of that RHR Loop inoperable. [1] **IF** the unit is in Mode 1,2 or 3 and it is desired to maintain LPCI Operability, **THEN N/A** this step. [1.1] **IF** Adequate Core Cooling is assured **OR** directed to cool the Suppression Pool irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI Injection Valve open interlock AS NECESSARY: • **PLACE** 2-HS-74-155A, LPCI SYS I OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS • **PLACE** 2-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 4 of 5

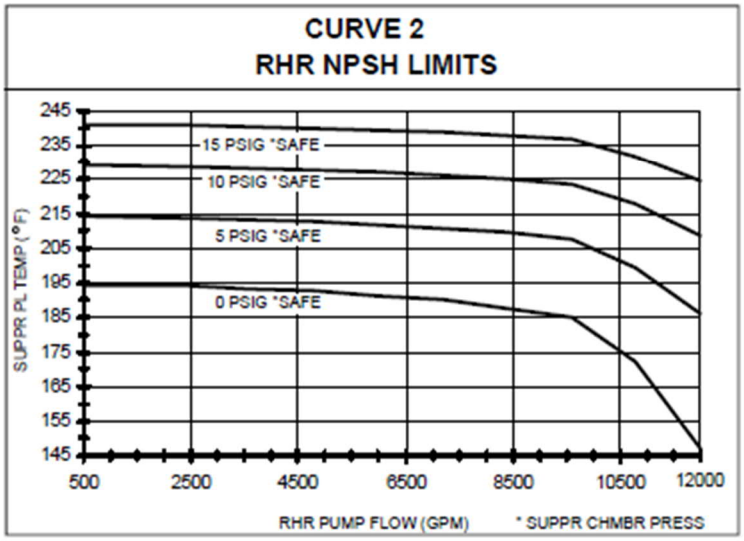
Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **PLACE** RHR SYSTEM I(II) in Suppression Pool Cooling as follows: [2.1] **ENSURE** at least one RHRSW Pump supplying each EECW header. [2.2] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s). [2.3] **THROTTLE** the following in service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW Flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VALVE • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VALVE • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE [2.4] **IF** Directed by SRO, **THEN PLACE** 2-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. [2.5] N/A [2.6] **IF** 2-FCV-74-53(67), RHR SYS I(II) LPCI INBD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 2-FCV-74-52(66), RHR SYS I(II) LPCI OUTBOARD INJECTION VALVE. [2.7] **OPEN** 2-FCV-74-57(71), RHR SYS I(II) SUPPRESSION CHAMBER/POOL ISOLATION VALVE. [2.8] **ENSURE** desired RHR Pump(s) for Suppression Pool Cooling are operating. [2.9] **THROTTLE OPEN** 2-FCV-74-59(73), RHR SYS I(II) SUPPRESSION POOL COOLING/TEST VALVE, to maintain EITHER of the following as indicated on 2-FI-74-50(64), RHR SYS I(II) FLOW: Between 7,000 and 10,000 gpm for one pump operation **OR** At or below 13,000 gpm for two pump operation [2.10] **ENSURE CLOSED** 2-FCV-74-7(30), RHR SYS I(II) MINIMUM FLOW VALVE. [2.11] **MONITOR** RHR Pump NPSH using Attachment 1.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 5 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	BOP	Attachment 1 (Page 1 of 1) **NPSH Monitoring** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. 
	BOP	[2.12] **NOTIFY** Chemistry that RHRSW is aligned to in service RHR Heat Exchangers. [2.13] **IF** Additional Suppression Pool Cooling flow is necessary, **THEN PLACE** additional RHR and RHRSW pumps in service using Steps 1.0[2.2] through 1.0[2.12].
	Driver	**When contacted as Chemistry, acknowledge any information or direction given.**
	NRC	**End of Event 9. When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Form 3.3-2 Required Operator Actions

Scenario Setup
UNIT 2

IC	28
Exam IC	289

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-65	56	ARP-9-3D	38	TS-3.5.1	334
OI-66	124	ARP-9-3F	37	EOI-1A	5
OI-67	126	ARP-9-23C	34	EOI-2	18
OI-68	168	ARP-9-23D	38	Appendix-1D	7
AOI-57-5A	62	GOI-100-1A	192	Appendix-1F	6
AOI-85-3	26	GOI-100-12	50	Appendix-8A	5
AOI-68-1	39	GOI-100-12A	125	Appendix-8B	9
AOI-100-1	121			Appendix-11A	5
				Appendix-17A	20

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	ILT 2304 NRC Scenario 5 UNIT 2.sch
Event Files(s):	ILT 2304 NRC Scenario 5 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ON LINE** 230	**Total Drywell Leakage (gpm)** 1.55	**Protected Equipment**
	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified		2A CRD Pump, 2C 4KV Unit Bd (due to 1B CRD Pump)
Rx Power 26%		**Floor Drain (gpm)** 0.11	
MWe 281		**Equipment Drain (gpm)** 1.44	

☐ Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐ CR Reviews Complete ☐ Leadership and Team Effectiveness

CHANGES IN LCOs

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2A Recirc Pump trip. All repairs and inspections are complete for restart.
1B CRD Pump is tagged to support breaker inspection.
IRM 'G' is bypassed to support maintenance from amplifier failure.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Start 2A Recirc Pump in accordance with 2-OI-68, Section 5.3, beginning at Step [12]. SR 3.4.9.3&4 is complete.
The SST will maintain SR 2.4.9.3&4 current.
Match speed of 2B Recirc Pump, then await instruction from the Reactor Engineer.

OPERATOR WORK AROUNDS OWAs – U0-1/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 0/U2 - 0

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-5
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 26% Reactor Power, Single Loop on 3B Recirc Pump, 3B CRD Pump is tagged for circuit breaker inspection, IRM 'G' is bypassed due to amplifier failure.

Turnover: Restore 3A Recirc Pump to service, match speed of 3B Recirc Pump, then await instructions from the Reactor Engineer. Support maintenance on 3B CRD Pump.

Critical Tasks:

1. When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the Recirc Pumps reaching zero speed to prevent possible fuel damage.
2. With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion inhibit, ADS or control Reactor Water Level such that no automatic ADS actuation occurs.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Start 3A Recirculation Pump
2.	N/A	R-OATC	Match Recirculation Pump Speeds
3.	HS-75-42A	C-BOP MC-BOP TS-NUSO	Inadvertent Core Spray Pump Start
4.	RD01A	C-OATC MC-OATC C-NUSO	Control Rod Drive (CRD) Pump Trip
5.	ED18A	C-BOP C-NUSO	Loss of I&C 'A'
6.	PC01C HS-65-69A/2	TS-NUSO	Standby Gas Train (SGT) Fails to Start
7.	TH03A TH03B	C-OATC C-NUSO	Dual Recirculation Pump Trip
8.	RD09A RD09B	M-ALL	Hydraulic Anticipated Transient Without SCRAM (ATWS)
9.	RP11	C-BOP C-NUSO	Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will restart 3A Recirculation Pump in accordance with 3-OI-68, Recirculation System.
2. The crew will raise speed of 3A Recirculation Pump to match 3B Recirculation Pump in accordance with 3-OI-68, and in doing so raise Reactor Power.
3. 3B Core Spray Pump will inadvertently start, requiring the crew to shut down 3B Core Spray Pump. When contacted, the Assistant Unit Operator (AUO) will report there is no visible oil level on 3B Core Spray Pump. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications
4. Control Rod Drive (CRD) Pump 3A will trip, requiring the crew to enter 3-AOI-85-3, CRD System Failure. The standby CRD Pump is tagged out and not available; therefore, the crew will dispatch an AUO to verify the status of the tripped pump and restart the tripped pump.
5. A Loss of I&C Bus A will occur, requiring the crew to respond in accordance with 3-AOI-57-5A, Loss of I&C Bus A.
6. The Immediate Actions of 3-AOI-57-5A, Loss of I&C Bus A, direct that Standby Gas Train(s) (SGT) be started; one SGT will not start when the crew attempts to start it, requiring the NUSO to reference Technical Specifications.
7. 3A Recirculation Pump will trip, requiring the crew to take action in accordance with 3-AOI-68-1, Recirc Pump Trip/Core Flow Decrease. During the immediate actions for the single Recirculation Pump trip, 3B Recirculation Pump will trip, requiring the crew to insert a manual Reactor SCRAM.
8. When Operator at the Controls enters a manual Reactor SCRAM due to the dual Recirc Pump trip, a hydraulic Anticipated Transient Without SCRAM (ATWS) will occur, requiring the crew to respond in accordance with 3-AOI-100-1, Reactor SCRAM, and 3-EOI-1A, ATWS RPV Control.
9. Following the Reactor SCRAM, the Main Steam Isolation Valves (MSIVs) will close due to an inadvertent isolation signal, requiring the crew to take action to restore Reactor Pressure and Reactor Water Level.

The Scenario ends when the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available high pressure injection system.

Critical Tasks 2

1. **When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the Recirc Pumps reaching zero speed to prevent possible fuel damage.**

   a. **Safety Significance**

   With thermal power being produced in the Reactor and actions to lower Reactor Power have not brought power out of the heating range, power oscillations and subsequent fuel damage may be generated.

   b. **Cues**

   The Reactor is SCRAMMED and ATWS actions are taken by the OATC and Reactor Power is greater than 5% or unknown.

   c. **Measured by**

   Reactor Power on APRM displays.

   d. **Feedback**

   Lowering Reactor Water Level.
   Lowering Reactor Power.

   e. **Critical Task Failure Criteria:**

   The operating crew fails to STOP and PREVENT injection and take action to lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds after Recirculation Flow is stopped.

2. **With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.**

   a. **Safety Significance**

   Precludes core damage due to an uncontrolled reactivity addition.

   b. **Cues**

   Procedural Compliance.

   c. **Measured by**

   ADS Logic inhibited prior to an automatic initiation.

   d. **Feedback**

   Reactor Pressure trend.
   Reactor Water Level trend.
   ADS LOGIC BUS A/B INHIBITED (3-9-3C, Window 18/31) annunciator status.

   e. **Critical Task Failure Criteria**

   ADS automatic initiation with Control Rods out.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1

Event Description: Start 3A Recirculation Pump

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**The crew has been briefed that another operator is responsible for 3-SR-3.4.9.3&4, Reactor Recirculation Pump Start Limitations.** **The crew has been briefed to start in 3-OI-68, Section 5.3 on Step [12].** **The crew has been pre-briefed on this procedure, but may elect to hold a re-focus brief.**
	NUSO	Directs the Balance of Plant Operator (BOP) to start 3A Recirculation Pump Speed.
	BOP	3-OI-68, Reactor Recirculation System Section 5.3, Recirc Pump Startup [10] **CHECK** RECIRC LOOP A(B) DIFFERENTIAL PRESS LOW annunciation, 3-XA-55-4A(4B), window 31 in ALARM. **NOTE** All faults and alarms should be reset prior to closing a Recirc Pump Discharge Valve to ensure the DRIVE READY light is obtained thus preventing the loop from cooling down causing the start limitations of the Recirc Pump to possibly be exceeded thus delaying the startup. [11] **DEPRESS** 3-HS-96-13, FAULT RESET, **AND CHECK** all Faults or Alarms are reset on ICS Screen VFDAAL. [11.1] **IF** all faults and alarms are clear, **THEN CONTINUE** Recirc Pump Startup at Step 5.3[12]. [11.2] N/A

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 2 of 6

Event Description: Start 3A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	**CAUTION** The closure circuit for 3-HS-68-3A, RECIRC PUMP 3A DISCHARGE VALVE, is a seal-in and is NOT to be held in the CLOSE position unless specifically directed. If the handswitch is held to the close position (approx. 2.5 seconds) after the valve has reached its full close valve position this will allow the reclosure circuit to engage. This reclosure causes the motor to re-engage at a rapid rate (approx. 10 times a second), which Can cause the motor to heat and also to exceed the recommended valve seat loading. [12] **ENSURE** 3-FCV-68-3, RECIRC PUMP 3A DISCHARGE VALVE, CLOSED. **NOTES** 1) The DRIVE READY light will only light after the FAULT RESET push button is depressed, if all the active faults are reset. ICS screen VFDAAL(VFDBAL) can be referred to verify the Number of Active Faults is zero after depressing the fault reset push buttons. Any faults listed after the FAULT RESET push button has been depressed, are keeping the drive from being ready to run. 2) Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161 source and large pumps are started. Unit SROs are to evaluate the need to place the Capacitor Banks in Manual prior to starting a Recirc pump from the Alternate feeder. The evaluation should consider placing a Caution order on the Recirc pump's start switches stating, "evaluate the need to place CAP Banks in Manual prior to starting Pumps from the ALTERNATE FEED." 3) The actions which occur after the START push-button is depressed are listed in Step [18]. 4) When depressing the switches which control the Recirc Drives these switches are to be firmly depressed to ensure all the contacts are made-up.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 3 of 6

Event Description: Start 3A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[13] IF the Recirc Drive startup sequence incomplete alarm is received and the Recirc Discharge Valve is jogging OPEN, THEN (N/A if alarm is not received) [13.1] **CLOSE** and **HOLD**, 3-FCV-68-3, RECIRC PUMP 3A DISCHARGE VALVE. [13.2] **OPEN** then **CLOSE** RECIRC PUMP A INSTR BKR 519(520) Panel 9-9 cabinet 5. [13.3] **RELEASE** 3-FCV-68-3, RECIRC PUMP 3A DISCHARGE VALVE. NOTES 1) Tech Spec LCO 3.4.1 should be reviewed prior to starting Recirc Pump 3A in Step 5.3[14] The applicability for the Tech Spec is Modes 1 & 2. 2) Step 5.3[14] is used to start Recirc Pump 3A under normal conditions. 3) Step 5.3[15] can be used to start Recirc Pump 3A during cold shutdown (Mode 4) when it is desired to start the pump and open the discharge valve without the jogging circuit. [14] **IF** starting Recirc Pump 3A, **THEN PERFORM** the following: [14.1] **CHECK** Tech Spec LCO 3.4.1 requirements are satisfied. [14.2] **DEPRESS** FAULT RESET, 3-HS-96-13. [14.3] **CHECK** DRIVE READY, 3-IL-96-37 is LIT. [14.4] **FIRMLY** DEPRESS DRIVE START, 3-HS-96-11. [14.5] **GO TO** Step 5.3[18]. [15] N/A [16] N/A [17] N/A

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 4 of 6

Event Description: Start 3A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[18] **CHECK** the following for Recirc Pump being started: • DRIVE RUNNING, 3-IL-96-41 is LIT • Pump DP rises to above 5 psid as indicated on 3-PDI-68-65, PUMP DP • Recirc Pump 3A(3B) Speed rises to ~345 RPM, as indicated on 3-SI-68-59(71), RECIRC PUMP 3A SPEED • RECIRC LOOP A(B) DIFF PRESS LOW 3-XA-55-4A, Window 31, is reset. • 3-FCV-68-3, RECIRC PUMP 3A DISCHARGE VALVE, begins to open by the automatic jogging circuit **NOTES** 1) In order to achieve balanced jet pump flows, the Recirc Pumps Speed may require a mismatch. 2) Recirc Pump Speed cannot be raised above 480 rpm (28%) until total Feedwater Flow is greater than 16 percent. Recirc Pump Speed can be controlled between ~345 RPM and ~480 RPM using the Recirc Drive/Recirc Pump Speed Controllers. 3) Recirc Pump A will trip 85 seconds after initiation of the automatic jogging sequence if 3-FCV-68-3, RECIRC PUMP A DISCHARGE VALVE, is less than 90 percent open. 4) Performance of 3-SR-3.4.2.1 is required 24 hours after reaching > 23% RTP and/or 4 hours after returning a Recirc Pump to service. 5) After Recirc loop(s) are in service, Technical Specifications Surveillance Requirements SR 3.4.1.1 and SR 3.4.2.1 have performance requirements which are to be referred to while performing Step 5.3[20]. [19] **CHECK OPEN** 3-FCV-68-3, RECIRC PUMP 3A DISCHARGE VALVE.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 5 of 6

Event Description: Start 3A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[20] **NOTIFY** Reactor Engineer to **CHECK** that the following has been performed, as applicable, depending on the number of operating Recirc loops, and **RECORD** the applicable requirements and their completion status in the narrative log. [20.1] 3-SR-3.4.1(DLO), Reactor Recirculation System Dual Loop Operation. OR [20.2] 3-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation.
	Driver	**If contacted as the Reactor Engineer, acknowledge any information or direction given. Report that all Surveillance Requirements (including 3-SR-3.4.2.1) are in progress or completed as required.**
	BOP	[21] **PERFORM** 3-SR-3.4.2.1 within the next 4 hours following placing a Recirc loop in service. (REFER TO Tech Specs 3.4.1 and 3.4.2.) [22] **DEPRESS** push-button 3-HS-68-32, RECIRC PUMP 3A(3B) RUNBACK RESET. [23] **MONITOR** Recirc Pump Seal Pressures during pressurization for proper operation. (Number 2 Seal Pressure should be approximately one half that of Number 1 Seal Pressure). [24] THROTTLE 3-THV-024-0976, VFD 3A HX OUTLET RCW THROTTLE VALVE, to maintain VFD inlet and outlet temperatures. **CAUTION** The Recirc System should be operated with balanced Jet Pump Flows to reduce hydraulic forces and vibration stresses on Jet Pumps and Retainers. [25] **ADJUST** Recirc Pump Speed 3A using 3-HS-96-15A(15B), RAISE SLOW (MEDIUM), or 3-HS-96-17A(17B), LOWER SLOW(MEDIUM), push buttons, to 480 rpm. [26] **WHEN** total Feedwater flow is greater than 16%, **THEN CHECK** reset, RECIRC LOOP A(B) FLOW LIMITER ENFORCING, 3-XA-55-4A, Window 35.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 1 Page 6 of 6

Event Description: Start 3A Recirculation Pump

Time	Position	Applicant's Actions or Behavior
	BOP	[27] N/A [28] **IF** this is the second Recirc Pump started, **THEN PERFORM** 3-SR-3.4.2.1 for the Jet Pump Mismatch requirements, within the next 24 hours (**REFER TO** Tech Specs 3.4.1.1). [29] **RAISE** Recirc Pumps 3A and/or 3B Speed as directed by 3-GOI-100-1A or 3-GOI-100-12, or 3-GOI-100-12A **REFER TO** Section 6.2.
	Driver	**If contacted as the Reactor Building Assistant Unit Operator (AUO), acknowledge any information or direction given.**
	NRC	**End of Event 1. Proceed to Event 2, Match Recirculation Pump Speeds.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 2 Page 1 of 1

Event Description: Match Recirculation Pump Speeds

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	NRC	**No action is required by the Driver to insert Event 2. Event 2 is cued by the turnover sheet.**
	NRC	**The crew will match Recirculation Pump Speeds by raising 3A Recirc Pump Speed in accordance with 3-OI-68 and the provided Reactivity Control Plan (RCP).**
	OATC	3-OI-68-1, Reactor Recirculation System [1] **IF** desired to control Recirc Pump 3A speed with Recirc Individual Control, **THEN PERFORM** the following; [1.1] Raise Recirc Pump 3A Speed using, 3-HS-96-15A(15B), RAISE SLOW (MEDIUM) (Otherwise N/A) [1.2] Lower Recirc Pump 3A Speed using 3-HS-96-17A(17B)(17C), SLOW (MEDIUM) (FAST) (Otherwise N/A) [2] N/A [3] N/A
	NRC	**When satisfied with the Reactor Power change, end of Event 2. Request that the Driver insert Event 3, Inadvertent Core Spray Pump Start.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 1 of 4

Event Description: Inadvertent Core Spray Pump Start

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. Core Spray Pump 3B will start without an initiation signal

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to start 3B Core Spray Pump.**
	BOP	Acknowledges and reports the following alarms: • CORE SPRAY SYS II PUMP B START, 3-9-3F, Window 1 • RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 3-9-3C, Window 10 • CORE SPRAY PUMP 3B RUNNING, 3-9-23C, Window 33 • RHR SERVICE WATER PUMP A3 RUNNING, 3-9-23A, Window 32 **(only received if the Core Spray Pump runs longer than 28 seconds)** Reports that 3B Core Spray Pump has started and that there are no valid signals that should result in Core Spray Pumps starting.
	NUSO	Directs the BOP to respond in accordance with Alarm Response Procedures.
	BOP	3-ARP-9-3F, Alarm Response Procedure CORE SPRAY SYS II PUMP B START, Window 1 Operator Action: A. **CHECK** auto start signals by multiple indications. B. **CHECK** Core Spray Pump 3B operation by motor amps, discharge pressure and flow. C. **IF** pump is NOT needed, **THEN STOP** Core Spray Pump 3B. D. N/A
	BOP	3-ARP-9-23C, Alarm Response Procedure CORE SPRAY PUMP 3B RUNNING, Window 33 Operator Action: A. **IF** accident signal is NOT present, **THEN** 1. CHECK panel 9-3 to verify pump is running.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 2 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior
	BOP	2. **INITIATE** actions to determine why pump is running. 3. **SECURE** pump.
	BOP	Alarm Response Procedure, 3-9-23A RHR SERVICE WATER PUMP A3 RUNNING, Window 32 Operator Action: A. **IF** accident signal is **NOT** present, **THEN PERFORM** the following: 1. **CHECK** Panel 3-9-3 to verify RHRSW Pump A3 is running. 2. **ENSURE** adequate Raw Cooling Water (RCW) Pressure PI-24-18, Panel 9-20. 3. **CHECK** Diesel Generator or Core Spray Pump running.
	Driver	**If informed as Unit 1 and 2, acknowledge that 3B Core Spray Pump inadvertently started, which caused A3 RHRSW Pump to start (if 3B Core Spray Pump is not secured in 28 seconds). If asked, neither Unit 1 nor Unit 2 requires that A3 RHRSW Pump remain running.** **If contacted as the Control Bay AUO to check the ATUs in the Aux. Instrument Room, acknowledge the direction. Wait 2 minutes and report that there are no ATUs tripped that are common to Core Spray Initiation Logic.** **When contacted as Maintenance, the Outside/Work Control SRO, or an AUO to investigate, wait 3 minutes and report that the oil level on 3B Core Spray Pump is below the sight glass.**
	BOP	Determines that 3B Core Spray Pump inadvertently started, recommends to the Nuclear Unit Senior Operator (NUSO) that 3B Core Spray Pump be secured, and secures 3B Core Spray Pump and A3 RHRSW Pump when directed.
	BOP	0-OI-67, Emergency Equipment Cooling Water System Section 7.1, Return to Standby Readiness of the North Header [1] **ENSURE** Chemistry treatment to associated RHRSW (EECW) Pump discharge line secured.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 3 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **ENSURE** there are NO emergency Diesel Generators running on any unit. [3] **ENSURE** there are NO Core Spray Pumps running on any unit. [4] **ENSURE** EECW system is **NOT** being used to supply cooling water to RBCCW. [5] **ENSURE** EECW System is **NOT** being used to supply cooling water to the Control Air Compressors. **NOTE** The RHRSW Pump Control Switches and amp meters are located at Control Room Panel 9-3, Unit 1, 2, and 3. **CAUTION** It is normal practice to keep one RHRSW Pump running on each of the North and South EECW headers to ensure the headers remain solid with no trapped air, and to ensure operability requirements. [6] **IF** Pump A3 is to be **SHUT DOWN**, **THEN PLACE** one of the following switches momentarily to the STOP position: • 0-HS-23-85A/3, RHRSW PUMP A3 EECW NORTH HDR, on Unit 3 [7] N/A [8] **IF** either pump remains in operation, **THEN CHECK** running current for the remaining pump is less than 61 amps using one of the following: • 0-EI-23-91/3, RHRSW PUMP C3 AMPS, on Unit 3
	NRC	**The candidate may reference Technical Specification 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation; however, given that the Driver reported as an AUO that there were no common ATUS for Core Spray Logic tripped in the Aux instrument Room, Technical Specification 3.3.5.1 is not required.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 3 Page 4 of 4

Event Description: Inadvertent Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior	
	NUSO	References Technical Specification 3.5.1, ECCS-Operating LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE. Applicability: MODE 1, MODES 2 and 3, except High Pressure Coolant Injection (HPCI) and ADS Valves are not required to be OPERABLE with Reactor Steam Dome Pressure ≤ 150 PSIG. <u>Condition:</u> A. One Low Pressure ECCS Injection/Spray subsystem inoperable. <u>OR</u> One Low Pressure Coolant Injection (LPCI) Pump in both LPCI subsystems inoperable.	
	NUSO	**REQUIRED ACTION:** A. Restore Low Pressure ECCS Injection/Spray subsystem(s) to OPERABLE status.	**COMPLETION TIME:** B.1 – 7 days
	Driver	**If contacted as the Outside / Work Control NUSO or as an AUO to hang a clearance on or open the circuit breaker for 3B Core Spray Pump, wait 2 minutes and insert Event 13. Contact the crew to inform them that the breaker for 3B Core Spray Pump is open.** **If contacted as Engineering to perform an evaluation, acknowledge the direction.** **If contacted as the Outside/Work Control SRO to address Protected Equipment acknowledge the direction.**	
	NRC	**End of Event 3. Request that the Driver insert Event 4, Control Rod Drive (CRD) Pump Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 4

Event Description: Control Rod Drive (CRD) Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3A CRD Pump will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to trip 3A CRD Pump.**
	NRC	**CRD Temperature alarms will be received in approximately 4 and a half minutes after the CRD Pump trip.**
	OATC	Informs the NUSO that 3A Control Rod Drive (CRD) Pump has tripped.
	NUSO	Directs the Operator at the Controls (OATC) to respond in accordance with 3-AOI-85-3, CRD System Failure.
	OATC	3-AOI-85-3, CRD System Failure 4.1 Immediate Actions: [1] N/A [2] N/A 4.2 Subsequent Actions **NOTES** 1) Operation with loss of CRD system should be minimized since CRD Mechanism Cooling Water, Reactor Recirc Pump Seal Purge Flow, Reactor Water Cleanup Recirc Pump Seal Water Flow, and CRD HCU Accumulator Charging Water will be lost. Technical Specification Section 3.1.3 and 3.1.5 should be consulted. 2) Do NOT attempt to start a CRD Pump which has tripped for unknown reasons before verifying locally that the pump, motor and breaker are in operating condition. 3) Steps 4.2[1] and 4.2[2] should be performed concurrently. [1] **IF** the operating CRD Pump has tripped **AND** backup CRD Pump is **NOT** available, **THEN PERFORM** the following: (Otherwise N/A) [1.1] **DISPATCH** personnel to locally verify the tripped pump, motor and breaker are in operating condition.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 4 Page 2 of 2

Event Description: Control Rod Drive (CRD) Pump Trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as the Reactor Building and Control Bay AUOs to verify 3A CRD Pump, motor, and breaker are in operating condition, acknowledge the direction.** **Wait 2 minutes and report that 3A CRD Pump, motor, and breaker are in operating condition. As the Reactor Building AUO report that there were scaffold builders in the area of 3A CRD Pump – you have directed them to report to the Control Room.**
	OATC	[1.2] **IF** the tripped pump is determined to be in operating condition AND with Unit supervisor permission, **THEN PERFORM** the following at Panel 3-9-5: (Otherwise N/A) [1.2.1] **PLACE** CRD SYSTEM FLOW CONTROL, 3-FIC-85-11, in MAN at minimum setting. [1.2.2] **ATTEMPT TO RESTART** tripped CRD Pump using one of the following: • CRD PUMP 3B, using 3-HS-85-2A • CRD Pump 3A, using 3-HS-85-1A [1.2.3] **ADJUST** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, to establish the following conditions: • 3-PDI-85-18A, CRD CLG WATER HEADER DP, between 10 psid and 20 psid • 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, between 40 and 65 gpm [1.2.4] **BALANCE** 3-FIC-85-11, CRD SYSTEM FLOW CONTROL, and **PLACE** in AUTO or BALANCE.
	NRC	**End of Event 4. Request that the Driver insert Event 5, Loss of I&C 'A'.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 1 of 4

Event Description: Loss of I&C 'A'

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. I&C 'A' will de-energize, causing a loss of loads and various alarms

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a loss of I&C 'A'.**
	NRC	**Numerous alarms are received during this event. Only some alarms are addressed in this guide.** **When Event 5 is inserted, a Reactor Building Ventilation Static Isolation will occur.**
	BOP	Acknowledges and reports the following alarm: • I&C BUS A VOLTAGE ABNORMAL, 3-9-8C, Window 21 • REACTOR BUILDING VENTILATION ABNORMAL, 3-9-3D, Window 3
	NUSO	Directs the BOP to respond in accordance with the Alarm Response Procedure and 3-AOI-57-5A, Loss of I&C Bus A.
	BOP	3-ARP-9-8C, Alarm Response Procedure I&C BUS A VOLTAGE ABNORMAL, Window 21 Operator Action: A. **CHECK** alarm by checking: • Loss of instrument power and remote position indication to Core Spray Div I and RHR Div I (Panel 9-3) • RWCU Filter Demin 3A isolation • Reactor Building/Refuel Zone Ventilation isolation B. **REFER TO** 3-AOI-57-5A and 0-GOI-300-2.
	BOP	3-AOI-57-5A, Loss of I&C Bus A 4.1 Immediate Actions – None 4.2 – Subsequent Actions [1] **ENSURE** Automatic Actions have occurred. [2] **IF** a Reactor Scram occurs, **THEN REFER** TO 3-AOI-100-1 and PERFORM it concurrently with this procedure.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 2 of 4

Event Description: Loss of I&C 'A'

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE** a flow path for Condensate System OR STOP the Condensate Pumps/Booster Pumps. **REFER TO** 3-OI-2, Condensate System. [4] **START** Standby Gas Train(s) and CHECK Reactor Building Pressure at or below 0.25" H2O vacuum (PDIC 64-1, Panel 25-219; PDIC 64-2, Panel 25-213). **REFER TO** Standby Gas Treatment System Manual Initiation section of 0-OI-65, Standby Gas Treatment System.
	NRC	**Based on their training in the Initial License Training Program, the crew may elect to either call Unit 1 or Unit 2 to start a Standby Gas Train (SGT) or go to Panel 9-25 and start the SGT themselves. The first SGT the crew attempts to start will not start. See Event 6 on page 18 of 43 for SGT actions.**
	BOP	[5] **ENSURE** Steam Jet Air Ejector (SJAE) 3B in service to maintain Condenser Vacuum. **REFER TO** 3-AOI-47-3, Loss of Condenser Vacuum.
	BOP	Places 3B Steam Jet Air Ejector in service in accordance with 3-OI-66/HARDCARD, Standby SJAE System Lineup Hard Card. 3.0 Operator Actions for SJAE 3B **NOTES** Radiation Protection should be notified prior to placing a SJAE in service. If time does not permit this due to plant conditions, then notification should be made when possible. 3-HS-001-0375, SJAE TRAIN PERMISSIVE (located on 3-LPNL-925-0105, U3 TB, el 586') should normally be in the position of the standby SJAE. If problems are encountered while placing a SJAE in service and time permits, operate this switch as required during the performance of this section.
	Driver	**If contacted as Radiation Protection, acknowledge any information or direction given.**
	BOP	[1] **ENSURE RESET** Off-Gas isolation using 3-HS-90-155, OFFGAS OUTLET/DRAIN ISOLATION VALVES. [2] **ENSURE OPEN** the following valves: • 3-HS-66-15, SJAE 3B INLET VALVE • 3-HS-1-156A, STEAM TO SJAE 3B

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 3 of 4

Event Description: Loss of I&C 'A'

Time	Position	Applicant's Actions or Behavior
	BOP	[3] **ENSURE** in AUTO/OPEN 3-HS-66-18, SJAE 3B OFFGAS OUTLET VALVE. [4] **PLACE** 3-HS-1-152, SJAE 3B PRESS CONTROLLER, in **CLOSE** and then in **OPEN**. [5] **ENSURE OPEN** the following valves (red light illuminated): • 3-PCV-1-153/167, STEAM TO SJAE 3B STAGES 1,2, AND 3. • 3-FCV-1-152, SJAE 3B INTMD CONDENSER DRAIN. [6] **MONITOR** Hotwell Pressure as indicated on recorder 3-XR-2-26, CONDENSATE, on Panel 3-9-6. [7] **FOR** the SJAE **NOT** being placed in service, **ENSURE CLOSED** the following valves: • 3-HS-66-14, SJAE 3A OFFGAS OUTLET VALVE • 3-HS-1-150, SJAE 3A PRESSURE CONTROLLER • 3-HS-1-155A STEAM TO SJAE 3A [8] **WHEN** Condenser Vacuum has stabilized, **THEN PROCEED** to 3-OI-66, Off-Gas System "Placing Standby SJAE in Operation" **AND PERFORM** the applicable steps to establish acceptable SJAE parameters to obtain stable and live pressure indications.
	BOP	(Continuing actions of 3-AOI-57-5A, Loss of I&C Bus A) [6] N/A [7] **WHEN** Reactor Water Level is normal, **THEN RESET** PCIS Group 6 Inboard Isolation and **RETURN** the affected systems to service or standby readiness. • **IF** a Reactor Scram HAS NOT occurred, **REFER TO** 3-AOI-64-2D, Ventilation System Isolation. [8] **REQUEST** Electrical Maintenance investigate I&C Bus 3A distribution malfunction(s).
	Driver	**When contacted as Electrical Maintenance, Work Control / Outside NUSO, or an AUO to investigate, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 5 Page 4 of 4

Event Description: Loss of I&C 'A'

Time	Position	Applicant's Actions or Behavior
	BOP	[9] **REQUEST** Unit SRO evaluate Attachment 1 for Limiting Conditions of Operation and Actions required for Technical Specification compliance.
	BOP	3-ARP-9-3D, Alarm Response Procedure REACTOR BUILDING VENTILATION ABNORMAL, Window 3 Operator Action: A. N/A B. **NOTIFY** the Unit 3 NUSO C. N/A D. **DISPATCH** personnel to check Bldg DP (PDIC 64-2, El 639 Rx Bldg.). E. **DISPATCH** personnel to check 480V AC Rx Bldg Vent Bd 3A and 3B. F. N/A
	NRC	**In the interest of time management, it is not expected that the crew address Technical Specifications or Fire Protection Requirements for this event.**
	NRC	**End of Event 5. See Event 6 for Driver Actions.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 6 Page 1 of 3

Event Description: Standby Gas Train (SGT) Fails to Start

Symptoms/Cues: Event is initiated by either the crew contacting another Unit for SGT Train start or automatically by Simulator setup if the crew attempts to manually start a SGT Train. One SGT Train will not be able to be started

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 6 is inserted as follows:** • **Method 1: When the crew attempts to start any train of SGT, it will fail. Ensure that the Event File is updated (this will automatically occur) to allow any other train to be started** • **Method 2: If contacted as Unit 1 or Unit 2 to start a train of SGT, acknowledge the direction. Wait 2 minutes, and insert Event 16 to start SGT 'B'. Report to the crew that 'C' SGT will not automatically start**
	NRC	**EXAMINER NOTE: See page 20 of 43 for actions if the crew contacts another Unit to start a SGT Train.**
	BOP	Method 1: 0-OI-65, Standby Gas Treatment System Section 5.2, Standby Gas Treatment System Manual Initiation **NOTE** When manually starting SGT, verify that the train with the highest run time is used so that all train beds are not exhausted at the same time.
	Driver	**If contacted as Unit 1 and asked which SGT Train should be started, inform the crew that all run hours are the same and any train can be started.**
	BOP	[1] **ENSURE** the following requirements are satisfied: • SGT Train A(B)(C) in standby readiness. **REFER TO** Section 4.0. • Main Stack Radiation Monitoring in Service. **REFER TO** 2-OI-90, Radiation Monitoring System. • As plant conditions permit, Normal ventilation has been in operation for a minimum of 2 hours following painting, fire, or chemical release for the affected zones from which SGT will take suction. [2] **REVIEW** the Precautions and Limitations in Section 3.0.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 6 Page 2 of 3

Event Description: Standby Gas Train (SGT) Fails to Start

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as Unit 2 to verify the Radiation Monitoring System is in service, report that it is in service in accordance with 2-OI-90.**
	BOP	[3] **ENSURE** suction path is aligned to SGT System as follows: [3.1] **IF** alignment to Containment Venting or Purge suction path is desired, **THEN REFER TO** 1(2)(3)-OI-64, Primary Containment System. [3.2] **IF** alignment to Reactor Zone Ventilation suction path is desired, **THEN ENSURE OPEN** the following dampers for the desired unit(s) to be aligned. • 3-HS-64-40 and 3-HS-64-41, REACTOR ZONE EXH TO SGTS Dampers, on Panel 3-9-25 [3.3] N/A [4] **START** SGT FAN A(B)(C) as follows: [4.1] N/A [4.2] N/A [4.3] **IF** starting SGT FAN A(B)(C) from Panel 3-9-25, **THEN DEPRESS** 0-HS-65-18A/3(40A/3)69A/3), SGTS TRAIN A(B)(C) FAN, push-button.
	NRC	**The first SGT Train that the BOP attempts to start will fail to start, but additional attempts to start different SGT Trains will be successful.**
	BOP	[5] **CHECK** SGT TRAIN A(B)(C) INLET DAMPER as follows: [5.1] **IF** SGT FAN A was started, **THEN CHECK** 0-HS-65-3A(0-IL-65-3), SGTS FILTER BANK A INLET DAMPER, indicates OPEN on Panel 1-9-25(3-9-25). [5.2] **IF** SGT FAN B was started, **THEN CHECK OPEN** 0-HS-65-25A(0-IL-65-25), SGTS FILTER BANK B INLET DAMPER, indicates OPEN on Panel 1-9-25(3-9-25). [5.3] **IF** SGT FAN C was started, **THEN CHECK OPEN** 0-HS-65-51A(0-IL-65-51), SGTS TRAIN C INLET DAMPER, indicates OPEN on Panel 3-9-25. [6] **CHECK** SGT TRAIN A(B)(C) RH CONTROL HTR as follows: [6.1] **IF** SGT FAN A(B) was started, **THEN CHECK** ENERGIZED 0-HS-65-12(34), SGTS TRAIN A(B) RH CONTROL HTR, on Panel 1-9-25. [6.2] **IF** SGT FAN C was started, **THEN CHECK** ENERGIZED 0-HS-65-60, SGTS TRAIN C RH CONTROL HTR, on Panel 2-9-25.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 6 Page 3 of 3

Event Description: Standby Gas Train (SGT) Fails to Start

Time	Position	Applicant's Actions or Behavior	
	Driver	**If contacted as Unit 1 or Unit 2 to verify inlet damper position or heater status, report that conditions are normal for SGT Train start.**	
	BOP	[7] Record start time and SGT Filter Bank Differential Pressure. [8] **DISPATCH** Operator to the Standby Gas Treatment building as soon as time allows to check for abnormal conditions (i.e. belt tightness, rubbing or vibration noises). [9] **MONITOR** Standby Gas Treatment Train operation. **REFER TO** Section 6.0	
	Driver	**If contacted as an AUO to monitor SGT Trains, acknowledge any information or direction given.**	
	BOP	Method 2: Contacts Unit 1 or Unit 2 to start a SGT Train.	
	Driver	**If contacted as Unit 1 or Unit 2 to start a SGT Train, acknowledge any information or direction given. Wait 2 minutes, insert Event 6 to start 'B' SGT, and then contact the crew and inform them that 'C' SGT would not start from Unit 1(2). Inform the crew that 'B' SGT is running.**	
	NUSO	References Technical Specification 3.6.4.3, Standby Gas Treatment (SGT) System LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE Applicability: MODES 1, 2, and 3. <u>Condition:</u> A. One SGT subsystem inoperable.	
		REQUIRED ACTION: A.1 Restore SGT subsystem to OPERABLE status.	**COMPLETION TIME:** A.1 – 7 days
	NRC	**End of Event 6. Request that the Driver insert Event 7, Dual Recirc Pump Trip.**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 7 Page 1 of 2

Event Description: Dual Recirc Pump Trip

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. 3A Recirc Pump will trip, and 90 seconds later 3B Recirc Pump will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to trip both Recirc Pumps.**
	OATC	Acknowledges and reports the following alarms: • RECIRC DRIVE 3A PROCESS ALARM, 3-9-4A, Window 28 • RECIRC DRIVE 3A DRIVE ALARM, 3-9-4A, Window 32 Reports to the NUSO that 3A Recirc Pump has tripped.
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedures and 3-AOI-68-1, Recirc Pump Trip/Core Flow Decrease.
	NRC	**Because the 3B Recirc Pump trips 90 seconds after 3A, the crew may not have time to perform many ARP Actions.**
	OATC	3-ARP-9-4A, Alarm Response Procedure RECIRC DRIVE 3A PROCESS ALARM, Window 28 Operator Action: A. N/A B. **REFER TO** ICS Screen "VFDAAL" and determine cause of alarm. C. N/A D. N/A E. N/A F. N/A G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	OATC	3-ARP-9-4A, Alarm Response Procedure RECIRC DRIVE 3A DRIVE ALARM, Window 32 Operator Action: A. **REFER TO** ICS Group Display "GD @VFDADA".

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 7 Page 2 of 2

Event Description: Dual Recirc Pump Trip

Time	Position	Applicant's Actions or Behavior
	OATC	B. **DETERMINE** cause of alarm. C. **IF** the VFD has shutdown, **THEN NOTIFY** engineering to determine if a random reboot has occurred by reviewing the fault logs.
	Driver	**If contacted as Engineering, acknowledge any information or direction given.**
	OATC	D. N/A E. N/A F. N/A G. N/A H. **NOTIFY** System Engineering for all other alarms or any problems encountered.
	OATC	When 3B Recirc Pump Trips, informs the NUSO that both Recirc Pumps have tripped.
	OATC	3-AOI-68-1, Recirc Pump Trip/Core Flow Decrease 4.1 Immediate Actions [1] **IF** both Recirc Pumps are tripped in MODES 1 or 2, **THEN**, SCRAM the Reactor.
	NUSO	Directs the OATC to insert a manual Reactor SCRAM.
	NRC	**End of Event 7. Proceed to Event 8, Hydraulic Anticipated Transient Without SCRAM (ATWS).**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 1 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Symptoms/Cues: Event is initiated automatically on Simulator setup. When a manual Reactor SCRAM is inserted for Event 7, all Control Rods will not go fully into the Core

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 8 is entered on Simulator setup. No action is required by the Driver to insert Event 8.**
	NRC	**EXAMINER NOTE: Event 9, Inadvertent Main Steam Isolation Valves (MSIVs) Isolation, will automatically be inserted 2 minutes after the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the Driver to insert Event 9. See page 37 of 43 for Event 9 actions.**
	NRC	**Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**
	OATC	3-AOI-100-1, Reactor SCRAM 4.1 Immediate Actions [1] **DEPRESS** 3-HS-99-5A/S3A, REACTOR SCRAM A and 3-HS-99-5A/S3B, REACTOR SCRAM B, on Panel 3-9-5. [2] **PLACE** 3-HS-99-5A/S1, REACTOR MODE SWITCH, in SHUTDOWN. [3] **IF** all Control Rods can NOT be verified fully inserted, **THEN INITIATE** ARI. [4] N/A **NOTES** 1) Perform steps 4.1[5.3] and4.1[5.4] in parallel. 2) Step 4.1[5.9] may be performed before step 4.1[5.8] if Reactor Water Level is slowly lowering. [5] **IF** Reactor Power is ABOVE 5% or unknown, **THEN PERFORM** the following: [5.1] **REPORT** the following to the UNIT SRO: • REACTOR SCRAM, MODE SWITCH in SHUTDOWN • Control Rods out • Reactor Power • Continuing with ATWS Actions

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 2 of 14

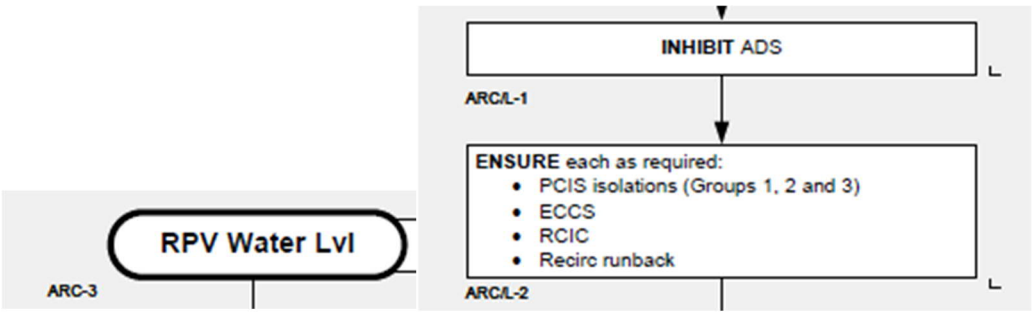
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC / BOP	[5.2] **ENSURE** shutdown both Recirc Pumps [5.3] **STOP** and **PREVENT** injection from CONDENSATE and FEEDWATER per APP 4. • Maintain Reactor Water Level -180 inches to -50 inches [5.4] **STOP** and **PREVENT** injection from HPCI, RHR, and Core Spray per EOI Appendix 4.
	CREW	**Critical Task: When Reactor Power is greater than 5% or unknown during an ATWS, STOP and PREVENT all injection into the Reactor except for RCIC, CRD, and SLC within 130 seconds of the Recirc Pumps reaching zero speed to prevent possible fuel damage.** **Critical Task Failure Criteria: The operating crew fails to STOP and PREVENT injection and take action to lower Reactor Water Level for Reactor Power control during an ATWS within 130 seconds after Recirculation Flow is stopped.**
	OATC / BOP	[5.5] **INITIATE** SLC and **ENSURE** injection. [5.6] **INHIBIT** ADS LOGIC.
	CREW	**Critical Task: With a Reactor SCRAM required and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.** **Critical Task Failure Criteria: ADS automatic initiation with Control Rods out.**
	OATC	[5.7] **BYPASS** Group 1 RPV Low-Low-Low Level isolation interlocks per EOI Appendix 8A. [5.8] **AFTER** Reactor Water Level reaches -50 inches, **THEN REPORT** "ATWS Actions Complete." • Reactor Water Level and trend • Reactor Power level • Reactor Pressure and trend • MSIV position • SLC injection status

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 3 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
		[5.9] **IF** Reactor Water Level is slowly lowering, **THEN** (Otherwise N/A) **REPORT** ATWS action complete. • Reactor Water Level and trend • Reactor Power level • Reactor Pressure and trend • MSIV position • SLC injection status
	OATC	[5.10] **IF** Any Main steam line is OPEN, **THEN CALL** for EOI Appendix 8E, Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation interlocks.
	NUSO	Enters 3-EOI-1A, ATWS RPV Control.
	Driver	**Insert the Events noted below as called for when contacted as the Work Control / Outside NUSO to perform the EOI Appendices.** • **3-EOI-Appendix-1F – Event 20** • **3-EOI-Appendix-2 – Event 21** • **3-EOI-Appendix-8E – Event 23** • **3-EOI-Appendix-8A – Event 22**
	NRC	**EXAMINER NOTE: Normally 3-EOI-Appendix-8A is performed in the Control Room at Panel 3-9-4 using keylock switches, but the crew may call for field performance.** **EOI Appendices are covered starting on page 33 of 43.**
	NRC	**EXAMINER NOTE: Due to the low Reactor Power Level for this ATWS and SLC Injection, the crew may not be able to lower Reactor Water Level to (-) 50 inches.**
	NUSO	

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 4 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	ARC/L-3

IF	THEN
Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7	**NO ACTION REQUIRED**

ARC/L-4

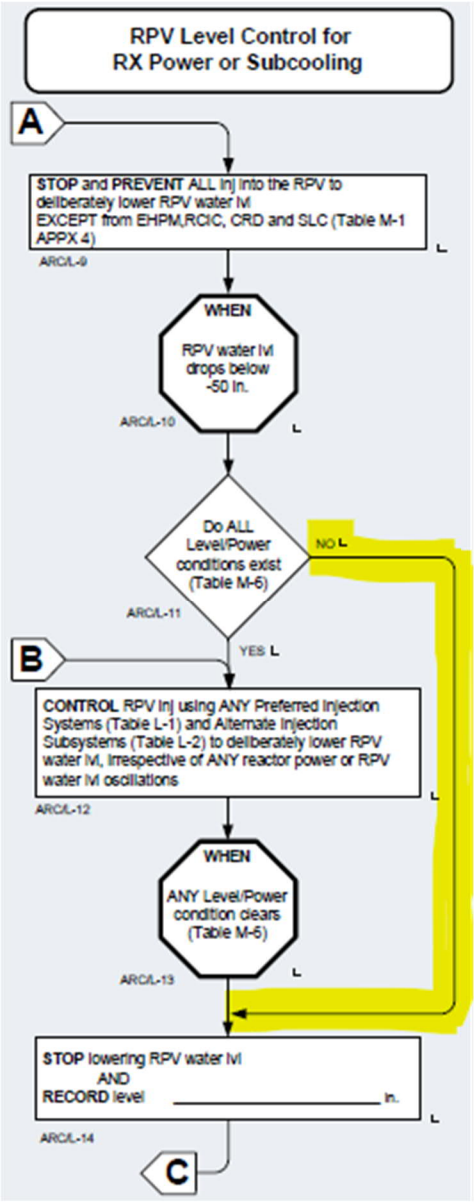
IF	THEN
Reactor Power is above 5% or unknown AND RPV Water Level is above (-) 50 in.	A
ALL Level/Power conditions exist (Table M-6)	**NO ACTION REQUIRED**
Reactor Power is above 10% or unknown AND RPV Water Level is below (-) 162 in. AND MSRV open/cycling OR Drywell Pressure above 2.4 psig	**NO ACTION REQUIRED**

Table M-6
Level/Power Conditions

- Suppression Pool Temperature is above 110°F ☐
- Reactor Power above 5% OR unknown ☐
- RPV Level above -162 in. ☐
- MSRV open/cycling OR DW pressure above 2.4 psig ☐

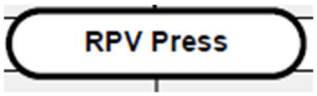
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 5 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	
	NUSO	Directs the crew to maintain Reactor Water Level in any band between (-) 50 and (-180) inches to keep Reactor Power low.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 6 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	 ARC/P-1 **IF**: **EMERGENCY DEPRESSURIZATION IS REQUIRED** **THEN**: **NO ACTION REQUIRED** ARC/P-2 **IF** ANY MSRV is cycling **THEN MANUALLY OPEN** MSRVs UNTIL RPV Pressure drops to the Pressure at which all Main Turbine Bypass Valves are fully open (PPX 11A) ARC/P-3 **IF**: **BORON INJECTION IS REQUIRED** AND The Main Condenser is available AND There has been no indication of a steam line break **THEN**: **NO ACTION REQUIRED (MSIV ISOLATION)** **IF**: RPV Depressurization will not result in loss of injection required for Adequate Core Cooling AND: • Suppression Pool Temperature and Level CANNOT be maintained in the safe area of Curve 3 at the existing RPV Pressure OR • Suppression Pool Level CANNOT be maintained in the safe area of Curve 4 **THEN**: **NO ACTION REQUIRED**

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 7 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	 ARC/P-4 **RESTORE** and **MAINTAIN** RPV Pressure between 1073 psig and 350 psig using the Main Turbine Bypass Valves (APPX 8B) and maintain cooldown rate below 100 ºF/hr ➢ Use Alternate RPV Pressure Control Systems (Table P-1), if necessary ➢ Crosstie CAD or MSRV carts to Drywell Control Air (APPX 8G, 20H), if necessary

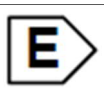
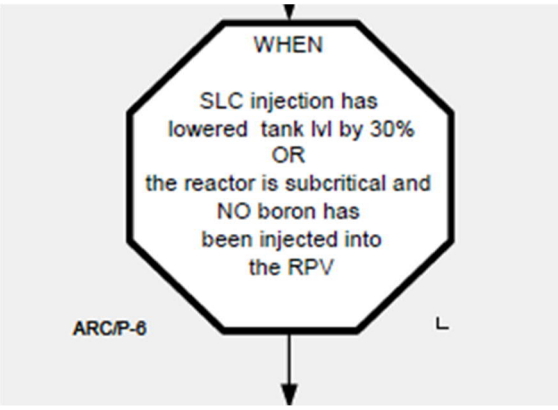
IF	THEN
Drywell Control Air is or becomes unavailable	**NO ACTION REQUIRED**

Table P-1
Alternate RPV Pressure Control Systems

Source	APPX
Main turb bypass vlvs	8B
MSRVs only if suppr lvl is above 5.5 ft IF "MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW" annunciator (XA-55-3D-18) is in alarm THEN **MINIMIZE** MSRV cycling by using sustained opening for depressurization	11A
HPCI with CST suction if available (2) (5) (7)	11C
RCIC with CST suction if available (2) (3) (5)	11B
RFPTs on min flow	11F
Main Steam system drains	11D
Steam seals	11G
SJAEs	11G
Off Gas preheater	11G
HPCI and RCIC drains	11J
RWCU if NO boron has been injected into the RPV	11E
MSRV Remote Operation only if suppr lvl is above 5.5 ft	EDMG-24 APPX. D FSI-3E

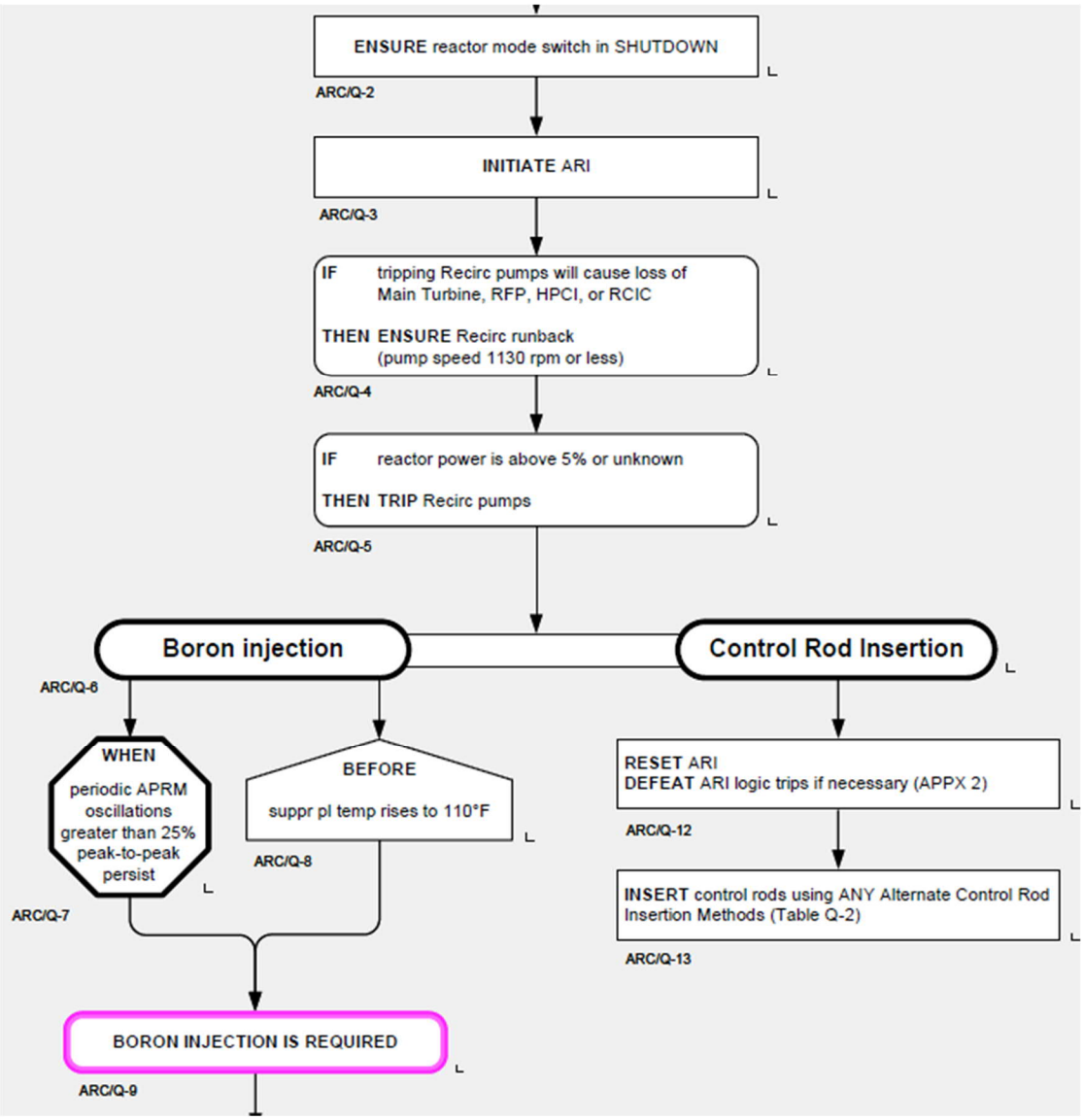
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 8 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>While MSIVs are OPEN, directs the crew to maintain Reactor Pressure using the Main Turbine (until it is required to be tripped at < 50MWE) and then using 3-EOI-Appendix-8B, Reopening MSIVs/Bypass Valve Operation.</td></tr>
<tr><td></td><td>NRC</td><td>When the MSIVs isolate, this method of Reactor Pressure Control will not be available. See Event 9 for Reactor Pressure Control actions following MSIV isolation.</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/P-5
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>The Reactor is NOT subcritical</td><td>  </td></tr>
<tr><td>It is anticipated that RPV Depressurization will result in a loss of injection for Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
</table>

</td></tr>
<tr><td></td><td>NUSO</td><td>ARC/Q-1
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>The Reactor is subcritical
AND
NO boron has been injected</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 9 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 10 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	Table Q-2 Alternate Control Rod Insertion Methods CONDITIONS / METHODS / APPX Scram valves failed to open: **DEENERGIZE** scram solenoids — 1A; **VENT** scram air header — 1B Scram valves opened but SDV is full: 1. **RESET** scram **DEFEAT** RPS logic if necessary 2. **DRAIN** SDV 3. **RECHARGE** accumulators 4. **INITIATE** scram — 1F Manual control rod insertion methods: **DRIVE** control rods **BYPASS** RWM and **RAISE** CRD drive water differential pressure if necessary — 1D; **RAISE** CRD cooling water header dp — 1G; **SCRAM** individual control rods — 1C; **VENT** control rod over piston volumes — 1E
	NUSO	1. **INITIATE** SLC 2. **INHIBIT** ADS 3. RECORD SLC tank level . IF / THEN Boron CANNOT be injected using SLC — **INJECT** boron into RPV using CRD (APPX 3B) SLC tank water lvl drops to 0% — **TRIP** the SLC pumps ARC/Q-10 **ENSURE** RWCU system isolation ARC/Q-11

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 11 of 14

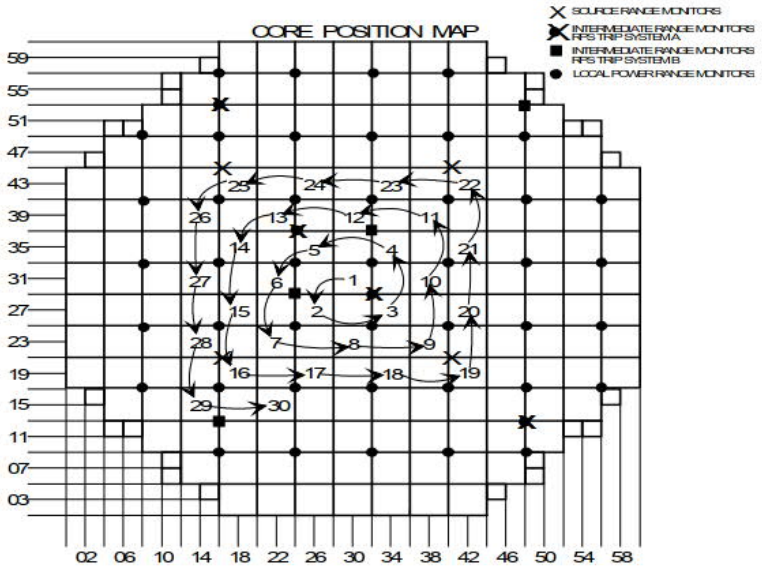
Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	BOP	3-EOI-APPENDIX-8A, Bypassing Group 1 Low Low Low Level Isolation Interlocks [1] **BYPASS** Group 1 RPV Low-Low-Low Level Isolation Interlocks as follows (Unit 3 Control Room, Panel 9-4): [1.1] **PLACE** keylock switch 3-HS-064-0056A, GRP 1 RPV LOW LEVEL BYPASS (SYS A1), in BYPASS. [1.2] **PLACE** keylock switch 3-HS-064-0056B, GRP 1 RPV LOW LEVEL BYPASS (SYS B1), in BYPASS. [1.3] **PLACE** keylock switch 3-HS-064-0056C, GRP 1 RPV LOW LEVEL BYPASS (SYS A2), in BYPASS. [1.4] **PLACE** keylock switch 3-HS-064-0056D, GRP 1 RPV LOW LEVEL BYPASS (SYS B2), in BYPASS. [1.5] **ENSURE** closed the following valves (Unit 3 Control Room, Panel 9-3): • 3-FCV-43-13, RX RECIRC SAMPLE INBOARD ISOLATION VALVE • 3-FCV-43-14, RX RECIRC SAMPLE OUTBOARD ISOLATION VALVE [2] N/A
	OATC	Drives Control Rods using 3-EOI-Appendix-1D, Insert Control Rods Using Reactor Manual Control System [1] **VERIFY** at least one CRD Pump in service. **NOTES** 1) Closing 3-85-586, CHARGING WATER ISOL valve may reduce the effectiveness of EOI Appendix 1A or 1B. 2) A ladder may be required to perform the following step. REFER TO Tools and Equipment, Attachment 1.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 12 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	[2] **IF** Reactor SCRAM or ARI CANNOT be reset, **THEN DISPATCH** personnel to close 3-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft). [3] **VERIFY** REACTOR MODE SWITCH in SHUTDOWN. [4] **BYPASS** Rod Worth Minimizer. [5] **REFER TO** Attachment 2 and INSERT Control Rods in the area of highest power as follows: [5.1] **SELECT** Control Rod. [5.2] **PLACE** CRD NOTCH OVERRIDE switch in EMERG ROD IN position UNTIL Control Rod is NOT moving inward. [5.3] **REPEAT** Steps 1.0[5.1] and 1.0[5.2] for each Control Rod to be inserted. [6] **WHEN** NO further Control Rod movement is possible or desired, **THEN DISPATCH** personnel to verify open 3-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft).
	OATC	**Attachment 2** **(Page 1 of 1)** **Core Position Map**  (EXAMPLE ONLY)

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 13 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	3-EOI-APPENDIX-1F, Manual SCRAM [1] **VERIFY** Reactor Scram and ARI reset. [1.1] **IF** ARI CANNOT be reset, **THEN EXECUTE** EOI Appendix 2, Defeating ARI Logic Trips concurrently with Step 1.0[1.2] of this procedure. [1.2] N/A for the Control Room, this step is performed locally in the Auxiliary Instrument Room [2] **WHEN** RPS Logic has been defeated, **THEN RESET** Reactor Scram. [3] **VERIFY** OPEN Scram Discharge Volume Vent and Drain Valves. [4] **DRAIN** SDV UNTIL the following annunciators clear: • WEST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 3-9-4, 3-XA-55-4A, Window 1) • EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 3-9-4, 3-XA-55-4A, Window 29)
	NRC	**After the SCRAM has been reset, the accumulators will drain in approximately 7 minutes and the alarms at Panel 3-9-4, Windows 1 and 29, will clear. The OATC may then attempt a Reactor SCRAM.**
	OATC	[5] **DISPATCH** personnel to **VERIFY OPEN** 3-SHV-085-0586, CHARGING WATER ISOLATION. **NOTES** 1) If EOI Appendix 2 has been executed, ARI initiation or reset will NOT be possible or necessary in Step 1.0[6]. 2) If Reactor Pressure is greater than 600 psig, NUSO may direct performance of step 1.0[6] prior to accumulators being fully recharged. [6] **WHEN** CRD Accumulators are recharged, **THEN INITIATE** manual Reactor SCRAM and ARI.
	NRC	**Control Rods will insert the first time the OATC attempts a Reactor SCRAM after the ATWS.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 8 Page 14 of 14

Event Description: Hydraulic Anticipated Transient Without SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	[7] **CONTINUE** to perform Steps 1.0[1] through 1.0[6] UNTIL ANY of the following exists: • ALL Control Rods are fully inserted, OR • NO inward movement of Control Rods is observed, OR • NUSO directs otherwise
	NRC	**End of Event 8. When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 1 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Symptoms/Cues: Event is initiated when the Reactor MODE SWITCH is placed in SHUTDOWN. The Main Steam Isolation Valves will close two (2) minutes after the Event is initiated

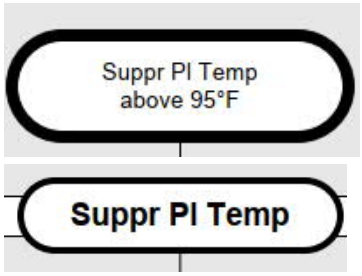
Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9 is automatically entered on Simulator setup. No action is required by the Driver to insert Event 9.**
	BOP	Informs the NUSO that MSIVs have closed.
	NUSO	Directs the BOP to maintain Reactor Pressure using 3-EOI-Appendix-11A, Alternate Pressure Control Systems MSRVs.
	BOP	3-EOI-APPENDIX-11A, Alternate RPV Pressure Control Systems MSRVs [1] N/A [2] N/A [3] **OPEN** MSRVs using the following sequence to control RPV pressure as directed by the NUSO:

1	3-PCV-1-179	MN STM LINE A RELIEF VALVE
2	3-PCV-1-180	MN STM LINE D RELIEF VALVE
3	3-PCV-1-4	MN STM LINE A RELIEF VALVE
4	3-PCV-1-31	MN STM LINE C RELIEF VALVE
5	3-PCV-1-23	MN STM LINE B RELIEF VALVE
6	3-PCV-1-42	MN STM LINE D RELIEF VALVE
7	3-PCV-1-30	MN STM LINE C RELIEF VALVE
8	3-PCV-1-19	MN STM LINE B RELIEF VALVE
9	3-PCV-1-5	MN STM LINE A RELIEF VALVE
10	3-PCV-1-41	MN STM LINE D RELIEF VALVE
11	3-PCV-1-22	MN STM LINE B RELIEF VALVE
12	3-PCV-1-18	MN STM LINE B RELIEF VALVE
13	3-PCV-1-34	MN STM LINE C RELIEF VALVE

Form 3.3-2 Required Operator Actions

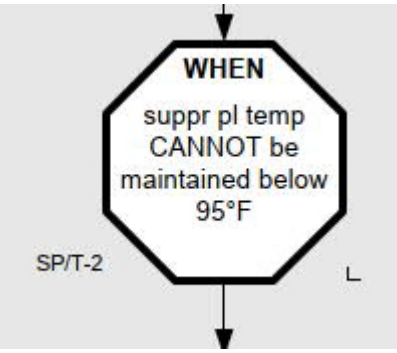
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 2 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	BOP	[4] N/A [5] N/A [6] N/A
	BOP	Acknowledges and reports the following alarm when received: • SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, 3-9-3E, Window 12
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	(If received) 3-ARP-9-3E, Alarm Response Procedure SUPPRESSION POOL AVERAGE TEMPERATURE HIGH, Window 12 Operator Action: A. **IF** alarm is valid, **THEN ENTER** 3-EOI-2, Primary Containment Control.
	NUSO	Enters 3-EOI-2, Primary Containment Control.
	NUSO	 SP/T-1 **MONITOR** and **CONTROL** Suppr Pool Temperature below 95°F using available Suppression Pool Cooling (APPX 17A)

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 3 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	NUSO	 SP/T-3 **OPERATE** all available Suppression Pool Cooling using only RHR Pumps NOT required for continuous injection (APPX 17A)
	NUSO	Directs the BOP to place Suppression Pool Cooling in service in accordance with 3-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling.
	BOP	3-EOI-Appendix-17A, RHR System Operation Suppression Pool Cooling **NOTE** Placing a BYPASS SEL switch in BYPASS in Step 1.0[1] below prevents automatic opening of the affected RHR loop's Outboard Injection Valve. This makes LPCI Mode of that RHR Loop inoperable. [1] **IF** the unit is in Mode 1,2 or 3 and it is desired to maintain LPCI Operability, **THEN N/A** this step. [1.1] **IF** Adequate Core Cooling is assured **OR** directed to cool the Suppression Pool irrespective of Adequate Core Cooling, **THEN BYPASS** LPCI Injection Valve open interlock AS NECESSARY: • **PLACE** 3-HS-74-155A, LPCI SYS I OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS • **PLACE** 3-HS-74-155B, LPCI SYS II OUTBOARD INJECTION VALVE BYPASS SELECT in BYPASS

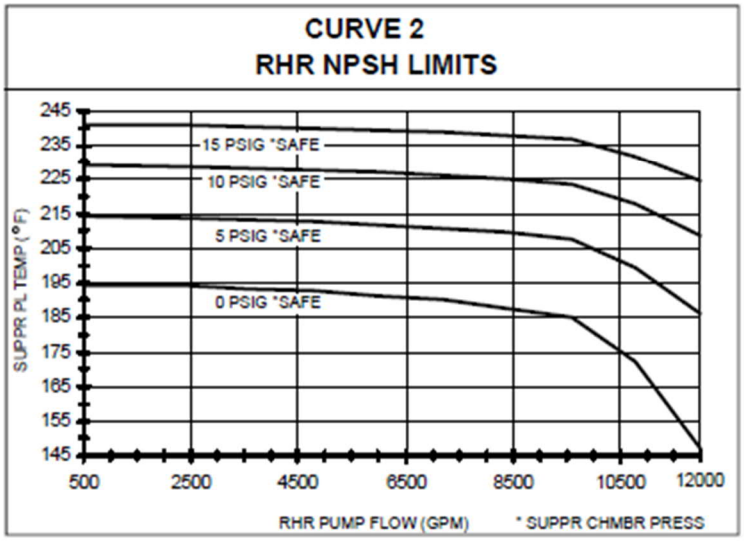
Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 4 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	BOP	[2] **PLACE** RHR SYSTEM I(II) in Suppression Pool Cooling as follows: [2.1] **ENSURE** at least one RHRSW Pump supplying each EECW header. [2.2] **ENSURE** RHRSW Pump supplying desired RHR Heat Exchanger(s). [2.3] **THROTTLE** the following in service RHRSW Outlet Valves to obtain 1700 to 4500 gpm RHRSW Flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VALVE • 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VALVE • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VALVE • 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VALVE [2.4] **IF** Directed by SRO, **THEN PLACE** 3-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. [2.5] N/A [2.6] **IF** 3-FCV-74-53(67), RHR SYS I(II) LPCI INBD INJECTION VALVE, is OPEN, **THEN ENSURE CLOSED** 3-FCV-74-52(66), RHR SYS I(II) LPCI OUTBOARD INJECTION VALVE. [2.7] **OPEN** 3-FCV-74-57(71), RHR SYS I(II) SUPPRESSION CHAMBER/POOL ISOLATION VALVE. [2.8] **ENSURE** desired RHR Pump(s) for Suppression Pool Cooling are operating. [2.9] **THROTTLE OPEN** 3-FCV-74-59(73), RHR SYS I(II) SUPPRESSION POOL COOLING/TEST VALVE, to maintain EITHER of the following as indicated on 3-FI-74-50(64), RHR SYS I(II) FLOW: Between 7,000 and 10,000 gpm for one pump operation **OR** At or below 13,000 gpm for two pump operation [2.10] **ENSURE CLOSED** 3-FCV-74-7(30), RHR SYS I(II) MINIMUM FLOW VALVE. [2.11] **MONITOR** RHR Pump NPSH using Attachment 1.

Op Test No.: 23-04 Scenario No. NRC-5 Event No.: 9 Page 5 of 5

Event Description: Inadvertent Main Steam Isolation Valves (MSIVs) Isolation

Time	Position	Applicant's Actions or Behavior
	BOP	**Attachment 1** **(Page 1 of 1)** **NPSH Monitoring** Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used. 
	BOP	[2.12] **NOTIFY** Chemistry that RHRSW is aligned to in service RHR Heat Exchangers. [2.13] **IF** Additional Suppression Pool Cooling flow is necessary, **THEN PLACE** additional RHR and RHRSW pumps in service using Steps 1.0[2.2] through 1.0[2.12].
	Driver	**When contacted as Chemistry, acknowledge any information or direction given.**
	NRC	**End of Event 9. When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel ((-) 162 inches) using high pressure systems, end of Scenario.**

Scenario Setup
UNIT 3

IC	28
Exam IC	267

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-65	56	ARP-9-3D	36	TS-3.5.1	334
OI-66	87	ARP-9-3F	38	EOI-1A	5
OI-67	126	ARP-9-23C	34	EOI-2	15
OI-68	106	ARP-9-23D	32	Appendix-1D	3
AOI-57-5A	47	GOI-100-1A	142	Appendix-1F	3
AOI-68-1	18	GOI-100-12	51	Appendix-8A	3
AOI-85-3	14	GOI-100-12A	77	Appendix-8B	6
AOI-100-1	80			Appendix-11A	2
				Appendix-17A	14

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	ILT 2304 NRC Scenario 5 UNIT 3.sch
Event Files(s):	ILT 2304 NRC Scenario 5 UNIT 3.evt

UNIT 3 SHIFT TURNOVER MEETING

Today			
MODE 1	**DAYS ON LINE** **230**	**Drywell Leakage (GPM)** **1.89**	**Protected Equipment** 3A CRD Pump, 3C 4KV Unit Bd (due to 3B CRD Pump)
Rx Power **26%**	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	**Floor Drain (GPM)** **0.31**	
MWe **281**		**Equipment Drain (GPM)** **1.58**	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

3A Recirc Pump trip. All repairs and inspections are complete for restart.
3B CRD Pump is tagged to support breaker inspection.
IRM 'G' is bypassed to support maintenance from amplifier failure.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Start 3A Recirc Pump in accordance with 3-OI-68, Section 5.3, beginning at Step [12]. SR 3.4.9.3&4 is complete.
The SST will maintain SR 3.4.9.3&4 current.
Match speed of 3B Recirc Pump, then await instruction from the Reactor Engineer.

OPERATOR WORK AROUNDS **OWAs – U0-1/U3-0** **Burdens - U0 – 0/U3-0** **Operator Challenges – U0 – 0/U3 - 0**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-6
Scenario Source: NEW Op-Test Number: 23-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 100% Reactor Power. Loop I of RHR is in Suppression Pool Cooling for a HPCI run on the previous shift. 2A RBCCW Pump is tagged for bearing replacement, the spare RBCCW Pump is aligned to Unit 2.

Turnover: Place Loop I of RHR in standby readiness.

Critical Tasks:
1. When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator.
2. When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the Nuclear Unit Senior Operator (NUSO) determines that Emergency Depressurization is required. The Operator initiates Emergency Depressurization as directed by the NUSO.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Secure Suppression Pool Cooling
2.	SCHEDULE STACK	C-OATC MC-OATC C-NUSO	Stack Dilution Fan Failure
3.	TH09A	I-OATC I-NUSO	Feedwater Level Control System (FWLCS) Input Failure (3-53)
4.	ZLOHS713A_1 ZLOHS713A_2 FCV-71-3	C-BOP MC-BOP TS-NUSO	Reactor Core Isolation Cooling (RCIC) System Isolation, One Valve Fails to Close
5.	TH31A	I-OATC TS-NUSO	Reactor High Pressure Instrument A1 Fails High / Half SCRAM
6.	MCO1A-F	C-BOP C-NUSO	Condenser Leakage / High Conductivity
7.	SCHEDULE TORUS	M-ALL	High Suppression Pool Water Level / Emergency Depressurization
8.	RP07	C-OATC MC-OATC C-NUSO	Electrical Anticipated Without Transient (ATWS)
9.	HS-1-22A	C-BOP C-NUSO	Steam Relief Valve (SRV) Fails to Open

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will place Suppression Pool Cooling in standby readiness in accordance with 2-OI-74, Residual Heat Removal Cooling System.
2. Stack Dilution Fan 2A will fail and the standby Stack Dilution Fan will fail to automatically start. The crew will take action to manually restore Stack Dilution Flow by starting 2B Stack Dilution Fan in accordance with 2-OI-66, Off-Gas System.
3. A Reactor Water Level input into the Feedwater Level Control System (LT-3-53) will fail low, requiring the crew to respond in accordance with Alarm Response Procedures and 2-OI-3, Reactor Feedwater System, to bypass the failed level indicator.
4. Reactor Core Isolation Cooling (RCIC) will receive an inadvertent isolation signal but not all valves will automatically close, requiring manual action to ensure RCIC is isolated. The Nuclear Senior Unit Operator (NUSO) will address Technical Specifications.
5. Reactor High Pressure A1 Channel Fails High/Half SCRAM (2-PIS-3-22AA), inserting a Reactor Half SCRAM. The crew will respond in accordance with Alarm Response Procedures and the NUSO will address Technical Specifications.
6. The Main Condenser will develop a tube leak in a waterbox, causing conductivity levels to rise. The crew will respond in accordance with 2-AOI-2-1, Reactor Coolant High Conductivity, and will isolate Main Condenser Waterboxes until the leak is isolated.
7. The crew will respond to a rising Suppression Pool Water Level in accordance with Emergency Operating Instruction 2-EOI-2, Primary Containment Control. Suppression Pool Water Level will not be able to be maintained in the safe area of Curve 4, SRV Tail Pipe Level Limit, requiring the crew to insert a manual Reactor SCRAM and eventually Emergency Depressurize the Reactor.
8. When the Operator at the Controls (OATC) inserts the Reactor SCRAM an Electrical Anticipated Transient Without SCRAM (ATWS) will occur, requiring the OATC to take manual action to insert the Control Rods.
9. When the crew attempts to Emergency Depressurize the Reactor, one (1) SRVs will fail to open, requiring the crew to open additional non-ADS SRVs for Emergency Depressurization.

The Scenario ends when the crew has Emergency Depressurized the Reactor, and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available low-pressure injection system.

Critical Tasks: 2

1. **When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator.**

   a. **Safety Significance**

   Prevent failure of Primary Containment from over pressurization.

   b. **Cues**

   Procedural Compliance.
   Suppression Pool Level indication.

   c. **Measured by**

   Observation – Both RPS SCRAM switches are depressed.

   d. **Feedback**

   Control Rods insert to their full in position.

   e. **Critical Task Failure Criteria**

   The operating crew fails to proceed without delay and in a controlled manner to insert a manual SCRAM before Suppression Pool Water Level enters the 'Action Required' area of Curve 4, SRV Tail Pipe Level Limit.

2. **When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the Nuclear Unit Senior Operator determines that Emergency Depressurization is required. The Balance of Plant Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.**

   a. **Safety Significance**

   Prevent failure of Primary Containment from over pressurization.

   b. **Cues**

   Procedural Compliance.
   RPV Pressure indication.
   Suppression Pool Level indication.

   c. **Measured by**

   Observation – the Nuclear Unit Senior Operator determines (as indicated by announcement or observable transition to RPV Emergency Depressurization), Emergency Depressurization is required at or before Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.
   AND
   Observation – the Nuclear Unit Senior Operator directs the Balance of Plant Operator to open 6 ADS valves

   d. **Feedback**

   Suppression Pool Water Level trend.
   RPV Pressure trend.
   MSRV status indication.

e. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time that Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 1 Page 1 of 3

Event Description: Secure Suppression Pool Cooling

Symptoms/Cues: Event is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not start Event 1, Secure Suppression Pool Cooling, after assuming the shift request that the driver contact the Unit Supervisor (SRO) as the Shift Manager and direct the crew to place Loop I of Residual Heat Removal (RHR) in standby readiness.**
	Driver	**If requested by the Chief Examiner, contact the SRO as the Shift Manager, and direct the crew to place Loop I of RHR in standby readiness.**
	NUSO	Directs the Balance of Plant Operator (BOP) to remove Suppression Pool Cooling from service in accordance with 2-OI-74, Residual Heat Removal System, Section 8.6, Shutdown of Loop I(II) Suppression Pool Cooling.
	BOP	2-OI-74, Residual Heat Removal System Section 8.6, Shutdown of Loop I(II) Suppression Pool Cooling **NOTE** 1) All operations are performed at Panel 2-9-3 unless otherwise noted. 2) RHR Flow should be monitored while in operation with multiple flow paths (e.g., LPCI and Suppression Pool Cooling together, etc.). During any evolution, total system flow as indicated on 2-FI-74-50(64), RHR SYSTEM I(II) FLOW, should remain between 7,000 to 10,000 GPM for 1 pump operation or between 10,000 and 20,000 GPM for 2-pump operation. [1] **ENSURE** Suppression Pool Cooling is in operation. **REFER TO** Section 8.5. [2] **REVIEW** the Precautions and Limitations in Section 3.0. [3] **NOTIFY** Radiation Protection of Suppression Pool Cooling Loop being removed from service. **RECORD** name and time of Radiation Protection representative notified in NOMS narrative log.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 1 Page 2 of 3

Event Description: Secure Suppression Pool Cooling

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as Radiation Protection, acknowledge that Suppression Pool Cooling is being removed from service and provide a name for the NOMS narrative log.** **If contacted as the Reactor Building Assistant Unit Operator (AUO) to go to the RHR Quad for securing RHR pumps, acknowledge any information or direction given.**
	BOP	**CAUTIONS** 1) To prevent draining an RHR Loop, at least one of the RHR System Test Valves must be closed before stopping RHR Pumps in the associated loop. 2) To prevent excessive vibration, RHR Pumps should not be allowed to operate for more than 3 minutes at minimum flow. 3) When closing throttle valves 2-FCV-74-59, RHR SYS I SUPPRESSION POOL COOLING/TEST VALVE from the Control Room, the handswitch should be held in the close position for approximately 6 seconds after the red light extinguishes. Failure to completely close these valves could provide a leak path to the suppression pool from the RHR discharge piping. [4] N/A [5] **CLOSE** 2-FCV-74-59(73), RHR SYSTEM I(II) SUPPRESSION POOL COOLING/TEST VALVE. [6] **WHEN** 2-FCV-74-59(73), RHR SYSTEM I(II) SUPPRESSION POOL COOLING/TEST VALVE is CLOSED, **THEN** **STOP** RHR PUMPS 2A(2B) or 2C(2D) using 2-HS-74-5A(28A) and/or 16A(39A). [7] **CLOSE** 2-FCV-74-57(71), RHR SYSTEM I(II) SUPPRESSION CHAMBER/POOL ISOLATION VALVE. [8] **CLOSE** 2-FCV-23-34(40), 2A(2C) RHR HEAT EXCHANGER RHRSW OUTLET VALVE(s) . [9] **IF** RHRSW for RHR Heat Exchanger(s) A and C is not required to support other unit operations, **THEN STOP** RHRSW Pump(s) for the Heat Exchanger(s) removed from service.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 1 Page 3 of 3

Event Description: Secure Suppression Pool Cooling

Time	Position	Applicant's Actions or Behavior
	BOP	[10] **CHECK** RHR System Discharge Header Pressure TRM 3.5.4 limit > 48 psig (> 35 psig), as indicated on 2-PI-74-51(65) RHR SYS I(II) DISCHARGE PRESSURE. [11] N/A
	Driver	**If contacted as Unit 1 or 3 that RHRSW Pumps are being secured, acknowledge any information or direction given.**
	NRC	**End of Event 1.** **EXAMINER NOTE: THE BALANCE OF PLANT OPERATOR (BOP) MUST BE REMOVED FROM THE CONTROL ROOM TO ALLOW THE OPERATOR AT THE CONTROLS (OATC) TO RESPOND TO THE NEXT EVENT.** **Request that the Driver insert Event 2, Stack Dilution fan failure.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 2 Page 1 of 2

Event Description: Stack Dilution Fan Failure

Symptoms/Cues: Event is initiated when requested by the Chief Examiner. 2A Stack Dilution Fan will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**CONTACT THE BOP AS REQUESTED BY THE CHIEF EXAMINER WITH A REASON TO LEAVE THE CONTROL ROOM.** **VERIFY THAT THE BOP HAS BEEN REMOVED FROM THE CONTROL ROOM PRIOR TO INSERTING EVENT 2.** **When requested by the Chief Examiner and the BOP is not in the Control Room, insert Event 2 to trip 2A Stack Dilution Fan.**
	OATC	Acknowledges and reports the following alarm: • STACK GAS DILUTION AIR FLOW LOW, 2-9-7A, Window 3
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	NRC	**It is acceptable if OATC elects to start 2B Stack Dilution Fan in accordance with 2-OI-66, Section 8.6, instead of using the ARP below.**
	OATC	2-ARP-9-7A, Alarm Response Procedure STACK GAS DILUTION AIR FLOW LOW, Window 3 Operator Action: A. **ENSURE** alternate fan ON and damper open, (red light illuminated) on Panel 2-9-7.
	OATC	Determines that the standby Stack Dilution Fan did not automatically start and manually starts 2B Stack Dilution Fan.
	NRC	**The BOP can be allowed to return to the Control Room at this point.**
	OATC	B. **DISPATCH** personnel to stack to check and report status of the following for both fans: 1. Fan motor. 2. Fan belts. 3. Damper stuck closed. C. **CHECK** breaker 5C on 480V Diesel Aux Bd A and B.
	Driver	**If contacted as an AUO, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 2 Page 2 of 2

Event Description: Stack Dilution Fan Failure

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-66, Off-Gas System Section 8.6, Standby Gas Dilution Fan Operation [1] **REVIEW** all Precautions and Limitations in Section 3.0. [2] N/A [3] **ENSURE** 2-XS-66-29, STACK DILUTION FAN SEL, in OFF. [4] **PLACE** applicable standby STACK DILUTION FAN control switch in START. • 2-HS-66-31A, STACK DILUTION FAN 2B [5] **IF** desired to operate only one stack dilution fan, **THEN PERFORM** the following for STACK DILUTION FAN 2A. (Otherwise N/A) [5.1] **PLACE** operating (or possibly tripped) 2-HS-66-29A, STACK DILUTION FAN 2A control switch, in STOP [5.2] **PLACE** the 2-XS-66-29, STACK DILUTION FAN SELECT control switch, in FAN A [representative of the non operating fan]. [5.3] **ENSURE** 2-HS-66-29A, STACK DILUTION FAN 2A control switch, in AUTO [6] N/A
	NRC	**End of Event 2. Request that the driver insert Event 3, Feedwater Level Control System (FWLCS) Input Failure (3-53).**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 3

Event Description: Feedwater Level Control System (FWLCS) Input Failure (3-53)

Symptoms/Cues: Event initiated when requested by the Chief Examiner. Feedwater Level Control Input LT-3-53, REACTOR WATER LEVEL 'A' will fail downscale

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to fail Feedwater Level Control Input LT-3-53, REACTOR WATER LEVEL 'A', downscale.**
	OATC	Acknowledges and reports the following alarm: • RFWCS INPUT FAILURE, 2-9-6C, Window 14
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure.
	OATC	2-ARP-9-6C, Alarm Response Procedure RFWCS INPUT FAILURE, Window 14 Operator Action: A. **CHECK** RFWCS continues to maintain Reactor Water Level. B. **IDENTIFY** bad/invalid signal by checking Control Room instrumentation and/or ICS. **REFER TO** ATTACHMENT 1 on next page for list of RFWCS instrumentation. **REFER TO** ICS RX FW LVL CONTROL SYS display (FWLCS). C. **REQUEST** assistance from Site Engineering. D. **BYPASS** the bad/invalid signal with Unit Supervisor approval. (REFER TO 2-OI-3, Reactor Feedwater System).
	Driver	**If contacted as Site Engineering, acknowledge any information or direction given.**

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 3 Page 2 of 3

Event Description: Feedwater Level Control System (FWLCS) Input Failure (3-53)

Time	Position	Applicant's Actions or Behavior
	OATC	**Attachment 1** RFWCS CONTROL ROOM INSTRUMENTATION * Denotes instrumentation with amber light indicating when the associated signal has been bypassed either manually or automatically by RFWCS. **REACTOR WATER LEVEL** *Reactor Water Level A - 2-LI-3-53 - Panel 2-9-5 *Reactor Water Level B - 2-LI-3-60 - Panel 2-9-5 *Reactor Water Level C - 2-LI-3-206 - Panel 2-9-5 *Reactor Water Level D - 2-LI-3-253 - Panel 2-9-5 Average Reactor Water Level - 2-XR-3-53 - Panel 2-9-5
	NRC	**The Nuclear Unit Senior Operator (NUSO) may reference Technical Specifications, but there are no LCOs associated with this event.**
	NUSO	Directs the Operator at the Controls (OATC) to bypass 2-LI-3-53, REACTOR WATER LEVEL 'A', in accordance with 2-OI-3, Reactor Feedwater System.
	OATC	2-OI-3, Reactor Feedwater System Section 8.2, Bypassing RFWCS Narrow Range Level Instruments **NOTES** 1) RFW Control System will allow up to three Narrow Range Reactor Level instruments to be bypassed at a time. Should the fourth level instrument fail with the other three instruments bypassed, 2-LIC-46-5, REACTOR WATER LEVEL CONTROL PDS, will trip to MANUAL mode. 2) Attachment 8 can be referred to for general information on RFWCS instrumentation. [1] **OBTAIN** approval from NUSO. [2] **BYPASS** desired Narrow Range Reactor Level instrument(s) on Panel 2-9-5 as follows: • To bypass 2-LI-3-53, LEVEL A, **DEPRESS** HS-3-53 BYPASS pushbutton.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 3 Page 3 of 3

Event Description: Feedwater Level Control System (FWLCS) Input Failure (3-53)

Time	Position	Applicant's Actions or Behavior
	OATC	[3] **CHECK** amber light illuminated on level instrument bypassed. [4] **IF** level instrument(s) was bypassed in Step 8.2[1] due to signal failure, **THEN** (Otherwise N/A): • **CHECK** RFWCS INPUT FAILURE, 2-XA-55-6C Window 14, will reset (N/A if annunciation was in alarm prior to signal failure). [5] **REQUEST** assistance from Engineering and IMs. [6] **ENSURE** RFW Control System continues to maintain Reactor Water Level.
	Driver	**If contacted as the Work Control / Outside NUSO, Engineering, Instrument Mechanics, or AUO, acknowledge any information or direction given.**
	NRC	**End of Event 3. Request that the Driver insert Event 4, Reactor Core Isolation Cooling (RCIC) System Isolation, One Valve Fails to Close.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 1 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. RCIC will receive an isolation signal, but 2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE, will not automatically close

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a RCIC System Isolation.**
	NRC	**2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE, will not automatically close when the isolation signal is received. The valve will close when an operator gives it a close signal manually.**
	BOP	Acknowledges and reports the following alarms to the NUSO: • RCIC TURBINE TRIPPED, 2-9-3B, Window 14 • RCIC TURBINE EXHAUST DISC RUPTURED, 2-9-3B, Window 35
	NUSO	Directs the BOP to respond in accordance with the appropriate ARPs.
	BOP	2-ARP-9-3B, Alarm Response Procedure RCIC TURBINE TRIPPED, Window 14 Operator Action: A. **DETERMINE** cause. B. **IF** valid alarm, **THEN REFER TO** 2-OI-71, Reactor Core Isolation Cooling System, Section 8.0. C. **REFER TO** Tech Spec 3.5.3.
	BOP	2-ARP-9-3B, Alarm Response Procedure RCIC TURBINE EXHAUST DISC RUPTURED, Window 35 Operator Action: A. **REFER TO** 2-AOI-64-2C, Group 5 Reactor Core Isolation Cooling Isolation

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	BOP	2-AOI-64-2C, Group 5 Reactor Core Isolation Cooling Isolation 3.0 AUTOMATIC ACTIONS • RCIC Turbine Trips • 2-FCV-71-2, RCIC STEAM LINE INBOARD ISOLATION VALVE CLOSES • 2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE CLOSES • 2-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE CLOSES • 2-FCV-71-9, RCIC TURBINE TRIP/THROTTLE VALVE CLOSES 4.1 Immediate Actions [1] **ENSURE** automatic actions occur.
	BOP	Determines that 2-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE did not automatically close and manually closes 2-FCV-71-3. Informs the NUSO concerning the status of 2-FCV-71-3.
	BOP	4.2 Subsequent Actions [1] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). [2] **DISPATCH** an operator to the RCIC Turbine room to investigate.
	Driver	**If contacted as an AUO, Mechanical Maintenance, or the Work Control / Outside NUSO acknowledge any information or direction given.**
	BOP	[3] **CHECK** the following monitors for a rise in activity or area temperature: [3.1] 2-RR-90-1, AREA RADIATION, Bank 2 Point 6 & 9 (2-RI-90-26 & 29), Panel 2-9-2. [3.2] 2-MON-90-50, AIR PARTICULATE MONITOR RADIATION (2-RM-90-57), Panel 2-9-2. [3.3] 2-RM-90-250, Reactor and Turbine Building Exhaust Radiation on CHEMISTRY CAM MONITOR CONTROLLER, 0-MON-90-361A, Panel 1-9-2.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 3 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as Unit 1, there is no apparent rise in 2-MON-90-250 activity.**
	BOP	[3.4] 2-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE (Panel 2-9-22) [4] **ENSURE** the following valves **OPEN** to drain RCIC steam lines: • 2-FCV-1-55 MAIN STEAM LINE INBOARD • 2-FCV-1-56, OUTBOARD DRAIN ISOLATION VALVE [5] **IF** the isolation occurred due to Exhaust Diaphragm high pressure, **THEN DISPATCH** an AUO to check 2-SHV-071-0014, RCIC TURB EXH SHUTOFF VALVE, DRAIN LINE SHUTOFF VALVE LOCKED OPEN. [6] N/A [7] N/A [8] N/A [9] **REFER TO** 2-OI-71, Reactor Core Isolation Cooling System and **PLACE** RCIC in the condition directed by the NUSO.
	Driver	**If contacted as the Work Control/Outside NUSO or AUO to investigate the cause, prepare tags, or address Protected Equipment Requirements acknowledge the direction.** **If contacted as an AUO to check ATUs in the Aux Instrument Room, acknowledge the direction.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 4 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	NUSO	Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs) LCO – Each PCIV, except Reactor Building to Suppression Chamber Vacuum Breakers, shall be OPERABLE Applicability: MODES 1, 2, and 3. NOTES: 1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be un-isolated intermittently under administrative controls. 2. Separate Condition entry is allowed for each penetration flow path. 3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs. 4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3. <u>Condition:</u> ---------- NOTE ---------- Only applicable to penetration flow paths with two PCIVs. ---------- A. One or more penetration flow paths with one PCIV inoperable except due to MSIV Leakage not within limits.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 5 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION
A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.
<u>AND</u>
A.2
----------NOTE----------
Isolation devices in high radiation areas may be verified by use of administrative means.

Verify the affected penetration flow path is isolated.</td><td>COMPLETION TIME
A.1 – 4 hours except for Main Steam Line
<u>AND</u>
8 hours for Main Steam Line

A.2 – Once per 31 days for isolation devices outside Primary Containment
<u>AND</u>
Prior to entering MODE 2 or 3 from MODE 4, if Primary Containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside Primary Containment</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Technical Specification 3.5.3, RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE

Applicability: MODES 1,
MODES 2, and 3 with Reactor Steam Dome Pressure > 150 psig

---------- NOTE ----------

LCO 3.0.4.b is not applicable to RCIC

<u>Condition:</u>
A. RCIC System inoperable.</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 6 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION** A.1 Verify by administrative means High Pressure Coolant Injection is OPERABLE. AND A.2 Restore RCIC System to OPERABLE status.	**COMPLETION TIME** A.1 – Immediately A.2 – 14 days
	NRC	**End of Event 4. Request that the Driver insert Event 5, Reactor High Pressure Instrument A1 Fails High / Half SCRAM**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 1 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. A Reactor half SCRAM will be initiated by a Reactor Pressure Instrument failing high

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to fail Reactor High Pressure Channel A1 high.**
	OATC	Acknowledges and reports the following alarms: • RX VESSEL PRESSURE HIGH HALF SCRAM, 2-9-4A, Window 9 • REACTOR CHANNEL 'A' AUTO SCRAM, 2-9-5B, Window 1 • RPS ANALOG TRIP UNIT TROUBLE, 2-9-5B, Window 23
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response Procedures.
	OATC	2-ARP-9-4A, Alarm Response Procedure RX VESSEL PRESSURE HIGH HALF SCRAM, 2-9-4A, Window 9 Operator Action: A. **CONFIRM** alarm by multiple indications. B. N/A C. **DISPATCH** personnel to the sensors to check for abnormal conditions. D. **IF** alarm is invalid, **THEN PERFORM** the following: 1. With SRO permission, **RESET** Half SCRAM. 2. **REFER TO** 2-OI-99, Reactor Protection System. E. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	BOP	2-ARP-9-5B, Alarm Response Procedure REACTOR CHANNEL 'A' AUTO SCRAM, 2-9-5B, Window 1 Operator Action: A. **CHECK** channel A relays dropped out by checking SCRAM solenoid and backup SCRAM valve lights extinguished.

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 2 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	OATC	B. **IF** any EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). C. N/A D. N/A E. With NUSO permission, **RESET** half-SCRAM. **REFER TO 2**-OI-99, Reactor Protection System. F. **CHECK** first-out printer to determine initiating cause. G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	OATC	2-ARP-9-5B, Alarm Response Procedure RPS ANALOG TRIP UNIT TROUBLE, 2-9-5B, Window 23 Operator Action: A. **DISPATCH** personnel to Aux Instrument Room to investigate cause. B. **IF** a power supply light on ATU Panel 2-9-83, -84, -85, or-86 is extinguished, **THEN DETERMINE** cause and **ATTEMPT** power restoration. C. **IF LATCH** light on Trip Unit is illuminated indicating GROSS FAIL, **THEN DETERMINE** cause and **ATTEMPT** reset. **NOTE** Other parameters will be masked from alarming while this alarm is sealed in. D. **FREQUENTLY MONITOR** other input parameters. E. **IF** necessary, **INITIATE** a Condition Report (CR) for Instrument Group to troubleshoot and repair. F. **IF** alarm is **NOT** valid, **THEN REFER TO** 0-OI-55, Annunciator System. G. With NUSO permission, **RESET** half-SCRAM. **REFER TO 2**-OI-99, Reactor Protection System.
	Driver	**When directed to investigate as the Control Bay Assistant Unit Operator (AUO), Work Control / Outside NUSO, wait 3 minutes and inform the crew that 2-PIS-3-22AA, Reactor Pressure A1 Channel is failed high with a Gross Failure. Investigation is in progress.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 3 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	NRC	**After the NUSO has addressed Technical Specifications, request that the driver contact the Control Room with the cause for the Reactor High Pressure trip.**
	NUSO	References Technical Specification 3.3, Instrumentation 3.3.1.1, Reactor Protection System (RPS) Instrumentation LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE. Applicability: According to Table 3.3.1.1-1 <u>Condition:</u> One or more required channels inoperable.
	NUSO	**REQUIRED ACTION** / **COMPLETION TIME** A.1 Place channel in trip. — A.1 – 12 hours <u>OR</u> A.2 — A.2 – 12 hours ----------NOTE---------- Not applicable to Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------- Place associated trip system in trip
	NUSO	Table 3.3.1.1-1 (page 2 of 3) Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 4 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, inform the crew that Instrument Mechanics in the Auxiliary Instrument Room reported that while performing the initial test setup for 2-SR-3.3.6.1.5(1B/A1), Primary Containment Isolation System Main Line Low Pressure Instrument Channel A1 Calibration, they inadvertently connected to 2-PIS-3-22AA, Reactor Pressure A1 channel, and tripped the ATU.** **If contacted as an Instrument Mechanic, report that no adjustments were made to 2-PIS-3-22AA, Reactor Pressure A1 Channel, that all test equipment has been removed, the instrument is within testing periodicity, and no retests are required.** **If contacted as the Shift Manager, acknowledge any report given. State to the applicant that demonstration of operability is not required – if asked for guidance, ask the candidate for a recommendation, and agree with any recommendation.**
	Driver	**If directed to reset the Gross Failure, insert Event 15 and Report that the Gross Failure is reset.**
	NUSO	Directs the OATC to reset the Half SCRAM in accordance with 2-OI-99, Reactor Protection System, Section 6.1, Reset of One RPS Trip Logic Channel.
	OATC	Resets the Half Reactor SCRAM on Panel 2-9-5. 2-OI-99, Reactor Protection System Section 6.1, Reset of One RPS Trip Logic Channel. [1] **ENSURE** Reactor Protection System in prestartup/standby readiness alignment in accordance with Section 4.0. [2] **REVIEW** all Precautions and Limitations in Section 3.0. [3] **ENSURE** RPS Bus for tripped channel ENERGIZED. Refer to sections 5.1, 5.2, 8.1, or 8.2. [4] **ENSURE** trip signals **NOT** present. [5] **MOMENTARILY PLACE** SCRAM RESET, 2-HS-99-5A/S5, as follows: [5.1] RESET FIRST. (Group 2/3) [5.2] RESET SECOND. (Group 1/4) [5.3] NORMAL

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 5 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	OATC	[6] **CHECK** the following conditions: A. All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED. B. The following four lights ILLUMINATED: • 2-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE • 2-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE C. Scram Discharge Volume vent and drain valves indicate OPEN. D. Points SOE033 (Channel A3 manual scram) and SOE035 (Channels A1&A2 Auto Scram) on ICS computer or on the First Out Printer reads "NOT TRIP" for RPS "A". E. Points SOE034 (Channel B3 manual scram) and SOE036 (Channels B1&B2 Auto Scram) on ICS computer or on the First Out Printer reads "NOT TRIP" for RPS "B".
	NRC	**End of Event 5. Request that the Driver insert Event 6, Condenser Leakage / High Conductivity.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 1 of 8

Event Description: Condenser Leakage / High Conductivity

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. A leak will develop in the Main Condenser, causing conductivity alarms

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause a Main Condenser Tube leak and High Conductivity.**
	BOP	Acknowledges and reports the following alarm: • CONDENSATE DEMIN ABNORMAL, Panel 2-9-6B, Window 6
	NUSO	Acknowledges the report and directs the BOP to respond in accordance with the Alarm Response Procedure and 2-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	2-ARP-9-6B, Alarm Response Procedure CONDENSATE DEMIN ABNORMAL, Window 6 Operator Action: A. **CHECK** system flow conductivity, and system DP, on Panel 2-9-6 for abnormalities. B. **REQUEST** personnel to determine problem with Condensate Demineralizers. C. **IF** High Conductivity exists, **THEN REFER TO** 2-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	2-AOI-2-1, Reactor Coolant High Conductivity 4.1 Immediate Actions: **None** 4.2 Subsequent Actions **CAUTION** Operations outside of the allowable regions shown on the recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. **REFER TO** 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, for required actions and monitoring to be performed during a power reduction.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 2 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **REDUCE** Reactor Power, as necessary, to maintain radiation levels in Main Steam Lines below the "Main Steam Line Radiation High-High" (2-XA-55-3A, Window 27) setpoint. **REFER TO** 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations. [2] **MONITOR** Reactor Vessel Water Conductivity on 2-CR-43-11A/12A, RWCU CONDUCTIVITY, on Panel 2-9-4. [3] **REQUEST** Chemistry to **SAMPLE** Reactor Water. **REFER TO** 2-SI-4.6.B.1-4, Reactor Coolant Chemistry.
	NRC	**Chemistry sample results will not be provided quickly enough to allow the SRO to address Technical Specifications/Technical Requirements Manual requirements. Therefore, Technical Specifications/Technical Requirements Manual evaluation for this event is not required and should not be used to evaluate the applicant's Technical Specifications/Technical Requirements Manual competency.**
	Driver	**If contacted as Chemistry to sample Reactor Water in accordance with 2-SI-4.6.B.1-4, Reactor Coolant Chemistry, acknowledge the direction. If contacted during this Event about the status of the sample, report that sampling is in progress.**
	BOP	[4] **IF** Condenser Hotwell in-leakage is suspected, **THEN PERFORM** the following: **CAUTION** When isolating a Condenser Waterbox, do NOT allow Condenser Vacuum to lower to less than (-) 25 inches Hg, as indicated on 2-XR-002-0026, CONDENSATE, with the Main Turbine on-line. [4.1] **ISOLATE** and **DRAIN** the Condenser Water Boxes (one at a time). **REFER TO** 2-OI-27, Condenser Circulating Water System.
	NRC	**Candidates may choose to isolate Waterboxes in any order. Therefore, when the first Waterbox is isolated, request that the driver insert Event 15 to delete the Condenser Tube Leak malfunction.**
	Driver	**When requested by the NRC insert Event 16 to delete the Condenser Tube Leak malfunction. Verify that Conductivity begins to lower.**

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 3 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	Isolates Waterboxes until Conductivity begins to lower. Informs the SRO when the leak has been isolated. [4.1.1] **OBSERVE** Reactor Water Conductivity and Off Gas Flow rate. (After the leaking water box is isolated and drained, Reactor Water Conductivity should begin to decrease within 30-45 minutes. Off-Gas Flow rate could raise.) [4.2] N/A [5] N/A [6] N/A
	BOP	**NOTE** If gross intrusion of resin into the Reactor Vessel, Hotwell or Condensate Storage Tank is suspected, the clean-up cannot be effectively done by normal system operation and flushing alone. The affected system needs to be shut down, opened, and physically cleaned to the extent practicable. Special cleaning methods and/or procedures will be developed for each particular case. [7] **IF** Control Room indications or sample requested from Chemistry in Step 4.2[3] shows Reactor Vessel Water Conductivity rising to 10 micromhos OR chloride concentration exceeds 0.5 ppm during power operation (Mode 1) 0R conductivity rises to 2 micromhos during Reactor Startup (Mode 2), **THEN PERFORM** the following steps: [7.1] **REFER TO** TRM 3.4.1, Coolant Chemistry. [7.2] **INITIATE** an orderly Reactor Shutdown. **REFER TO** 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations.
	BOP	Acknowledges and reports the alarms as received: • CONDENSATE PUMP DISCHARGE CONDUCTIVITY HIGH, 2-9-6B, Window 19 • RWCU FILTER OUTLET CONDUCTIVITY HIGH, 2-9-4B, Window 3

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 4 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	2-ARP-9-6B, Alarm Response Procedure CNDS PUMP DISCHARGE CNDCT HIGH, Window 19 Operator Action: A. **CHECK** 2-CR-43-1A on Panel 2-9-6 for changes in the conductivity reading and identify Condensate and/or Booster Pump Discharge Conductivity above 0.2 µS/cm. B. **REQUEST** Chemistry to verify Condensate and/or Booster Pump Discharge Conductivity. C. **CHECK** rate at which the conductivity is rising as a possible clue to what is causing the rise. D. **REFER TO** 2-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	2-ARP-9-4B, Alarm Response Procedure RWCU OUTLET CONDUCTIVITY HIGH, Window 3 Operator Action: A. **CONFIRM** alarm: • **CHECK** recorder 2-CR-43-12A, Panel 2-9-4 • **REQUEST** Chem lab to validate alarm B. **IF** conductivity is rising on 2-CR-43-11A, **THEN CHECK** 2-CR-43-1A on Panel 2-9-6. C. **REFER TO** Tech Spec Section 3.4.6, TRM 3.4.1. D. **IF** valid, **THEN REFER TO** 2-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	2-OI-27, Condenser Circulating Water System Section 8.17, Removing Condenser Waterbox From Service For Leak Check **CAUTIONS** 1) Due to WO 122007449, which requires reduced torque loading on FCV-27-47, COND 2B2 INLET SHUTOFF VALVE, the 2B2 waterbox should not be reoved from service unless absolutely necessary. 2) Only two Waterboxes should be isolated at a time. 2) Prior to removing 2B1 or 2B2 Water Boxes from service, HOTWELL LOW LEVEL MAKEUP CONTROL, 2-LC-2-6, should be PLACED in MANUAL to prevent erratic makeup flow control. 3) Do **NOT** Isolate 2 water boxes side by side.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 5 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **IF** time permits, **THEN COLLECT** the Amertap System balls and **SHUTDOWN** the Amertap System from the Waterboxes to be removed. **REFER TO** 2-OI-27B, Amertap Condenser Tube Cleaning System.
	Driver	**If the crew contacts the Turbine Building AUO or Work Control / Outside SRO to perform 2-OI-27B, Amertap Condenser Tube Cleaning System, in preparation for isolating any Waterbox:** • **Acknowledge the direction to collect Amertap Balls and shutdown the Amertap System** • **State that time compression is being applied and that the Amertap Balls have been collected and the Amertap system is shut down.** **If contacted as the Turbine Building AUO to shut down the debris filter in accordance with 2-OI-27, Condenser Circulating Water System, Section 7.2, Shutdown Of The Debris Filter, acknowledge the direction.**
	BOP	[2] N/A [3] **CLOSE** the CNDR 2A1(2A2)(2B1)(2B2)(2C1)(2C2), OFFGAS SHUTOFF VALVE, to the waterbox to be removed from service using 2-HS-66-2A(3A)(4A)(5A)(6A)(7A) on Panel 2-9-8. **NOTES** 1) CONDENSER 2A, 2B, 2C CCW OUTLET SHUTOFF VALVE travel can be stopped by depressing the associated STOP pushbutton on Panel 2-25-116. 2) Outlet valve may need to be opened to control H2O level rise in OOS water box. 3) Condenser Vacuum shall be maintained greater than 25 inches Hg at all times.

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 6 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **WHILE** closely monitoring Condenser Vacuum, **CLOSE** the appropriate CNDR 2A(2B)(2C) CCW OUTLET SHUTOFF VALVES using the OPEN/CLOSE/STOP push-buttons for the associated handswitches listed below on Panel 2-25-116. (N/A valves not corresponding to the desired waterbox being removed.) • 2-HS-027-0046D, CONDENSER 2A SS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0038D, CONDENSER 2A NS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0062D, CONDENSER 2B SS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0054D, CONDENSER 2B NS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0078D, CONDENSER 2C SS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0070D, CONDENSER 2C NS CCW OUTLET SHUTOFF VALVE
	Driver	**If directed as the Turbine Building AUO to close CCW Outlet Shutoff Valves at Panel 2-25-116, enter event(s) as follows:** **Event 17 - 2-HS-027-0046, 2A SS Outlet** **Event 18 - 2-HS-027-0038, 2A NS Outlet** **Event 19 - 2-HS-027-0062, 2B SS Outlet** **Event 20 - 2-HS-027-0054, 2B NS Outlet** **Event 21 - 2-HS-027-0078, 2C SS Outlet** **Event 22 - 2-HS-027-0070, 2C NS Outlet** **Inform the crew when the chosen valve has been closed.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 7 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[5] As necessary to maintain Condenser Vacuum greater than 25 inches Hg, **SLOWLY THROTTLE OPEN** the remaining CONDENSER 2A(2B)(2C) CCW OUTLET SHUTOFF VALVES by using the following OPEN/CLOSE/STOP pushbutton on Panel 25-116: • 2-HS-027-0046D, CONDENSER 2A SS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0038D, CONDENSER 2A NS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0062D, CONDENSER 2B SS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0054D, CONDENSER 2B NS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0078D, CONDENSER 2C SS CCW OUTLET SHUTOFF VALVE • 2-HS-027-0070D, CONDENSER 2C NS CCW OUTLET SHUTOFF VALVE
	NUSO	[6] **IF** necessary, **LOWER** LOAD and **MAINTAIN** condenser vacuum greater than 25 inches Hg (Otherwise N/A).
	BOP	[7] IF 2B2 (NS) waterbox is being removed from service, THEN PERFORM the following due to WO 122007449. [7.1] ENSURE CLOSED 2-HS-27-54D, CNDR 2B NS CCW OUTL SHUTOFF VLV [7.2] 2-BYV-27-564, OPEN CCW INLET VLV 2-FCV-27-47 BYP
	Driver	**If contacted as an AUO, Work Control /Outside NUSO to throttle open Condenser Shutoff Valves, acknowledge the direction and report that the requested valve has been throttled.**
	NRC	**Inform the Driver when the Inlet Shutoff Valve for the chosen Waterbox is closed and request that the Driver insert Event 16 to delete the leaking tube malfunction.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 8 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[8] **CLOSE** the CONDENSER 2A(2B)(2C) CCW INLET SHUTOFF VALVE for the waterbox to be removed from service using the appropriate switch on Panel 2-9-20: • CONDENSER 2A SS CCW INLET SHUTOFF VALVE, 2-HS-27-39A • CONDENSER 2A NS CCW INLET SHUTOFF VALVE, 2-HS-27-31A • CONDENSER 2B SS CCW INLET SHUTOFF VALVE, 2-HS-27-55A • CONDENSER 2B NS CCW INLET SHUTOFF VALVE, 2-HS-27-47A • CONDENSER 2C SS CCW INLET SHUTOFF VALVE, 2-HS-27-71A • CONDENSER 2C NS CCW INLET SHUTOFF VALVE, 2-HS-27-63A [8.1] IF 2B2 (NS) waterbox is being removed from service, THEN PERFORM the following due to WO 122007449. [8.1.1] **CLOSE** 2-BYV-27-564, CCW INLET VLV 2-FCV-27-47 BYP [9] **ADJUST** the remaining CONDENSER CCW OUTLET SHUTOFF VALVES to optimize CCW System performance by performing Section 6.1. [10] **IF** required, **THEN START** Amertap System on the unaffected Condenser Waterboxes when CCW System pressures and temperatures have stabilized. **REFER TO** 2-OI-27B, Amertap Condenser Tube Cleaning System. [11] N/A
	Driver	**If contacted as an AUO or the Work Control / Outside NUSO to start the Amertap System on any Waterbox that is un-isolated, acknowledge the direction and report that the Amertap System has been started.**
	NRC	**End of Event 6. Request that the Driver insert Event 7, High Suppression Pool Water Level / Emergency Depressurization.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 1 of 9

Event Description: High Suppression Pool Water Level

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. A leak into the Suppression Pool will cause Water Level to rise

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7, to cause Suppression Pool Water Level to rise.**
	NRC	**Event 8, Electrical Anticipated Transient Without SCRAM, and Event 9, Steam Relief Valve Fails to Open, are automatically entered upon Simulator setup. No action is required to enter these events. See page 37 of 40 for Event 8, and page 38 of 40 for Event 9.**
	BOP	Acknowledges and reports the following alarms as they are received: • DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, 2-9-3B, Window 26 • SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, 2-9-3B, Window 15
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedures.
	BOP	2-ARP-9-3B, Alarm Response Procedure DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, Window 26 Operator Action: A. **CHECK** alarm by checking Drywell to Suppression Chamber DP. B. **REFER TO** 2-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell. C. **REFER** TO Tech Spec Section 3.6.2.6, Drywell-to-Suppression Chamber Differential Pressure.
	NRC	**Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant's Tech Spec competency.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 2 of 9

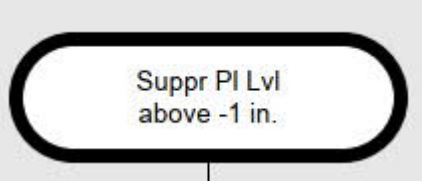
Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 2-ARP-9-3B SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, Window 15 Operator Action: A. **CHECK** Suppression Pool Level using multiple indications. B. N/A C. **IF** level is high, **THEN CHECK** for RCIC, HPCI, Core Spray, or RHR draining to Suppression Pool, and **CHECK** 2-TR-64-161 and -162. D. **REFER TO** 2-OI-74, Residual Heat Removal System, Sections 8.0. E. **REFER TO** Tech Spec Section 3.6.2.2. F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN ENTER** 2-EOI-2 , Primary Containment Control Flowchart. G. N/A
	BOP	Determines that the cause for the Drywell to Suppression Chamber DP alarm is rising Suppression Pool Water Level and informs the NUSO.
	NUSO	Directs the BOP to monitor Suppression Pool Water Level and to provide an update when level reaches (-) 1 inch.
	BOP	Acknowledges and reports the following alarm to the NUSO when received: • SUPPRESSION POOL LEVEL HIGH, 2-9-3F, Window 12
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	2-ARP-9-3F, Alarm Response Procedure SUPPRESSION POOL LEVEL HIGH, Window 12 A. **CHECK** CST 2 and Suppression Pool Level using multiple indications. B. **ENSURE** HPCI Suction automatically transfers to the Suppression Pool. C. N/A D. **REFER TO** Tech Spec 3.5.1, 3.5.2.
	NRC	**Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant's Tech Spec competency.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 3 of 9

Event Description: High Suppression Pool Water Level

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>When appropriate, enters 2-EOI-2, Primary Containment Control on high Suppression Pool Water Level (level above (-) 1 inch).</td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-2, Primary Containment Control
  </td></tr>
<tr><td></td><td>NUSO</td><td>SP/L-1
MONITOR and CONTROL Suppression Pool Water Level (-) 6 inches to (-) 1 inch.
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Suppression Pool Water Level CANNOT be maintained below (-) 1 inch.</td><td>A</td></tr><tr><td>Suppression Pool Water Level CANNOT be maintained above (-) 6 inches.</td><td>NO ACTION REQUIRED</td></tr></table></td></tr>
<tr><td></td><td>NUSO</td><td>A 
SP/L-3
MAINTAIN Suppression Pool Water Level below 19 ft. (APPX 18, 20K)
SP/L-4
WHEN Suppression Pool Level CANNOT be maintained below (APPX 9) 19 feet
SP/L-5
STOP DW Sprays</td></tr>
</table>

Form 3.3-2 Required Operator Actions

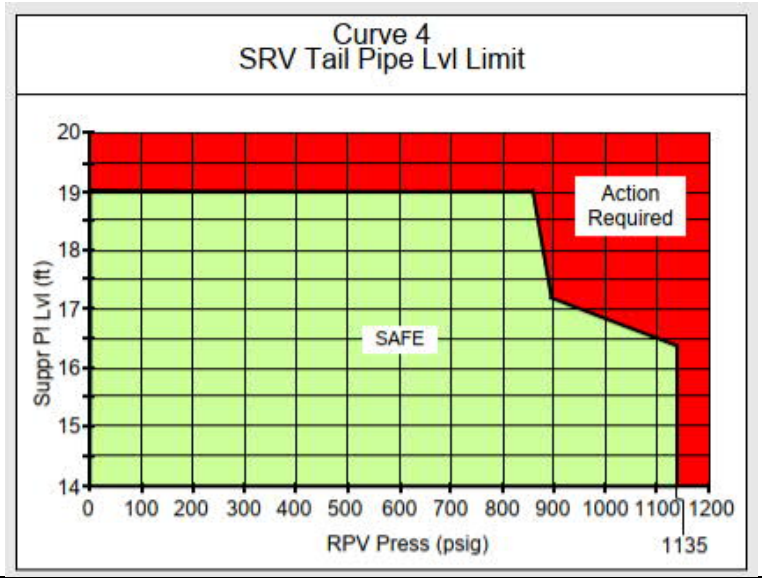
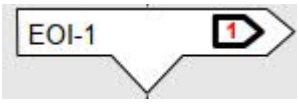
Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 4 of 9

Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the BOP to control Suppression Pool Water Level in accordance with 2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup.
	Driver	**If contacted as an AUO to perform any steps locally per 2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup, acknowledge any direction given.** **If contacted as an AUO to verify any valves closed, acknowledge any information or direction given.**
	BOP	2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup [1] N/A [2] N/A [3] **IF** directed by the NUSO, **THEN REMOVE** water from the Suppression Pool as follows: [3.1] **DISPATCH** personnel to perform the following (Unit 2 RB, El 519 ft, Torus Area): [3.1.1] **ENSURE OPEN** 2-SHV-074-0786A (B), RHR DRAIN PUMP 2A(2B) DISCHARGE SHUTOFF VALVE. [3.1.2] **OPEN** the following valves: • 2-SHV-074-0564A(B), RHR DRAIN PUMP A(B) SEAL WATER SUPPLY • 2-SHV-074-0529A (B), RHR DRAIN PUMP A (B) SHUTOFF VALVE [3.1.3] **UNLOCK** and **OPEN** 2-SHV-074-0765A (B), RHR DRAIN PUMP A(B) DISCHARGE. [3.1.4] **NOTIFY** Unit Operator that RHR Drain Pump 2A (2B) is lined up to remove water from Suppression Pool. [3.1.5] **REMAIN** at torus area UNTIL Unit 2 Operator directs starting of RHR Drain Pump 2A (2B).
	Driver	**After 2 minutes, report that the outside portions of 2-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup are complete. If directed to start the RHR Drain Pump, report that the RHR Drain Pump has been started.** **If contacted as the Rad Waste Operator, acknowledge any reports or direction given.**

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 5 of 9

Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	 SP/L-9 **MAINTAIN** Suppression Pool Level within the safe area of Curve 4 (APPX 18, 20K) 
	NUSO	SP/L-10 **WHEN** Suppression Pool Level CANNOT be maintained within the safe area of Curve 4 (APPX 9) 

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 6 of 9

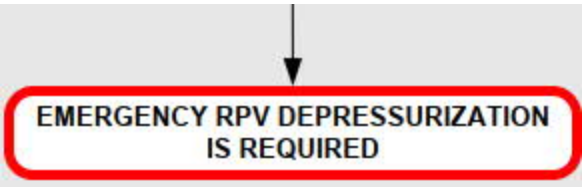
Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	CREW	**Critical Task:** **When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator (NUSO).** **Critical Task Failure Criteria:** **The operating crew fails to proceed without delay and in a controlled manner to insert a manual SCRAM before Suppression Pool Water Level enters the 'Action Required' area of Curve 4, SRV Tail Pipe Level Limit.**
	NUSO	Directs the OATC to insert a manual Reactor SCRAM and directs the crew to enter 2-AOI-100-1, Reactor SCRAM.
	NRC	**Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**
	OATC	2-AOI-100-1, Reactor SCRAM Immediate Actions: [1] **DEPRESS** REACTOR SCRAM A and B, 2-HS-99-5A/S3A and 2-HS-99-5A/S3B, on Panel 2-9-5. [2] **PLACE** REACTOR MODE SWITCH, 2-HS-99-5A/S1, in SHUTDOWN.
	NRC	**When the OATC attempts to insert a manual Reactor SCRAM, Event 8, Electrical Anticipated Transient Without Scram (ATWS), will occur. No action is required by the Driver to insert Event 8. See page 37 of 40 for Event 8 actions.**
	NUSO	Enters 2-EOI-1, ATWS RPV Control, and then exits to 2-EOI-1, RPV Control MODES 1-3 when Control Rods are inserted by ARI.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 7 of 9

Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	When all Control Rods are inserted, directs the crew to perform the following: • Maintain Reactor Pressure to ensure that Suppression Pool Level is maintained within the safe are of Curve 4 in accordance with 2-EOI-Appendix-8B, Reopening MSIVs/Bypass Valve Operation • Maintain Reactor Water Level in accordance with 2-EOI Appendix- 5A, Injection System Lineup Condensate/Feedwater
	NUSO	(Continuing actions of 2-EOI-2, Primary Containment Control) SP/L-11 **WHEN** Suppression Pool Level and RPV Pressure CANNOT be maintained within the safe area of Curve 4 (APPX 9). SP/L-12 **STOP** injection into RPV from sources external to Primary Containment **EXCEPT** from systems required to assure Adequate Core Cooling or shut down the Reactor SP/L-13 **WHEN** Suppression Pool Level and RPV Pressure CANNOT be restored and maintained within the safe area of Curve 4 

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 8 of 9

Event Description: High Suppression Pool Water Level

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>CREW</td><td>Critical Task:
When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the NUSO determines that Emergency Depressurization is required. The Operator initiates Emergency Depressurization as directed by the NUSO.

Critical Task Failure Criteria:
The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time that Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.</td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-1, RPV Control MODES 1-3

Emergency RPV Depressurization

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that RPV depressurization will result in loss of injection required for Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 feet</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 9 of 9

Event Description: High Suppression Pool Water Level

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
RC/P-7
<table>
<tr><td>IF</td><td>Drywell Pressure is above 2.45 psig</td></tr>
<tr><td>THEN</td><td>PREVENT injection from ONLY those Core Spray and LPCI pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr>
</table>
RC/P-8
<table>
<tr><td colspan="2">EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Water Level is above 5.5 feet
THEN OPEN 6 MSRVs (ADS Valves preferred)
• OK to exceed 100 F/hr Cooldown Rate</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Control Air becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Less than 4 MSRVs can be opened
AND
RPV Pressure is 80 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NRC</td><td>See page 38 of 40 for Event 9, Steam Relief Valve Fails to Open actions.</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 7. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 8 Page 1 of 1

Event Description: Electrical Anticipated Transient Without Scram (ATWS)

Symptoms/Cues: Event is automatically initiated when the crew inserts a manual Reactor SCRAM; Rods will remain out when a Reactor SCRAM is inserted

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 8, Electrical Anticipated Transient Without Scram (ATWS), is automatically entered on simulator setup. No action is required to insert this event.**
	OATC	2-AOI-100-1, Reactor SCRAM [3] **IF** all Control Rods can **NOT** be verified fully inserted, **THEN INITIATE** Alternate Rod Insertion (ARI).
	Driver	**Verify that the ATWS malfunction clears, and Control Rods insert when the OATC initiates ARI.**
	OATC	After initiating ARI, verifies that Control Rods insert and informs the NUSO. Continues with Reactor SCRAM actions. [4] **IF** Reactor Power is 5% or BELOW, **THEN**: (otherwise MARK N/A) **REPORT** the following to the NUSO: • Reactor SCRAM • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Reactor Power Level Performs Subsequent Actions in accordance with 2-AOI-100-1, Reactor SCRAM, Section 4.2.
	NRC	**End of Event 8.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 9 Page 1 of 1

Event Description: Steam Relief Valve (SRV) Fails to Open

Symptoms/Cues: Event is automatically inserted on Simulator setup. One Steam Relief Valve will fail to open during Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9 is automatically inserted on Simulator setup. No action is required by the Driver to insert Event 9. Steam Relief Valve 1-22 will not open during Emergency Depressurization.**
	BOP	Emergency Depressurizes the Reactor by opening six (6) SRVs, ADS Valves are preferred. Determines that SRV PCV-1-22 did not open, and opens one non-ADS Valve. Informs the NUSO that 2 ADS Valves failed to open, but there are six (6) SRVs open for Emergency Depressurization.
	NRC	**End of Event 9, end of Scenario. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.**

Scenario Setup
UNIT 2

IC	290
Exam IC	290

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-3	163	ARP-6B	22	T.S. 3.5.3	334
OI-27	102	ARP-6C	27	T.S. 3.6.1.3	253
OI-66	124	ARP-7A	35		
OI-74	195	AOI-2-1	23		
OI-99	100	AOI-64C	18		
ARP-3B	40	AOI-100-1	121		
ARP-3F	42	EOI-1	21		
ARP-4A	52	EOI-2	18		
ARP-4B	51	APPX-18	10		
ARP-5B	33	T.S. 3.3.1.1	258		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Danger Tag on 2A RBCCW Pump Hang Protected Equipment Tag on 1B RBCCW Pump
Schedule Files(s):	ILT 2304 NRC Scenario 6 UNIT 2.sch
Event Files(s):	ILT 2304 NRC Scenario 6 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ON LINE	Drywell Leakage (GPM)	Protected Equipment
	230		1B RBCCW Pump
		1.89	
Rx Power	PRA (Phoenix) -Green	Floor Drain (GPM)	
100%	500Kv GRID - Qualified	0.31	
MWe	161Kv Grid -Qualified	Equipment Drain (GPM)	
1307		1.58	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

HPCI PMT run completed.
2A RBCCW Pump tagged for bearing replacement.
Spare RBCCW Pump aligned to Unit 2.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Return RHR Loop I to standby readiness.
Support maintenance on 2A RBCCW Pump.

OPERATOR WORK AROUNDS **OWAs – U0-1/U2-0** **Burdens - U0 – 0/U2-0** **Operator Challenges – U0 – 12/U2 - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-6
Scenario Source: NEW Op-Test Number: 23-04
Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 100% Reactor Power. Loop I of RHR is in Suppression Pool Cooling for a HPCI run on the previous shift. 3A RBCCW Pump is tagged for bearing replacement, the spare RBCCW Pump is aligned to Unit 3.

Turnover: Place Loop I of RHR in standby readiness.

Critical Tasks:
1. When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator (NUSO).
2. When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the Nuclear Unit Senior Operator (NUSO) determines that Emergency Depressurization is required. The Operator initiates Emergency Depressurization as directed by the NUSO.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Secure Suppression Pool Cooling
2.	SCHEDULE STACK	C-OATC MC-OATC C-NUSO	Stack Dilution Fan Failure
3.	TH09A	I-OATC I-NUSO	Feedwater Level Control System (FWLCS) Input Failure (3-53)
4.	ZLOHS713A_1 ZLOHS713A_2 FCV-71-3	C-BOP MC-BOP TS-NUSO	Reactor Core Isolation Cooling (RCIC) System Isolation, One Valve Fails to Close
5.	TH31A	I-OATC TS-NUSO	Reactor High Pressure Instrument A1 Fails High / Half SCRAM
6.	MCO1A-F	C-BOP C-NUSO	Condenser Leakage / High Conductivity
7.	SCHEDULE TORUS	M-ALL	High Suppression Pool Water Level / Emergency Depressurization
8.	RP07	C-OATC MC-OATC C-NUSO	Electrical Anticipated Without Transient (ATWS)
9.	HS-1-22A	C-BOP C-NUSO	Steam Relief Valve (SRV) Fails to Open

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will place Suppression Pool Cooling in standby readiness in accordance with 3-OI-74, Residual Heat Removal Cooling System.
2. Stack Dilution Fan 3A will fail and the standby Stack Dilution Fan will fail to automatically start. The crew will take action to manually restore Stack Dilution Flow by starting 3B Stack Dilution Fan in accordance with 3-OI-66, Off-Gas System.
3. A Reactor Water Level input into the Feedwater Level Control System (LT-3-53) will fail low, requiring the crew to respond in accordance with Alarm Response Procedures and 3-OI-3, Reactor Feedwater System, to bypass the failed level indicator.
4. Reactor Core Isolation Cooling (RCIC) will receive an inadvertent isolation signal but not all valves will automatically close, requiring manual action to ensure RCIC is isolated. The Nuclear Senior Unit Operator (NUSO) will address Technical Specifications.
5. Reactor High Pressure A1 Channel Fails High/Half SCRAM (3-PIS-3-22AA), inserting a Reactor Half SCRAM. The crew will respond in accordance with Alarm Response Procedures and the Nuclear Unit Senior Operator (NUSO) will address Technical Specifications.
6. The Main Condenser will develop a tube leak in a waterbox, causing conductivity levels to rise. The crew will respond in accordance with 3-AOI-2-1, Reactor Coolant High Conductivity, and will isolate Main Condenser Waterboxes until the leak is isolated.
7. The crew will respond to a rising Suppression Pool Water Level in accordance with Emergency Operating Instruction 3-EOI-2, Primary Containment Control. Suppression Pool Water Level will not be able to be maintained in the safe area of Curve 4, SRV Tail Pipe Level Limit, requiring the crew to insert a manual Reactor SCRAM and eventually Emergency Depressurize the Reactor.
8. When the Operator at the Controls (OATC) inserts the Reactor SCRAM an Electrical Anticipated Transient Without SCRAM (ATWS) will occur, requiring the OATC to take manual action to insert the Control Rods.
9. When the crew attempts to Emergency Depressurize the Reactor, one (1) SRV will fail to open, requiring the crew to open an additional non-ADS SRVs for Emergency Depressurization.

The Scenario ends when the crew has Emergency Depressurized the Reactor, and has control of Reactor Water Level above the Top of Active Fuel (TAF (-) 162 inches) using any available low-pressure injection system.

Critical Tasks: 2

1. **When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator (NUSO).**

   a. **Safety Significance**

   Prevent failure of Primary Containment from over pressurization.

   b. **Cues**

   Procedural Compliance.
   Suppression Pool Level indication.

   c. **Measured by**

   Observation – Both RPS SCRAM switches are depressed.

   d. **Feedback**

   Control Rods insert to their full in position.

   e. **Critical Task Failure Criteria**

   The operating crew fails to proceed without delay and in a controlled manner to insert a manual SCRAM before Suppression Pool Water Level enters the 'Action Required' area of Curve 4, SRV Tail Pipe Level Limit.

2. **When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the Nuclear Unit Senior Operator determines that Emergency Depressurization is required. The Balance of Plant Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator.**

   a. **Safety Significance**

   Prevent failure of Primary Containment from over pressurization.

   b. **Cues**

   Procedural Compliance.
   RPV Pressure indication.
   Suppression Pool Level indication.

   c. **Measured by**

   Observation – the Nuclear Unit Senior Operator determines (as indicated by announcement or observable transition to RPV Emergency Depressurization), Emergency Depressurization is required at or before Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.
   AND
   Observation – the Nuclear Unit Senior Operator directs the Balance of Plant Operator to open 6 ADS valves

   d. **Feedback**

   Suppression Pool Water Level trend.
   RPV Pressure trend.
   MSRV status indication.

e. **Critical Task Failure Criteria**

The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time that Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 1 Page 1 of 3

Event Description: Secure Suppression Pool Cooling

Symptoms/Cues: Event is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	Driver	**PRIOR to placing the simulator in RUN, start CPERF to record critical parameters.**
	NRC	**If the crew does not start Event 1, Secure Suppression Pool Cooling, after assuming the shift request that the driver contact the Unit Supervisor (SRO) as the Shift Manager and direct the crew to place Loop I of Residual Heat Removal (RHR) in standby readiness.**
	Driver	**If requested by the Chief Examiner, contact the SRO as the Shift Manager, and direct the crew to place Loop I of RHR in standby readiness.**
	NUSO	Directs the Balance of Plant Operator (BOP) to remove Suppression Pool Cooling from service in accordance with 3-OI-74, Residual Heat Removal System, Section 8.6, Shutdown of Loop I(II) Suppression Pool Cooling.
	BOP	3-OI-74, Residual Heat Removal System Section 8.6, Shutdown of Loop I(II) Suppression Pool Cooling **NOTE** 1) All operations are performed at Panel 3-9-3 unless otherwise noted. 2) RHR Flow should be monitored while in operation with multiple flow paths (e.g., LPCI and Suppression Pool Cooling together, etc.). During any evolution, total system flow as indicated on 3-FI-74-50(64), RHR SYSTEM I(II) FLOW, should remain between 7,000 to 10,000 GPM for 1 pump operation or between 10,000 and 20,000 GPM for 2-pump operation. [1] **ENSURE** Suppression Pool Cooling is in operation. **REFER TO** Section 8.5. [2] **REVIEW** the Precautions and Limitations in Section 3.0. [3] **NOTIFY** Radiation Protection of Suppression Pool Cooling Loop being removed from service. **RECORD** name and time of Radiation Protection representative notified in NOMS narrative log.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 1 Page 2 of 3

Event Description: Secure Suppression Pool Cooling

Time	Position	Applicant's Actions or Behavior
	Driver	**When contacted as Radiation Protection, acknowledge that Suppression Pool Cooling is being removed from service and provide a name for the NOMS narrative log.** **If contacted as the Reactor Building Assistant Unit Operator (AUO) to go to the RHR Quad for securing RHR pumps, acknowledge any information or direction given.**
	BOP	**CAUTIONS** 1) To prevent draining an RHR Loop, at least one of the RHR System Test Valves must be closed before stopping RHR Pumps in the associated loop. 2) To prevent excessive vibration, RHR Pumps should not be allowed to operate for more than 3 minutes at minimum flow. 3) When closing throttle valves 3-FCV-74-59, RHR SYS I SUPPRESSION POOL COOLING/TEST VALVE from the Control Room, the handswitch should be held in the close position for approximately 6 seconds after the red light extinguishes. Failure to completely close these valves could provide a leak path to the suppression pool from the RHR discharge piping. [4] N/A [5] **CLOSE** 3-FCV-74-59(73), RHR SYSTEM I(II) SUPPRESSION POOL COOLING/TEST VALVE. [6] **STOP** and/or 3-HS-74-16A, RHR PUMP 3A and 3-HS-74-5A, RHR PUMP 3C. [7] **CLOSE** 3-FCV-74-57(71), RHR SYSTEM I(II) SUPPRESSION CHAMBER/POOL ISOLATION VALVE. [8] **CLOSE** 3-FCV-23-34, and 3--FCV-23-40, 3A RHR HEAT EXCHANGER RHRSW OUTLET VALVE and 3C RHR HEAT EXCHANGER RHRSW OUTLET VALVE. [9] **IF** RHRSW for RHR Heat Exchanger(s) A and C is not required to support other unit operations, **THEN STOP** RHRSW Pump(s) for the Heat Exchanger(s) removed from service.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 1 Page 3 of 3

Event Description: Secure Suppression Pool Cooling

Time	Position	Applicant's Actions or Behavior
	BOP	[10] **CHECK** RHR System Discharge Header Pressure > 48 psig, as indicated on 3-PI-74-51, RHR SYS I DISCHARGE PRESSURE, and > 35 psig, as indicated on 3-PI-74-65, RHR SYS II DISCHARGE PRESSURE. [11] N/A
	Driver	**If contacted as Unit 1 or 2 that RHRSW Pumps are being secured, acknowledge any information or direction given.**
	NRC	**End of Event 1.** **EXAMINER NOTE: THE BALANCE OF PLANT OPERATOR (BOP) MUST BE REMOVED FROM THE CONTROL ROOM TO ALLOW THE OPERATOR AT THE CONTROLS (OATC) TO RESPOND TO THE NEXT EVENT.** **Request that the Driver insert Event 2, Stack Dilution fan failure.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 2 Page 1 of 2

Event Description: Stack Dilution Fan Failure

Symptoms/Cues: Event is initiated when requested by the Chief Examiner. 3A Stack Dilution Fan will trip

Time	Position	Applicant's Actions or Behavior
	Driver	**CONTACT THE BOP AS REQUESTED BY THE CHIEF EXAMINER WITH A REASON TO LEAVE THE CONTROL ROOM.** **VERIFY THAT THE BOP HAS BEEN REMOVED FROM THE CONTROL ROOM PRIOR TO INSERTING EVENT 2.** **When requested by the Chief Examiner and the BOP is not in the Control Room, insert Event 2 to trip 3A Stack Dilution Fan.**
	OATC	Acknowledges and reports the following alarm: • STACK GAS DILUTION AIR FLOW LOW, 3-9-7A, Window 3
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	NRC	**It is acceptable if OATC elects to start 3B Stack Dilution Fan in accordance with 3-OI-66, Section 8.6, instead of using the ARP below.**
	OATC	3-ARP-9-7A, Alarm Response Procedure STACK GAS DILUTION AIR FLOW LOW, Window 3 Operator Action: A. **CHECK** alternate fan ON and damper open, (red light illuminated) on Panel 3-9-7.
	OATC	Determines that the standby Stack Dilution Fan did not automatically start and manually starts 3B Stack Dilution Fan.
	NRC	**The BOP can be allowed to return to the Control Room at this point.**
	OATC	B. **DISPATCH** personnel to stack to check and report status of the following for both fans: 1. Fan motor. 2. Fan belts. 3. Damper stuck closed. C. **CHECK** breaker 5E on 480V Diesel Aux Bd A and B.
	Driver	**If contacted as an AUO, acknowledge any information or direction given.**

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 2 Page 2 of 2

Event Description: Stack Dilution Fan Failure

Time	Position	Applicant's Actions or Behavior
	OATC	3-OI-66, Off-Gas System Section 8.6, Standby Gas Dilution Fan Operation [1] **REVIEW** all Precautions and Limitations in Section 3.0. [2] N/A [3] **PLACE** the 3-XS-66-29, STACK DILUTION FAN SEL, in OFF. [4] **PLACE** applicable standby STACK DILUTION FAN control switch in START. • 3-HS-66-31A, STACK DILUTION FAN 3B [5] **IF** desired to operate only one stack dilution fan, **THEN PERFORM** the following for STACK DILUTION FAN 3A. (Otherwise N/A) [5.1] **PLACE** operating (or possibly tripped) 3-HS-66-29A, STACK DILUTION FAN 3A control switch, in STOP (spring return to center position (auto)). [5.2] **PLACE** the 3-XS-66-29, STACK DILUTION FAN SELECT control switch, inFAN A. [5.3] **ENSURE** 3-HS-66-29A,STACK DILUTION FAN 3A control switch, in AUTO (spring return to center position (auto)). [6] N/A
	NRC	**End of Event 2. Request that the driver insert Event 3, Feedwater Level Control System (FWLCS) Input Failure (3-53).**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 3 Page 1 of 3

Event Description: Feedwater Level Control System (FWLCS) Input Failure (3-53)

Symptoms/Cues: Event initiated when requested by the Chief Examiner. Feedwater Level Control Input LT-3-53, REACTOR WATER LEVEL 'A' will fail downscale

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to fail Feedwater Level Control Input LT-3-53, REACTOR WATER LEVEL 'A', downscale.**
	OATC	Acknowledges and reports the following alarm: • RFWCS INPUT FAILURE, 3-9-6C, Window 14
	NUSO	Directs the OATC to respond in accordance with the Alarm Response Procedure.
	OATC	3-ARP-9-6C, Alarm Response Procedure RFWCS INPUT FAILURE, Window 14 Operator Action: A. **CHECK** RFWCS continues to maintain Reactor Water Level. B. **IDENTIFY** bad/invalid signal by checking Control Room instrumentation and/or ICS. **REFER TO** ATTACHMENT 1 on next page for list of RFWCS instrumentation. **REFER TO** ICS RX FW LVL CONTROL SYS display (FWLCS). C. **REQUEST** assistance from Site Engineering. D. **BYPASS** the bad/invalid signal with Unit Supervisor approval. (REFER TO 3-OI-3, Reactor Feedwater System).
	Driver	**If contacted as Site Engineering, acknowledge any information or direction given.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 3 Page 2 of 3

Event Description: Feedwater Level Control System (FWLCS) Input Failure (3-53)

Time	Position	Applicant's Actions or Behavior
	OATC	**Attachment 1** RFWCS CONTROL ROOM INSTRUMENTATION * Denotes instrumentation with amber light indicating when the associated signal has been bypassed either manually or automatically by RFWCS. **REACTOR WATER LEVEL** *Reactor Water Level A - 3-LI-3-53 - Panel 3-9-5 *Reactor Water Level B - 3-LI-3-60 - Panel 3-9-5 *Reactor Water Level C - 3-LI-3-206 - Panel 3-9-5 *Reactor Water Level D - 3-LI-3-253 - Panel 3-9-5 Average Reactor Water Level - 3-XR-3-53 - Panel 3-9-5
	NRC	**The Nuclear Unit Senior Operator (NUSO) may reference Technical Specifications, but there are no LCOs associated with this event.**
	NUSO	Directs the Operator at the Controls (OATC) to bypass 3-LI-3-53, REACTOR WATER LEVEL 'A', in accordance with 3-OI-3, Reactor Feedwater System.
	OATC	3-OI-3, Reactor Feedwater System Section 8.2, Bypassing RFWCS Narrow Range Level Instruments **NOTES** 1) RFW Control System will allow up to three Narrow Range Reactor Level instruments to be bypassed at a time. Should the fourth level instrument fail with the other three instruments bypassed, 3-LIC-46-5, REACTOR WATER LEVEL CONTROL PDS, will trip to MANUAL mode. 2) Attachment 8 can be referred to for general information on RFWCS instrumentation. [1] **OBTAIN** approval from NUSO. [2] **BYPASS** desired Narrow Range Reactor Level instrument(s) on Panel 3-9-5 as follows: • To bypass 3-LI-3-53, LEVEL A, **DEPRESS** HS-3-53 BYPASS pushbutton.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 3 Page 3 of 3

Event Description: Feedwater Level Control System (FWLCS) Input Failure (3-53)

Time	Position	Applicant's Actions or Behavior
	OATC	[3] **CHECK** amber light illuminated on level instrument bypassed. [4] **IF** level instrument(s) was bypassed in Step 8.2[1] due to signal failure, **THEN** (Otherwise N/A): • **CHECK** RFWCS INPUT FAILURE, 3-XA-55-6C Window 14, will reset (N/A if annunciation was in alarm prior to signal failure). [5] **REQUEST** assistance from Engineering and IMs. [6] **ENSURE** RFW Control System continues to maintain Reactor Water Level.
	Driver	**If contacted as the Work Control / Outside NUSO, Engineering, Instrument Mechanics, or AUO, acknowledge any information or direction given.**
	NRC	**End of Event 3. Request that the Driver insert Event 4, Reactor Core Isolation Cooling (RCIC) System Isolation, One Valve Fails to Close.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 1 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. RCIC will receive an isolation signal, but 3-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE, will not automatically close

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a RCIC System Isolation.**
	NRC	**3-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE, will not automatically close when the isolation signal is received. The valve will close when an operator gives it a close signal manually.**
	BOP	Acknowledges and reports the following alarms to the NUSO: • RCIC TURBINE TRIPPED, 3-9-3B, Window 14 • RCIC TURBINE EXHAUST DISC RUPTURED, 3-9-3B, Window 35
	NUSO	Directs the BOP to respond in accordance with the appropriate ARPs.
	BOP	3-ARP-9-3B, Alarm Response Procedure RCIC TURBINE TRIPPED, Window 14 Operator Action: A. **DETERMINE** cause. B. **IF** valid alarm, **THEN REFER TO** 3-OI-71, Reactor Core Isolation Cooling System, Section 8.0. C. **REFER TO** Tech Spec 3.5.3.
	BOP	3-ARP-9-3B, Alarm Response Procedure RCIC TURBINE EXHAUST DISC RUPTURED, Window 35 Operator Action: A. **REFER TO** 3-AOI-64-2C, Group 5 Reactor Core Isolation Cooling Isolation

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 2 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	BOP	3-AOI-64-2C, Group 5 Reactor Core Isolation Cooling Isolation 3.0 AUTOMATIC ACTIONS • RCIC Turbine Trips • 3-FCV-71-2, RCIC STEAM LINE INBOARD ISOLATION VALVE CLOSES • 3-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE CLOSES • 3-FCV-71-34, RCIC PUMP MINIMUM FLOW VALVE CLOSES • 3-FCV-71-9, RCIC TURBINE TRIP/THROTTLE VALVE CLOSES 4.1 Immediate Actions [1] **ENSURE** automatic actions occur.
	BOP	Determines that 3-FCV-71-3, RCIC STEAM LINE OUTBOARD ISOLATION VALVE did not automatically close and manually closes 3-FCV-71-3. Informs the NUSO concerning the status of 3-FCV-71-3.
	BOP	4.2 Subsequent Actions [1] **IF** ANY EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). [2] **DISPATCH** an operator to the RCIC Turbine room to investigate.
	Driver	**If contacted as an AUO, Mechanical Maintenance, or the Work Control / Outside NUSO acknowledge any information or direction given.**
	BOP	[3] **CHECK** the following monitors for a rise in activity or area temperature: [3.1] 3-RR-90-1, AREA RADIATION, Bank 2 Point 6 & 9 (3-RI-90-26 & 29), Panel 3-9-2. [3.2] 3-MON-90-50, AIR PARTICULATE MONITOR RADIATION (3-RM-90-57), Panel 3-9-2. [3.3] 3-RM-90-250, Reactor and Turbine Building Exhaust Radiation on CHEMISTRY CAM MONITOR CONTROLLER, 0-MON-90-361A, Panel 1-9-2.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 3 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	Driver	**If contacted as Unit 1, there is no apparent rise in 2-MON-90-250 activity.**
	BOP	[3.4] 3-TR-69-29, LEAK DETECTION SYSTEM TEMPERATURE (Panel 3-9-21) [4] **ENSURE** the following valves **OPEN** to drain RCIC steam lines: • 3-FCV-1-55 MAIN STEAM LINE INBOARD • 3-FCV-1-56, OUTBOARD DRAIN ISOLATION VALVE [5] **IF** the isolation occurred due to Exhaust Diaphragm high pressure, **THEN DISPATCH** an AUO to check 3-SHV-071-0014, RCIC TURB EXH SHUTOFF VALVE, DRAIN LINE SHUTOFF VALVE LOCKED OPEN. [6] N/A [7] N/A [8] N/A [9] **REFER TO** 3-OI-71, Reactor Core Isolation Cooling System and **PLACE** RCIC in the condition directed by the NUSO.
	Driver	**If contacted as the Work Control/Outside NUSO or AUO to investigate the cause, prepare tags, or address Protected Equipment Requirements acknowledge the direction.** **If contacted as an AUO to check ATUs in the Aux Instrument Room, acknowledge the direction.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 4 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior
	NUSO	Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs) LCO – Each PCIV, except Reactor Building to Suppression Chamber Vacuum Breakers, shall be OPERABLE Applicability: NOTES: 1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be un-isolated intermittently under administrative controls. 2. Separate Condition entry is allowed for each penetration flow path. 3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs. 4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3. <u>Condition:</u> ---------- NOTE ---------- Only applicable to penetration flow paths with two PCIVs. ---------- A. One or more penetration flow paths with one PCIV inoperable except due to MSIV Leakage not within limits.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 5 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

<table>
<tr><th>Time</th><th>Position</th><th colspan="2">Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>REQUIRED ACTION
A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.
<u>AND</u>
A.2
----------NOTE----------
Isolation devices in high radiation areas may be verified by use of administrative means.

Verify the affected penetration flow path is isolated.</td><td>COMPLETION TIME
A.1 – 4 hours except for Main Steam Line
<u>AND</u>
A.2 – Once per 31 days for isolation devices outside Primary Containment
<u>AND</u>
Prior to entering MODE 2 or 3 from MODE 4, if Primary Containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside Primary Containment</td></tr>
<tr><td></td><td>NUSO</td><td colspan="2">Technical Specification 3.5.3, RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE

---------- NOTE ----------
LCO 3.0.4.b is not applicable to RCIC

<u>Condition:</u>
A. RCIC System inoperable.</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 4 Page 6 of 6

Event Description: Reactor Core Isolation Cooling (RCIC) Isolation, One Valve Fails to Automatically Close

Time	Position	Applicant's Actions or Behavior	
	NUSO	**REQUIRED ACTION** A.1 Verify by administrative means High Pressure Coolant Injection is OPERABLE. AND A.2 Restore RCIC System to OPERABLE status.	**COMPLETION TIME** A.1 – Immediately A.2 – 14 days
	NRC	**End of Event 4. Request that the Driver insert Event 5, Reactor High Pressure Instrument A1 Fails High / Half SCRAM**	

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 1 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. A Reactor half SCRAM will be initiated by a Reactor Pressure Instrument failing high

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to fail Reactor High Pressure Channel A1 high.**
	OATC	Acknowledges and reports the following alarms: • RX VESSEL PRESSURE HIGH HALF SCRAM, 3-9-4A, Window 9 • REACTOR CHANNEL 'A' AUTO SCRAM, 3-9-5B, Window 1 • RPS ANALOG TRIP UNIT TROUBLE, 3-9-5B, Window 23
	NUSO	Directs the OATC to respond in accordance with the appropriate Alarm Response Procedures.
	OATC	3-ARP-9-4A, Alarm Response Procedure RX VESSEL PRESSURE HIGH HALF SCRAM, 3-9-4A, Window 9 Operator Action: A. **CONFIRM** alarm by multiple indications. B. N/A C. **DISPATCH** personnel to the sensors to check for abnormal conditions. D. **IF** alarm is NOT valid or initiating condition is corrected, **THEN PERFORM** the following: 1. **RESET** Half SCRAM with SRO permission. 2. **REFER TO** 3-OI-99, Reactor Protection System. E. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	BOP	3-ARP-9-5B, Alarm Response Procedure REACTOR CHANNEL 'A' AUTO SCRAM, 3-9-5B, Window 1 Operator Action: A. **CONFIRM** channel A relays dropped out by checking SCRAM solenoid and backup SCRAM valve lights extinguished.

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 2 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	OATC	B. **IF** any EOI entry condition is met, **THEN ENTER** the appropriate EOI(s). C. N/A D. N/A E. With NUSO permission, **RESET** half-SCRAM. **REFER TO** 3-OI-99, Reactor Protection System. F. **CHECK** first-out printer to determine initiating cause. G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5, Equipment Important to Emergency Response.
	OATC	3-ARP-9-5B, Alarm Response Procedure RPS ANALOG TRIP UNIT TROUBLE, 3-9-5B, Window 23 Operator Action: A. **DISPATCH** personnel to Aux Instrument Room to investigate cause. B. **IF** a power supply light on ATU Panel 3-9-83, -84, -85, or-86 is extinguished, **THEN DETERMINE** cause and **ATTEMPT** power restoration. C. **IF LATCH** light on Trip Unit is illuminated indicating GROSS FAIL, **THEN DETERMINE** cause and **ATTEMPT** reset. **NOTE** Other parameters will be masked from alarming while this alarm is sealed in. D. **FREQUENTLY MONITOR** other input parameters. E. **IF** necessary, **INITIATE** a Condition Report (CR) for Instrument Group to troubleshoot and repair. F. **IF** alarm is **NOT** valid, **THEN REFER TO** 0-OI-55, Annunciator System.
	Driver	**When directed to investigate as the Control Bay Assistant Unit Operator (AUO), Work Control / Outside NUSO, wait 3 minutes and inform the crew that 3-PIS-3-22AA, Reactor Pressure A1 Channel is failed high with a Gross Failure. Investigation is in progress.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 3 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	NRC	**After the NUSO has addressed Technical Specifications, request that the driver contact the Control Room with the cause for the Reactor High Pressure trip.**
	NUSO	References Technical Specification 3.3, Instrumentation 3.3.1.1, Reactor Protection System (RPS) Instrumentation LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE. Applicability: According to Table 3.3.1.1-1 <u>Condition:</u> One or more required channels inoperable.
	NUSO	**REQUIRED ACTION** / **COMPLETION TIME** A.1 Place channel in trip. — A.1 – 12 hours <u>OR</u> A.2 — A.2 – 12 hours ----------NOTE---------- Not applicable to Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------- Place associated trip system in trip
	NUSO	Table 3.3.1.1-1 (page 2 of 3) Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 4 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	Driver	**When directed by the Chief Examiner, inform the crew that Instrument Mechanics in the Auxiliary Instrument Room reported that while performing the initial test setup for 3-SR-3.3.6.1.5(1B/A1), Primary Containment Isolation System Main Line Low Pressure Instrument Channel A1 Calibration, they inadvertently connected to 3-PIS-3-22AA, Reactor Pressure A1 channel, and tripped the ATU.** **If contacted as an Instrument Mechanic, report that no adjustments were made to 3-PIS-3-22AA, Reactor Pressure A1 Channel, that all test equipment has been removed, the instrument is within testing periodicity, and no retests are required.** **If contacted as the Shift Manager, acknowledge any report given. State to the applicant that demonstration of operability is not required – if asked for guidance, ask the candidate for a recommendation, and agree with any recommendation.**
	Driver	**If directed to reset the Gross Failure, insert Event 15 and Report that the Gross Failure is reset.**
	NUSO	Directs the OATC to reset the Half SCRAM in accordance with 3-OI-99, Reactor Protection System, Section 6.1, Reset of One RPS Trip Logic Channel.
	OATC	Resets the Half Reactor SCRAM on Panel 3-9-5. 3-OI-99, Reactor Protection System Section 6.1, Reset of One RPS Trip Logic Channel. [1] **ENSURE** Reactor Protection System in prestartup/standby readiness alignment in accordance with Section 4.0. [2] **REVIEW** all Precautions and Limitations in Section 3.0. [3] **ENSURE** RPS Bus for tripped channel ENERGIZED. [4] **ENSURE** trip signals **NOT** present. [5] **MOMENTARILY PLACE** SCRAM RESET, 3-HS-99-5A/S5, as follows: [5.1] RESET FIRST. (Group 2/3) [5.2] RESET SECOND. (Group 1/4) [5.3] NORMAL

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 5 Page 5 of 5

Event Description: Reactor High Pressure Instrument A1 Fails High / Half SCRAM

Time	Position	Applicant's Actions or Behavior
	OATC	[6] **CHECK** the following conditions: A. All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED. B. The following four lights ILLUMINATED: • 3-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE • 3-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE C. Scram Discharge Volume vent and drain valves indicate OPEN. D. Points SOE033 (Channel A3 manual scram) and SOE035 (Channels A1&A2 Auto Scram) on ICS computer or on the First Out Printer reads "NOT TRIP" for RPS "A". E. Points SOE034 (Channel B3 manual scram) and SOE036 (Channels B1&B2 Auto Scram) on ICS computer or on the First Out Printer reads "NOT TRIP" for RPS "B".
	NRC	**End of Event 5. Request that the Driver insert Event 6, Condenser Leakage / High Conductivity.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 1 of 8

Event Description: Condenser Leakage / High Conductivity

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. A leak will develop in the Main Condenser, causing conductivity alarms

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause a Main Condenser Tube leak and High Conductivity.**
	BOP	Acknowledges and reports the following alarm: • CONDENSATE DEMIN ABNORMAL, Panel 3-9-6B, Window 6
	NUSO	Acknowledges the report and directs the BOP to respond in accordance with the Alarm Response Procedure and 3-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	3-ARP-9-6B, Alarm Response Procedure CONDENSATE DEMIN ABNORMAL, Window 6 Operator Action: A. **CHECK** system flow conductivity, and system DP, on Panel 3-9-6 for abnormalities. B. **DISPATCH** personnel to determine problem with Condensate Demineralizers. C. **IF** High Conductivity exists, **THEN REFER TO** 3-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	3-AOI-2-1, Reactor Coolant High Conductivity 4.1 Immediate Actions: **None** 4.2 Subsequent Actions **CAUTION** Operations outside of the allowable regions shown on the recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. **REFER TO** 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, for required actions and monitoring to be performed during a power reduction.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 2 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **LOWER** Reactor Power, as necessary, to maintain radiation levels in Main Steam Lines below the "Main Steam Line Radiation High-High" (3-XA-55-3A, Window 27) setpoint. **REFER TO** 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations. [2] **MONITOR** Reactor Vessel Water Conductivity on 3-CR-43-11A/12A, RWCU CONDUCTIVITY, on Panel 3-9-4. [3] **NOTIFY** Chemistry to sample Reactor Water. **REFER TO** 3-SI-4.6.B.1-4, Reactor Coolant Chemistry.
	NRC	**Chemistry sample results will not be provided quickly enough to allow the SRO to address Technical Specifications/Technical Requirements Manual requirements. Therefore, Technical Specifications/Technical Requirements Manual evaluation for this event is not required and should not be used to evaluate the applicant's Technical Specifications/Technical Requirements Manual competency.**
	Driver	**If contacted as Chemistry to sample Reactor Water in accordance with 3-SI-4.6.B.1-4, Reactor Coolant Chemistry, acknowledge the direction. If contacted during this Event about the status of the sample, report that sampling is in progress.**
	BOP	[4] **IF** Condenser Hotwell in-leakage is suspected, **THEN PERFORM** the following: **CAUTION** When isolating a Condenser Waterbox, do NOT allow Condenser Vacuum to lower to less than (-) 25 inches Hg, as indicated on 3-XR-002-0026, CONDENSATE, with the Main Turbine on-line. [4.1] **ISOLATE** and **DRAIN** the Condenser Water Boxes (one at a time). **REFER TO 3**-OI-27, Condenser Circulating Water System.
	NRC	**Candidates may choose to isolate Waterboxes in any order. Therefore, when the first Waterbox is isolated, request that the driver insert Event 15 to delete the Condenser Tube Leak malfunction.**
	Driver	**When requested by the NRC insert Event 16 to delete the Condenser Tube Leak malfunction. Verify that Conductivity begins to lower.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 3 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	Isolates Waterboxes until Conductivity begins to lower. Informs the SRO when the leak has been isolated. [4.1.1] **OBSERVE** Reactor Water Conductivity and Off Gas Flow rate. (After the leaking water box is isolated and drained, Reactor Water Conductivity should begin to decrease within 30-45 minutes. Off-Gas Flow rate could raise.) [4.2] N/A [5] N/A [6] N/A
	BOP	**NOTE** If gross intrusion of resin into the Reactor Vessel, Hotwell or Condensate Storage Tank is suspected, the clean-up cannot be effectively done by normal system operation and flushing alone. The affected system needs to be shut down, opened, and physically cleaned to the extent practicable. Special cleaning methods and/or procedures will be developed for each particular case. [7] **IF** Control Room indications or sample requested from Chemistry in Step 4.2[3] shows Reactor Vessel Water Conductivity rising to 10 micromhos OR chloride concentration exceeds 0.5 ppm during power operation (Mode 1) 0R conductivity rises to 2 micromhos during Reactor Startup (Mode 2), **THEN PERFORM** the following steps: [7.1] **REFER TO** TRM 3.4.1, Coolant Chemistry. [7.2] **INITIATE** an orderly Reactor Shutdown. **REFER TO** 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations.
	BOP	Acknowledges and reports the alarms as received: • CONDENSATE PUMP DISCHARGE CONDUCTIVITY HIGH, 3-9-6B, Window 19 • RWCU FILTER OUTLET CONDUCTIVITY HIGH, 3-9-4B, Window 3

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 4 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	3-ARP-9-6B, Alarm Response Procedure CNDS PUMP DISCHARGE CNDCT HIGH, Window 19 Operator Action: A. **CHECK** 3-CR-43-1A on Panel 3-9-6 for changes in the conductivity reading and identify Condensate and/or Booster Pump Discharge Conductivity above 0.2 µS/cm. B. **NOTIFY** Chemistry to verify Condensate and/or Booster Pump Discharge Conductivity. C. **CHECK** rate at which the conductivity is rising as a possible clue to what is causing the rise. D. **REFER TO** 3-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	3-ARP-9-4B, Alarm Response Procedure RWCU OUTLET CONDUCTIVITY HIGH, Window 3 Operator Action: A. **CONFIRM** alarm: • **CHECK** recorder 3-CR-43-11A/12A, Panel 3-9-4 • **REQUEST** Chem lab to validate alarm B. **IF** conductivity is rising on 3-CR-43-11A/12A, **THEN CHECK** 3-CR-43-1A on Panel 3-9-6. C. **REFER TO** Tech Spec Section 3.4.6, TRM 3.4.1. D. **IF** valid, **THEN REFER TO** 3-AOI-2-1, Reactor Coolant High Conductivity.
	BOP	3-OI-27, Condenser Circulating Water System Section 8.15, Removing Condenser Waterbox From Service For Leak Check **CAUTIONS** 1) Only two Waterboxes should be isolated at a time. 2) Before removing 3B1(SS) or 3B2(NS) Water Boxes from service, HOTWELL LOW LEVEL MAKEUP CONTROL, 3-LC-2-6, should be PLACED in MANUAL to prevent erratic makeup flow control. 3) Do **NOT** Isolate 2 water boxes side by side.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 5 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[1] **IF** time permits, **THEN COLLECT** the Amertap System balls and **SHUTDOWN** the Amertap System from the Waterboxes to be removed. **REFER TO** 3-OI-27B, Amertap Condenser Tube Cleaning System.
	Driver	**If the crew contacts the Turbine Building AUO or Work Control / Outside SRO to perform 3-OI-27B, Amertap Condenser Tube Cleaning System, in preparation for isolating any Waterbox:** • **Acknowledge the direction to collect Amertap Balls and shutdown the Amertap System** • **State that time compression is being applied and that the Amertap Balls have been collected and the Amertap system is shut down.** **If contacted as the Turbine Building AUO to shut down the debris filter in accordance with 3-OI-27, Condenser Circulating Water System, Section 7.2, Shutdown Of The Debris Filter, acknowledge the direction.**
	BOP	[2] N/A [3] **CLOSE** the CNDR 3A1(3A2)(3B1)(3B2)(3C1)(3C2), OFFGAS SHUTOFF VALVE, to the waterbox to be removed from service using 3-HS-66-2A(3A)(4A)(5A)(6A)(7A) on Panel 3-9-8. **NOTES** 1) CONDENSER 3A, 3B, 3C CCW OUTLET SHUTOFF VALVE travel can be stopped by depressing the associated STOP pushbutton on Panel 3-25-116. 2) Outlet valve may need to be opened to control H2O level rise in OOS water box. 3) Condenser Vacuum shall be maintained greater than 25 inches Hg at all times.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 6 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[4] **WHILE** closely monitoring Condenser Vacuum, **CLOSE** the appropriate CNDR 3A(3B)(3C) CCW OUTLET SHUTOFF VALVES using the OPEN/CLOSE/STOP push-buttons for the associated handswitches listed below on Panel 3-25-116. (N/A valves not corresponding to the desired waterbox being removed.) • 3-HS-027-0046D, CONDENSER 3A SS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0038D, CONDENSER 3A NS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0062D, CONDENSER 3B SS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0054D, CONDENSER 3B NS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0078D, CONDENSER 3C SS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0070D, CONDENSER 3C NS CCW OUTLET SHUTOFF VALVE
	Driver	**If directed as the Turbine Building AUO to close CCW Outlet Shutoff Valves at Panel 3-25-116, enter event(s) as follows:** **Event 17 - 3-HS-027-0046, 3A SS Outlet** **Event 18 - 3-HS-027-0038, 3A NS Outlet** **Event 19 - 3-HS-027-0062, 3B SS Outlet** **Event 20 - 3-HS-027-0054, 3B NS Outlet** **Event 21 - 3-HS-027-0078, 3C SS Outlet** **Event 22 - 3-HS-027-0070, 3C NS Outlet** **Inform the crew when the chosen valve has been closed.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 7 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[4.1] As necessary to maintain Condenser Vacuum greater than 25 inches Hg, **SLOWLY THROTTLE OPEN** the remaining CONDENSER 3A(3B)(3C) CCW OUTLET SHUTOFF VALVES by using the following OPEN/CLOSE/STOP pushbutton on Panel 25-116: • 3-HS-027-0046D, CONDENSER 3A SS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0038D, CONDENSER 3A NS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0062D, CONDENSER 3B SS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0054D, CONDENSER 3B NS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0078D, CONDENSER 3C SS CCW OUTLET SHUTOFF VALVE • 3-HS-027-0070D, CONDENSER 3C NS CCW OUTLET SHUTOFF VALVE
	NUSO	[4.2] **IF** necessary, **LOWER** load to **MAINTAIN** Condenser Vacuum greater than 25 inches Hg. (Otherwise N/A)
	Driver	**If contacted as an AUO, Work Control /Outside NUSO to throttle open Condenser Shutoff Valves, acknowledge the direction and report that the requested valve has been throttled.**
	NRC	**Inform the Driver when the Inlet Shutoff Valve for the chosen Waterbox is closed and request that the Driver insert Event 16 to delete the leaking tube malfunction.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 6 Page 8 of 8

Event Description: Condenser Leakage / High Conductivity

Time	Position	Applicant's Actions or Behavior
	BOP	[5] **CLOSE** the CONDENSER 3A(3B)(3C) CCW INLET SHUTOFF VALVE for the waterbox to be removed from service using the appropriate switch on Panel 3-9-20: • CONDENSER 3A SS CCW INLET SHUTOFF VALVE, 3-HS-27-39A • CONDENSER 3A NS CCW INLET SHUTOFF VALVE, 3-HS-27-31A • CONDENSER 3B SS CCW INLET SHUTOFF VALVE, 3-HS-27-55A • CONDENSER 3B NS CCW INLET SHUTOFF VALVE, 3-HS-27-47A • CONDENSER 3C SS CCW INLET SHUTOFF VALVE, 3-HS-27-71A • CONDENSER 3C NS CCW INLET SHUTOFF VALVE, 3-HS-27-63A [6] N/A [7] **ADJUST** the remaining CONDENSER CCW OUTLET SHUTOFF VALVES to optimize CCW System performance by performing Section 6.1. [8] **IF** required, **THEN START** Amertap System on the unaffected Condenser Waterboxes when CCW System pressures and temperatures have stabilized. **REFER TO** 3-OI-27B, Amertap Condenser Tube Cleaning System. [9] N/A
	Driver	**If contacted as an AUO or the Work Control / Outside NUSO to start the Amertap System on any Waterbox that is un-isolated, acknowledge the direction and report that the Amertap System has been started.**
	NRC	**End of Event 6. Request that the Driver insert Event 7, High Suppression Pool Water Level / Emergency Depressurization.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 1 of 9

Event Description: High Suppression Pool Water Level

Symptoms/Cues: Event is initiated by the Simulator driver when requested by the Chief Examiner. A leak into the Suppression Pool will cause Water Level to rise

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7, to cause Suppression Pool Water Level to rise.**
	NRC	**Event 8, Electrical Anticipated Transient Without SCRAM, and Event 9, Steam Relief Valve Fails to Open, are automatically entered upon Simulator setup. No action is required to enter these events. See page 37 of 40 for Event 8, and page 38 of 40 for Event 9.**
	BOP	Acknowledges and reports the following alarms as they are received: • DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, 3-9-3B, Window 26 • SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, 3-9-3B, Window 15
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedures.
	BOP	3-ARP-9-3B, Alarm Response Procedure DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, Window 26 Operator Action: A. **CHECK** alarm by checking Drywell to Suppression Chamber DP. B. **REFER TO** 3-OI-64, Primary Containment System. C. **REFER** TO Tech Spec Section 3.6.2.6, Drywell-to-Suppression Chamber Differential Pressure.
	NRC	**Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant's Tech Spec competency.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 2 of 9

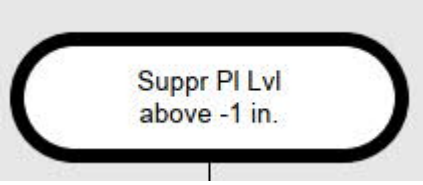
Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	BOP	Alarm Response Procedure, 3-ARP-9-3B SUPPRESSION CHAMBER WATER LEVEL ABNORMAL, Window 15 Operator Action: A. **CHECK** level using multiple indications. B. N/A C. **IF** level is high, **THEN CHECK** for RCIC, HPCI, Core Spray, or RHR draining to Suppression Pool, and **CHECK** 3-TR-64-161 and -162. D. **REFER TO** 3-OI-74, Residual Heat Removal System, Sections 8.2, 8.3, and 8.4. E. **REFER TO** Tech Spec Section 3.6.2.2. F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN ENTER** 3-EOI-2 , Primary Containment Control Flowchart. G. N/A
	BOP	Determines that the cause for the Drywell to Suppression Chamber DP alarm is rising Suppression Pool Water Level and informs the NUSO.
	NUSO	Directs the BOP to monitor Suppression Pool Water Level and to provide an update when level reaches (-) 1 inch.
	BOP	Acknowledges and reports the following alarm to the NUSO when received: • SUPPRESSION POOL LEVEL HIGH, 3-9-3F, Window 12
	NUSO	Directs the BOP to respond in accordance with the appropriate Alarm Response Procedure.
	BOP	3-ARP-93F, Alarm Response Procedure SUPPRESSION POOL LEVEL HIGH, Window 12 A. **CHECK** CST 3 and Suppression Pool level using multiple indications. B. **ENSURE** HPCI Suction automatically transfers to the Suppression Pool. C. N/A D. **REFER TO** Tech Spec 3.5.1, 3.5.2 and 3.6.2.2.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 3 of 9

Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NRC	**Due to time constraints, Tech Spec evaluation for this event is not required and should not be used to evaluate the applicant's Tech Spec competency.**
	NUSO	When appropriate, enters 3-EOI-2, Primary Containment Control on high Suppression Pool Water Level (level above (-) 1 inch).
	NUSO	3-EOI-2, Primary Containment Control 
	NUSO	SP/L-1 **MONITOR** and **CONTROL** Suppression Pool Water Level (-) 6 inches to (-) 1 inch. **IF** / **THEN** Suppression Pool Water Level CANNOT be maintained below (-) 1 inch. → A Suppression Pool Water Level CANNOT be maintained above (-) 6 inches. → **NO ACTION REQUIRED**
	NUSO	A  SP/L-3 **MAINTAIN** Suppression Pool Water Level below 19 ft. (APPX 18, 20K)

Form 3.3-2 Required Operator Actions

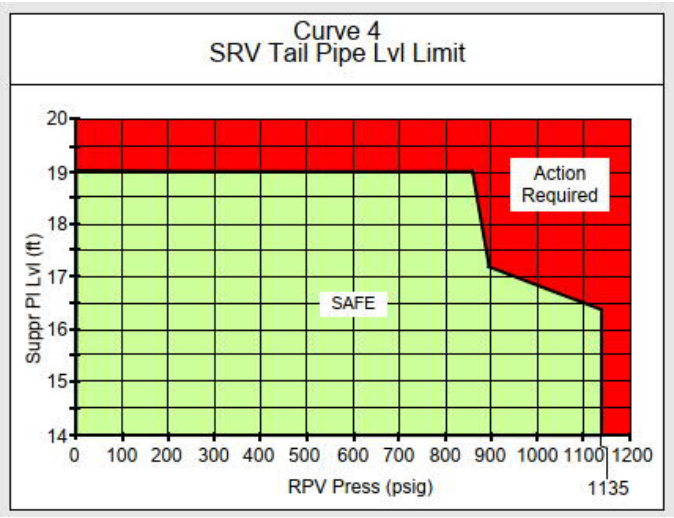
Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 4 of 9

Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	SP/L-4 **WHEN** Suppression Pool Level CANNOT be maintained below (APPX 9) 19 feet SP/L-5 **STOP** DW Sprays
	NUSO	Directs the BOP to control Suppression Pool Water Level in accordance with 3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup.
	Driver	**If contacted as an AUO to perform any steps locally per 3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup, acknowledge any direction given.** **If contacted as an AUO to verify any valves closed, acknowledge any information or direction given.**
	BOP	3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup [1] N/A [2] N/A [3] **IF** directed by the NUSO, **THEN REMOVE** water from the Suppression Pool as follows: [3.1] **DISPATCH** personnel to perform the following (Unit 3 RB, El 519 ft, Torus Area): [3.1.1] **ENSURE OPEN** 3-SHV-074-0786A (B), RHR DRAIN PUMP 3A(3B) DISCHARGE SHUTOFF VALVE. [3.1.2] **OPEN** the following valves: • 3-SHV-074-0564A(B), RHR DRAIN PUMP A(B) SEAL WATER SUPPLY • 3-SHV-074-0529A (B), RHR DRAIN PUMP A (B) SHUTOFF VALVE [3.1.3] **UNLOCK** and **OPEN** 3-SHV-074-0765A (B), RHR DRAIN PUMP A(B) DISCHARGE.

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 5 of 9

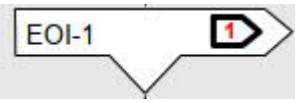
Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	BOP	[3.1.4] **NOTIFY** Unit Operator that RHR Drain Pump 3A (3B) is lined up to remove water from Suppression Pool. [3.1.5] **REMAIN** at torus area UNTIL Unit 3 Operator directs starting of RHR Drain Pump 3A (3B).
	Driver	**After 2 minutes, report that the outside portions of 3-EOI-Appendix-18, Suppression Pool Water Inventory Removal and Makeup are complete. If directed to start the RHR Drain Pump, report that the RHR Drain Pump has been started.** **If contacted as the Rad Waste Operator, acknowledge any reports or direction given.**
	NUSO	 SP/L-9 **MAINTAIN** Suppression Pool Level within the safe area of Curve 4 (APPX 18, 20K) 

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 6 of 9

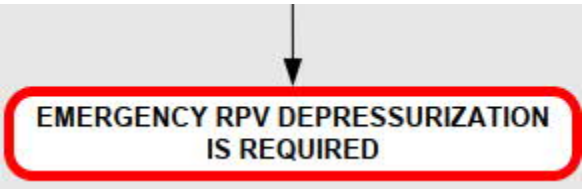
Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	SP/L-10 **WHEN** Suppression Pool Level CANNOT be maintained within the safe area of Curve 4 (APPX 9) 
	CREW	**Critical Task:** **When Suppression Pool Water Level cannot be maintained within the safe area of Curve 4, SRV Tail Pipe Level Limit, the crew will insert a manual Reactor SCRAM as directed by the Nuclear Unit Senior Operator (NUSO).** **Critical Task Failure Criteria:** **The operating crew fails to proceed without delay and in a controlled manner to insert a manual SCRAM before Suppression Pool Water Level enters the 'Action Required' area of Curve 4, SRV Tail Pipe Level Limit.**
	NUSO	Directs the OATC to insert a manual Reactor SCRAM and directs the crew to enter 3-AOI-100-1, Reactor SCRAM.
	NRC	**Only the Immediate Actions for the Reactor SCRAM are listed below due to the large number of subsequent actions in the Abnormal Operating Instruction.**
	OATC	3-AOI-100-1, Reactor SCRAM Immediate Actions: [1] **DEPRESS** REACTOR SCRAM A and B, 3-HS-99-5A/S3A and 3-HS-99-5A/S3B, on Panel 3-9-5. [2] **PLACE** REACTOR MODE SWITCH, 3-HS-99-5A/S1, in SHUTDOWN.
	NRC	**When the OATC attempts to insert a manual Reactor SCRAM, Event 8, Electrical Anticipated Transient Without Scram (ATWS), will occur. No action is required by the Driver to insert Event 8. See page 37 of 40 for Event 8 actions.**
	NUSO	Enters 3-EOI-1, ATWS RPV Control, and then exits to 3-EOI-1, RPV Control MODES 1-3 when Control Rods are inserted by ARI.

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 7 of 9

Event Description: High Suppression Pool Water Level

Time	Position	Applicant's Actions or Behavior
	NUSO	When all Control Rods are inserted, directs the crew to perform the following: • Maintain Reactor Pressure to ensure that Suppression Pool Level is maintained within the safe are of Curve 4 in accordance with 3-EOI-Appendix-8B, Reopening MSIVs/Bypass Valve Operation Maintain Reactor Water Level in accordance with 3-EOi-Appendix5a, Injection System Lineup Condensate/Feedwater
	NUSO	(Continuing actions of 3-EOI-2, Primary Containment Control) SP/L-11 **WHEN** Suppression Pool Level and RPV Pressure CANNOT be maintained within the safe area of Curve 4 (APPX 9). SP/L-12 **STOP** injection into RPV from sources external to Primary Containment **EXCEPT** from systems required to assure Adequate Core Cooling or shut down the Reactor SP/L-13 **WHEN** Suppression Pool Level and RPV Pressure CANNOT be restored and maintained within the safe area of Curve 4 

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 8 of 9

Event Description: High Suppression Pool Water Level

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td>CREW</td><td>Critical Task:
When Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4, the NUSO determines that Emergency Depressurization is required. The Balance of Plant Operator initiates Emergency Depressurization as directed by the Nuclear Unit Senior Operator (NUSO).

Critical Task Failure Criteria:
The operating crew fails to proceed without delay and in a controlled manner to initiate Emergency Depressurization from the time that Suppression Pool Water Level and RPV Pressure cannot be restored and maintained within the safe area of Curve 4.</td></tr>
<tr><td></td><td>NUSO</td><td>3-EOI-1, RPV Control MODES 1-3

Emergency RPV Depressurization

RC/P-6
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV Water Level CANNOT be determined</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that RPV depressurization will result in loss of injection required for Adequate Core Cooling</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is anticipated that Containment Water Level will rise above 44 feet</td><td>NO ACTION REQUIRED</td></tr>
</table></td></tr>
</table>

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 7 Page 9 of 9

Event Description: High Suppression Pool Water Level

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
RC/P-7
<table>
<tr><td>IF</td><td>Drywell Pressure is above 2.45 psig</td></tr>
<tr><td>THEN</td><td>PREVENT injection from ONLY those Core Spray and LPCI pumps NOT required to assure Adequate Core Cooling (APPX 4)</td></tr>
</table>
RC/P-8
<table>
<tr><td colspan="2">EMERGENCY DEPRESSURIZE the RPV
IF Suppression Pool Water Level is above 5.5 feet
THEN OPEN 6 MSRVs (ADS Valves preferred)
• OK to exceed 100 F/hr Cooldown Rate</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Drywell Control Air becomes unavailable</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Less than 4 MSRVs can be opened
AND
RPV Pressure is 80 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table>
</td></tr>
<tr><td></td><td>NRC</td><td>See page 38 of 40 for Event 9, Steam Relief Valve Fails to Open actions.</td></tr>
<tr><td></td><td>NRC</td><td>End of Event 7. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 8 Page 1 of 1

Event Description: Electrical Anticipated Transient Without Scram (ATWS)

Symptoms/Cues: Event is automatically initiated when the crew inserts a manual Reactor SCRAM; Rods will remain out when a Reactor SCRAM is inserted

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 8, Electrical Anticipated Transient Without Scram (ATWS), is automatically entered on simulator setup. No action is required to insert this event.**
	OATC	3-AOI-100-1, Reactor SCRAM [3] **IF** all Control Rods can **NOT** be verified fully inserted, **THEN INITIATE** Alternate Rod Insertion (ARI).
	Driver	**Verify that the ATWS malfunction clears, and Control Rods insert when the OATC initiates ARI.**
	OATC	After initiating ARI, verifies that Control Rods insert and informs the NUSO. Continues with Reactor SCRAM actions. [4] **IF** Reactor Power is 5% or BELOW, **THEN**: (otherwise MARK N/A) **REPORT** the following to the NUSO: • Reactor SCRAM • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering) • Reactor Pressure and trend • MSIV position (Open or Closed) • Reactor Power Level Performs Subsequent Actions in accordance with 3-AOI-100-1, Reactor SCRAM, Section 4.2.
	NRC	**End of Event 8.**

Form 3.3-2 Required Operator Actions

Op Test No.: 23-04 Scenario No. NRC-6 Event No.: 9 Page 1 of 1

Event Description: Steam Relief Valve (SRV) Fails to Open

Symptoms/Cues: Event is automatically inserted on Simulator setup. One Steam Relief Valve will fail to open during Emergency Depressurization

Time	Position	Applicant's Actions or Behavior
	NRC	**Event 9 is automatically inserted on Simulator setup. No action is required by the Driver to insert Event 9. Steam Relief Valve 1-22 will not open during Emergency Depressurization.**
	BOP	Emergency Depressurizes the Reactor by opening six (6) SRVs, ADS Valves are preferred. Determines that SRV PCV-1-22 did not open, and opens one non-ADS Valve. Informs the NUSO that 2 ADS Valves failed to open, but there are six (6) SRVs open for Emergency Depressurization.
	NRC	**End of Event 9, end of Scenario. When the crew has inserted all Control Rods, Emergency Depressurized the Reactor, and has control of Reactor Water level above the Top of Active Fuel ((-) 162 inches) using low pressure systems, end of Scenario.**

Form 3.3-2 Required Operator Actions

Scenario Setup
UNIT 3

IC	28
Exam IC	274

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-3	115	ARP-6C	30		
OI-27	85	ARP-7A	30		
OI-66	87	AOI-2-1	14		
OI-74	142	AOI-64C	6		
OI-99	68	AOI-100-1	80		
ARP-3B	25	EOI-1	16		
ARP-3F	38	EOI-2	15		
ARP-4A	51	APPX-18	3		
ARP-4B	54	T.S. 3.3.1.1	221		
ARP-5B	33	T.S. 3.5.3	294		
ARP-6B	21	T.S. 3.6.1.3	294		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN Hang Danger Tag on 3A RBCCW Pump Hang Protected Equipment Tag on 3B RBCCW Pump
Schedule Files(s):	ILT 2304 NRC Scenario 6 UNIT 3.sch
Event Files(s):	ILT 2304 NRC Scenario 6 UNIT 3.evt

UNIT 3 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ON LINE	Drywell Leakage (GPM)	Protected Equipment
	230	1.89	3B RBCCW Pump
Rx Power 100%	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	Floor Drain (GPM) 0.31	
MWe 1307		Equipment Drain (GPM) 1.58	

☐Review logs ☐Qualifications ☐Review RCP/Rx Brief ☐Review LCO/OWA Actions ☐Walkdown Panels/Verify EOOS
☐CR Reviews Complete ☐Leadership and Team Effectiveness

CHANGES IN LCOs

LCOs OF 72 HOURS OR LESS

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

HPCI PMT run completed.
3A RBCCW Pump tagged for bearing replacement.
Spare RBCCW Pump aligned to Unit 3.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Return RHR Loop I to standby readiness.
Support maintenance on 3A RBCCW Pump.

OPERATOR WORK AROUNDS **OWAs – U0-1/U3-0** **Burdens - U0 – 0/U3-0** **Operator Challenges – U0 – 12/U3 - 6**

ODMIs/ACMPs

ONEAs

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

SCHEDULED ITEMS NOT COMPLETED

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Perform a CRD Low Drive Water DP Flush Per 2-OI-85, Control Rod Drive System
JPM NUMBER:	726-U2	REVISION:	1

TASK APPLICABILITY:	☒SRO-U	☒SRO-1	☒UO
TASK NUMBER / TASK TITLE(S):	U-085-NO-37 / Perform a CRD Flush to Un-Block Restricted Cooling Water Flow		
K/A RATINGS:	RO / SRO 3.7		
K/A No. &STATEMENT:	201001 Control Rod Drive Hydraulic System A1.01: Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: CRD drive water header pressure		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	1		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ _____
Validator *Date*

Approved by: _____ _____
Facility Representative *Date*

NRC Approved: _____ _____
Facility Author *Date*

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 726-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate will perform a flush of a CRD in accordance with 2-OI-85, Control Rod Drive System, Section 8.15, CRDM Low Drive Water DP Flush.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-OI-85

VALIDATION TIME: 5 min

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	08/12/19	All	Initial issue
1	02/08/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-OI-85	156

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to IC 28** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 2-OI-85, Control Rod Drive System, Section 8.15, CRDM Low Drive Water DP Flush**

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

Schedule File(s): None
Event File(s): None

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Operator at the Controls on Unit 2.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to perform a CRDM Low Drive Water DP Flush on Control Rod 34-27 for at least 10 seconds in accordance with 2-OI-85, Control Rod Drive System, Section 8.15.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 2-OI-85, Control Rod Drive System, Section 8.15, CRDM Low Drive Water DP Flush. 2-OI-85, Control Rod Drive System Section 8.15, CRDM Low Drive Water Flush Step 1: [1] **REVIEW** all Precautions and Limitations in Section 3.6 and 3.7. Expected Action(s): Completed during pre-brief, but the candidate may review Precautions and Limitations.	_____ SAT _____UNSAT _____N/A
NOTE The following steps will flush the CRD without the possibility of moving a Control Rod. Step 2: [2] **LOWER** the 2-PDI-85-17A, CRD DRIVE WATER HEADER DP, to less than 60 psid, using 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE. Expected Action(s): Candidate lowers Control Rod Drive Water DP to less than 60 psid using 2-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 3: [3] **PLACE** 2-HS-85-46, CRD POWER, in ON. Expected Action(s): Candidate verifies that 2-HS-85-46, CRD POWER, is ON.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **SELECT** the desired Control Rod by depressing the appropriate 2-XS-85-40, CRD ROD SELECT pushbutton. Expected Action(s): Candidate selects Control Rod 34-27.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [5] **PLACE** 2-HS-85-48, CRD CONTROL SWITCH, in ROD IN and **HOLD** to flush the Control Rod. Expected Action(s): Candidate places and holds 2-HS-85-48, CRD CONTROL SWITCH, in ROD IN to flush the Control Rod.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 6: [5.1] **IF** the Control Rod moves, **THEN RELEASE** 2-HS-85-48, CRD CONTROL SWITCH, to STOP rod motion. (Otherwise N/A) [5.1.1] **NOTIFY** Unit SRO and Reactor Engineer for further instructions. Expected Action(s): Candidate verifies there is no Control Rod motion and marks this step as N/A.	_____ SAT _____UNSAT _____N/A
Step 7: [6] **CHECK** 2-FI-85-15A, CRD DRIVE WATER HEADER FLOW, has stabilized. (Control Rod should **NOT** insert) Expected Action(s): Candidate checks that 2-FI-85-15A, CRD DRIVE WTR HDR FLOW, has stabilized. Verifies that Control Rod 34-27 is not inserting.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 8: [7] **RELEASE** 2-HS-85-48, CRD CONTROL SWITCH. Expected Action(s): Candidate releases 2-HS-85-48, CRD CONTROL SWITCH.	_____ SAT _____UNSAT _____N/A
Step 9: [8] **PLACE AND HOLD** 2-HS-85-47, CRD NOTCH OVERRIDE, in NOTCH OVERRRIDE. Expected Action(s): Candidate places and holds 2-HS-85-47, CRD NOTCH OVERRIDE, in NOTCH OVERRIDE.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 10: [9] **PLACE AND HOLD** 2-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH. Expected Action(s): Candidate places and holds 2-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 11: [9.1] **IF** Control Rod moves, **THEN RELEASE** 2-HS-85-47, CRD NOTCH OVERRIDE and 2-HS-85-48, CRD CONTROL SWITCH to STOP rod motion. (Otherwise N/A) [9.1.1] **NOTIFY** Unit SRO and Reactor Engineer for further instructions. Expected Action(s): Candidate verifies there is no Control Rod motion and marks this step as N/A.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 12: [10] **CHECK** 2-FI-85-15A, CRD DRIVE WATER HEADER FLOW, has stabilized. (Control Rod should **NOT** withdraw) Expected Action(s): Candidate checks 2-FI-85-15A, CRD DRIVE WATER HEADER FLOW, has stabilized. Verifies that Control Rod 34-27 is not withdrawing.	_____ SAT _____UNSAT _____N/A
Step 13: [11] **RELEASE** both 2-HS-85-47, CRD NOTCH OVERRIDE, and 2-HS-85-48, CRD CONTROL SWITCH. Expected Action(s): Candidate releases 2-HS-85-47, CRD NOTCH OVERRIDE, and 2-HS-85-48, CRD CONTROL SWITCH.	_____ SAT _____UNSAT _____N/A
CUE: Once JPM Step 13 has been completed, inform the candidate "Another Operator will finish this procedure. This completes your task".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Operator at the Controls on Unit 2.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to perform a CRDM Low Drive Water DP Flush on Control Rod 34-27 for at least 10 seconds in accordance with 2-OI-85, Control Rod Drive System, Section 8.15.

Job Performance Measure (JPM)

SITE: BFN		JPM TITLE:	Perform a CRD Low Drive Water DP Flush Per 3-OI-85, Control Rod Drive System
JPM NUMBER:	726-U3	REVISION:	2

TASK APPLICABILITY:	☒SRO-U	☒SRO-1	☒UO
TASK NUMBER / TASK TITLE(S):	U-085-NO-37 / Perform a CRD Flush to Un-Block Restricted Cooling Water Flow		
K/A RATINGS:	RO / SRO 3.7		
K/A No. &STATEMENT:	201001 Control Rod Drive Hydraulic System A1.01: Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: CRD drive water header pressure		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	1		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ _____
Validator *Date*

Approved by: _____ _____
Facility Representative *Date*

NRC Approved: _____ _____
Facility Author *Date*

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 726-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate will perform a flush of a CRD in accordance with 3-OI-85, Control Rod Drive System, Section 8.15, CRDM Low Drive Water DP Flush.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 3-OI-85

VALIDATION TIME: 5 min

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	08/12/19	All	Initial issue
1	9/18/19	6	Updated due to 3-OI-85 revision 87, 8.15[2] changed from 75 psid to 60 psid.
2	02/08/23	All	JPM update

Procedure Revisions

Procedure	Revision
3-OI-85	99

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to IC 28** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 3-OI-85, Control Rod Drive System, Section 8.15, CRDM Low Drive Water DP Flush**

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

Schedule File(s): None
Event File(s): None

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Operator at the Controls on Unit 3.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to perform a CRDM Low Drive Water DP Flush on Control Rod 22-15 for at least 10 seconds in accordance with 3-OI-85, Control Rod Drive System, Section 8.15.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 3-OI-85, Control Rod Drive System, Section 8.15, CRDM Low Drive Water DP Flush. 3-OI-85, Control Rod Drive System Section 8.15, CRDM Low Drive Water Flush Step 1: [1] **REVIEW** all Precautions and Limitations in Section 3.6 and 3.7. Expected Action(s): Completed during pre-brief, but the Candidate may review Precautions and Limitations.	_____ SAT _____UNSAT _____N/A
NOTE The following steps will flush the CRD without the possibility of moving a Control Rod. Step 2: [2] **LOWER** 3-PDI-85-17A, CRD DRIVE WATER HEADER DP, to less than 60 PSID, using 3-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE. Expected Action(s): Candidate lowers Control Rod Drive Water DP to less than 60 PSID using 3-HS-85-23A, CRD DRIVE WATER PRESS CONTROL VALVE.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [3] **PLACE** 3-HS-85-46, CRD POWER, in ON. Expected Action(s): Candidate verifies that 3-HS-85-46, CRD POWER, is ON.	_____ SAT _____UNSAT _____N/A
Step 4: [4] **SELECT** the desired Control Rod by depressing the appropriate 3-XS-85-40, CRD ROD SELECT pushbutton. Expected Action(s): Candidate selects Control Rod 22-15.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [5] **PLACE** 3-HS-85-48, CRD CONTROL SWITCH, in ROD IN and **HOLD** to flush the Control Rod. Expected Action(s): Candidate places and holds 3-HS-85-48, CRD CONTROL SWITCH, in ROD IN to flush the Control Rod.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 6: [5.1] **IF** the Control Rod moves, **THEN RELEASE** 3-HS-85-48, CRD CONTROL SWITCH, to STOP rod motion. (Otherwise N/A) [5.1.1] **NOTIFY** Unit SRO and Reactor Engineer for further instructions. Expected Action(s): Candidate verifies there is no Control Rod motion and marks this step as N/A.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [6] **CHECK** 3-FI-85-15A, CRD DRIVE WATER HEADER FLOW, has stabilized. (Control Rod should **NOT** insert) Expected Action(s): Candidate checks that 3-FI-85-15A, CRD DRIVE WATER HEADER FLOW, has stabilized. Verifies that Control Rod 22-15 is not inserting.	_____ SAT _____UNSAT _____N/A
Step 8: [7] **RELEASE** 3-HS-85-48, CRD CONTROL SWITCH. Expected Action(s): Candidate releases 3-HS-85-48, CRD CONTROL SWITCH.	_____ SAT _____UNSAT _____N/A
Step 9: [8] **PLACE AND HOLD** 3-HS-85-47, CRD NOTCH OVERRIDE, in NOTCH OVERRIDE. Expected Action(s): Candidate places and holds 3-HS-85-47, CRD NOTCH OVERRIDE, in NOTCH OVERRRIDE.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 10: [9] **PLACE AND HOLD** 3-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH. Expected Action(s): Candidate places and holds 3-HS-85-48, CRD CONTROL SWITCH, in ROD OUT NOTCH.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 11: [9.1] **IF** Control Rod moves, **THEN RELEASE** 3-HS-85-47 CRD NOTCH OVERRIDE and 3-HS-85-48, CRD CONTROL SWITCH 3-HS-85-48 to STOP rod motion. (Otherwise N/A) [9.1.1] **NOTIFY** Unit SRO and Reactor Engineer for further instructions. Expected Action(s): Candidate verifies there is no Control Rod motion and marks this step as N/A.	_____ SAT _____UNSAT _____N/A
Step 12: [10] **CHECK** 3-FI-85-15A, CRD DRIVE WATER HEADER FLOW, has stabilized. (Control Rod should **NOT** withdraw) Expected Action(s): Candidate checks 3-FI-85-15A, CRD DRIVE WATER HEADER FLOW, has stabilized. Verifies that Control Rod 22-15 is not withdrawing.	_____ SAT _____UNSAT _____N/A
Step 13: [11] **RELEASE** both 3-HS-85-47, CRD NOTCH OVERRIDE, and 3-HS-85-48, CRD CONTROL SWITCH. Expected Action(s): Candidate releases 3-HS-85-47, CRD NOTCH OVERRIDE, and 3-HS-85-48, CRD CONTROL SWITCH.	_____ SAT _____UNSAT _____N/A
CUE: Once JPM Step 13 has been completed, inform the candidate "Another Operator will finish this procedure. This completes your task".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Operator at the Controls on Unit 3.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to perform a CRDM Low Drive Water DP Flush on Control Rod 22-15 for at least 10 seconds in accordance with 3-OI-85, Control Rod Drive System, Section 8.15.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Remove 2A RFPT From Service
JPM NUMBER:	274A-U2	REVISION:	2

TASK APPLICABILITY:	☐ SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-003-NO-07 / Shutdown the Reactor Feedwater System		
K/A RATINGS:	RO/SRO: 3.8		
K/A No. &STATEMENT:	259001 SF2 Reactor Feedwater System A3.03; System flow: Ability to monitor automatic operation of the Feedwater System, including: System flow (3.8)		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	2		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

Validated by: _____ _____
Validator *Date*

Approved by: _____ _____
Facility Representative *Date*

NRC Approved: _____ _____
Facility Author *Date*

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 274A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to remove 2A Reactor Feedwater Pump from service in accordance with 2-OI-3, Reactor Feedwater System and respond to a malfunction of the Discharge Check Valve.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 A Solid Understanding of Plant Design, Engineering, and Sciences.

REFERENCES/PROCEDURES NEEDED: 2-OI-3

VALIDATION TIME: 13 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	12/18/19	All	Initial issue
1	1/14/19	All	JPM update; Format update
2	2/22/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-OI-3	163

SIMULATOR SETUP

IC	28
Exam IC	280

Console Operator Instructions	• **Reset to IC 280** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 2-OI-3, Reactor Feedwater System Section 7.1, RFPT/RFPT Shutdown** • **Run schedule file 2304 NRC JPM B.sch** • **Verify event file 2304 NRC JPM B.sch**

Malfunctions	Description	Event	Severity	Delay	Initial set
FW08A	**RFP 3A DISCHARGE CHECK VALVE STICKS**	**N/A**	**87%**	**N/A**	**N/A**

Overrides	Description	Event	Severity	Delay	Initial set
N/A					

Schedule File(s): 2304 NRC JPM B.SCH
Event File(s): 2304 NRC JPM B.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2. A significant oil leak has developed on 2A Reactor Feedwater Pump (RFPT), requiring 2A RFPT to be shut down for emergent repair. Reactor Power has been reduced to 94% in preparation for removing 2A RFPT from service.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to remove 2A RFPT from service in accordance with 2-OI-3, Reactor Feedwater System, Section 7.1, RFPT/RFPT Shutdown.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 2-OI-3, Reactor Feedwater System Section 7.1, RFPT/RFPT Shutdown.	
Step 1: **7.1 RFPT/RFPT Shutdown** **NOTES** 1) Reactor power should be verified less than or equal to 95% prior to starting or stopping a Condensate Booster Pump or Reactor Feedwater Pump. REFER TO P&L 3.0VV if performing 0-TI-704. 2) There is NOT adequate flow/NPSH to maintain 100% CLTP when one Condensate Pump is out of service. To maintain adequate NPSH available to the Condensate Booster Pumps, Condensate flow is limited to 14.1 M lbm/hr. 3) When operating with one Condensate and/or Condensate Booster Pump out of service, verify that condensate system flow measured on CONDENSATE, 2-XR-002-0026 (Point 4), is less than 12 mlbm/hr prior to removing a Reactor Feedwater Pump from service. 4) If Feedwater control becomes unstable, it may be necessary to switch to SINGLE ELEMENT mode from the THREE ELEMENT mode earlier than recommended in this instruction. **CAUTIONS** 1) FAILURE to monitor SJAE/OG CNDR CNDS FLOW, 2-FI-2-42, on Panel 2-9-6 for proper flow (between 2 x 10^6 and 3 x 10^6 lbm/hr) may only result in SJAE poor performance. The SJAE's will NOT trip on Condensate System low pressure. 2) Changes in Condensate System flow may require adjustment to SPE CNDS BYPASS, 2-FCV-002-0190. 3) When isolating the Reactor Feedwater Pump(s) for maintenance, the associated injection water should also be isolated to prevent high seal differential pressure and allow the RFW Pump shafts to rotate freely. (BFNPER123395) 4) When a Reactor Feed Pump is isolated (suction, discharge, and minimum flow valve closed) with injection water aligned to the pump, there is a potential of rising pump casing pressure and seal water leakoff flows reaching the point where seal water drains are overcome and seal water is forced into the oil system through the bearing housings. Therefore, the time that a RFP is isolated with injection water aligned to the pump should be minimized. [1] **REFER TO** Section 3.0 and **REVIEW** Precautions and Limitations. Expected Action(s): This step should have been completed during the pre-brief, but the candidate may review the Precautions and Limitations.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] **IF** Two Condensate Pumps are in service, **THEN ENSURE** Reactor Power is ≤ 85%. (**REFER TO** P&L 3.0WW). Expected Action(s): Candidate determines that 3 Condensate Pumps are in operation, and marks this step as N/A.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **IF** any Condensate or Condensate Booster Pump is NOT in service, **THEN** (Otherwise N/A). **ENSURE** that Condensate System Flow measured on CONDENSATE, 2-XR-002-0026 (Point 4), is less than 12 Mlbm/hr (approximately 85% power) prior to removing a Reactor Feedwater Pump from service. Expected Action(s): Candidate determines that all Condensate and Condensate Booster Pumps are in service and marks this step as N/A.	_____ SAT _____UNSAT _____N/A
Step 4: [4] **IF** RFPT being removed from service is last operating RFPT, **THEN DEPRESS** 2-HS-46-6/1, SINGLE ELEMENT Pushbutton. (Otherwise N/A) Expected Action(s): Candidate determines that 2A RFPT is not the last operating RFPT and marks this step as N/A.	_____ SAT _____UNSAT _____N/A
Step 5: [5] **ENSURE** in AUTO, RFPT Turning Gear Motor. • 2-HS-3-101A, RFPT 2A TURNING GEAR MOTOR Expected Action(s): Candidate ensures 2-HS-3-101A, RFPT 2A TURNING GEAR MOTOR is in AUTO.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 6: **NOTES** 1) Column 1 (PV) on individual RFPT Speed Control Panel Display Stations (PDS) displays actual pump speed and is not controlled in any mode. 2) When selected, Column 2 (SP) on individual RFPT Speed Control PDS displays pump flow bias and is changed with the Ramp Raise/Ramp Lower pushbuttons with the controller in AUTO (blue light illuminated). 3) When selected, Column 3 (CO) on individual RFPT Speed Control PDS displays RFPT speed demand and is changed with the Ramp Raise/Ramp Lower pushbuttons with the controller in MANUAL (amber light illuminated). 4) Attachment 2 can be referred to for additional information on the RFPT Speed Control PDSs. **EXAMINER NOTE: The candidate may elect to use either the RFPT Manual Governor Switch or the PDS in Manual to remove 2A RFPT from service. Either method is acceptable.** • **If using Manual Governor Switch, Step 7.1[7] go to Step 7** • **If using PDS in Manual, Step 7.1[8] go to Step 10** [6] **LOWER** speed of RFPT being removed from service by either of the following methods: • **IF** Using individual RFPT Manual Governor Switch, **THEN GO TO** Step 7.1[7] • **IF** Using individual RFPT Speed Control PDS in MANUAL, **THEN GO TO** Step 7.1[8] Expected Action(s): Candidate selects either Step 7.1[7] or 7.1[8] to lower RFPT speed.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: **EXAMINER NOTE: The below step (Step 7.1[7]) is followed if the candidate chose to use the RFPT Manual Governor Switch.** [7] **SLOWLY LOWER** speed of individual RFPT on Panel 2-9-5, by performing the following: [7.1] **CONTROL** RFPT 2A [7.1.1] **DEPRESS** 2-HS-46-8A, RFPT 2A SPEED CONTROL RAISE/LOWER SWITCH to MANUAL GOVERNOR. • **CHECK** amber light at switch illuminated. Expected Action(s): Candidate depresses 2-HS-46-8A, RFPT 2A SPEED CONTROL RAISE/LOWER SWITCH to MANUAL GOVERNOR and checks the amber light is lit. **(N/A and NOT CRITICAL if Step 7.1[8] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 8: [7.1.2] **SLOWLY LOWER** RFPT speed, by placing RFPT Speed Control switch in RAISE and LOWER positions, as necessary. Expected Action(s): Candidate lowers 2A RFPT Speed. **(N/A and NOT CRITICAL if Step 7.1[8] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 9: [7.1.3] **IF** this is **NOT** the last operating Feed Pump, **THEN OBSERVE** rise in speed of any remaining RFPT operating in AUTO. Expected Action(s): Candidate observes the response of 2B and 2C RFPT. **(N/A if Step 7.1[8] is performed)**	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 10: **EXAMINER NOTE: The below step (Step 7.1[8]) is followed if the candidate chose to use the PDS in Manual.** [8] **SLOWLY LOWER** RFPT Speed Control PDS, on Panel 2-9-5 by performing the following: [8.1] **CONTROL** RFPT 2A [8.1.1] **PLACE** PDS 2-SIC-46-8 in MANUAL AND **ENSURE** Column 3 (CO) is selected. Expected Action(s): Candidate places 2A PDS 2-SIC-46-8 in MANUAL and verifies that Column 3 is selected. **(N/A and NOT CRITICAL if Step 7.1[7] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 11: [8.1.2] **SLOWLY LOWER** RFPT Speed, using Ramp Raise/Ramp Lower pushbuttons as necessary. Expected Action(s): Candidate lowers 2A RFPT Speed. **(N/A and NOT CRITICAL if Step 7.1[7] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 12: [8.1.3] **IF** this is **NOT** the last operating feed pump, **THEN OBSERVE** rise in speed of any remaining RFPT operating in AUTO. Expected Action(s): Candidate observes the response of 2B and 2C RFPT. **(N/A if Step 7.1[7] is performed)**	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Begin Alternate Path. As the speed of 2A RFPT lowers, the Discharge Check Valve will fail open. This malfunction is entered on simulator setup. No action is required by the driver for the Alternate Path.	

STEP / STANDARD	SAT / UNSAT
Step 13: **CAUTION** RFP Discharge Check Valve may fail to close when removing a RFP from service. [9] **IF** at any time RFPT Discharge Check Valve failure is experienced while removing RFPT from service, **THEN** (Otherwise N/A) **REFER TO** Step 7.1[15]. Expected Action(s): Candidate determines that 2A RFPT Discharge Check Valve is failed open and proceeds to Step 7.1[15].	`_____ SAT _____UNSAT _____N/A
Step 14: [15] **IF** RFPT Discharge Check Valve failure is experienced, **THEN** (Otherwise N/A): [15.1] **DEPRESS** 2-HS-3-94A, RFPT 2A DISCHARGE TESTABLE CHECK VALVE pushbutton. • **ENSURE** Discharge Check Valve closed. Expected Action(s): Candidate depresses 2-HS-3-94A, RFPT 2A DISCHARGE TESTABLE CHECK VALVE pushbutton and determines that the Check Valve now indicates partially closed (expected behavior).	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 15: [10] **CONTINUE** to slowly lower RFPT speed to minimum Speed setting (approximately 600 rpm) using Ramp RAISE / Ramp LOWER pushbuttons as necessary. Expected Action(s): Candidate continues to lower RFPT Speed.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: Inform the operator "another operator will continue this procedure".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2. A significant oil leak has developed on 2A Reactor Feedwater Pump (RFPT), requiring 2A RFPT to be shut down for emergent repair. Reactor Power has been reduced to 94% in preparation for removing 2A RFPT from service.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to remove 2A RFPT from service in accordance with 2-OI-3, Reactor Feedwater System, Section 7.1, RFPT/RFPT Shutdown.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Remove 3A RFPT From Service
JPM NUMBER:	274A-U3	REVISION:	2

TASK APPLICABILITY:	☐ SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-003-NO-07 / Shutdown the Reactor Feedwater System		
K/A RATINGS:	RO/SRO: 3.8		
K/A No. &STATEMENT:	259001 SF2 Reactor Feedwater System A3.03; System flow: Ability to monitor automatic operation of the Feedwater System, including: System flow (3.8)		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	2		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

Validated by: _____ _____
Validator *Date*

Approved by: _____ _____
Facility Representative *Date*

NRC Approved: _____ _____
Facility Author *Date*

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 274A-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to remove 3A Reactor Feedwater Pump from service in accordance with 3-OI-3, Reactor Feedwater System and respond to a malfunction of the Discharge Check Valve.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 A Solid Understanding of Plant Design, Engineering, and Sciences.

REFERENCES/PROCEDURES NEEDED: 3-OI-3

VALIDATION TIME: 13 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	12/18/19	All	Initial issue
1	1/14/19	All	JPM update; Format update
2	2/22/23	All	JPM update

Procedure Revisions

Procedure	Revision
3-OI-3	115

SIMULATOR SETUP

IC	28
Exam IC	280

Console Operator Instructions	• **Reset to IC 280** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 3-OI-3, Reactor Feedwater System Section 7.1, RFPT/RFPT Shutdown** • **Run schedule file 2304 NRC JPM B.sch** • **Verify event file 2304 NRC JPM B.sch**

Malfunctions	Description	Event	Severity	Delay	Initial set
FW08A	**RFP 3A DISCHARGE CHECK VALVE STICKS**	**N/A**	**87%**	**N/A**	**N/A**

Overrides	Description	Event	Severity	Delay	Initial set
N/A					

Schedule File(s): 2304 NRC JPM B.SCH
Event File(s): 2304 NRC JPM B.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3. A significant oil leak has developed on 3A Reactor Feedwater Pump (RFPT), requiring 3A RFPT to be shut down for emergent repair. Reactor Power has been reduced to 94% in preparation for removing 3A RFPT from service.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to remove 3A RFPT from service in accordance with 3-OI-3, Reactor Feedwater System, Section 7.1, RFPT/RFPT Shutdown.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 3-OI-3, Reactor Feedwater System Section 7.1, RFPT/RFPT Shutdown.	
Step 1: **7.1 RFPT/RFPT Shutdown** **CAUTIONS** 1) There is NOT adequate flow/NPSH to maintain 100% (3952) when one Condensate Pump is out of service. To maintain adequate NPSH available to the Condensate Booster Pumps, Condensate flow is limited to 14.1 Mlbm/hr. 2) At 100% (3952), Reactor power should be verified less than or equal to 95% prior to starting or stopping a Condensate Booster Pump or Reactor Feedwater Pump. If testing in accordance with 0-TI-704, (Condensate Pump, Condensate Booster Pump and Feedwater Pump Testing), then adjust Reactor power as required to support 0-TI-704 prior to stopping a Condensate Booster Pump or Reactor Feedwater Pump. 3) FAILURE to monitor SJAE/OG CNDR CNDS FLOW, 3-FI-2-42, on Panel 3-9-6 for proper flow (between 2 x 10^6 and 3 x 10^6 lbm/hr) may result in SJAE poor performance. The SJAE's will NOT trip on Condensate system low pressure. 4) Changes in Condensate System flow may require adjustment to SPE CNDS BYPASS, 3-FCV-002-0190. Refer to P&L 3.0L as necessary. 5) When isolating the Reactor Feedwater Pump(s) for maintenance, the associated injection water should also be isolated to prevent high seal differential pressure and allow the RFW Pump shafts to rotate freely. (BFNPER123395) 6) When a Reactor Feed Pump is isolated (suction, discharge and minimum flow valve closed) with injection water aligned to the pump, there is a potential of rising pump casing pressure and seal water leakoff flows reaching the point where seal water drains are overcome and seal water is forced into the oil system through the bearing housings. Therefore, the time that a RFP is isolated with injection water aligned to the pump should be minimized [1] **REFER TO** Section 3.0 and **REVIEW** Precautions and Limitations. Expected Action(s): This step should have been completed during the pre-brief, but the candidate may review the Precautions and Limitations.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: [2] **IF** Two Condensate Pumps are in service, **THEN ENSURE** Reactor Power is ≤ 85%. (**REFER TO** P&L 3.0 OO). Expected Action(s): Candidate determines that 3 Condensate Pumps are in operation, and marks this step as N/A.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **IF** any Condensate and/or Condensate Booster Pump is NOT in service, **THEN ENSURE** that Condensate System Flow measured on 3-XR-002-0026, CONDENSATE, is less than 12 Mlbm/hr (75% Reactor Power) prior to removing a Reactor Feedwater Pump from service. (Otherwise, N/A) Expected Action(s): Candidate determines that all Condensate and Condensate Booster Pumps are in service and marks this step as N/A.	_____ SAT _____UNSAT _____N/A
Step 4: **NOTE** It may be necessary to switch to SINGLE ELEMENT mode from THREE ELEMENT mode earlier than recommended if Feedwater control becomes unstable. [4] **IF** RFPT being removed from service is last operating RFPT **OR IF** at any time Feedwater control becomes unstable**, THEN DEPRESS** SINGLE ELEMENT pushbutton, 3-HS-46-6/1. (Otherwise N/A) • **ENSURE** green backlight for pushbutton illuminated. Expected Action(s): Candidate determines that 3A RFPT is not the last operating RFPT and marks this step as N/A.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [5] **ENSURE** in AUTO: • 3-HS-3-101A, RFPT 3A TURNING GEAR MOTOR Expected Action(s): Candidate verifies 3-HS-3-101A, RFPT 3A TURNING GEAR MOTOR is in AUTO.	_____ SAT _____UNSAT _____N/A
Step 6: **NOTES** 1) Column 1 (PV) on individual RFPT Speed Control Panel Display Stations (PDS) displays actual pump speed and is NOT controlled in any mode. 2) When selected, then Column 2 (SP) on individual RFPT Speed Control PDS displays pump flow bias and is changed with Ramp RAISE/Ramp LOWER pushbuttons with controller in AUTO (blue light illuminated). 3) When selected, then Column 3 (CO) on individual RFPT Speed Control PDS displays RFPT speed demand and is changed with Ramp RAISE/Ramp LOWER pushbuttons with controller in MANUAL (amber light illuminated). 4) Attachment 2 has additional information on RFPT Speed Control PDS. **EXAMINER NOTE: The candidate may elect to use either the RFPT Manual Governor Switch or the PDS in Manual to remove 3A RFPT from service. Either method is acceptable.** • **If using Manual Governor Switch, Step 7.1[7] go to Step 7** • **If using PDS in Manual, Step 7.1[8] go to Step 10** [6] **LOWER** speed of RFPT being removed from service by either of the following methods: • **IF** Using individual RFPT Manual Governor Switch, **THEN GO TO** Step 7.1[7] • **IF** Using individual RFPT Speed Control PDS in MANUAL, **THEN GO TO** Step 7.1[8] Expected Action(s): Candidate selects either Step 7.1[6] or 7.1[7] to lower speed.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: **EXAMINER NOTE: The below step (Step 7.1[6]) is followed if the candidate chose to use the RFPT Manual Governor Switch.** [7] **LOWER** speed of RFPT using individual RFPT SPEED CONT RAISE/LOWER switch as follows (Panel 3-9-5): • 3-HS-46-8A, RFPT 3A SPEED CONT RAISE/LOWER Switch [7.1] **DEPRESS** RFPT Speed Control Raise/Lower switch to MANUAL GOVERNOR. [7.2] **ENSURE** illuminated amber light at switch. Expected Action(s): Candidate depresses 3-HS-46-8A, RFPT 3A SPEED CONT RAISE/LOWER SWITCH to MANUAL GOVERNOR and checks the amber light is lit. **(N/A and NOT CRITICAL if Step 7.1[7] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 8: [7.3] **SLOWLY LOWER** RFPT speed, by placing RFPT Speed Control Switch in RAISE and LOWER positions, as necessary. Expected Action(s): Candidate lowers 3A RFPT Speed. **(N/A and NOT CRITICAL if Step 7.1[7] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 9: [7.4] **IF** this is **NOT** the last operating RFPT, **THEN OBSERVE** rise in speed of any operating RFPT in auto as RFW Control System maintains Reactor Water Level. Expected Action(s): Candidate observes the response of 3B and 3C RFPT. **(N/A if Step 7.1[7] is performed)**	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 10: **EXAMINER NOTE: The below step (Step 7.1[8]) is followed if the candidate chose to use the PDS in Manual.** [8] **LOWER** speed of RFPT using individual RFPT SPEED CONTROL PDS as follows (Panel 3-9-5): • 3-SIC-46-8, RFPT 3A SPEED CONTROL (PDS) [8.1] **PLACE** PDS in MANUAL and **ENSURE** Column 3 (CO) selected. Expected Action(s): Candidate places 3A PDS 3-SIC-46-8 in MANUAL and verifies that Column 3 is selected. **(N/A and NOT CRITICAL if Step 7.1[6] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 11: [8.2] **SLOWLY LOWER** RFPT speed using Ramp Up/Ramp Down pushbuttons as necessary. Expected Action(s): Candidate lowers 3A RFPT Speed. **(N/A and NOT CRITICAL if Step 7.1[6] is performed)**	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 12: [8.3] **IF** this is **NOT** the last operating RFPT, **THEN OBSERVE** rise in speed of any operating RFPT in auto as RFW Control System maintains Reactor water level. Expected Action(s): Candidate observes the response of 3B and 3C RFPT. **(N/A if Step 7.1[6] is performed)**	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Begin Alternate Path. As the speed of 3A RFPT lowers, the Discharge Check Valve will fail open. This malfunction is entered on simulator setup. No action is required by the driver for the Alternate Path.	

STEP / STANDARD	SAT / UNSAT
Step 13: **CAUTION** RFP Discharge Check Valve failure may be experienced while removing RFP from service. [9] **IF** at any time RFPT Discharge Check Valve failure is experienced while removing RFPT from service, **THEN PERFORM** the following: (Otherwise N/A) [9.1] **DEPRESS** RFPT Discharge Testable Check Valve Pushbutton for approximately ten seconds (Panel 3-9-6). Expected Action(s): Candidate depresses RFPT Discharge Testable Check Valve Pushbutton for approximately ten seconds.	_____ SAT _____UNSAT _____N/A
Step 14: [9.2] **CHECK** reverse flow through the Check Valve has stopped. Expected Action(s): Candidate checks that reverse flow through the Check Valve has stopped and determines that the Check Valve now indicates partially closed (expected behavior).	_____ SAT _____UNSAT _____N/A
Step 15: [10] **CONTINUE** to slowly lower RFPT Speed to minimum speed setting (approximately 600 rpm) using the manual governor in step 7.1[7] or the Ramp RAISE / Ramp LOWER pushbuttons in step 7.1[8] as necessary. Expected Action(s): Candidate continues to lower RFPT Speed.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: Inform the operator "another operator will continue this procedure".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3. A significant oil leak has developed on 3A Reactor Feedwater Pump (RFPT), requiring 3A RFPT to be shut down for emergent repair. Reactor Power has been reduced to 94% in preparation for removing 3A RFPT from service.

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to remove 3A RFPT from service in accordance with 3-OI-3, Reactor Feedwater System, Section 7.1, RFPT/RFPT Shutdown.

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Match Recirculation Pump Speeds and Respond to a Loss of Speed Control per 3-OI-68-3, Recirculation Loop A or B Speed Control Failure
JPM NUMBER:	763A-U2	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-068-AB-03 / Respond to Recirculation Loop A or B Speed Control Failure		
K/A RATINGS:	RO/SRO: 3.9		
K/A STATEMENT:	202001 Recirculation System A4.14; Ability to manually operate and/or monitor in the control room: Variable frequency drives		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	4		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by:

_____ _____

Validator *Date*

Approved by:

_____ _____

Facility Representative *Date*

NRC Approved:

_____ _____

Facility Author *Date*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 763A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to perform the actions necessary in accordance with 3-OI-68, Reactor Recirculation System, to match Recirculation Pump Speeds and to respond to a Loss of Recirc Speed Control in accordance with 3-AOI-68-3, Recirculation Loop A or B Speed Control Failure.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-OI-68, 2-AOI-68-3

VALIDATION TIME: 5 Minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/22/23	ALL	Initial issue

Procedure Revisions

Procedure	Revision
2-OI-68	168
2-AOI-68-3	20

SIMULATOR SETUP

IC	28
Exam IC	281

Console Operator Instructions	• Reset to IC 281 • Place the simulator in RUN to ensure stable conditions • Ensure the candidate has been pre-briefed on 3-OI-68 and is provided RCP • Run schedule file 2304 NRC JPM C UNIT 3.SCH • Verify event file 2304 NRC JPM C UNIT 3.EVT

Overrides	Description	Event	Severity	Delay	Initial set
HS-96-15B	RECIRC PUMP 3A RAISE MEDIUM	1	RMEDIUM	N/A	N/A

Schedule File(s): 2304 NRC JPM C UNIT 2.SCH
Event File(s): 2304 NRC JPM C UNIT 2.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2, with the following conditions:

- Reactor Power was reduced to 90% for maintenance, which has been completed
- Recirculation Pump Speed has been split in accordance with 2-OI-68, Reactor Recirculation System

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to perform the following:

- Raise 2A Recirculation Pump Speed to match 2B in accordance with 2-OI-68, Reactor Recirculation System
- Then raise Reactor Power in accordance with 2-GOI-100-1A, Unit Startup and Power Operation, and RCP-2304U2-JPM-C

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: 2-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown in 0-TI-248, Unit 2 Core Operating Limits Report (COLR) and 2-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor Pump Speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pumps 2A and/or 2B Speed with Recirc Individual Control, **THEN PERFORM** the following: • **RAISE** Recirc Pump 2A using 2-HS-96-15A(B), RAISE SLOW (MEDIUM) Expected Action(s): Candidate raises 2A Recirc Pump Speed in accordance with the Reactivity Control Plan.	_____SAT _____UNSAT _____N/A
EXAMINER NOTE: Begin alternate path. When 2A Recirculation Pump Speed is ≥ 1260 RPM, Pump Speed will begin to raise without Operator action. No action is required by the driver to insert this malfunction.	

STEP / STANDARD	SAT / UNSAT
Step 2: 2-AOI-68-3, Recirculation Loop A or B Speed Control Failure 4.1 Immediate Actions: None 4.2 Subsequent Actions **NOTE** Depressing Recirc Master Control pushbuttons changes the speed of both Recirc Pumps. **CAUTIONS** 1) Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. 2) Single Recirculation Loop operation is prohibited in the MELLLA+ operating domain. **REFERENCE** Tech Spec 3.4.1. [1] **ATTEMPT** to control Recirc Pump 2A(2B) Speed using other speed control push buttons. REFERENCE 2-OI-68, Reactor Recirculation System, Adjusting Recirc Flow to Adjust Reactor Power. Expected Action(s): Candidate attempts to control 2A Recirc Pump Speed using other speed control methods.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [2] **IF** Recirc Pump Speed is raising or lowering uncontrollably, OR lowering of Recirc Pump Speed is immediately required, **THEN**: [2.1] **SHUT DOWN** the Recirc Pump by depressing 2-HS-96-19, RECIRC DRIVE 2A SHUTDOWN. **REFERENCE** 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease. Expected Action(s): **EXAMINER NOTE: 2A Recirc Pump will not be able to be shut down using 2-HS-96-19.** Candidate depresses 2-HS-96-19 in an attempt to shutdown 2A Recirc Pump and determines that 2A Recirc Pump did not shut down. Continues in 2-AOI-68-3.	_____SAT _____UNSAT _____N/A
Step 4: [2.2] **IF** Step 4.2 [2.1] failed to shut down Recirc Drive 2A, **THEN PERFORM** the following: [2.2.1] **DEPRESS** push-button 2-HS-96-29, RECIRC DRIVE 2A AUTO TO MANUAL TRIP. Expected Action(s): Candidate depresses 2-HS-96-29.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [2.2.2] **ENSURE** in MANUAL, 2-XS-57-16A, RECIRC DRIVE 2A AUTO/MAN SELECT. Expected Action(s): Candidate verifies 2-XS-57-16A is in manual.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 6: [2.2.3] **PLACE** in TRIP 2-HS-57-17, RECIRC DRIVE 2A NORMAL FEEDER or 2-HS-57-15, RECIRC DRIVE 2A ALTERNATE FEEDER. Expected Action(s): Candidate places 2-HS-57-17 in trip.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: Once JPM Step 6 has been completed, inform the candidate "Another Operator will finish this procedure. This completes your task".	

STOP TIME _____

Form 3.2-3 Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2, with the following conditions:

- Reactor Power was reduced to 90% for maintenance, which has been completed
- Recirculation Pump Speed has been split in accordance with 2-OI-68, Reactor Recirculation System

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to perform the following:

- Raise 2A Recirculation Pump Speed to match 2B in accordance with 2-OI-68, Reactor Recirculation System
- Then raise Reactor Power in accordance with 2-GOI-100-1A, Unit Startup and Power Operation, and RCP-2304U2-JPM-C

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Match Recirculation Pump Speeds and Respond to a Loss of Speed Control per 3-OI-68-3, Recirculation Loop A or B Speed Control Failure
JPM NUMBER:	763A-U3	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-068-AB-03 / Respond to Recirculation Loop A or B Speed Control Failure		
K/A RATINGS:	RO/SRO: 3.9		
K/A STATEMENT:	202001 Recirculation System A4.14; Ability to manually operate and/or monitor in the control room: Variable frequency drives		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	4		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by:

_____ _____

Validator *Date*

Approved by:

_____ _____

Facility Representative *Date*

NRC Approved:

_____ _____

Facility Author *Date*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 763A-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to perform the actions necessary in accordance with 3-OI-68, Reactor Recirculation System, to match Recirculation Pump Speeds and to respond to a Loss of Recirc Speed Control in accordance with 3-AOI-68-3, Recirculation Loop A or B Speed Control Failure.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 3-OI-68, 3-AOI-68-3

VALIDATION TIME: 5 Minutes

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/23/23	All	Initial issue

Procedure Revisions

Procedure	Revision
3-OI-68	106
3-AOI-68-3	11

SIMULATOR SETUP

IC	28
Exam IC	281

Console Operator Instructions	• Reset to IC 281 • Place the simulator in RUN to ensure stable conditions • Ensure the candidate has been pre-briefed on 3-OI-68 and is provided RCP • Run schedule file 2304 NRC JPM C UNIT 3.SCH • Verify event file 2304 NRC JPM C UNIT 3.EVT

Malfunctions	Description	Event	Severity	Delay	Initial set
HS-96-15B	RECIRC PUMP 3A RAISE MEDIUM	1	RMEDIUM	N/A	N/A

Schedule File(s): 2304 NRC JPM C UNIT 3.SCH
Event File(s): 2304 NRC JPM C UNIT 3.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3, with the following conditions:

- Reactor Power was reduced to 90% for maintenance, which has been completed
- Recirculation Pump Speed has been split in accordance with 3-OI-68, Reactor Recirculation System

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to perform the following:

- Raise 3A Recirculation Pump Speed to match 3B in accordance with 3-OI-68, Reactor Recirculation System
- Then raise Reactor Power in accordance with 3-GOI-100-1A, Unit Startup and Power Operation, and RCP-2304U3-JPM-C

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: 3-OI-68, Reactor Recirculation System Section 6.2, Adjusting Recirc Flow **NOTES** 1) Thermal Limits are shown on 0-TI-248, Unit 3 Core Operating Limits Report (COLR) and 3-SR-2, Instrument Checks and Observations. 2) Recirc Flow changes made during the later part of the operating cycle (Coastdown) could cause Core Flow values to approach or exceed the allowable values of the Increased Core Flow (ICF) Region of the Power to Flow Map. Instruments used to monitor pump speed and Core Flow should be identified in the Reactivity Control Plan. These values should be recorded prior to reducing Core Flow and used as a benchmark to reestablish the previous conditions when returning to power. Increased caution should be used when changes in Recirc Flow are made in this area. [1] **IF** desired to control Recirc Pump 3A speed with Recirc Individual Control, **THEN PERFORM** the following: [1.1] Raise Recirc Pump 3A using, 3-HS-96-15A(15B), RAISE SLOW (MEDIUM). Expected Action(s): Candidate raises 3A Recirc Pump Speed in accordance with the Reactivity Control Plan.	_____SAT _____UNSAT _____N/A
EXAMINER NOTE: Begin alternate path. When 3A Recirculation Pump Speed is ≥ 1260 RPM, Pump Speed will begin to raise without Operator action. No action is required by the driver to insert this malfunction.	

STEP / STANDARD	SAT / UNSAT
Step 2: 3-AOI-68-3, Recirculation Loop A or B Speed Control Failure 4.1 Immediate Actions: None 4.2 Subsequent Actions **NOTE** Depressing Recirc Master Control pushbuttons will change the speed of both Recirc Pumps. **CAUTIONS** 1) Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. 2) Single Recirculation Loop operation is prohibited in the MELLLA+ operating domain. **REFERENCE** Tech Spec 3.4.1. [1] **ATTEMPT** to control Recirc Pump 3A(3B) Speed using other speed control push buttons in accordance with 3-OI-68, Reactor Recirculation System. Expected Action(s): Candidate attempts to control 3A Recirc Pump Speed using other speed control methods.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [2] **IF** Recirc Pump Speed is raising or lowering uncontrollably OR lowering of Recirc Pump Speed is immediately required, **THEN**: [2.1] **SHUT DOWN** the Recirc Pump by depressing 3-HS-96-19, RECIRC DRIVE 3A SHUTDOWN and **REFERENCE** 3-AOI-68-1, Recirc Pump Trip/Core Flow Decrease. Expected Action(s): **EXAMINER NOTE: 3A Recirc Pump will not be able to be shut down using 3-HS-96-19.** Candidate depresses 3-HS-96-19 in an attempt to shutdown 3A Recirc Pump and determines that 3A Recirc Pump did not shut down. Continues in 3-AOI-68-3.	_____SAT _____UNSAT _____N/A
Step 4: [2.2] **IF** Step 4.2 [2.1] failed to shut down Recirc Drive 3A, **THEN PERFORM** the following: [2.2.1] **DEPRESS** push-button 3-HS-96-29, RECIRC DRIVE 3A AUTO TO MANUAL TRIP. Expected Action(s): Candidate depresses 3-HS-96-29.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [2.2.2] **ENSURE** in MANUAL, 3-XS-57-16A, RECIRC DRIVE 3A AUTO/MAN SELECT. Expected Action(s): Candidate verifies 2-XS-57-16A is in manual.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 6: [2.2.3] **PLACE** in TRIP 3-HS-57-17, RECIRC DRIVE 3A NORMAL FEEDER or 3-HS-57-15, RECIRC DRIVE 3A ALTERNATE FEEDER. Expected Action(s): Candidate places 3-HS-57-17 in trip.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: Once JPM Step 6 has been completed, inform the candidate "Another Operator will finish this procedure. This completes your task".	

STOP TIME _____

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3, with the following conditions:

- Reactor Power was reduced to 90% for maintenance, which has been completed
- Recirculation Pump Speed has been split in accordance with 3-OI-68, Reactor Recirculation System

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to perform the following:

- Raise 3A Recirculation Pump Speed to match 3B in accordance with 3-OI-68, Reactor Recirculation System
- Then raise Reactor Power in accordance with 3-GOI-100-1A, Unit Startup and Power Operation, and RCP-2304U3-JPM-C

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Purge the Primary Containment per 2-EOI Appendix-14A, N2 Makeup to Primary Containment
JPM NUMBER:	669-U2	REVISION:	1

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-77 Lineup Nitrogen Makeup to Primary Containment in accordance with EOI Appendix -14A		
K/A RATINGS:	RO/SRO: 3.8		
K/A STATEMENT:	223001 Primary Containment System and Auxiliaries A2.04**;** Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High containment/drywell hydrogen concentration		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	5		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	12 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 669-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate will perform operations necessary to purge the Primary Containment with Nitrogen in accordance with 2-EOI Appendix-14A.

Operator Fundamental evaluated:
OF-1 Monitoring plant conditions and indications closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-EOI Appendix-14A

VALIDATION TIME: 8 Minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	10/09/15	All	Initial issue
1	2/23/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-EOI Appendix-14A	9

SIMULATOR SETUP

IC	28
Exam IC	284

Console Operator Instructions	• Reset to IC 284 • Place the simulator in RUN to ensure stable conditions • Set XR 76 110 to “OVERVIEW”

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

Schedule File(s): N/A
Event File(s): N/A

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2. An accident has occurred with the following conditions:

- The Reactor was SCRAMMED due to a Recirculation System leak
- Fuel damage has occurred
- 2-EOI Appendix-12, Primary Containment Venting, has been completed
- 2-EOI Appendix-19, H2/O2 Analyzer Operation, has been completed

Note: Another Operator will be controlling Reactor Water Level using RCIC

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to lower Hydrogen Concentration by performing 2-EOI Appendix-14A, Nitrogen Makeup to Primary Containment.

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **IF** PCIS Group 6 Isolation signal exists, **THEN PERFORM** 2-EOI Appendix-8E concurrently with this procedure. Expected Action(s): It is provided in the Candidate Cue Sheet that 2-EOI Appendix-8E is complete. Candidate continues on in this procedure. **DRIVER: When contacted to perform 2-EOI Appendix-8E, insert EVENT 1 and report 2-EOI Appendix-8E is complete.**	_____SAT _____UNSAT _____N/A
Step 2: [2] **MONITOR** Drywell and Suppression Chamber H2 and O2 concentrations on one or more of the following displays (2-EOI Appendix-19): • 2-MON-76-110, H2/O2 ANALYZER (Panel 2-9-55) • 2-XR-76-110, H2/O2 CONCENTRATION (Panel 2-9-54) Expected Action(s): It is provided in the Candidate Cue Sheet that 2-EOI Appendix-19 is complete. Candidate continues on in this procedure.	_____SAT _____UNSAT _____N/A
Step 3: [3] **IF** Drywell / Suppression Chamber H2 or O2 analyzer is or becomes inoperable, **THEN NOTIFY** Chem Lab to sample Drywell or Suppression Chamber for H2 and O2 using CI-644, Primary Gas Determination of Hydrogen and Oxygen. Expected Action(s): The H2/O2 Analyzer is operable, so this step is marked as N/A. Candidate continues on in this procedure.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **WHEN** Drywell or Suppression Chamber H2 or O2 concentration is increasing, **THEN CONTINUE** in this procedure. Expected Action(s): Candidate continues on in this procedure.	_____ SAT _____UNSAT _____N/A
Step 5: **NOTE** This procedure assumes that the normal makeup is from N2 Storage Tank B. [5] **DISPATCH** personnel to N2 Storage Tank 'B' to **CHECK** 0-PI-76-331, N2 TANK 'B' PRESSURE INDICATOR, between 160 and 190 psig. Expected Action(s): Candidate contacts an Assistant Unit Operator (AUO) to check N2 Bank 'B' Pressure. **EXAMINER CUE: N2 Bank 'B' Pressure is 175 psig.**	_____ SAT _____UNSAT _____N/A
Step 6: **EXAMINER NOTE: If Candidate ask if 2-EOI Appendix-19, H2/O2 Analyzer Operation, has been aligned and sampling for 10 minutes or greater or if requested to be performed (already given as complete in the initial conditions), state 2-EOI Appendix-19 was performed more than 10 minutes ago.** [6] **DETERMINE** area with highest H2 and/or O2 concentrations and **INFORM** the NUSO. Expected Action(s): Candidate determines that Drywell concentrations are higher than the Suppression Chamber. **EXAMINER NOTE: If the Candidate notifies the NUSO that Drywell H2 concenrtraion is higher than the Suppression Chamber, acknowledge the report.**	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [7] **CHECK** PCIS RESET (Panel 3-9-4). Expected Action(s): Candidate resets Group 6 PCIS.	_____ SAT _____UNSAT _____N/A
Step 8: [8] **IF** it is desired to makeup N2 to the Suppression Chamber, **THEN CONTINUE** in this procedure at Step 1.0[10]. Expected Action(s): Candidate marks this step as N/A and continues with N2 makeup to the Drywell.	_____ SAT _____UNSAT _____N/A
Step 9: [9] **CONTROL** Drywell H2 and O2 as follows: A. **OPEN** the following valves to admit N2 to Drywell (Panel 2-9-3): • 2-FCV-76-18, DRYWELL N2 MAKEUP INBOARD ISOLATION VALVE Expected Action(s): Candidate opens 2-FCV-76-18.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 10: • 2-FCV-76-17, PRIMARY CONTAINMENT N2 MAKEUP OUTBOARD ISOLATION VALVE Expected Action(s): Candidate opens 2-FCV-76-17	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 8: B. **SLOWLY** ADJUST 2-PC-76-14, DRYWELL/SUPPRESSION CHAMBER N2 MAKEUP PRESSURE CONTROL (Panel 2-9-3), to maintain between 55 and 60 scfm, or as directed by SRO if SAMG execution is in progress. Expected Action(s): Candidate adjusts 2-PC-76-14 to achieve 55-60 scfm.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: Once JPM Step 8 has been completed, inform the candidate "Another Operator will finish this procedure. This completes your task".	

STOP TIME _____

Form 3.2-3 Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2. An accident has occurred with the following conditions:

- The Reactor was SCRAMMED due to a Recirculation System leak
- Fuel damage has occurred
- 2-EOI Appendix-12, Primary Containment Venting, has been completed
- 2-EOI Appendix-19, H2/O2 Analyzer Operation, has been completed

Note: Another Operator will be controlling Reactor Water Level using RCIC

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to lower Hydrogen Concentration by performing 2-EOI Appendix-14A, Nitrogen Makeup to Primary Containment.

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Purge the Primary Containment per 3-EOI Appendix-14A, N2 Makeup to Primary Containment
JPM NUMBER:	669-U3	REVISION:	1

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-77 Lineup Nitrogen Makeup to Primary Containment in accordance with EOI Appendix-14A		
K/A RATINGS:	RO/SRO: 3.8		
K/A STATEMENT:	223001 Primary Containment System and Auxiliaries A2.04**;** Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High containment/drywell hydrogen concentration		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	5		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	12 minutes		

Validated by: _____ _____
Validator *Date*

Approved by: _____ _____
Facility Representative *Date*

NRC Approved: _____ _____
Facility Author *Date*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 669-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate will perform operations necessary to purge the Primary Containment with Nitrogen in accordance with 3-EOI Appendix-14A.

Operator Fundamental evaluated:
OF-1 Monitoring plant conditions and indications closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 3-EOI Appendix-14A

VALIDATION TIME: 8 Minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	10/09/15	All	Initial issue
1	2/23/23	All	JPM update

Procedure Revisions

Procedure	Revision
3-EOI Appendix-14A	6

SIMULATOR SETUP

IC	28
Exam IC	278

Console Operator Instructions	• Reset to IC 278 • Place the simulator in RUN to ensure stable conditions • Set XR 76 110 to “OVERVIEW”

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A					

Schedule File(s): N/A
Event File(s): N/A

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3. An accident has occurred with the following conditions:

- The Reactor was SCRAMMED due to a Recirculation System leak
- Fuel damage has occurred
- 3-EOI Appendix-12, Primary Containment Venting, has been completed
- 3-EOI Appendix-19, H2/O2 Analyzer Operation, has been completed

Note: Another Operator will be controlling Reactor Water Level using RCIC

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to lower Hydrogen Concentration by performing 3-EOI Appendix-14A, Nitrogen Makeup to Primary Containment.

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **IF** PCIS Group 6 Isolation signal exists, **THEN PERFORM** 3-EOI Appendix-8E concurrently with this procedure. Expected Action(s): It is provided in the Candidate Cue Sheet that 3-EOI Appendix-8E is complete. Candidate continues on in this procedure. **DRIVER: When contacted to perform 3-EOI Appendix-8E, insert EVENT 1 and report 3-EOI Appendix-8E is complete.**	_____SAT _____UNSAT _____N/A
Step 2: [2] **MONITOR** Drywell and Suppression Chamber H2 and O2 concentrations on one or more of the following displays (3-EOI Appendix 19): • 3-MON-76-110, H2/O2 ANALYZER (Panel 3-9-55) • 3-XR-76-110, H2/O2 CONCENTRATION (Panel 3-9-54) Expected Action(s): It is provided in the Candidate Cue Sheet that 3-EOI Appendix-19 is complete. Candidate continues on in this procedure.	_____SAT _____UNSAT _____N/A
Step 3: [3] **IF** Drywell / Suppression Chamber H2 or O2 analyzer is or becomes inoperable, **THEN NOTIFY** Chem Lab to sample Drywell or Suppression Chamber for H2 and O2 using CI-644, Primary Gas Determination of Hydrogen and Oxygen. Expected Action(s): The H2/O2 Analyzer is operable, so this step is marked as N/A. Candidate continues on in this procedure.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **WHEN** Drywell or Suppression Chamber H2 or O2 concentration is increasing, **THEN CONTINUE** in this procedure. Expected Action(s): Candidate continues on in this procedure.	_____ SAT _____UNSAT _____N/A
Step 5: **NOTE** This procedure assumes that the normal makeup is from N2 Storage Tank 'B'. [5] **DISPATCH** personnel to N2 Storage Tank 'B' to **CHECK** 0-PI-76-331, N2 TANK 'B' PRESSURE INDICATOR, between 160 and 190 psig. Expected Action(s): Candidate contacts an Assistant Unit Operator (AUO) to check N2 Bank 'B' Pressure. **EXAMINER CUE: N2 Bank 'B' Pressure is 175 psig.**	_____ SAT _____UNSAT _____N/A
Step 6: **EXAMINER NOTE: If Candidate ask if 3-EOI Appendix-19, H2/O2 Analyzer Operation, has been aligned and sampling for 10 minutes or greater or if requested to be performed (already given as complete in the initial conditions), state 3-EOI Appendix-19 was performed more than 10 minutes ago.** [6] **DETERMINE** area with highest H2 and/or O2 concentrations and **INFORM** the NUSO. Expected Action(s): Candidate determines that Drywell concentrations are higher than the Suppression Chamber. **EXAMINER NOTE: If the Candidate notifies the NUSO that Drywell H2 concenrtraion is higher than the Suppression Chamber, acknowledge the report.**	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [7] **CHECK** PCIS RESET (Panel 3-9-4). Expected Action(s): Candidate resets Group 6 PCIS.	_____ SAT _____UNSAT _____N/A
Step 8: [8] **IF** it is desired to makeup N2 to the Suppression Chamber, **THEN CONTINUE** in this procedure at Step 1.0[10]. Expected Action(s): Candidate marks this step as N/A and continues with N2 makeup to the Drywell.	_____ SAT _____UNSAT _____N/A
Step 9: [9] **CONTROL** Drywell H2 and O2 as follows: A. **OPEN** the following valves to admit N2 to Drywell (Panel 3-9-3): • 3-FCV-76-18, DRYWELL N2 MAKEUP INBOARD ISOLATION VALVE Expected Action(s): Candidate opens 3-FCV-76-18.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 10: • 3-FCV-76-17, PRIMARY CONTAINMENT N2 MAKEUP OUTBOARD ISOLATION VALVE Expected Action(s): Candidate opens 3-FCV-76-17	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 8: B. **SLOWLY** ADJUST 3-PC-76-14, DRYWELL/SUPPRESSION CHAMBER N2 MAKEUP PRESSURE CONTROL (Panel 3-9-3), to maintain between 55 and 60 scfm, or as directed by SRO if SAMG execution is in progress. Expected Action(s): Candidate adjusts 3-PC-76-14 to achieve 55-60 scfm.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: Once JPM Step 8 has been completed, inform the candidate "Another Operator will finish this procedure. This completes your task".	

STOP TIME _____

Form 3.2-3 Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3. An accident has occurred with the following conditions:

- The Reactor was SCRAMMED due to a Recirculation System leak
- Fuel damage has occurred
- 3-EOI Appendix-12, Primary Containment Venting, has been completed
- 3-EOI Appendix-19, H2/O2 Analyzer Operation, has been completed

Note: Another Operator will be controlling Reactor Water Level using RCIC

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to lower Hydrogen Concentration by performing 3-EOI Appendix-14A, Nitrogen Makeup to Primary Containment.

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Parallel an Emergency Diesel Generator to a Shutdown Board per 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability Test	
JPM NUMBER:		759A-U2	REVISION:	0

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-082-NO-07 / Parallel and load D/G at panel 9-23		
K/A RATINGS:	RO / SRO: 3.5		
K/A No. &STATEMENT:	264000 SF6 EGE Emergency Generators A1.10; Ability to predict and/or monitor changes in parameters associated with operation of the Emergency Generators, including: Lights and alarms		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	6		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

Validated by: _____ _____
Validator *Date*

Approved by: _____ _____
Facility Representative *Date*

NRC Approved: _____ _____
Facility Author *Date*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 759A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to perform operations necessary to parallel an Emergency Diesel Generator (D EDG) to its associated 4KV Shutdown Board in accordance with 0-OI-82, and respond to a Low Lube Oil Pressure alarm.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-SR-3.8.1.1(D), 1/2-ARP-9-23D

VALIDATION TIME: 12 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/28/23	All	Initial issue

Procedure Revisions

Procedure	Revision
0-OI-82, ATT 1, 2	180
0-SR-3.8.1.1(D)	60
1/2-ARP-9-23D	38

SIMULATOR SETUP

IC	28
Exam IC	282

Console Operator Instructions	• **Reset to IC 282** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 0-SR-3.8.1.1(D) and 0-OI-82 - ATT 1, 2 to review (another Operator will be filling it out)** • **Verify that Remote Function ED32D (4KV SD BD 3ED MODE SWITCH) is ENABLED** • **Ensure that a stopwatch and the procedure stand are staged** • **Start schedule file 2304 NRC JPM E UNIT 2.sch** • **Verify event file 2304 NRC JPM E UNIT 2.evt**

Malfunctions	Description	Event	Severity	Delay	Initial
0-XA-55-23D_4	DIESEL GEN D LUBE OIL ABNORMAL	1	ON	30	OFF
0-XA-55-23D_2	DIESEL GEN D TROUBLE	1	ON	60	OFF

Overrides	Description	Event	Severity	Delay	Initial
ZLO0IL8213D_1	LOW OIL PRESSURE DG D	1	ON	2	OFF

Schedule File(s): 2304 NRC JPM E UNIT 2.SCH
Event File(s): 2304 NRC JPM E UNIT 2.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 2. 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability Test, is in progress with the following conditions:

- An Assistant Unit Operator (AUO) is standing by at 'D' Diesel Generator
- 'D' Diesel Generator has been started and is running
- 0-SR-3.8.1.1(D), is complete up to Section 6.9, Step [4]

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to parallel 'D' Diesel Generator to the 'D' 4KV Shutdown Board in accordance with 0-SR-3.8.1.1(D) and load the Diesel to 2300 kW, starting at Section 6.9, Step [4].

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability Test. Step 1: [4] On Panel 0-9-23-8 **PREPARE** the Diesel Generator to be paralleled as follows: [4.1] **PLACE** 0-25-211-D/20A, DG D BKR 1816 SYNC, to the "ON" position, if necessary. Expected Action(s): Candidate places 0-25-211-D/20A in the ON position.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [4.2] **ENSURE** 0-XS-211-D, 4KV SD BOARD D VOLTAGE SELECT, in SHUTDOWN BOARD D. Expected Action(s): Candidate places 0-XS-211-D in the SHUTDOWN BOARD D position.	_____ SAT _____UNSAT _____N/A
Step 3: [4.3] **CHECK** Shutdown Board D voltage is 3950 to 4400 VOLTS and NOT undergoing abnormal voltage transients using 0-EI-211-D, 4KV SD BD D VOLTS. [4.4] **CHECK** 0-SI-211-CD, SYSTEM SYNC FREQUENCY, is 59 to 61 HZ and not undergoing abnormal frequency transients. Expected Action(s): Candidate verifies electrical parameters are within limits and are stable as required.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: **EXAMINER NOTE: Step [4.5] is a Concurrent Verification (CV) step. The candidate may state that the step cannot be completed without CV. Inform the candidate that verification is being completed as required.** [4.5] **PULL OUT** and **PLACE** 0-HS-82-D/5A, DG D MODE SELECT, in PARALLELED WITH SYSTEM. [4.6] **CHECK** PARALLELED WITH SYSTEM light ILLUMINATED. Expected Action(s): **EXAMINER NOTE: Verifying that the PARALLELED WITH SYSTEM light is illuminated is NOT a Critical Step.** Candidate places 0-HS-82-D/5A in PARALLELED WITH SYSTEM.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [4.7] **ADJUST** Diesel Generator D Frequency, using 0-HS-82-D/3A, DG D GOVERNOR CONTROL, to obtain a Synchroscope needle rotating one revolution every 15 to 20 seconds in the FAST direction. Expected Action(s): Candidate adjusts Diesel Generator frequency to cause the synchroscope to rotate in the FAST direction as required.	_____ SAT _____UNSAT _____N/A
Step 6: [4.8] **ADJUST** Diesel Generator D Voltage (0-EI-82-CD, GEN SYNC REF VOLTAGE) to match Shutdown Board D Voltage (0-EI-211-CD, SYSTEM SYNC REF VOLTAGE) using 0-HS-82-D/2A, DG D VOLT REGULATOR CONTROL. Expected Action(s): Candidate adjusts Diesel Generator Voltage to match Shutdown Board D Voltage.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [5] On Diesel Generator D Protective Relay Cabinet, Panel 25-47D. (D/G Room D) **CHECK** Phase voltages for the PT Transformers. Expected Action(s): **DRIVER CUE: When contacted as the DG AUO to perform Step [5], report when requested that all three switch positions indicate 4.2kV.** Candidate contacts the Diesel Generator AUO for voltage readings.	_____ SAT _____UNSAT _____N/A
Step 8: Section 6.10, Paralleling and Loading D/G to the Shutdown Board [1] **PARALLEL** the Diesel Generator to the D Shutdown Board by performing the following steps concurrently: [1.1] **WHEN** the synchroscope needle is approximately 2 minutes on the left hand side of the 12 o'clock position, **THEN CLOSE** Diesel Generator D Output Breaker 1816 using 0-HS-211-D/20A, DG D BKR 1816. Expected Action(s): Candidate closes the 'D' Diesel Generator Breaker to parallel with 4kV Shutdown Board D.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 9: [1.2] **RAISE** kW load using 0-HS-82-D/3A, DG D GOVERNOR CONTROL, to obtain approximately 400 kW as indicated on 0-JI-82-D/A, DG D KILOWATTS. [1.3] **RAISE** kVAR load using 0-HS-82-D/2A, DG D VOLTAGE REGULATOR CONTROL, to obtain approximately 300 kVAR lagging as indicated on 0-VAR-82-D/A, DG D KILOVARS. Expected Action(s): Candidate picks up load on 'D' DG using voltage and governor controls.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Begin Alternate Path. When load on 'D' Diesel Generator exceeds 500kW in the next step, a loss of Lube Oil will occur.	
Step 10: [2] **WHILE** maintaining approximately 0.8 lagging power factor, **PERFORM** the following concurrently to raise the Load of the Diesel Generator as follows: • **SLOWLY RAISE** kW load using 0-HS-82-D/3A, DG D GOVERNOR CONTROL, to obtain a load ≥ 2295 kW and ≤ 2550 kW as indicated on 0-JI-82-D/A, DG D KILOWATTS • **SLOWLY RAISE** kVAR load using 0-HS-82-D/2A, DG D VOLTAGE REGULATOR CONTROL, to obtain a load of at least 1725 kVAR lagging but NOT greater than 1900 kVAR lagging as indicated on 0-VAR-82-D/A, DG D KILOVARS Expected Action(s): Candidate raises kW and kVAR load on 'D' Diesel Generator.	_____ SAT _____UNSAT _____N/A
Step 11: **EXAMINER NOTE: Candidate may elect to allow the DG to continue running while referencing the ARP below.** Alarm Response Procedure, 1/2-ARP-9-23D DIESEL GENERATOR D LUBE OIL ABNORMAL, Window 4 Operator Action: **A. CHECK** Panel 0-9-23-8 to see if Low Low Lube Oil Pressure light is illuminated. Expected Action(s): Candidate checks that the 'D' Diesel Generator Low Oil Pressure Light is illuminated.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 12: B. **IF** Low Low Lube Oil pressure light illuminates continuously, **THEN SHUT DOWN** the Diesel Generator with Emergency Stop Pushbutton. Expected Action(s): Candidate observes that the 'D' Diesel Generator Low Lube Oil Pressure light is illuminated and shuts down the Diesel Generator using the Emergency Stop pushbutton.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: Once JPM Step 12 is completed, inform the candidate "Another Operator will continue this procedure."	

STOP TIME: _____

Form 3.2-3 Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 2. 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability Test, is in progress with the following conditions:

- An Assistant Unit Operator (AUO) is standing by at 'D' Diesel Generator
- 'D' Diesel Generator has been started and is running
- 0-SR-3.8.1.1(D), is complete up to Section 6.9, Step [4]

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to parallel 'D' Diesel Generator to the 'D' 4KV Shutdown Board in accordance with 0-SR-3.8.1.1(D) and load the Diesel to 2300 kW, starting at Section 6.9, Step [4].

Form 3.2-3 Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Parallel an Emergency Diesel Generator to a Shutdown Board per 3-SR-3.8.1.1(D), Diesel Generator 3D Monthly Operability Test
JPM NUMBER:	759A-U3	REVISION:	0

TASK APPLICABILITY:	☐ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-082-NO-07 / Parallel and load D/G at panel 9-23		
K/A RATINGS:	RO / SRO: 3.5		
K/A No. &STATEMENT:	264000 SF6 EGE Emergency Generators A1.10; Ability to predict and/or monitor changes in parameters associated with operation of the Emergency Generators, including: Lights and alarms		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	6		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

Validated by: _____ _____
Validator *Date*

Approved by: _____ _____
Facility Representative *Date*

NRC Approved: _____ _____
Facility Author *Date*

Form 3.2-3 Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 759A

RO _____ SRO _____ DATE: _____

TASK STANDARD: Candidate is expected to perform operations necessary to parallel an Emergency Diesel Generator (3D EDG) to its associated 4KV Shutdown Board in accordance with 3-OI-82, and respond to a Low Lube Oil Pressure alarm.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 3-SR-3.8.1.1(3D), 3-ARP-9-23D

VALIDATION TIME: 12 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/28/23	All	Initial issue

Procedure Revisions

Procedure	Revision
3-SR-3.8.1.1(D)	67
3-ARP-9-23D	32

SIMULATOR SETUP

IC	28
Exam IC	284

Console Operator Instructions	• **Reset to IC 284** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 3-SR-3.8.1.1(3D) and 3-OI-82 - ATT 1, 2 to review (another Operator will be filling it out)** • **Verify that Remote Function ED32D (4KV SD BD 3ED MODE SWITCH) is ENABLED** • **Ensure that a stopwatch and the procedure stand are staged** • **Start schedule file 2304 NRC JPM E UNIT 3.sch** • **Verify event file 2304 NRC JPM E UNIT 3.evt**

Malfunctions	Description	Event	Severity	Delay	Initial
0-XA-55-23D_4	DIESEL GEN D LUBE OIL ABNORMAL	1	ON	30	OFF
0-XA-55-23D_2	DIESEL GEN D TROUBLE	1	ON	60	OFF

Overrides	Description	Event	Severity	Delay	Initial
ZLO0IL8213D_1	LOW OIL PRESSURE DG D	1	ON	2	OFF

Schedule File(s): 2304 NRC JPM E UNIT 3.SCH
Event File(s): 2304 NRC JPM E UNIT 3.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 3. 3-SR-3.8.1.1(3D), Diesel Generator 3D Monthly Operability Test, is in progress with the following conditions:

- An Assistant Unit Operator (AUO) is standing by at 3D Diesel Generator
- 3D Diesel Generator has been started and is running
- 3-SR-3.8.1.1(3D), is complete up to Section 6.9, Step [4]

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to parallel 3D Diesel Generator to the 3D 4KV Shutdown Board in accordance with 3-SR-3.8.1.1(3D) and load the Diesel to 2300 kW, starting at Section 6.9, Step [4].

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Ensure the candidate has been pre-briefed on 3-SR-3.8.1.1(D), Diesel Generator 3D Monthly Operability Test. Step 1: [4] On Panel 3-9-23-8 **PREPARE** the Diesel Generator for paralleling as follows: [4.1] **PLACE** 3-25-211-3ED/10A, DG 3D BKR 1836 SYNC SWITCH, to ON, if necessary. Expected Action(s): Candidate places 3-25-211-3ED/10A in the ON position.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [4.2] **ENSURE** 3-XS-211-3ED, 4KV SHUTDOWN BOARD 3ED VOLTAGE SELECT, in SHUTDOWN BOARD 3ED. Expected Action(s): Candidate places 3-XS-211-3ED in the SHUTDOWN BOARD 3ED position.	_____ SAT _____UNSAT _____N/A
Step 3: [4.3] **CHECK** Shutdown Board 3ED Voltage is 3950 to 4400 VOLTS and not undergoing abnormal voltage transients using 3-EI-211-3ED, 4KV SHUTDOWN BOARD 3ED VOLTS. [4.4] **CHECK** 3-SI-211-3ECD/B, SYSTEM SYNC FREQUENCY, is 59 to 61 HZ and NOT undergoing abnormal frequency transients. Expected Action(s): Candidate verifies electrical parameters are within limits and are stable as required.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: **EXAMINER NOTE: Step [4.5] is a Concurrent Verification (CV) step. The candidate may state that the step cannot be completed without CV. Inform the candidate that verification is being completed as required.** [4.5] **PULL OUT** and **PLACE** 3-HS-82-3D/5A, DG 3D MODE SELECT, in PARALLELED WITH SYSTEM. [4.6] **CHECK** PARALLELED WITH SYSTEM light ILLUMINATED. Expected Action(s): **EXAMINER NOTE: Verifying that the PARALLELED WITH SYSTEM light is illuminated is NOT a Critical Step.** Candidate places 3-HS-82-3D/5A in PARALLELED WITH SYSTEM.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [4.7] **ADJUST** Diesel Generator 3D Frequency, using 3-HS-82-3D/3A, DG 3D GOVERNOR CONTROL, to obtain a Synchroscope needle rotating one revolution every 15 to 20 seconds in the FAST direction. Expected Action(s): Candidate adjusts Diesel Generator frequency to cause the synchroscope to rotate in the FAST direction as required.	_____ SAT _____UNSAT _____N/A
Step 6: [4.8] **ADJUST** Diesel Generator 3D Voltage (3-EI-82-3CD, GEN SYNC REF VOLTAGE) to match Shutdown Board 3ED Voltage (3-EI-211-3ECD/B, SYSTEM SYNC REF VOLTAGE) using 3-HS-82-3D/2A, DG 3D VOLT REGULATOR CONTROL. Expected Action(s): Candidate adjusts Diesel Generator Voltage to match Shutdown Board 3ED Voltage.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [5] On Diesel Generator D Protective Relay Cabinet, Panel 25-47D. (D/G Room 3D) **CHECK** Phase voltages for the PT Transformers. Expected Action(s): **DRIVER CUE: When contacted as the DG AUO to perform Step [5], report when requested that all three switch positions indicate 4.2kV.** Candidate contacts the Diesel Generator AUO for voltage readings.	_____ SAT _____UNSAT _____N/A
Step 8: Section 6.10, Paralleling and Loading D/G to the Shutdown Board [1] **PARALLEL** the Diesel Generator to the 3ED Shutdown Board by performing the following steps concurrently: [1.1] **WHEN** the synchroscope needle is approximately 2 minutes on the left hand side of the 12 o'clock position, **THEN CLOSE** Diesel Generator 3D Output Breaker 1836 using 3-HS-211-3ED/10A, DG 3D BKR 1836. Expected Action(s): Candidate closes the 3D Diesel Generator Breaker to parallel with 4kV Shutdown Board 3ED.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 9: [1.2] **RAISE** kW load using 3-HS-82-3D/3A, DG 3D GOVERNOR CONTROL, to obtain approximately 400 kW as indicated on 0-JI-82-3D/A, DG 3D KILOWATTS. [1.3] **RAISE** kVAR load using 3-HS-82-D/2A, DG 3D VOLTAGE REGULATOR CONTROL, to obtain approximately 300 kVAR lagging as indicated on 3-VAR-82-3D/A, DG 3D KILOVARS. Expected Action(s): Candidate picks up load on 3D DG using voltage and governor controls.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Begin Alternate Path. When load on 3D Diesel Generator exceeds 500kW in the next step, a Loss of Lube Oil will occur.	
Step 10: [2] **WHILE** maintaining approximately 0.8 lagging power factor, **PERFORM** the following concurrently to raise the Load of the Diesel Generator as follows: • **SLOWLY RAISE** kW load using 3-HS-82-3D/3A, DG 3D GOVERNOR CONTROL, to obtain a load ≥ 2295 kW and ≤ 2550 kW as indicated on 3-JI-82-3D/A, DG 3D KILOWATTS • **SLOWLY RAISE** kVAR load using 3-HS-82-3D/2A, DG 3D VOLTAGE REGULATOR CONTROL, to obtain a load of at least 1725 kVAR lagging but NOT greater than 1900 kVAR lagging as indicated on 3-VAR-82-3D/A, DG 3D KILOVARS Expected Action(s): Candidate raises kW and kVAR load on 3D Diesel Generator.	_____ SAT _____UNSAT _____N/A
Step 11: **EXAMINER NOTE: Candidate may elect to allow the DG to continue running while referencing the ARP below.** Alarm Response Procedure, 3-ARP-9-23D DIESEL GENERATOR D LUBE OIL ABNORMAL, Window 4 Operator Action: **A. CHECK** Panel 3-9-23 to see if Low Low Lube Oil Pressure light is illuminated. Expected Action(s): Candidate checks that the 3D Diesel Generator Low Oil Pressure Light is illuminated.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 12: B. **IF** Low Low Lube Oil pressure light illuminates continuously, **THEN SHUT DOWN** the Diesel Generator with Emergency Stop Pushbutton. Expected Action(s): Candidate observes that the 3D Diesel Generator Low Lube Oil Pressure light is illuminated and shuts down the Diesel Generator using the Emergency Stop pushbutton.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: Once JPM Step 12 is completed, inform the candidate "Another Operator will continue this procedure."	

STOP TIME: _____

Form 3.2-3 Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are the Balance of Plant Operator on Unit 3. 3-SR-3.8.1.1(3D), Diesel Generator 3D Monthly Operability Test, is in progress with the following conditions:

- An Assistant Unit Operator (AUO) is standing by at 3D Diesel Generator
- 3D Diesel Generator has been started and is running
- 3-SR-3.8.1.1(3D), is complete up to Section 6.9, Step [4]

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to parallel 3D Diesel Generator to the 3D 4KV Shutdown Board in accordance with 3-SR-3.8.1.1(3D) and load the Diesel to 2300 kW, starting at Section 6.9, Step [4].

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Alternate Fuel Pool Cooling Pumps per 2-OI-78, Fuel Pool Cooling System	
JPM NUMBER:	765A-U2	REVISION:	0	

TASK APPLICABILITY:	☒ SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-078-AB-01 / Respond to Fuel Pool Cleanup System Failure		
K/A RATINGS:	RO / SRO: 3.1 / 2.8		
K/A No. &STATEMENT:	233000 Fuel Pool Cooling and Cleanup A2.14: Ability to (a) predict the impacts of the following on the Fuel Pool Cooling and Cleanup and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Low system flow		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	9		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	7 minutes		

Validated by: _____ Validator _____ Date

Approved by: _____ Facility Representative _____ Date

NRC Approved: _____ Facility Author _____ Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 765A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to alternate Fuel Pool Cooling Pumps in accordance with 2-OI-78, Fuel Pool Cooling System, and respond to a Fuel Pool Cooling Abnormal alarm.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 2-OI-78, 2-ARP-9-4C

VALIDATION TIME: 5 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	03/02/23	ALL	Initial issue

Procedure Revisions

Procedure	Revision
2-OI-78	69
2-ARP-9-4C	38

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to Exam IC 28** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 2-OI-78** • **Start schedule file 2304 NRC JPM H.SCH** • **Verify event file 2304 NRC JPM H.EVT**

Malfunctions	Description	Event	Severity	Delay	Initial set
XA-55-4C_4	FUEL POOL SKIMMER SURGE TK LVL LO/LO-LO	1	ON	15	OFF
XA-55-4C_1	FUEL POOL SYSTEM ABNORMAL	2	ON	N/A	OFF

Schedule File(s): 2304 NRC JPM H.SCH
Event File(s): 2304 NRC JPM H.EVT

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2, with the following plant conditions:

- 2B Fuel Pool Cooling Pump maintenance is complete and is ready for PMT
- 2-SHV-078-538, Fuel Pool Cooling Pump 2B Suction Valve, is open
- Both Fuel Pool Cooling Heat Exchangers are in service
- The Reactor Building Assistant Unit Operator (AUO) is standing by for pump start

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) directs you to alternate Fuel Pool Cooling Pumps in accordance with 2-OI-78, Fuel Pool Cooling System, Section 8.7.

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **NOTIFY** Radiation Protection of impending action to swap Fuel Pool Cooling Pumps. **RECORD** time Radiation Protection notified in NOMS Narrative Log. Expected Action(s): Candidate notifies Radiation Protection.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: As Radiation Protection acknowledge any information given.	
Step 2: [2] **ENSURE** 2-SHV-078-0538, PUMP 2B SUCTION, is OPEN. Expected Action(s): It is given in the initial conditions that 2B Fuel Pool Cooling Suction Valve is open. Candidate continues in the procedure.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **ENSURE** both FPC HXs in service. Expected Action(s): It is given in the initial conditions that both Fuel Pool Cooling Heat Exchangers are in service. Candidate continues in the procedure.	_____ SAT _____UNSAT _____N/A
Step 4: [4] **START** 2-HS-78-15A, FUEL POOL COOLING PUMP 2B, on Panel 2-9-4. Expected Action(s): Candidate starts 2B Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Begin Alternate Path – when the candidate stops 2A Fuel Pool Cooling Pump in the next step, a leak from 2B Fuel Pool Cooling Pump will be simulated.	
Step 5: [5] **STOP** 2-HS-78-10A, FUEL POOL COOLING PUMP 2A, on Panel 2-9-4. Expected Action(s): Candidate stops 2A Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 6: [6] **CHECK** system parameters normal (within approximately 10 minutes of starting Fuel Pool Cooling Pump). [7] **CHECK** idle Fuel Pool Cooling Pump is NOT rotating and continue monitoring pump for rotation for the next several hours. Expected Action(s): Candidate checks system parameters and directs the AUO to verify system operation.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: If contacted as the Reactor Building AUO, acknowledge any information or direction given.	
Step 7: Alarm Response Procedure. FUEL POOL SKIMMER SURGE TANK LEVEL LO/LO-LO (2-ARP-9-4C, Window 4) A. **ENSURE** personnel are aware of Condensate Transfer Pump status AND **DISPATCH** personnel to 2-LPNL-925-0015 (EL 621', R-12, S-Line) to CHECK if the level is LOW using 2-IL-078-001B/CA, or LO-LO using 2-IL-078-001D/FA(001E/GA). B. N/A Expected Action(s): Candidate contacts the Reactor Building AUO to check Surge Tank Level.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Acknowledge any information or direction given. Report that Surge Tank Level is low. **When ready to cause the next Fuel Pool Cooling System Alarm, request that the Driver insert Event 2.**	
DRIVER NOTE: When requested by the Examiner, insert Event 2 to cause the Fuel Pool System Abnormal Alarm, Panel 2-9-4C, Window 1.	
Step 8: Alarm Response Procedure, FUEL POOL SYSTEM ABNORMAL (3-ARP-9-4C, Window 1) A. **DISPATCH** personnel to Panels 25-15 and 25-16 to determine the cause of the alarm. Expected Action(s): Candidate contacts the Reactor Building AUO to determine the cause.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: When contacted by the candidate to investigate the cause of the alarm, report that there is a significant leak from the discharge of 2B Fuel Pool Cooling Pump.	
Step 9: Steps B – E are not applicable to this JPM. Expected Action(s): Candidate marks these steps as N/A.	_____ SAT _____UNSAT _____N/A
Step 10: F. **IF** Fuel Pool Cooling Pump Discharge Pressure Low Alarm is valid **THEN SWAP** Fuel Pool Cooling Pumps, **REFER TO** 2-OI-78. 2-OI-78, Fuel Pool Cooling System Section 8.7, Swapping Fuel Pool Cooling System [4] **START** 2-HS-78-10A, FUEL POOL COOLING PUMP 2A on Panel 2-9-4. Expected Action(s): Candidate starts 2A Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 11: [5] **STOP** 2-HS-78-15A, FUEL POOL COOLING PUMP 2B on Panel 2-9-4. Expected Action(s): Candidate stops 2B Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: "Another Operator will continue this procedure".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2, with the following plant conditions:

- 2B Fuel Pool Cooling Pump maintenance is complete and is ready for PMT
- 2-SHV-078-538, Fuel Pool Cooling Pump 2B Suction Valve, is open
- Both Fuel Pool Cooling Heat Exchangers are in service
- The Reactor Building Assistant Unit Operator (AUO) is standing by for pump start

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) directs you to alternate Fuel Pool Cooling Pumps in accordance with 2-OI-78, Fuel Pool Cooling System, Section 8.7.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Alternate Fuel Pool Cooling Pumps per 3-OI-78, Fuel Pool Cooling System	
JPM NUMBER:	765A-U3	REVISION:	0	

TASK APPLICABILITY:	☐ SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-078-AB-01 / Respond to Fuel Pool Cleanup System Failure		
K/A RATINGS:	RO / SRO: 3.1 / 2.8		
K/A No. &STATEMENT:	233000 Fuel Pool Cooling and Cleanup A2.14: Ability to (a) predict the impacts of the following on the Fuel Pool Cooling and Cleanup and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Low system flow		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	9		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	7 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 765A-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to alternate Fuel Pool Cooling Pumps in accordance with 3-OI-78, Fuel Pool Cooling System, and respond to a Fuel Pool Cooling Abnormal alarm.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 3-OI-78, 3-ARP-9-4C

VALIDATION TIME: 5 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	03/02/23	ALL	Initial issue

Procedure Revisions

Procedure	Revision
3-OI-78	64
3-ARP-9-4C	46

SIMULATOR SETUP

IC	28
Exam IC	N/A

Console Operator Instructions	• **Reset to Exam IC 28** • **Place the simulator in RUN to ensure stable conditions** • **Ensure the candidate has been pre-briefed on 3-OI-78** • **Start schedule file 2304 NRC JPM H.SCH** • **Verify event file 2304 NRC JPM H.EVT**

Malfunctions	Description	Event	Severity	Delay	Initial set
XA-55-4C_4	FUEL POOL SKIMMER SURGE TK LVL LO/LO-LO	1	ON	15	OFF
XA-55-4C_1	FUEL POOL SYSTEM ABNORMAL	2	ON	N/A	OFF

Schedule File(s): 2304 NRC JPM H.SCH
Event File(s): 2304 NRC JPM H.EVT

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3, with the following plant conditions:

- 3B Fuel Pool Cooling Pump maintenance is complete and is ready for PMT
- 3-SHV-078-538, Fuel Pool Cooling Pump 3B Suction Valve, is open
- Both Fuel Pool Cooling Heat Exchangers are in service
- The Reactor Building Assistant Unit Operator (AUO) is standing by for pump start

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) directs you to alternate Fuel Pool Cooling Pumps in accordance with 3-OI-78, Fuel Pool Cooling System, Section 8.7.

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **NOTIFY** Radiation Protection of impending action to swap Fuel Pool Cooling Pumps. **RECORD** time Radiation Protection notified in NOMS Narrative Log. Expected Action(s): Candidate notifies Radiation Protection.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: As Radiation Protection acknowledge any information given.	
Step 2: [2] **ENSURE** 3-SHV-078-0538, PUMP 3B SUCTION, is OPEN. Expected Action(s): It is given in the initial conditions that 3B Fuel Pool Cooling Suction Valve is open. Candidate continues in the procedure.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **ENSURE** both FPC HXs in service. Expected Action(s): It is given in the initial conditions that both Fuel Pool Cooling Heat Exchangers are in service. Candidate continues in the procedure.	_____ SAT _____UNSAT _____N/A
Step 4: [4] **START** 3-HS-78-15A, FUEL POOL COOLING PUMP 3B, on Panel 3-9-4. Expected Action(s): Candidate starts 3B Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Begin Alternate Path – when the candidate stops 3A Fuel Pool Cooling Pump in the next step, a leak from 3B Fuel Pool Cooling Pump will be simulated.	
Step 5: [5] **STOP** 3-HS-78-10A, FUEL POOL COOLING PUMP 3A, on Panel 3-9-4. Expected Action(s): Candidate stops 3A Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 6: [6] **CHECK** system parameters normal (within approximately 10 minutes of starting Fuel Pool Cooling Pump). [7] **CHECK** idle Fuel Pool Cooling Pump is **NOT** rotating and continue monitoring pump for rotation for the next several hours. Expected Action(s): Candidate checks system parameters and directs the AUO to verify system operation.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: If contacted as the Reactor Building AUO, acknowledge any information or direction given.	
Step 7: Alarm Response Procedure, FUEL POOL SKIMMER SURGE TANK LEVEL LO/LO-LO (3-ARP-9-4C, Window 4) A. **ENSURE** personnel are aware of Condensate Transfer Pump status AND **DISPATCH** personnel to 3-LPNL-925-0015 (EL 621', R-19, S-Line) to CHECK if the level is LOW using 3-IL-078-001B/CA, or LO-LO using 3-IL-078-001D/FA(001E/GA). B. N/A Expected Action(s): Candidate contacts the Reactor Building AUO to check Surge Tank Level.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Acknowledge any information or direction given. Report that Surge Tank Level is low. **When ready to cause the next Fuel Pool Cooling System Alarm, request that the Driver insert Event 2.**	
DRIVER NOTE: When requested by the Examiner, insert Event 2 to cause the Fuel Pool System Abnormal Alarm, Panel 3-9-4C, Window 1.	
Step 8: Alarm Response Procedure, FUEL POOL SYSTEM ABNORMAL (3-ARP-9-4C, Window 1) A. **DISPATCH** personnel to Panels 25-15 and 25-16 to determine the cause of the alarm. Expected Action(s): Candidate contacts the Reactor Building AUO to determine the cause.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: When contacted by the candidate to investigate the cause of the alarm, report that there is a significant leak from the discharge of 3B Fuel Pool Cooling Pump.	
Step 9: Steps B – E are not applicable to this JPM. Expected Action(s): Candidate marks these steps as N/A.	_____ SAT _____UNSAT _____N/A
Step 10: F. **IF** Fuel Pool Cooling Pump Discharge Pressure Low Alarm is valid **THEN SWAP** Fuel Pool Cooling Pumps, **REFER TO** 3-OI-78. 3-OI-78, Fuel Pool Cooling System Section 8.7, Swapping Fuel Pool Cooling System [4] **START** 3-HS-78-10A, FUEL POOL COOLING PUMP A on Panel 3-9-4. Expected Action(s): Candidate starts 3A Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 11: [5] **STOP** 3-HS-78-15A, FUEL POOL COOLING PUMP 3B on Panel 3-9-4. Expected Action(s): Candidate stops 3B Fuel Pool Cooling Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: "Another Operator will continue this procedure".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3, with the following plant conditions:

- 3B Fuel Pool Cooling Pump maintenance is complete and is ready for PMT
- 3-SHV-078-538, Fuel Pool Cooling Pump 3B Suction Valve, is open
- Both Fuel Pool Cooling Heat Exchangers are in service
- The Reactor Building Assistant Unit Operator (AUO) is standing by for pump start

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) directs you to alternate Fuel Pool Cooling Pumps in accordance with 3-OI-78, Fuel Pool Cooling System, Section 8.7.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Start a Condenser Circulating Water Pump per OI-27, Condenser Circulating Water System
JPM NUMBER:	757A-U2	REVISION:	0

TASK APPLICABILITY:	☐ SRO-U	☐ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-027-NO-05 / Coordinate Returning a CCW Pump to Service		
K/A RATINGS:	RO / SRO 3.3		
K/A No. &STATEMENT:	400000 SF8 CCW Component Cooling Water System A3.07: Ability to monitor automatic operations of the Component Cooling Water System, including: System realignments		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	8		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	6 minutes		

Validated by: _____ Validator _____ Date

Approved by: _____ Facility Representative _____ Date

NRC Approved: _____ Facility Author _____ Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 757A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate will start a Condenser Circulating Water (CCW) Pump in accordance with 2-OI-27, Condenser Circulating Water System, and respond to a failure of the pump discharge valve to automatically open.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 2-OI-27

VALIDATION TIME: 4 min

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	022323	All	Initial issue

Procedure Revisions

Procedure	Revision
2-OI-27	102

SIMULATOR SETUP

IC	**28**
Exam IC	**279**

Console Operator Instructions	• **Reset to IC 279** • **Ensure Candidate has been pre-briefed on 2-OI-27** • **Place the simulator in RUN to ensure stable conditions** • **Run schedule file 2304 NRC JPM G.SCH** • **Verify event file 2304 NRC JPM G.EVT**

Malfunctions	Description	Event	Severity	Delay	Initial set
FCV-27-13	MOTOR_OPERATED_VALVE CCW PUMP A DISCH	N/A	FAIL_POWER_NOW	0	N/A
ZLOHS2713A_1	GREEN CCW PUMP 3A DISCH ISOL VALVE	N/A	ON	0	N/A
ZLOHS2713A_2	RED CCW PUMP 3A DISCH ISOL VALVE	N/A	OFF	0	N/A

Schedule File(s): 2304 NRC JPM G.SCH
Event File(s): 2304 NRC JPM G.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2, with the following conditions:

- 2A CCW Pump was shut down for maintenance and is ready for restart
- Cooling Towers are NOT in service

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to start 2A CCW Pump in accordance with 2-OI-27, Condenser Circulating Water System, Section 8.2, Returning a CCW Pump to Service, starting at Step [5] (Steps [1] – [4] are complete).

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [5] **ENSURE CLOSED** 2-FCV-27-13, CCW PUMP 2A DISCHARGE ISOLATION VALVE, on Panel 2-9-20. Expected Action(s): Candidate ensures that 2-FCV-27-12 is closed.	_____ SAT _____UNSAT _____N/A
Step 2: [6] **IF** Cooling Towers are in service, **THEN ESTABLISH** communications between Control Room and Cooling Tower Operators. Warm water channel level will be affected by changes in CCW System flow. Cooling Tower Operators must be allowed time to respond to these changing conditions. **REFER TO** 0-OI-27C, Cooling Tower System. Expected Action(s): It is given in the operator cues that Cooling Towers are not in service, therefore this step is marked as N/A.	_____ SAT _____UNSAT _____N/A
Step 3: [7] **START** CCW PUMP 2A using 2-HS-27-10A on Panel 2-9-20. Expected Action(s): Candidate starts 2A CCW Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Begin Alternate Path – when 2A CCW Pump is started, the CCW Pump Discharge Valve will fail to automatically open.	
Step 4: [7 (continued)] **ENSURE** the respective 2-FCV-27-13, CCW PUMP 2A DISCHARGE VALVE, automatically travels to the fully open position. Expected Action(s): Candidate determines that 2-FCV-27-13 did not automatically open, and manually opens the 2A CCW Pump Discharge Valve.	**Critical Step** _____ SAT _____UNSAT _____N/A
Cue: "Another Operator will continue this procedure".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 2, with the following conditions:

- 2A CCW Pump was shut down for maintenance and is ready for restart
- Cooling Towers are NOT in service

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to start 2A CCW Pump in accordance with 2-OI-27, Condenser Circulating Water System, Section 8.2, Returning a CCW Pump to Service, starting at Step [5] (Steps [1] – [4] are complete).

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Start a Condenser Circulating Water Pump per OI-27, Condenser Circulating Water System
JPM NUMBER:	757A-U3	REVISION:	0

TASK APPLICABILITY:	☐ SRO-U	☐ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-027-NO-05 / Coordinate Returning a CCW Pump to Service		
K/A RATINGS:	RO / SRO 3.3		
K/A No. &STATEMENT:	400000 SF8 CCW Component Cooling Water System A3.07: Ability to monitor automatic operations of the Component Cooling Water System, including: System realignments		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	8		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☒ YES	☐ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	6 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 757A-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate will start a Condenser Circulating Water (CCW) Pump in accordance with 3-OI-27, Condenser Circulating Water System, and respond to a failure of the pump discharge valve to automatically open.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

REFERENCES/PROCEDURES NEEDED: 3-OI-27

VALIDATION TIME: 4 min

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	030123	All	Initial issue

Procedure Revisions

Procedure	Revision
3-OI-27	85

SIMULATOR SETUP

IC	28
Exam IC	253

Console Operator Instructions	• **Reset to IC 253** • **Ensure Candidate has been pre-briefed on 3-OI-27** • **Place the simulator in RUN to ensure stable conditions** • **Run schedule file 2304 NRC JPM G.SCH** • **Verify event file 2304 NRC JPM G.EVT**

Malfunctions	Description	Event	Severity	Delay	Initial set
FCV-27-13	MOTOR_OPERATED_VALVE CCW PUMP A DISCH	N/A	FAIL_POWER_NOW	0	N/A
ZLOHS2713A_1	GREEN CCW PUMP 3A DISCH ISOL VALVE	N/A	ON	0	N/A
ZLOHS2713A_2	RED CCW PUMP 3A DISCH ISOL VALVE	N/A	OFF	0	N/A

Schedule File(s): 2304 NRC JPM G.SCH
Event File(s): 2304 NRC JPM G.EVT

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3, with the following conditions:

- 3A CCW Pump was shut down for maintenance and is ready for restart
- Cooling Towers are NOT in service

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to start 3A CCW Pump in accordance with 3-OI-27, Condenser Circulating Water System, Section 8.2, Returning a CCW Pump to Service, starting at Step [5] (Steps [1] – [4] are complete).

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [5] **ENSURE CLOSED** 3-FCV-27-13, CCW PUMP 3A DISCHARGE ISOLATION VALVE, on Panel 3-9-20. Expected Action(s): Candidate ensures that 3-FCV-27-12 is closed.	_____ SAT _____UNSAT _____N/A
Step 2: [6] **IF** Cooling Towers are in service, **THEN ESTABLISH** communications between Control Room and Cooling Tower Operators. Warm water channel level will be affected by changes in CCW System flow. Cooling Tower Operators must be allowed time to respond to these changing conditions. **REFER TO** 0-OI-27C, Cooling Tower System. Expected Action(s): It is given in the operator cues that Cooling Towers are not in service, therefore this step is marked as N/A.	_____ SAT _____UNSAT _____N/A
Step 3: [7] **START** CCW PUMP 3A using 3-HS-27-10A on Panel 3-9-20. Expected Action(s): Candidate starts 3A CCW Pump.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: Begin Alternate Path – when 3A CCW Pump is started, the CCW Pump Discharge Valve will fail to automatically open.	
Step 4: [7 (continued)] **ENSURE** the respective 3-FCV-27-13, CCW PUMP 3A DISCHARGE VALVE, automatically travels to the fully open position. Expected Action(s): Candidate determines that 3-FCV-27-13 did not automatically open, and manually opens the 3A CCW Pump Discharge Valve.	**Critical Step** _____ SAT _____UNSAT _____N/A
Cue: "Another Operator will continue this procedure".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct" (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 3, with the following conditions:

- 3A CCW Pump was shut down for maintenance and is ready for restart
- Cooling Towers are NOT in service

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) has directed you to start 3A CCW Pump in accordance with 3-OI-27, Condenser Circulating Water System, Section 8.2, Returning a CCW Pump to Service, starting at Step [5] (Steps [1] – [4] are complete).

Job Performance Measure (JPM)

SITE: BFN		JPM TITLE:	Recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus
JPM NUMBER:	748-U2	REVISION:	1

TASK APPLICABILITY:	☐ SRO-U	☐ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-099-AB-01 / Respond to Loss of Power to One RPS Bus		
K/A RATINGS:	RO/SRO: 3.9		
K/A STATEMENT:	212000 Reactor Protection System A2.01: Ability to (a) predict the impacts of the following on the Reactor Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: RPS motor-generator set failure		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	7		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ Validator _____ Date

Approved by: _____ Facility Representative _____ Date

NRC Approved: _____ Facility Author _____ Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 748-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 2-AOI-99-1

VALIDATIONTIME: 6 min

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	11/9/2020	All	Initial issue
1	2/23/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-AOI-99-1	31

SIMULATOR SETUP

IC	28
Exam IC	283

Console Operator Instructions	• Reset to IC 283 • Place the simulator in RUN to ensure stable conditions

Malfunctions	Description	Event	Severity	Delay	Initial set
NONE					

Schedule File(s): N/A
Event File(s): N/A

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 2
- Unit 2 is operating at 100% Reactor Power
- 2A RPS Motor Generator tripped, and 2A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 2 Nuclear Unit Senior Operator (NUSO) has directed you to continue with restoring affected systems to normal in accordance with 2-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: 2-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12] [12] RESET the RPS trip logic half SCRAM at Panel 2-9-5 as follows: **NOTE** The eight CONTROL ROD TEST SCRAM SOLENOID GROUP A and B LIGHTS SHOULD ILLUMINATE. [12.1] **MOMENTARILY PLACE** 2-HS-99-5A-S5, SCRAM RESET as follows: [12.2] RESET FIRST position. (Group 2/3). [12.3] RESET SECOND position. (Group 1/4). [12.4] NORMAL position. Expected Action(s): Candidate resets the RPS trip logic half SCRAM at Panel 2-9-5 by placing 2-HS-99-5A-S5, SCRAM RESET in RESET FIRST position (Group 2/3), then RESET SECOND position (Group 1/4), and finally back to the NORMAL position.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [13] **VERIFY** the following: [13.1] All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED. Expected Action(s): Candidate verifies all eight SCRAM solenoid logic reset lights are illuminated.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [13.2] The following four lights ILLUMINATED: [13.2.1] 2-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE. [13.2.2] 2-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE. Expected Action(s): Candidate verifies that the A and B Backup SCRAM valve lights are illuminated.	_____SAT _____UNSAT _____N/A
Step 4: [13.3] Scram Discharge Volume Vent and Drain Valves indicate OPEN. Expected Action(s): Candidate verifies the eight (8) Scram Discharge Volume Vent and Drain Valves indicate open.	_____ SAT _____UNSAT _____N/A
Step 5: [14] **RESET** PCIS trip logic at Panel 2-9-4 as follows: [14.1] **MOMENTARILY** PLACE 2-HS-64-16A-S32, PCIS DIV I RESET, to left and right RESET positions. [14.2] **VERIFY** the following red lights ILLUMINATED: [14.2.1] 2-IL-64-A1, MSIV GROUP A1. [14.2.2] 2-IL-64-B1, MSIV GROUP B1. **EXAMINER NOTE: Verifying the red lights for 2-IL-64-A1, MSIV GROUP A1 and 2-IL-64-B1, MSIV GROUP B1 are ILLUMINATED, is NOT a Critical Step.** Expected Action(s): Candidate places 2-HS-64-16A-S32, PCIS DIV I RESET, on Panel 2-9-4, to left and right RESET positions.	**Critical Step** _____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 6: [14.3] **MOMENTARILY** PLACE 2-HS-64-16A-S33, PCIS DIV II RESET, to left and right RESET positions. [14.4] **VERIFY** the following red lights ILLUMINATED: [14.4.1] 2-IL-64-A2, MSIV GROUP A2. [14.4.2] 2-IL-64-B2, MSIV GROUP B2. **EXAMINER NOTE: Verifying the red lights for 2-IL-64-A2, MSIV GROUP A2 and 2-IL-64-B2, MSIV GROUP B2 are ILLUMINATED, is NOT a Critical Step.** Expected Action(s): Candidate places 2-HS-64-16A-S33, PCIS DIV II RESET, on Panel 2-9-4, to left and right RESET positions.	**Critical Step** _____SAT _____UNSAT _____N/A
EXAMINER CUE: Once Step [14.4] is completed, inform the Candidate "Another Operator will continue the procedure. "	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 2
- Unit 2 is operating at 100% Reactor Power
- 2A RPS Motor Generator tripped, and 2A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 2 Nuclear Unit Senior Operator (NUSO) has directed you to continue with restoring affected systems to normal in accordance with 2-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus
JPM NUMBER:	748-U3	REVISION:	1

TASK APPLICABILITY:	☐ SRO-U	☐ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-099-AB-01 / Respond to Loss of Power to One RPS Bus		
K/A RATINGS:	RO/SRO: 3.9		
K/A STATEMENT:	212000 Reactor Protection System A2.01: Ability to (a) predict the impacts of the following on the Reactor Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: RPS motor-generator set failure		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	7		

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

EVALUATION LOCATION:	☐ In-Plant	☒ Simulator	☐ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 748-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Candidate is expected to recover from a loss of RPS in accordance with AOI-99-1, Loss of Power to One RPS Bus.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling plant evolutions precisely.

PRA: NA

REFERENCES/PROCEDURES NEEDED: 3-AOI-99-1

VALIDATIONTIME: 6 min

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	11/9/2020	All	Initial issue
1	2/23/23	All	JPM update

Procedure Revisions

Procedure	Revision
3-AOI-99-1	20

SIMULATOR SETUP

IC	28
Exam IC	283

Console Operator Instructions	• Reset to IC 283 • Place the simulator in RUN to ensure stable conditions

Malfunctions	Description	Event	Severity	Delay	Initial set
NONE					

Schedule File(s): N/A
Event File(s): N/A

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 3
- Unit 3 is operating at 100% Reactor Power
- 3A RPS Motor Generator tripped, and 3A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 3 Nuclear Unit Senior Operator (NUSO) has directed you to continue with restoring affected systems to normal in accordance with 3-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: 3-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12] [12] RESET the RPS trip logic half SCRAM at Panel 3-9-5 as follows: [12.1] **MOMENTARILY PLACE** 3-HS-99-5A-S5, SCRAM RESET as follows: [12.2] RESET FIRST position. (Group 2/3). [12.2] RESET SECOND position. (Group 1/4). [12.4] NORMAL. Expected Action(s): Candidate resets the RPS trip logic half SCRAM at Panel 3-9-5 by placing 3-HS-99-5A-S5, SCRAM RESET in RESET FIRST position (Group 2/3), then RESET SECOND position (Group 1/4), and finally back to the NORMAL position	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [13] **CHECK** the following conditions: [13.1] All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED. Expected Action(s): Candidate verifies all eight SCRAM solenoid logic reset lights are illuminated.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [13.2] The following four lights ILLUMINATED: [13.2.1] 3-IL-99-5A/AB, SYSTEM A BACKUP SCRAM VALVE. [13.2.2] 3-IL-99-5A/CD, SYSTEM B BACKUP SCRAM VALVE. Expected Action(s): Candidate verifies that the A and B Backup SCRAM valve lights are illuminated.	_____SAT _____UNSAT _____N/A
Step 4: **EXAMINER NOTE: The candidate may elect to locate the ICS computer points SOE33 and SOE035 via various menus (any method is acceptable).** [13.3] Scram Discharge Volume Vent and Drain Valves indicate OPEN. [13.4] Points SOE033 (Channel A3 manual scram) and SOE035 (Channels A1&A2 Auto Scram) on ICS computer or on the First Out Printer reads “NOT TRIP” for RPS “A”. [13.5] N/A Expected Action(s): Candidate verifies the eight (8) Scram Discharge Volume Vent and Drain Valves indicate open. Checks ICS computer points SOE033 and SOE035 or the First Out Printer reads ‘NOT TRIP’ for RPS 2A.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 5: [14] **RESET** PCIS trip logic at Panel 3-9-4 as follows: [14.1] **MOMENTARILY PLACE** 3-HS-64-16A-S32, PCIS DIV I RESET, to left and right RESET positions. [14.2] **CHECK** the following red lights ILLUMINATED: [14.2.1] 3-IL-64-A1, MSIV GROUP A1. [14.2.2] 3-IL-64-B1, MSIV GROUP B1. **EXAMINER NOTES: Verifying the red lights for 3-IL-64-A1, MSIV GROUP A1 and 3-IL-64-B1, MSIV GROUP B1 are ILLUMINATED, is NOT a Critical Step.** Expected Action(s): Candidate places 3-HS-64-16A-S32, PCIS DIV I RESET, on Panel 3-9-4, to left and right RESET positions.	**Critical Step** _____SAT _____UNSAT _____N/A
Step 6: [14.3] **MOMENTARILY PLACE** 3-HS-64-16A-S33, PCIS DIV II RESET, to left and right RESET positions. [14.4] **CHECK** the following red lights ILLUMINATED: [14.4.1] 3-IL-64-A2, MSIV GROUP A2. [14.4.2] 3-IL-64-B2, MSIV GROUP B2. **EXAMINER NOTES: Verifying the red lights for 3-IL-64-A2, MSIV GROUP A2 and 3-IL-64-B2, MSIV GROUP B2 are ILLUMINATED, is NOT a Critical Step.** Expected Action(s): Candidate places 3-HS-64-16A-S33, PCIS DIV II RESET, on Panel 3-9-4, to left and right RESET positions.	**Critical Step** _____SAT _____UNSAT _____N/A
EXAMINER CUE: **Once Step [14.4] is completed, inform the Candidate "Another Operator will continue the procedure. "**	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-SIMULATOR: I will explain the initial conditions and state the task to be performed. I will provide initiating cues and reports on other actions when directed by you. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's Correct". (OR "That's Incorrect", if applicable). When you have completed your assigned task, you will say, "my task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

- You are an Operator on Unit 3
- Unit 3 is operating at 100% Reactor Power
- 3A RPS Motor Generator tripped, and 3A RPS Bus has been placed on alternate

INITIATING CUES: The Unit 3 Nuclear Unit Senior Operator has directed you to continue with restoring affected systems to normal in accordance with 3-AOI-99-1, Loss of Power to One RPS Bus, Section 4.2 [12].

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	1-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS
JPM NUMBER:	758-U1	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-86 / Bypass RCIC High RPV Water Level Turbine Trip, EOI APPENDIX-16H		
K/A RATINGS:	RO / SRO: 3.9		
K/A No. &STATEMENT:	295008 High Reactor Water Level AA1.05: Ability to operate and/or monitor the following as they apply to High Reactor Water Level: RCIC		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	2		

EVALUATION LOCATION:	☒ In-Plant	☐ Simulator	☐ Classroom
METHOD OF TESTING:	☒ Simulated Performance	☐ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ Validator _____ Date

Approved by: _____ Facility Representative _____ Date

NRC Approved: _____ Facility Author _____ Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 758-U1

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Examinee is expected to perform 1-EOI Appendix-16H, Bypassing Reactor Core Isolation (RCIC) High RPV Water Level Shutdown Interlocks, to use RCIC for injection.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 1-EOI APPENDIX-16H

VALIDATIONTIME: 8 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/24/23	All	Initial issue

Procedure Revisions

Procedure	Revision
1-EOI APPENDIX-16H	0

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
LOGISTICS	Staff escort candidate between staging and exam location

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 1 with the following conditions:

- Unit 1 Reactor has SCRAMMED due to a leak in the Drywell
- Reactor Water Level **CANNOT** be determined
- 1-EOI-1, RPV CONTROL MODES 1-3, has been entered and followed to C4-RPV FLOODING, Step C4-5 RPV Injection
- 1-EOI Appendix-7H, Alternate RPV Injection System Lineup RCIC Using Auxiliary Steam, is being performed
- You already have a hand-held radio

INITIATING CUES:

The NUSO directs you to perform 1-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: **Examinee has demonstrated obtaining procedures on the simulator, therefore, just hand him/her the procedure, however, verify that they locate the EOI Equipment Storage Box (Nearest to Unit 1 Reactor Building, 250V RMOV Board 1C)**	
Step 1: 1-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS **1. DISPATCH** an Operator to 1-BKR-071-0008, RCIC TURBINE STEAM SUPPLY VALVE FCV-71-8 to perform the following (250V RMOV Board 1C, Compartment 4B): a. WHEN……..Requested by Unit Operator, THEN……**PLACE** 1-HS-071-0008C, RCIC TURBINE STEAM SUPPLY VALVE, in the OPEN position. Expected action(s): **EXAMINER CUE: Once the Examinee locates Unit 1 Reactor Building Elevation 565', 250V RMOV Board 1C, Compartment 4B, then 1-BKR-071-0008, RCIC TURBINE STEAM SUPPLY VALVE FCV-71-8, using the simulated hand-held radio, state:** **The Unit 1 Operator directs you to PLACE 1-HS-071-0008C, RCIC TURBINE STEAM SUPPLY VALVE, in the OPEN position.** **EXAMINER CUE: As switches are simulated, inform candidate that the switch has been positioned as performed. When correctly simulated, the green valve position indicating lamp for 1-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated. The Unit Operator will be notified in JPM Step 3 below.** Examinee proceeds to the Unit 1 Reactor Building Elevation 565' (NW corner on West wall), 250V RMOV Board 1C, locates Compartment 4B, then 1-BKR-071-0008, RCIC TURBINE STEAM SUPPLY VALVE FCV-71-8. Examinee then simulates positioning switches as required by procedure to place 1-HS-071-0008C, RCIC TURBINE STEAM SUPPLY VALVE, in the OPEN position.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: b. **PLACE** 1-XS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE TRANSFER SWITCH, to EMERG position. Expected action(s): **EXAMINER CUE: As switch is simulated being taken to EMERG, VALVE MOTOR ON red light will illuminate while motor is energized.** Examinee maintains location on the Unit 1 Reactor Building Elevation 565', 250V RMOV Board 1C, Compartment 4B, then simulates positioning switches as required by procedure to place 1-XS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE TRANSFER SWITCH, to the EMERG position.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 3: **2.** WHEN……. 1-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN, THEN…… **NOTIFY** Unit Operator Expected action(s): **EXAMINER CUE: When correctly simulated, the green valve position indicating lamp for 1-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated.** **Once notified by the Examinee, acknowledge as Unit 1 Operator that 1-FCV-071-8, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN position.** Examinee observes that 1-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE, is in the OPEN position by verifying that the green valve position indicating lamp for 1-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated. Once verified, the Examinee notifies the Unit 1 Operator that 1-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN position.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: After completion of JPM Step 3, inform the examinee, this completes your task.	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 1 with the following conditions:

- Unit 1 Reactor has SCRAMMED due to a leak in the Drywell
- Reactor Water Level **CANNOT** be determined
- 1-EOI-1, RPV CONTROL MODES 1-3, has been entered and followed to C4-RPV FLOODING, Step C4-5 RPV Injection
- 1-EOI Appendix-7H, Alternate RPV Injection System Lineup RCIC Using Auxiliary Steam, is being performed
- You already have a hand-held radio

INITIATING CUES:

The NUSO directs you to perform 1-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	2-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS
JPM NUMBER:	758-U2	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-86 / Bypass RCIC High RPV Water Level Turbine Trip, EOI APPENDIX-16H		
K/A RATINGS:	RO / SRO: 3.9		
K/A No. &STATEMENT:	295008 High Reactor Water Level AA1.05: Ability to operate and/or monitor the following as they apply to High Reactor Water Level: RCIC		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	2		

EVALUATION LOCATION:	☒ In-Plant	☐ Simulator	☐ Classroom
METHOD OF TESTING:	☒ Simulated Performance	☐ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ Validator _____ Date

Approved by: _____ Facility Representative _____ Date

NRC Approved: _____ Facility Author _____ Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 758-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Examinee is expected to perform 2-EOI Appendix-16H, Bypassing Reactor Core Isolation (RCIC) High RPV Water Level Shutdown Interlocks, to use RCIC for injection.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 2-EOI APPENDIX-16H

VALIDATIONTIME: 8 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/24/23	All	Initial issue

Procedure Revisions

Procedure	Revision
2-EOI APPENDIX-16H	6

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
LOGISTICS	Staff escort candidate between staging and exam location

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or "That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following conditions:

- Unit 2 Reactor has SCRAMMED due to a leak in the Drywell
- Reactor Water Level **CANNOT** be determined
- 2-EOI-1, RPV CONTROL MODES 1-3, has been entered and followed to C4-RPV FLOODING, Step C4-5 RPV Injection
- 2-EOI Appendix-7H, Alternate RPV Injection System Lineup RCIC Using Auxiliary Steam, is being performed
- You already have a hand-held radio

INITIATING CUES:

The NUSO directs you to perform 2-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: **Examinee has demonstrated obtaining procedures on the simulator, therefore, just hand him/her the procedure, however, verify that they locate the EOI Equipment Storage Box (Nearest to Unit 2 Reactor Building, 250V RMOV Board 2C)**	
Step 1: 2-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS **1. PLACE** 2-XS-071-0008, RCIC TURBINE STEAM SUPPLY EMERGENCY TRANSFER SWITCH, to EMERG position. (250V RMOV Board 2C, Compartment 4B). Expected action(s): **EXAMINER CUE: As switch is simulated being taken to EMERG, VALVE MOTOR ON red light will illuminate while motor is energized.** Examinee proceeds to Unit 2 Reactor Building Elevation 565' (NW corner on West wall), 250V RMOV Board 2C, locates Compartment 4B, then simulates positioning switches as required by procedure to place 2-XS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE TRANSFER SWITCH, to the EMERG position.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: **EXAMINER CUE: Once the Examinee locates Unit 2 Reactor Building Elevation 565' (NW corner on West wall), 250V RMOV Board 2C, Compartment 4B, then 2-BKR-071-0008, RCIC TURBINE STEAM SUPPLY VALVE FCV-71-8, using the simulated hand-held radio, state:** **The Unit 2 Operator directs you to PLACE 2-HS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE, in the OPEN position.** **EXAMINER CUE: As switches are simulated, inform candidate that the switch has been positioned as performed. When correctly simulated, the green valve position indicating lamp for 2-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated. The Unit Operator will be notified in JPM Step 3 below.** **2. WHEN** Requested by Unit Operator, **THEN** **OPEN** 2-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE (250V RMOV Board 2C, Compartment 4B) Expected action(s): Examinee proceeds to the Unit 2 Reactor Building Elevation 565' (NW corner on West wall), 250V RMOV Board 2C, locates Compartment 4B, then 2-BKR-071-0008, RCIC TURBINE STEAM SUPPLY VALVE FCV-71-8. Examinee then simulates positioning switches as required by procedure to place 2-HS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE, in the OPEN position.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: **3. WHEN** 2-FCV-071-8, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN, **THEN**…… **NOTIFY** Unit Operator Expected action(s): **EXAMINER CUE: When correctly simulated, the green valve position indicating lamp for 2-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated.** **Once notified by the Examinee, acknowledge as Unit 2 Operator that 2-FCV-071-8, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN position.** Examinee observes that 2-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE, is in the OPEN position by verifying that the green valve position indicating lamp for 2-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated. Once verified, the Examinee notifies the Unit 2 Operator that 2-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN position.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: After completion of JPM Step 3, inform the examinee, this completes your task.	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 2 with the following conditions:

- Unit 2 Reactor has SCRAMMED due to a leak in the Drywell
- Reactor Water Level **CANNOT** be determined
- 2-EOI-1, RPV CONTROL MODES 1-3, has been entered and followed to C4-RPV FLOODING, Step C4-5 RPV Injection
- 2-EOI Appendix-7H, Alternate RPV Injection System Lineup RCIC Using Auxiliary Steam, is being performed
- You already have a hand-held radio

INITIATING CUES:

The NUSO directs you to perform 2-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	3-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS
JPM NUMBER:	758-U3	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒ SRO-I	☒ UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-86 / Bypass RCIC High RPV Water Level Turbine Trip, EOI APPENDIX-16H		
K/A RATINGS:	RO / SRO: 3.9		
K/A No. &STATEMENT:	295008 High Reactor Water Level AA1.05: Ability to operate and/or monitor the following as they apply to High Reactor Water Level: RCIC		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	2		

EVALUATION LOCATION:	☒ In-Plant	☐ Simulator	☐ Classroom
METHOD OF TESTING:	☒ Simulated Performance	☐ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ Validator _____ Date

Approved by: _____ Facility Representative _____ Date

NRC Approved: _____ Facility Author _____ Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 758-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Examinee is expected to perform 3-EOI Appendix-16H, Bypassing Reactor Core Isolation (RCIC) High RPV Water Level Shutdown Interlocks, to use RCIC for injection.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 3-EOI APPENDIX-16H

VALIDATIONTIME: 8 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/24/23	All	Initial issue

Procedure Revisions

Procedure	Revision
3-EOI APPENDIX-16H	2

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
LOGISTICS	Staff escort candidate between staging and exam location

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following conditions:

- Unit 3 Reactor has SCRAMMED due to a leak in the Drywell
- Reactor Water Level **CANNOT** be determined
- 3-EOI-1, RPV CONTROL MODES 1-3, has been entered and followed to C4-RPV FLOODING, Step C4-5 RPV Injection
- 3-EOI Appendix-7H, Alternate RPV Injection System Lineup RCIC Using Auxiliary Steam, is being performed
- You already have a hand-held radio

INITIATING CUES:

The NUSO directs you to perform 3-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: **Examinee has demonstrated obtaining procedures on the simulator, therefore, just hand him/her the procedure, however, verify that they locate the EOI Equipment Storage Box (Nearest to Unit 3 Reactor Building, 250V RMOV Board 3C)**	
Step 1: 3-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS **1. PLACE** 3-XS-071-0008, RCIC TURBINE STEAM SUPPLY EMERGENCY TRANSFER SWITCH, to EMERG position. (250V RMOV Board 3C, Compartment 4B). Expected action(s): **EXAMINER CUE: As switch is simulated being taken to EMERG, VALVE MOTOR ON red light will illuminate while motor is energized.** Examinee proceeds to Unit 3 Reactor Building Elevation 565' (NW corner on West wall), 250V RMOV Board 3C, locates Compartment 4B, then simulates positioning switches as required by procedure to place 3-XS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE TRANSFER SWITCH, to the EMERG position.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 2: **EXAMINER CUE: Once the Examinee locates Unit 3 Reactor Building Elevation 565' (NW corner on West wall), 250V RMOV Board 3C, Compartment 4B, then 3-BKR-071-0008, RCIC TURBINE STEAM SUPPLY VALVE FCV-71-8, using the simulated hand-held radio, state:** **The Unit 3 Operator directs you to PLACE 3-HS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE, in the OPEN position.** **EXAMINER CUE: As switches are simulated, inform candidate that the switch has been positioned as performed. When correctly simulated, the green valve position indicating lamp for 3-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated. The Unit Operator will be notified in JPM Step 3 below.** **2. WHEN** Requested by Unit Operator, **THEN** **OPEN** 3-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE (250V RMOV Board 3C, Compartment 4B) Expected action(s): Examinee proceeds to the Unit 3 Reactor Building Elevation 565' (NW corner on West wall), 250V RMOV Board 3C, locates Compartment 4B, then 3-BKR-071-0008, RCIC TURBINE STEAM SUPPLY VALVE FCV-71-8. Examinee then simulates positioning switches as required by procedure to place 3-HS-071-0008, RCIC TURBINE STEAM SUPPLY VALVE, in the OPEN position.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: **3. WHEN** 3-FCV-071-8, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN, **THEN**…… **NOTIFY** Unit Operator Expected action(s): **EXAMINER CUE: When correctly simulated, the green valve position indicating lamp for 3-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated.** **Once notified by the Examinee, acknowledge as Unit 3 Operator that 3-FCV-071-8, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN position.** Examinee observes that 3-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE, is in the OPEN position by verifying that the green valve position indicating lamp for 3-FCV-71-8 is extinguished and the red valve position indicating lamp is illuminated. Once verified, the Examinee notifies the Unit 2 Operator that 3-FCV-071-008, RCIC TURBINE STEAM SUPPLY VALVE, is OPEN position.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: After completion of JPM Step 3, inform the examinee, this completes your task.	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do **NOT** operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

****

INITIAL CONDITIONS:

You are an Operator on Unit 3 with the following conditions:

- Unit 3 Reactor has SCRAMMED due to a leak in the Drywell
- Reactor Water Level **CANNOT** be determined
- 3-EOI-1, RPV CONTROL MODES 1-3, has been entered and followed to C4-RPV FLOODING, Step C4-5 RPV Injection
- 3-EOI Appendix-7H, Alternate RPV Injection System Lineup RCIC Using Auxiliary Steam, is being performed
- You already have a hand-held radio

INITIATING CUES:

The NUSO directs you to perform 3-EOI APPENDIX-16H, BYPASSING RCIC HIGH RPV WATER LEVEL SHUTDOWN INTERLOCKS.

CAUTION:

DO NOT OPERATE ANY PLANT EQUIPMENT!

PANELS WILL NOT BE OPENED!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	EDG 3A Right/Left Bank Air Compressor Crosstie Operation	
JPM NUMBER:		764-3A	REVISION:	0

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	EDG 3A Right/Left Bank Air Compressor Crosstie Operation			
K/A RATINGS:	RO: 3.8 SRO: 3.9			
K/A No. &STATEMENT:	264000 SF6 EGE Emergency Generators, A2.13; Ability to (a) predict the impacts of the following on the Emergency Generators and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of starting air			
RELATED PRA INFORMATION:	N/A			
SAFETY FUNCTION:	6			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 12 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 764-3A

RO _____ SRO _____ DATE: _____

TASK STANDARD: Examinee is expected to perform operations necessary to manually crosstie EDG 3A Left Bank Air Compressor to Right Bank Receivers in accordance with 3-OI-82, Standby Diesel Generator System.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely
OF-5 Having a solid understanding of plant design, engineering principles, and sciences

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 3-OI-82, Standby Diesel Generator System

VALIDATION TIME: 12 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/28/23	All	Initial issue

Procedure Revisions

Procedure	Revision
3-OI-82	157

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do NOT operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an extra Licensed Operator assigned to Work Control.

During performance of 3-SR-3.8.1.1(3A), EDG 3A Monthly Operability Test, 3-CKV-086-0500A, EDG 3A RIGHT BANK AIR COMPRESSOR DISCHARGE CHECK VALVE, failed to seat, failing Acceptance Criteria. This resulted in 3-CKV-086-0500A being declared INOPERABLE for In Service Testing (IST) inspection criteria.

INITIATING CUES:

The 3A EDG remains OPERABLE, but due to the failure of 3-CKV-086-0500A, the NUSO has directed you to perform 3-OI-82, Standby Diesel Generator System, Section 8.33, EDG 3A Right/Left Bank Air Compressor Crosstie Operation, [1] to align for crosstie operation of EDG 3A Left Bank Air Compressor to Right Bank Receivers.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: The respective doors to 3A/3B/3C/3D Diesel Generator (EDG) Rooms all share one common hallway.	
Step 1: 3-OI-82, Standby Diesel Generator System, Section 8.33, DG 3A Right/Left Bank Air Compressor Crosstie Operation, [1] to align for crosstie operation of DG 3A Left Bank Air Compressor to Right Bank Receivers **NOTES** 1) All steps in this section are performed locally in 3A Diesel Generator Room. 2) DIESEL GEN 3A TROUBLE annunciator (3-XA-55-23A, window 2) and the NOT AUTO alarm light on Diesel Generator 3A Engine Control Cabinet will alarm during performance of the following step. Alarm should clear once this section is complete and 3-HS-086-0668A is taken back to AUTO. 3) IF crosstie operation is desired due to low DG receiver air pressure, an attempt to first raise pressure by swapping the air dryer tower of the affected air bank via section 8.9 OR 8.10 is preferred per system engineer. Expected Action(s): Examinee is expected to transit from the designated briefing/staging location to EDG 3A Diesel Generator Room where they will go inside the room and properly locate EDG 3A Right Bank Air Compressor area.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [1.1] **PLACE** 3-HS-086-0668A, EDG 3A RIGHT BANK AIR COMPRESSOR MOTOR switch, in OFF. Expected Action(s): Examinee simulates placing 3-HS-086-0668A, EDG 3A RIGHT BANK AIR COMPRESSOR MOTOR switch, in OFF position.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: JPM Step 2 - After the Examinee simulates placing in OFF position, state 3-HS-086-0668A is OFF.	

STEP / STANDARD	SAT / UNSAT
Step 3: [1.2] **CLOSE** 3-SHV-086-503A, EDG 3A RIGHT BANK AIR COMPRESSOR DISCHARGE VALVE. Expected Action(s): Examinee simulates closing 3-SHV-086-503A, EDG 3A RIGHT BANK AIR COMPRESSOR DISCHARGE VALVE in the clockwise position.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: JPM Step 3 - After the Examinee simulates closing in the clockwise direction, state 3-SHV-086-503A is CLOSED.	
Step 4: [1.3] **OPEN** 3-SHV-086-515A, EDG 3A RIGHT BANK TO LEFT BANK AIR HEADER CROSSTIE VALVE. Expected Action(s): Examinee simulates opening 3-SHV-086-515A, EDG 3A RIGHT BANK TO LEFT BANK AIR HEADER CROSSTIE VALVE in the counterclockwise position.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: JPM Step 4 - After the Examinee simulates opening in the counterclockwise direction, state 3-SHV-086-515A is OPEN.	
Step 5: [1.4] **UNLOCK and OPEN** 3-SHV-086-514A, EDG 3A LEFT BANK TO RIGHT BANK AIR HEADER CROSSTIE VALVE. Expected Action(s): Examinee simulates unlocking and opening 3-SHV-086-514A, EDG 3A LEFT BANK TO RIGHT BANK AIR HEADER CROSSTIE VALVE in the counterclockwise position.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: JPM Step 5 - After the Examinee simulates unlocking and opening in the counterclockwise position multiple times, state 3-SHV-086-514A is OPEN.	
CUE: Once JPM Step 5 is complete, state "Another Operator will be tasked with completing the procedure".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do NOT operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an extra Licensed Operator assigned to Work Control.

During performance of 3-SR-3.8.1.1(3A), EDG 3A Monthly Operability Test, 3-CKV-086-0500A, EDG 3A RIGHT BANK AIR COMPRESSOR DISCHARGE CHECK VALVE, failed to seat, failing Acceptance Criteria. This resulted in 3-CKV-086-0500A being declared INOPERABLE for In Service Testing (IST) inspection criteria.

INITIATING CUES:

The 3A EDG remains OPERABLE, but due to the failure of 3-CKV-086-0500A, the NUSO has directed you to perform 3-OI-82, Standby Diesel Generator System, Section 8.33, EDG 3A Right/Left Bank Air Compressor Crosstie Operation, [1] to align for crosstie operation of EDG 3A Left Bank Air Compressor to Right Bank Receivers.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Manual Start of 'A' And 'C' Fire Pumps (Switchgear)
JPM NUMBER:	673	REVISION:	2

TASK APPLICABILITY:	☒SRO	☐STA	☒UO	☐NAUO
TASK NUMBER / TASK TITLE(S):	MONITOR AUTO START OF FIRE PUMP			
K/A RATINGS:	RO: 3.7 SRO: 3.3			
K/A No. &STATEMENT:	286000 Fire Protection System A2.08; Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to actuate when required			
RELATED PRA INFORMATION:	Risk Significant			
SAFETY FUNCTION:	8			

EVALUATION LOCATION:	☒In-Plant	☐ Simulator	☐ Control Room	☐ Lab
	☐ Other - List			

APPLICABLE METHOD OF TESTING: ☐ Discussion ☒ Simulate/Walkthrough ☐ Perform

TIME FOR COMPLETION: 15 min TIME CRITICAL (Y/N) N ALTERNATE PATH (Y/N) N

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 673

RO _____ SRO _____ DATE: _____

TASK STANDARD: The Examinee is expected to perform operations necessary to manually start 'A' and 'C' Fire Pumps in accordance with 0-OI-26, High Pressure Fire Protection System from 4160V Switchgear.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-5 Having a solid understanding of plant design, engineering principles, and sciences.

PRA: N/A

REFERENCES/PROCEDURES NEEDED: 0-OI-26

VALIDATIONTIME: 15 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records.(Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	04/28/15	All	Initial issue
1	09/17/18	All	JPM update
2	02/28/23	All	JPM update

Procedure Revisions

Procedure	Revision
0-OI-26	109

PLANT STAGING INSTRUCTIONS

LOCATION	Candidates staged in TSC or designated location
CAUTIONS	Inform SM, check protected equipment
LOGISTICS	Staff escort candidate between staging and exam location

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do NOT operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- Reactor Power is 70%
- A fire exists on 1B Reactor Feed Pump Lube Oil Reservoir
- Fire OPS has been dispatched to the scene
- Fire Pump B is Tagged for Maintenance
- A and C Fire Pumps failed to Auto Start

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to manually start A and C Electric Fire Pumps per 0-OI-26, High Pressure Fire System, Section 5.4.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:_____

<table>
<tr><td>STEP / STANDARD</td><td>SAT / UNSAT</td></tr>
<tr><td colspan="2">EXAMINER NOTE: Applicant may elect to start either Fire Pump A or C first. JPM Steps 1-4 cover manually starting Fire Pump A – JPM steps for Fire Pump C start on page 6.</td></tr>
<tr><td>Step 1:

0-OI-26, High Pressure Fire Protection System
Section 5.4, Manual Start of an Electric Fire Pump from the 4160V Switchgear

[1] PLACE 0-XS-26-1, FIRE PUMP A TRANSFER switch, in the EMERGENCY position at the respective switchgear.

<table><tr><th>Pump</th><th>Board</th><th>Compt</th></tr><tr><td>A</td><td>4160V Shutdown Bd A</td><td>11</td></tr></table>
Expected Action(s):

Examinee simulates placing 0-XS-26-1, FIRE PUMP A TRANSFER switch, in the EMERGENCY position at 4160 Shutdown Board A, Compartment 11.</td><td>Critical Step

_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">CUE: After JPM Step 1 is complete, state 0-XS-26-1, FIRE PUMP A TRANSFER, is in the EMERGENCY position.</td></tr>
<tr><td>Step 2:

[2] CLOSE the breaker by placing 0-HS-26-1C, FIRE PUMP A CONTROL switch, to CLOSE.

Expected Action(s):

Examinee simulates placing switch 0-HS-26-1C, FIRE PUMP A CONTROL, in the CLOSE position.</td><td>Critical Step

_____ SAT

_____UNSAT

_____N/A</td></tr>
<tr><td colspan="2">CUE: After JPM Step 2 is complete, state Breaker Red indicating light is ON, Green light is OFF, Amber light is ON, Breaker Flag is RED.</td></tr>
</table>

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>Step 3:

[3] CHECK pump ammeter at breaker indicates between 25 and 40 amps.

Expected Action(s):

Examinee verifies pump ammeter at breaker indicates between 25 and 40 amps.</td><td>_____ SAT
_____UNSAT
_____N/A</td></tr>
<tr><td colspan="2">CUE: After JPM Step 3 is complete, state FIRE PUMP A ammeter at breaker indicates 30 amps.</td></tr>
<tr><td>Step 4:

[4] PLACE the 0-XS-26-1, FIRE PUMP A TRANSFER switch, in NORMAL.

Expected Action(s):

Examinee places 0-XS-26-1, FIRE PUMP A TRANSFER switch, in NORMAL.</td><td>_____ SAT
_____UNSAT
_____N/A</td></tr>
<tr><td colspan="2">CUE: After JPM Step 4 is complete, state FIRE PUMP A TRANSFER Switch is in NORMAL. Acknowledge Section 5.4 [5] and [6] if Applicant directs AUO actions….</td></tr>
<tr><td colspan="2">EXAMINER NOTE: Applicant may elect to start either Fire Pump A or C first. Steps 5-8 cover manually starting Fire Pump C.</td></tr>
<tr><td>Step 5:

[1] PLACE 0-XS-26-3, FIRE PUMP C TRANSFER switch, in the EMERGENCY position at the respective switchgear.

<table><tr><th>Pump</th><th>Board</th><th>Compt</th></tr><tr><td>C</td><td>4160V Shutdown Bd C</td><td>10</td></tr></table>
Expected Action(s):

Examinee simulates placing 0-XS-26-3, FIRE PUMP C TRANSFER switch, in the EMERGENCY position at 4160 Shutdown Board C, Compartment 10.</td><td>Critical Step

_____ SAT
_____UNSAT
_____N/A</td></tr>
<tr><td colspan="2">CUE: After JPM Step 5 is complete, state 0-XS-26-3, FIRE PUMP C TRANSFER, is in the EMERGENCY position.</td></tr>
<tr><td colspan="2">RECOMMEND ENDING JPM HERE SINCE IT REPEATS STEPS-NEW LOCATION</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Step 6: [2] **CLOSE** the breaker by placing 0-HS-26-3C, FIRE PUMP C CONTROL switch, to CLOSE. Expected Action(s): Examinee simulates placing 0-HS-26-3C, FIRE PUMP C, in the CLOSE position.	**Critical Step** _____ SAT _____UNSAT _____N/A
CUE: After JPM Step 6 is complete, state Breaker Red indicating light is ON, Green light is OFF, Amber light is ON, Breaker Flag is RED.	
Step 7: [3] **CHECK** pump ammeter at breaker indicates between 25 and 40 amps. Expected Action(s): Examinee verifies pump ammeter at breaker indicates between 25 and 40 amps	_____ SAT _____UNSAT _____N/A
CUE: After JPM Step 7 is complete, state FIRE PUMP C ammeter at breaker indicates 30 amps.	
Step 8: [4] **PLACE** the 0-XS-26-3, FIRE PUMP c TRANSFER switch, in NORMAL. Expected Action(s): Examinee places 0-XS-26-3, FIRE PUMP C TRANSFER switch, in NORMAL.	_____ SAT _____UNSAT _____N/A
CUE: After JPM Step 8 is complete, state FIRE PUMP C TRANSFER Switch is in NORMAL. Acknowledge Section 5.4 [5] and [6] if Applicant directs AUO actions….	
EXAMINER NOTE: (Once both A and C Fire Pumps have been started), state "Another Operator will be tasked with completing the procedure". **OR AFTER JPM Step 5 above**	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

IN-PLANT: I will explain the initial conditions and state the task to be performed. ALL STEPS WILL BE SIMULATED. Do NOT operate any plant equipment. Touch STAAR may be carried out to the point of touching a label. If it becomes necessary to physically touch a control switch, use a non-conductive pointing device. I will provide initiating cues and indicate any steps to be discussed. When you complete the task successfully, the objective for this job performance measure will be satisfied. When your task is given, you will repeat the task and I will acknowledge "That's correct" (or “That's incorrect", if applicable). When you have completed your assigned task, you will say, "My task is complete" and I will acknowledge that your task is complete.

INITIAL CONDITIONS:

You are an Operator on Unit 1.

- Reactor Power is 70%
- A fire exists on 1B Reactor Feed Pump Lube Oil Reservoir
- Fire OPS has been dispatched to the scene
- Fire Pump B is Tagged for Maintenance
- A and C Fire Pumps failed to Auto Start

INITIATING CUES:

The Nuclear Unit Senior Operator (NUSO) directs you to manually start A and C Electric Fire Pumps per 0-OI-26, High Pressure Fire System, Section 5.4.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine Adequate Performance of License Reactivation
JPM NUMBER:	661	REVISION:	4

TASK APPLICABILITY:	☐ SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	RO 3.3		
K/A No. &STATEMENT:	2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.		
RELATED PRA INFORMATION:	None		
SAFETY FUNCTION:	CONDUCT OF OPERATIONS-ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: __661__

RO ____ SRO ____ DATE: _____

TASK STANDARD: Using OPDP-10, License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions and a provided table, the Examinee is expected to determine which Unit Operator has performed the necessary steps to return their license to active status.

Operator Fundamental evaluated:
OF-4 Working Effectively as a Team

PRA: NA

REFERENCES/PROCEDURES NEEDED: OPDP-10

VERIFICATION TIME: __15 min__

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
2	08/16/17	ALL	Converted JPM to new format
3	02/10/2022	ALL	Updated JPM
4	5/03/2022	ALL	Updated JPM to match OPDP-10 rev.12 (no technical changes)

Procedure Revisions

Procedure	Revision
OPDP-10	12

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

Three (3) off-shift licensed personnel are returning to shift from rotating assignments and are returning their licenses to active status. The following table gives information about hours worked under direction of an activated licensee, tours performed, etc.

License	Pre-activation Meeting	Shift 1	Shift 2	Shift 3	Shift 4	Shift 5	Shift 6	Plant Tour
RO1	Ops Training Manager And Shift Manager	12 hrs U-3 OATC	12 hrs U-3 OATC	12 hrs U-1 OATC	12 hrs tagging SST RO	12 hrs U-2 OATC	12 hrs U-2 OATC	completed plant tour with STA (SRO)
RO2	Ops Training Manager And Ops Superintendent	12 hrs U-2 BOP	12 hrs U-2 BOP Called for Random Drug test during shift – missed end of shift turnover	12 hrs U-1 BOP	12 hrs U-3 BOP	12hrs U-3 BOP	N/A	completed plant tour with SST (RO)
RO3	Ops Training Manager And Ops Superintendent	12 hrs U-2 OATC	12 hrs U-2 OATC	12 hrs U-3 OATC	12 hrs U-3 OATC	12 hrs U-1 OATC	N/A	completed plant tour with U1 RB AUO

INITIATING CUES:

The Shift Manager has tasked you to determine which of these personnel, if any, have completed the requirements for returning their license to active status.

If any personnel do not meet the requirements for returning their license to active status, state the reason(s) why.

KEY

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

Three (3) off-shift licensed personnel are returning to shift from rotating assignments and are returning their licenses to active status. The following table gives information about hours worked under direction of an activated licensee, tours performed, etc.

License	Pre-activation Meeting	Shift 1	Shift 2	Shift 3	Shift 4	Shift 5	Shift 6	Plant Tour
RO1	Ops Training Manager And Shift Manager	12 hrs U-3 OATC	12 hrs U-3 OATC	12 hrs U-1 OATC	12 hrs tagging SST RO	12 hrs U-2 OATC	12 hrs U-2 OATC	completed plant tour with STA (SRO)
RO2	Ops Training Manager And Ops Superintendent	12 hrs U-2 BOP	12 hrs U-2 BOP Called for Random Drug test during shift – missed end of shift turnover	12 hrs U-1 BOP	12 hrs U-3 BOP	12hrs U-3 BOP	N/A	completed plant tour with SST (RO)
RO3	Ops Training Manager And Ops Superintendent	12 hrs U-2 OATC	12 hrs U-2 OATC	12 hrs U-3 OATC	12 hrs U-3 OATC	12 hrs U-1 OATC	N/A	completed plant tour with U1 RB AUO

INITIATING CUES:

The Shift Manager has tasked you to determine which of these personnel, if any, have completed the requirements for returning their license to active status.

If any personnel do not meet the requirements for returning their license to active status, state the reason(s) why.

Answer: **ONLY** RO2 has completed the requirements for license reactivation.
RO1 did **NOT** interview with the Operations Superintendent, contrary to OPDP-10. The Shift Manager is **NOT** an authorized alternative for the Operations Superintendent.

RO3 performed the plant tour with a non-licensed Operator (U1 RB AUO), contrary to OPDP-10.

KEY

NPG Standard Department Procedure	License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions	OPDP-10 Rev. 0012 Page 7 of 26

3.2 Instructions

3.2.1 Active License Status Maintenance

A. To maintain an active status, the licensee shall actively perform the functions of an SRO or RO for a minimum of seven 8-hour shifts, or five 12-hour shifts, per calendar quarter, in a position credited for watch-standing proficiency.

B. To maintain the supervisory portion of an SRO license active, an SRO must stand at least one complete watch per calendar quarter in a shift crew position credited for SRO-only supervisory licensed duties. The remainder of complete watches (to meet the required minimum of seven 8-hour or five 12-hour shifts per calendar quarter) may be performed in either a credited SRO or RO position.

C. It is the license holders responsibility to maintain cognizance of his/her license status. All licensed personnel are responsible to ensure TVA Medical is notified of any changes associated with the medical condition or medicine provided, as described in this procedure, and that an updated NRC Form 396 is required to be initiated and routed to the NRC within 30 days of confirmation of a new or changed medical restriction.

D. The Licensing Department ensures an updated NRC Form 396 is initiated and routed to the NRC within 30 days of confirmation of a new medical restriction, or any changes associated with medical restrictions, after being notified by the medical staff or Operations Training.

E. Each site will track license status using Learning Management System (LMS). If an individual's license is not listed as active, he or she shall not perform in a Technical Specifications (T.S.) licensed position until the reason is evaluated and corrected. If the licensee has met all requirements but is not showing active due to an error or delay with LMS or a supporting function, a Condition Report (CR) shall be initiated to document the reason, and he or she may then hold a licensed position.

F. A licensee who is reactivating a license shall work 40 hours performing licensed duties in tandem with an active licensed person (Under Instruction).

G. When a licensee is issued a new NRC License number, the licensee is immediately active. The licensee is not required to perform 40 hours Under Instruction that calendar quarter and is not required to stand five 12-hour shifts or seven 8-hour shifts in the calendar quarter that the new license number was issued. The licensee must stand the required shifts in the next calendar quarter to maintain an active license.

3.2.2 Nuclear Plant Requirements for Maintaining Active License Status

This section provides administrative instructions to comply with 10CFR55.53 (e), ... "actively performing the functions of an operator or senior operator." 10CFR55.4 defines this as "an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that the individual carries out and is responsible for the duties covered by that position."

KEY

3.2.3 Medical Requirements (continued)

1. This shall be done in compliance with the thirty (30) day reporting period under 10CFR55.25.

2. If the diagnosis is not final, a follow-up letter will be sent as required by 10CFR55.25 later.

3.2.4 Nuclear Plant Requirements for Returning an Inactive License to Active Status

This section is intended to provide guidance to return a licensed individual to an active status.

A. The Code of Federal Regulation, 10CFR55.53 f(2) specifies returning a license to active status. The intent of the code is to ensure proficiency in the conduct of licensed activities prior to assuming licensed duties. The following requirements are addressed as part of this code:

1. The qualifications and status of the licensee are current and valid. This requirement ensures the licensee has completed all required requalification training, including plant modifications and industry events; and secondly, that all conditions of his/her license are still being met.

2. This licensee has completed a minimum of 40 hours of shift functions under the direction of a reactor operator or senior operator, as appropriate, and in the position to which the individual will be assigned. This ensures that an active license is directing or performing the manipulations of plant controls, and allows the inactive individual to obtain proficiency at his/her watch station. Included within the minimum of 40 hours is the following:

a. A complete review of turnover procedures by the reactor operator or senior reactor operator as appropriate for the position, to ensure that the licensee is familiar with current shift turnover practices.

b. A complete tour of the plant, accompanied by an active licensed RO or SRO, as appropriate. Plant tour should be of similar detail and thoroughness as AUO rounds.

B. All licensed personnel who maintain a license shall comply with these requirements to return to active status. The Operations Superintendent or designee is responsible for administering the process.

C. The following guidelines are to be used when reactivating a license:

1. Prior to standing the minimum of 40 hours of shift functions, the licensed individual shall meet with the Operations Training Manager and the Operations Superintendent to discuss his/her current status and any standards or expectations. Additional requirements may be imposed at the discretion of the Operations Superintendent. This shall be documented on Attachment 1.

KEY

NPG Standard Department Procedure	License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions	OPDP-10 Rev. 0012 Page 14 of 26

3.2.4 Nuclear Plant Requirements for Returning an Inactive License to Active Status (continued)

2. The individual shall be under the direct supervision of an active licensed individual in the position (SM or Unit Supervisor as applicable for SROs and licensed Unit Operator for ROs) to which the individual will be assigned.

3. The individual shall make a narrative log entry at the start of the shift which will include the following at a minimum:

   a. Name and time of assuming shift

   b. Shift Position (as identified in 3.2.2A) assumed under direction

   c. Name of the operators (Board and Desk), Control Room SRO, or Shift Manager providing supervision.

4. A copy of the completed log shall be printed for each shift and submit as documentation with Attachment 1.

5. If an individual moves from one unit to another unit during the same shift for the purpose of breaking-in on the other unit, the individual shall make a log entry indicating that they are moving to the other unit to continue their break-in. Another entry will be made when the individual goes under instruction on the new unit's log. This requirement is not applicable to a Shift Manager since the break-in would still be under the same individual.

6. The individual shall review the turnover procedures with an active reactor operator or senior reactor operator, as applicable. The SM or Operations Superintendent shall determine the minimum procedures to be reviewed.

7. As a minimum, the following shall be completed to satisfy the plant tour requirement:

   a. Prior to beginning the tour, a discussion should be held with the Shift Manager to obtain guidance on which areas to focus on during the plant tour. The tour should be similar in detail and thoroughness as AUO rounds.

   b. Tour accessible plant areas listed on the site-specific section of Attachment 1. The licensee shall be accompanied by an active licensed RO or SRO, as applicable. The tours can be performed on different days and with different licenses providing oversight.

   c. Document the completion of all tours on the cover page.

   d. Obtain a copy of door logs and submit with Attachment 1 for documentation.

   e. Attachment 1, Return to Active Status, contains the documentation required for reactivating a license. The unused pages for other plants tours can be discarded. The remaining document shall be sent to Training to become part of the training record and to LMS.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: Uses OPDP-10, License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions, to analyze the information provided to determine which personnel meet the requirements for license reactivation. Expected Action(s): Examinee determines that **ONLY** RO2 meets the requirements for returning their license to active status in accordance with OPDP-10.	**Critical Step** ___ SAT ___UNSAT ___N/A
Step 2: States the reason why RO1 does **NOT** meet the requirements for returning their license to active status. Expected Action(s): RO1 did **NOT** interview with the Operations Superintendent, contrary to OPDP-10. The Shift Manager is **NOT** an authorized alternative for the Operations Superintendent.	**Critical Step** ___ SAT ___UNSAT ___N/A
Step 3: States the reason why RO3 does **NOT** meet the requirements for returning their license to active status. Expected Action(s): RO3 performed the plant tour with a non-licensed Operator (U1 RB AUO), contrary to OPDP-10.	**Critical Step** ___ SAT ___UNSAT ___N/A
EXAMINER CUE: Once the Operator completes JPM Step 3 above, state this completes your task.	
END OF TASK	

STOP TIME:_____

Provide to Applicant

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

Three (3) off-shift licensed personnel are returning to shift from rotating assignments and are returning their licenses to active status. The following table gives information about hours worked under direction of an activated licensee, tours performed, etc.

License	Pre-activation Meeting	Shift 1	Shift 2	Shift 3	Shift 4	Shift 5	Shift 6	Plant Tour
RO1	Ops Training Manager And Shift Manager	12 hrs U-3 OATC	12 hrs U-3 OATC	12 hrs U-1 OATC	12 hrs tagging SST RO	12 hrs U-2 OATC	12 hrs U-2 OATC	completed plant tour with STA (SRO)
RO2	Ops Training Manager And Ops Superintendent	12 hrs U-2 BOP	12 hrs U-2 BOP Called for Random Drug test during shift – missed end of shift turnover	12 hrs U-1 BOP	12 hrs U-3 BOP	12hrs U-3 BOP	N/A	completed plant tour with SST (RO)
RO3	Ops Training Manager And Ops Superintendent	12 hrs U-2 OATC	12 hrs U-2 OATC	12 hrs U-3 OATC	12 hrs U-3 OATC	12 hrs U-1 OATC	N/A	completed plant tour with U1 RB AUO

INITIATING CUES:

The Shift Manager has tasked you to determine which of these personnel, if any, have completed the requirements for returning their license to active status.

If any personnel do not meet the requirements for returning their license to active status, state the reason(s) why.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Work Hour Limitations
JPM NUMBER:	551	REVISION:	4

TASK APPLICABILITY:	☐ SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	RO 2.9		
K/A No. &STATEMENT:	2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.		
RELATED PRA INFORMATION:	None		
SAFETY FUNCTION:	CONDUCT OF OPERATIONS-ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 551

RO _____ SRO _____ DATE: _____

TASK STANDARD: Using NPG-SPP-03.21, Fatigue Rule and Work Hour Limits, and given an Operator's work schedule, the Examinee is expected to determine that the **ONLY** work hour limit **NOT** met is the maximum of 72 work hours in any 7-day (168-hour) rolling period and that the schedule needs to be revised or a waiver granted.

Operator Fundamental evaluated:
OF-3 Operating the Plant With A Conservative Bias

PRA: N/A

REFERENCES/PROCEDURES NEEDED: NPG-SPP-03.21

VALIDATION TIME: 15 minutes

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Rev No.	Effective Date	Pages Affected	Description
2	09/11/18	All	Updated Format
3	9/17/19	All	Updated due to revision of NPG-SPP-03.21
4	2/09/23	All	Revised to alter limit violated (>72 work hours in any 7-day (168-hour) rolling period), and to reflect revision to NPG-SPP-03.21.

Procedure Revisions

Procedure	Revision
NPG-SPP-03.21	29

****

Classroom

****

INITIAL CONDITIONS: You are a Reactor Operator.

Unit 1, 2 and 3 are all operating at 100% Rated Thermal Power.

A shift group swap is being planned and the normal running Unit schedule is being adjusted to allow for the swap to occur.

You were in compliance with all work hour limits during your **LAST** shift cycle.

You are assigned as a running Unit RO for the upcoming 5 week shift cycle, and below is your required work schedule. For each of the 12 hour scheduled shifts, assume that you will arrive 30 minutes prior to the start of your shift for shift turnover each work day. Assume each of the 10 hour shifts are normally scheduled LOR training.

Note: The eSOMS NFR program is not available.

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/7	8/8	8/9	8/10	8/11	8/12	8/13
0700-1900	0700-1900	0700-1900	0700-1900	OFF	OFF	OFF

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/14	8/15	8/16	8/17	8/18	8/19	8/20
OFF	OFF	OFF	OFF	1900-0700	1900-0700	1900-0700

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/21	8/22	8/23	8/24	8/25	8/26	8/27
1900-0700	OFF	OFF	OFF	0700-1900	0700-1900	0700-1900

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/29	8/30	8/31	9/1	9/2	9/3	9/4
OFF	1900-0700	1900-0700	1900-0700	OFF	OFF	0700-1900

Mon	Tues	Wed	Thu	Fri	Sat	Sun
9/5	9/6	9/7	9/8	9/9	9/10	9/11
0700-1700	0700-1700	0700-1700	0700-1700	0700-1900	0700-1900	OFF

INITIATING CUES: Review the required work schedule to verify all work hour limits will be satisfied.

If any limits are not met, then identify what administrative measures are required.

KEY

3.2.5 Work Hour Limits and Breaks

A. Work hour controls for individuals performing covered work consist of the following:

1. Maximum of 16 work hours in any 24-hour rolling period. [R-4]
2. Maximum of 26 work hours in any 48-hour rolling period. [R-4]
3. Maximum of 72 work hours in any 7-day (168-hour) rolling period.[R-4]
4. Minimum 10-hour break between successive work periods, except that an 8-hour break is allowed when necessary to accommodate a crew's scheduled transition between work schedules or shifts. The 8-hour application is for a crew's transition and not an individual's transition. [R-5]
5. Minimum 34-hour break in any 9-day period. [R-5]
6. Online work hours are no more than 54-hours per week on average for the current initial averaging or rolling averaging periods. [R-6]
7. Minimum days off (MDO, when in outage).

B. Limits 3.2.5.A.1. through 3.2.5.A.5. apply during online and outage plant conditions. The limit in 3.2.5.A.6, 54-work hour averaging, is utilized when the plant is online and may be utilized when the plant is offline. The 3.2.5.A.7. limit, MDO, may be utilized when the plant is offline and only for individuals working on outage activities.

C. For on-line work, the Site Department Administrator establishes the 54-work hour averaging period. Averaging calculations are modeled in the approved NFR tracking software:

1. A covered worker may not work more than a weekly average of 54-work hours calculated using an averaging period of up to six weeks.
2. The averaging period advances (rolls) by one week at a time (by seven consecutive calendar days) at the finish of every averaging period while dropping off the first week of the previous averaging period.

   Example: An operator has a six week rolling averaging period. The first rolling averaging period begins at 0000 on Monday and includes weeks one through six and ends at 0000 Monday. The second averaging period starts at the start of week seven (week one drops off) and encompasses weeks two through seven. The second six week averaging period begins with the start of the seventh week when the first week drops off.
3. The date and time of the week when an averaging period starts is 0000 Monday.

D. The periods of "24" hours," "48 hours," and "7 days" are considered rolling time periods. Rolling means the period is not re-zeroed or reset following a day off or after obtaining authorization to exceed 10 CFR 26 overtime limits. The "24 hours," "48 hours," and "7 days" periods do not restart after a day off; the period continues to roll.

KEY

3.2.8 Work Hour Scheduling (continued)

4. Training requirements

5. Vacation scheduling

D. To determine if an individual is eligible for a call in work period use the guidelines:

1. Evaluation of periodic overtime for covered individuals' work hours should be conducted with respect to an expected average of 54 hours per week, or less.

2. Hours worked should be evaluated to determine if a limit will be exceeded based on the work schedule by a backward look at the number of hours that have or will have been worked based on a time in the future (if the individual works at time T, how many hours will be worked in the 24 hours/48 hours/ 7 days preceding time T relative to 16 hour/48 hour/7 day work hour limits and 10 hour and 34 hour break requirements and the online 54-work hour average weekly limit or outage minimum days off.

E. When entering an unplanned unit outage, unplanned security outage or increased threat condition, compliance is maintained if work hour limits (16/24, 26/48, 72/7), break requirements and the maximum average work hour limit / minimum days off requirements were met in the last seven day period.

F. An individual is considered "reset" from deviation, whether under a waiver or otherwise, when the work hour limits (16/24, 26/48, 72/7), break requirements and maximum average work hour limit / minimum days off requirements were met in the last seven day period.

3.2.9 The Waiver Process

A. The waiver process consists of two components:

1. Documentation of plant conditions or circumstances that warrant the use of a waiver.

2. Documentation the affected covered individual(s) is (are) fit-for-duty, with respect to fatigue, for any work period (shift) during which one or more work hour limits will be exceeded.[R-21]

a. A fatigue assessment of the worker(s) is conducted in accordance with section 3.2.11 to document the worker's fitness for duty regarding fatigue.

b. The work hour limit(s) to be exceeded for the assessed individual are determined by entering the proposed schedule change in the work hour tracking system.

B. The waiver process provides prior authorization for covered workers to exceed work hour limits while performing covered work.

START TIME: _____

STEP / STANDARD	SAT / UNSAT
Step 1: Proposed work schedule reviewed against 10CFR26 limits (found in NPG-SPP-03.21, Nuclear Fatigue Management Program, Section 3.2.5 Work Hour Limits and Breaks). Expected Action(s): Examinee evaluates schedule and determines that the **ONLY** work hour limit **NOT** met is the maximum of 72 work hours in any 7-day (168-hour) rolling period. This is determined from evaluating the 9/4 - 9/10 work days which total up to 76 work hours straight in a 7-day rolling period.	**Critical Step** ___SAT ___UNSAT ___N/A
Step 2: Determines required actions for exceeding the maximum of 72 work hours in any 7-day (168-hour) rolling period in accordance with NPG-SPP-03.21, Section 3.2.9 Waiver Process. Expected Action(s): Examinee determines individual cannot work schedule as written without an approved waiver.	**Critical Step** ___SAT ___UNSAT ___N/A
EXAMINER NOTE: Examinee may elect to also provide additional measures beyond an approved waiver, however this is not required for the Critical Step in JPM Step 2 above. They may state the need to take an admin day off and/or come in 4 hours late or leave 4 hours early during one of the days from 9/4 – 9/10.	
END OF TASK	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

****

Classroom

****

INITIAL CONDITIONS: You are a Reactor Operator.

Unit 1, 2 and 3 are all operating at 100% Rated Thermal Power.

A shift group swap is being planned and the normal running Unit schedule is being adjusted to allow for the swap to occur.

You were in compliance with all work hour limits during your **LAST** shift cycle.

You are assigned as a running Unit RO for the upcoming 5 week shift cycle, and below is your required work schedule. For each of the 12 hour scheduled shifts, assume that you will arrive 30 minutes prior to the start of your shift for shift turnover each work day. Assume each of the 10 hour shifts are normally scheduled LOR training.

Note: The eSOMS NFR program is not available.

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/7	8/8	8/9	8/10	8/11	8/12	8/13
0700-1900	0700-1900	0700-1900	0700-1900	OFF	OFF	OFF

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/14	8/15	8/16	8/17	8/18	8/19	8/20
OFF	OFF	OFF	OFF	1900-0700	1900-0700	1900-0700

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/21	8/22	8/23	8/24	8/25	8/26	8/27
1900-0700	OFF	OFF	OFF	0700-1900	0700-1900	0700-1900

Mon	Tues	Wed	Thu	Fri	Sat	Sun
8/29	8/30	8/31	9/1	9/2	9/3	9/4
OFF	1900-0700	1900-0700	1900-0700	OFF	OFF	0700-1900

Mon	Tues	Wed	Thu	Fri	Sat	Sun
9/5	9/6	9/7	9/8	9/9	9/10	9/11
0700-1700	0700-1700	0700-1700	0700-1700	0700-1900	0700-1900	OFF

INITIATING CUES: Review the required work schedule to verify all work hour limits will be satisfied.

If any limits are not met, then identify what administrative measures are required.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Main Generator Voltage Regulator Operation
JPM NUMBER:	760	REVISION:	0

TASK APPLICABILITY:	☐ SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-047-NO-35 / Transfer the Generator Voltage Regulator to Manual/Auto		
K/A RATINGS:	RO 4.2		
K/A No. &STATEMENT:	2.2.44 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.		
RELATED PRA INFORMATION:	None		
SAFETY FUNCTION:	EQUIPMENT CONTROL-ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	10 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: __760__

RO ____ DATE: _____

TASK STANDARD: Using 2-ARP-9-8A, 2-OI-47, Turbine-Generator System, or 0-GOI-300-4, Switchyard Manual, the Examinee is expected to determine notification requirements for operating the Main Generator Voltage Regulator in MANUAL. Additionally, determine if the Generator Kilovar Limitations Capability Curve parameters are being exceeded, if so, what action(s) are required.

Operator Fundamental evaluated: OF-3 Operating the Plant with a Conservative Bias

PRA: NA

REFERENCES/PROCEDURES NEEDED:
2-OI-35
2-OI-47
2-ARP-9-8A
0-GOI-300-4
0-AOI-57-1E

VERIFICATION TIME: __10 min.____

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	02/07/2023	ALL	Initial Revision

Procedure Revisions

Procedure	Revision
2-OI-35	75
2-OI-47	204
2-ARP-9-8A	49
0-GOI-300-4	130
0-AOI-57-1E	14

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

Unit 2 is operating at 100% Rated Thermal Power (RTP) with severe storms in the area.

At 0800:

- 4KV Board Voltage oscillations are observed in Unit 2 Main Control Room
- 0-AOI-57-1E, Grid Instability, is entered by the Nuclear Unit Senior Operator (NUSO)
- 2-XR-57-57, GENERATOR LOAD RECORDER, is 1280 MW
- 2-EI-57-51, GENERATOR MVAR, is (+) 100 MVARs (outgoing)
- 2-PI-35-17A, GENERATOR HYDROGEN PRESSURE, is 58 psig

At 0802:

- 2-HS-57-27, VOLTAGE REGULATOR MANUAL / AUTO SELECT, **AUTOMATICALLY** tripped to MANUAL

INITIATING CUE:

For the **CURRENT** conditions and in accordance with applicable procedure(s), the Nuclear Unit Senior Operator (NUSO) directs you as the Balance of Plant Operator (BOP) to determine:

1. Who, if anyone, is required to be notified?
2. If any Generator Calculated Capability Curve parameters require adjustment?
    - Yes or No
    - If Yes, list parameter(s) and action(s) to be taken?

_____
_____
_____
_____
_____
_____

KEY

BFN Unit 2	Panel 9-8 2-XA-55-8A	2-ARP-9-8A Rev. 0049 Page 7 of 45

EXCITATION SYSTEM XFER TO MANUAL
2-EA-57-132

3

(Page 1 of 2)

Sensor/Trip Point:
D011E
(digital output)

Operates whenever voltage regulator is automatically tripped to manual.

Sensor Location: Panel 2-9-8
Control Bay 1C

Probable Cause:
A. Failure of automatic mode in both Voltage Regulator channels.
B. Volts/Hertz level 1 (greater than 107% for 45 seconds).

Automatic Action: None

Operator Action:
A. **ENSURE** VOLTAGE REGULATOR MAN/AUTO SEL, 2-HS-57-27, in MANUAL (Amber light illuminated, blue light extinguished).
B. **ADJUST** VOLTAGE REGULATOR LOWER/RAISE ADJUST, 2-HS-57-26, to maintain the following:
- GENERATOR VOLTS, 2-EI-57-39, between 20,900V and 23,100V
- GENERATOR MVARs, 2-EI-57-51, within the generator capability curve. **REFER TO** 2-OI-47, Illustration 7.

Continued on Next Page

KEY

BFN Unit 2	Panel 9-8 2-XA-55-8A	2-ARP-9-8A Rev. 0049 Page 8 of 45

EXCITATION SYSTEM XFER TO MANUAL 2-EA-57-132, Window 3
(Page 2 of 2)

Operator Action: (Continued)

NOTE

Transmission Operator notification is required within 30 minutes of the event (NERC Standard VAR-002).

C. **NOTIFY** Transmission Operator of the following:
- Time of the event
- Cause of the event (if known)
- Duration of the event (if known)

D. **CHECK** the following displays on ICS to identify the cause:
- AVR ALARM CODES (AVRALM)
- AVR FAULT CODES (AVRFLT)

E. **DISPATCH** personnel to Main Generator Voltage Regulator on Turb Bldg El 604' for the following:
1. **CHECK** for Volts/Hertz relay actuation.
2. **USE** touch screen to investigate cause and reset alarms/faults. **REFER TO** 2-OI-47.

F. **CHECK** for blown fuse using GENERATOR VOLTAGE/METER POT switch on Panel 2-9-8.

G. **REQUEST** assistance, as required, from the following:
- Site Engineering
- Maintenance

References:

2-45E1573-2	45N2611-1	2-45E1550
2-45E620-11	2-45E1573-1	

KEY

BFN Unit 2	Turbine-Generator System	2-OI-47 Rev. 0204 Page 19 of 278

3.4 Generator Operation (continued)

E. Generator operation using Air as the cooling medium

1. If generator is being rolled to 1800 rpm with air instead of hydrogen in generator, DO NOT close field breaker and DO NOT flash generator field.

2. If generator is operated at any time with air instead of hydrogen as cooling medium, air pressure in generator is required to be maintained above atmospheric pressure to avoid drawing oil from shaft seals into generator along the shaft.

F. Generator Isophase Bus Duct Cooling

1. If bus duct cooling is not in service, limit bus amperage to less than 16,500 amps.

2. [NRC/C] The Bus Duct Cooling System is required to be in service at all times except when required to be out of service for maintenance or modification. This preventative maintenance measure provides positive air pressure on bus duct to prevent dust, dirt, and moisture intrusion. [LER 259/89012]

3. Plant Engineering should be contacted to evaluate balancing damper positioning for proper air distribution when the Bus Duct Cooling configuration is changed.

G. [NER/C] The Neutral ground voltmeter should be monitored during startup and initial loading. In the event of upward trending approaching 10 volt trip point, load should be reduced until neutral voltage lowers [NER/C SQN II-90-072]

H. While the Main Generator is tied to the Grid:

1. The Transmission Operator (TOp) is required to be notified of any status change (Manual/Auto/Loss of Function) of the Voltage Regulator associated with the Main Generator, as soon as practical but within 30 minutes. [NERC VAR-002-2] [SOER 99-1]

2. The Transmission Operator is required to be notified prior to a planned Voltage Regulator transfer from Manual or Automatic. [NERC VAR-002] [SOER 99-1]

3. All position changes (Auto or Manual) of the Voltage Regulator are required to be entered into the Narrative Log along with the date, time of position change, reasons, anticipated duration, and notifications made. [NERC VAR-002-3] [SOER 99-1]

I. Operation of the Main Generator without automatic voltage control could impact grid voltage requirements as referenced in Sections 6.1[10] and 8.10.1[5]

KEY

3.4 Generator Operation (continued)

J. When the Main Generator is online, the Voltage Regulator is required to be operated in Automatic, unless coordinated with the Transmission Operator (TOp).

K. If the Main Generator Voltage Regulator is operated in the Manual mode of regulation, operators should monitor MVARs loading to ensure limits are maintained. The preferred method to tie the Main Generator to the Electrical Grid is with the Voltage Regulator in Auto. It is NOT desired to tie the generator to the Electrical Grid with the voltage regulator in manual.

L. During normal operation 480V Unit Bd 2A, Compt. 4C, which is powering the Excitation Transformer T02.1, will always be the normal/primary source for power the Voltage Regulator; 480V Common Bd. 2, Compt. 5D, which is powering Excitation Transformer T02.2, should always remain the alternate power source. During each refueling outage the default channel will be alternated and the manual selector switches, 2-XSW-242-0002/Q801 and 2-XSW-242-0002/Q802, should be changed over so that the primary power supply will power the current default channel of the Voltage Regulator. Switch manipulation is done when the Voltage Regulator is NOT in operation.

M. The power source transfer at Unit Bd. 2A or Common Bd 2 (i.e. the power sources feeding these boards being transferred between normal and alternate) is NOT a bumpless transfer. Recent OE at BFN has shown that transferring power sources to these 480V boards causes a loss of channel fault at the Voltage Regulator. Even though the power returns almost immediately, the channel that read the loss of power cannot be returned to service until the fault is cleared. Therefore, transferring power to these boards should only occur when the board in question is powering the alternate Voltage Regulator channel and the ECT is functioning properly so that the fault may be cleared after the transfer takes place.

3.5 Temperatures and Temperature Monitoring

A. Shell Warming is required anytime the turbine high-pressure shell has cooled to less than 250°F.

B. Maintain turbine first stage shell metal temperature heat-up/cool down rate less than 150°F per hour.

C. Turbine first stage shell inner surface to outer surface temperature differential should not exceed 75°F.

KEY

8.10 Transferring the Voltage Regulator Between Auto and Manual

NOTES

1) Each BFN Unit has a maximum outgoing (lagging) Mvar limit of +175 Mvar. This limit applies to both 500 kV and 161 kV offsite power source qualifications. If the outgoing Mvar limit is exceeded and is not corrected with 15 minutes, then both offsite power sources are disqualified for the unit that is exceeding the limit.
2) The AVR has a dynamic limiter when operating in Automatic. In Manual, the limiter is not applicable. There is a dynamic protection setpoint that will trip the exciter field breaker in Manual or Automatic.
3) Unless otherwise directed, the normal band for Generator MVARS should be + 50 Mvars to + 150 Mvars while regulation is in manual.
4) The Transmission Operator (TOp) is required to be notified of any status change (Manual/Auto/Loss of Function) of the Voltage Regulator, as soon as practical but within 30 minutes. [NERC VAR-002-2] [SOER 99-1]
5) The Transmission Operator is required to be notified prior to a planned Voltage Regulator transfer from Manual or Automatic. [NERC VAR-002-2] [SOER 99-1]
6) Operation of the Main Generator without automatic control could impact grid voltage requirements Refer to Sections , 6.1[10], 8.10.1[5]
7) When the Main Generator is online, the Voltage Regulator is required to be operated in Automatic, unless coordinated with the Transmission Operator (TOp).
8) All position changes (Auto or Manual) of the Voltage Regulator are required to be entered into the Narrative long along with the date, time of position change, reasons, anticipated duration, and notifications made. [NERC VAR-002-3] [SOER 99-1]

CAUTION

Frequent monitoring of GENERATOR MVARS (2-EI-57-51) is required as long as the VOLTAGE REGULATOR MAN/AUTO SEL switch is in MAN.

8.10.1 Transfer of the Voltage Regulator From Auto to Manual

[1] **IF** Voltage Regulator transfer is of a non-emergency nature and time permits, **THEN:**

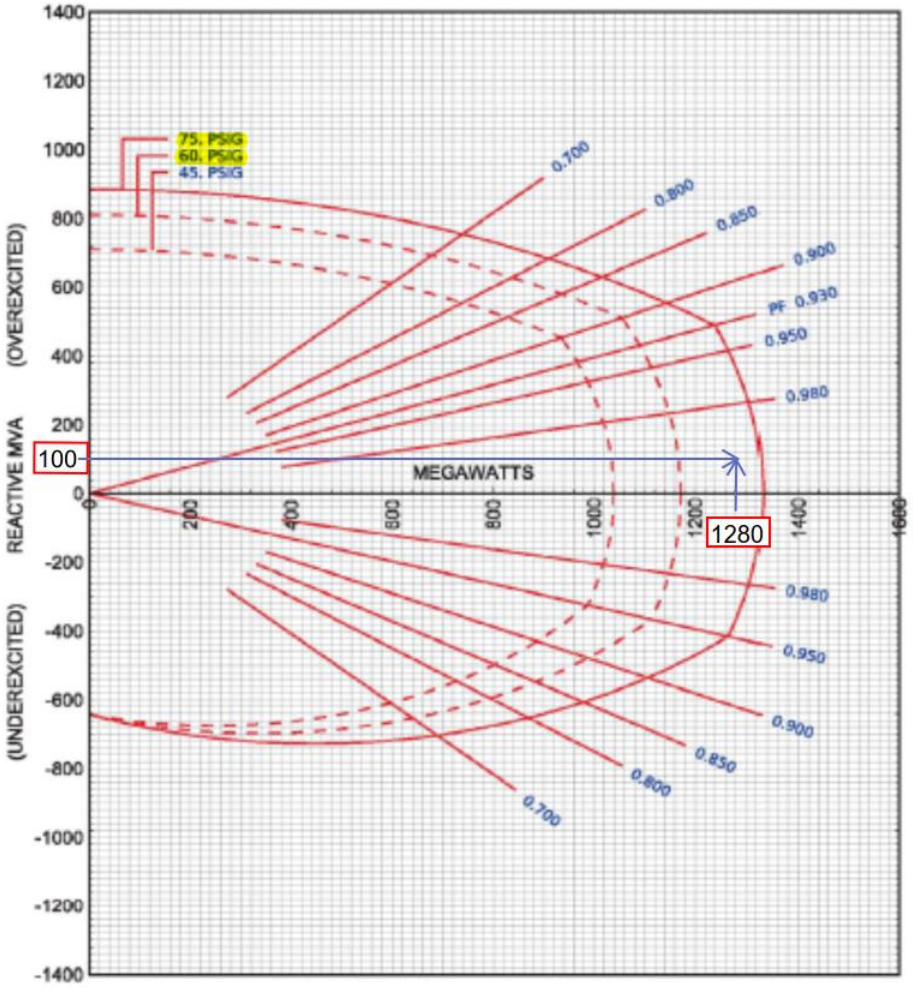
NOTIFY the Transmission Operator of the planned transfer of the Voltage Regulator from Automatic to Manual. [SOER 99-1]

KEY

BFN Unit 2	Turbine-Generator System	2-OI-47 Rev. 0204 Page 269 of 278

Attachment 7
(Page 2 of 2)

SIEMENS ENERGY, INC CURVE NO. 090506GLC
CALCULATED CAPABILITY CURVES
AT 100 PERCENT VOLTAGE (22.0 kV)
WITH WATER COOLED STATOR
HYDROGEN INNER-COOLED TURBINE GENERATOR



1332.0 MVA PF 0.930 22.0 kV 34955 A
3 PHASE 60 Hz 1800 RPM SCR 0.58 75 PSIG
RATED COLD GAS 46 °C RATED STATOR WATER 50°C
DATE - 2015-06-26 CURVE NO. - 090506GLC

KEY

BFN Unit 0	Switchyard Manual	0-GOI-300-4 Rev. 0130 Page 17 of 170

3.1 General (continued)

NN. The BFN Generator Operator must inform the TOp of the status of the Voltage Regulator (Automatic or manual).

OO. Entry into this procedure requires evaluation of the situation per EOOS Management, NPG-SPP-09.11.1

PP. Protective relays should NOT be removed from service without first contacting the Transmission Operator (TOp) for risk evaluation. Reference the following: TRANS-TSO-SOP-30.104, Guidelines on 161-kv Protection System Outages and TRANS-TSO-SOP-30.187, Guidelines on 500-kv Protection System Outages.

QQ. Transfer Trip Switches for the 500kV lines defeat the remote trip functions for both PCBs on that line (e.g., if the transfer switch for 5288 is placed in test, the remote trip function is defeated not only for PCB 5288, but also 5284).

RR. NERC Standard MOD-025-2, Verification and Data Reporting of Generator Real and Reactive Power Capability and Synchronous Condenser Reactive Power Capability, requires the verification of operational data for each Facility within 12 calendar months following the discovery that its Real Power or Reactive Power capability has changed by more than 10 percent of the last reported verified capability and is expected to last more than six months. Refer to MOD-025-2 for more information.

KEY

BFN Unit 2	Generator Hydrogen Cooling System	2-OI-35 Rev. 0075 Page 37 of 95

6.2 Daily Checks During Normal Operation (continued)

[3.12] **MOMENTARILY DEPRESS** RESET pushbutton inside generator condition monitor AND **ENSURE** alarm "GEN CONDITION MONITOR ABNORMAL" on 2-XA-55-8B-28 is reset.

[3.13] **NOTIFY** Unit Operator that draining of Oil/Moisture trap is complete, and Generator Condition Monitor is back in service.

NOTES

1) GEN H2 FLOW TRANSMITTER 2-FIT-35-102, outputs a 12-20mA signal which correlates to a flow range of 0-8000 SCFD.
2) ICS High Alarm is Less than or equal to 1500 SCFD
3) ICS High High Alarm is less than or equal to 2000 SCFD.

[4] **ENSURE** Gen H2 system is in Auto Makeup control and hydrogen pressure is greater than 73 psig and less than 79 psig.

[5] **CHECK** ICS GEN H2 FLOW TRANSMITTER 2-FIT-35-102, is less than 1200 SCFD.

[5.1] **IF** Gen H2 Flow is greater than 1200 SCFD, **THEN**

PERFORM the following: (Otherwise N/A)

[5.1.1] **NOTIFY** System Engineer to evaluate hydrogen consumption on system. REF P&L 3.0L.

[5.2] **IF** Gen H2 Flow is greater than 1500 SCFD, **THEN**

PERFORM the following: (Otherwise N/A)

[5.2.1] **MONITOR** GEN H2 FLOW as necessary to stay within Attachment 1; Generator Hydrogen Pressure Versus Load Curve.

[5.2.2] **NOTIFY** System Engineer for Evaluation.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: The Candidate may elect to reference 0-AOI-57-1E, Grid Instability, since it is given in the initial conditions, however it is not required.	
EXAMINER CUE: ONLY If/After the Candidate requests a copy of 2-OI-47, Turbine-Generator System, Attachment 7 (Page 2 of 2) – CALCULATED CAPABILITY CURVE, provide them a hard copy.	
Step 1: Any of the following may be referenced: Alarm Response Procedure for EXCITATION SYSTEM TRANSFER TO MANUAL (2-9-8A, Window 3) **Or** 2-OI-47, Turbine-Generator System, Section 3.4 H.1, H.3, K and/or Section 8.10, Transferring the Voltage Regulator Between Auto and Manual **Or** 0-GOI-300-4, Switchyard Manual, Section 3.1 NN. The Transmission Operator (TOp) is required to be notified of the status of the Voltage Regulator (Automatic or Manual). Expected Action(s): Examinee determines that the Transmission Operator (TOp) must be notified that 2-HS-57-27, VOLTAGE REGULATOR MANUAL / AUTO SELECT, **AUTOMATICALLY** tripped to MANUAL.	**Critical Step** ____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: ONLY If/After the Candidate requests a copy of 2-OI-47, Turbine-Generator System, Attachment 7 (Page 2 of 2) – CALCULATED CAPABILITY CURVE, provide them a hard copy.	
Step 2: When 2-HS-57-27, VOLTAGE REGULATOR MANUAL / AUTO SELECT, **AUTOMATICALLY** tripped to MANUAL, EXCITATION SYSTEM TRANSFER TO MANUAL (2-9-8A, Window 3) alarms which references 2-OI-47, Turbine-Generator System, Attachment 7. The use of 2-OI-47, Generator Kilovar Limitations Capability Curve - Attachment 7, ensures Main Turbine Generator operating parameters such as MVARs, Hydrogen Pressure, Power Factor (pf), and Megawatts are within their normal operating band. Additionally, 2-OI-47 references 2-OI-35, Generator Hydrogen Cooling System, which states the Hydrogen Pressure normal band is 73 – 79 psig. Expected Action(s): Examinee references the Capability Curve and determines that the given 2-PI-35-17A, GENERATOR HYDROGEN PRESSURE, of 58 psig is below the normal operating Hydrogen Pressure band of 73 to 79 psig for the given 2-XR-57-57, GENERATOR LOAD RECORDER, of 1280 MW. The Examinee determines **either or both** of the following action(s) are required: • Lower Generator Load of 1280 MW since it exceeds permissible operation by being greater than the 60 psig Hydrogen Pressure curve • Raise Hydrogen Pressure to allow for proper generator cooling for 1280 MW operation	**Critical Step** ____ SAT _____UNSAT ____N/A
EXAMINER CUE: Once the Operator completes Step 2 above, the task is complete.	
END OF TASK	

STOP TIME:_____

Job Performance Measure (JPM)

Provide to Applicant

****

CLASSROOM: I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

Unit 2 is operating at 100% Rated Thermal Power (RTP) with severe storms in the area.

At 0800:

- 4KV Board Voltage oscillations are observed in Unit 2 Main Control Room
- 0-AOI-57-1E, Grid Instability, is entered by the Nuclear Unit Senior Operator (NUSO)
- 2-XR-57-57, GENERATOR LOAD RECORDER, is 1280 MW
- 2-EI-57-51, GENERATOR MVAR, is (+) 100 MVARs (outgoing)
- 2-PI-35-17A, GENERATOR HYDROGEN PRESSURE, is 58 psig

At 0802:

- 2-HS-57-27, VOLTAGE REGULATOR MANUAL / AUTO SELECT, **AUTOMATICALLY** tripped to MANUAL

INITIATING CUE:

For the **CURRENT** conditions and in accordance with applicable procedure(s), the Nuclear Unit Senior Operator (NUSO) directs you as the Balance of Plant Operator (BOP) to determine:

1. Who, if anyone, is required to be notified?
2. If any Generator Calculated Capability Curve parameters require adjustment?
    - Yes or No
    - If Yes, list parameter(s) and action(s) to be taken?

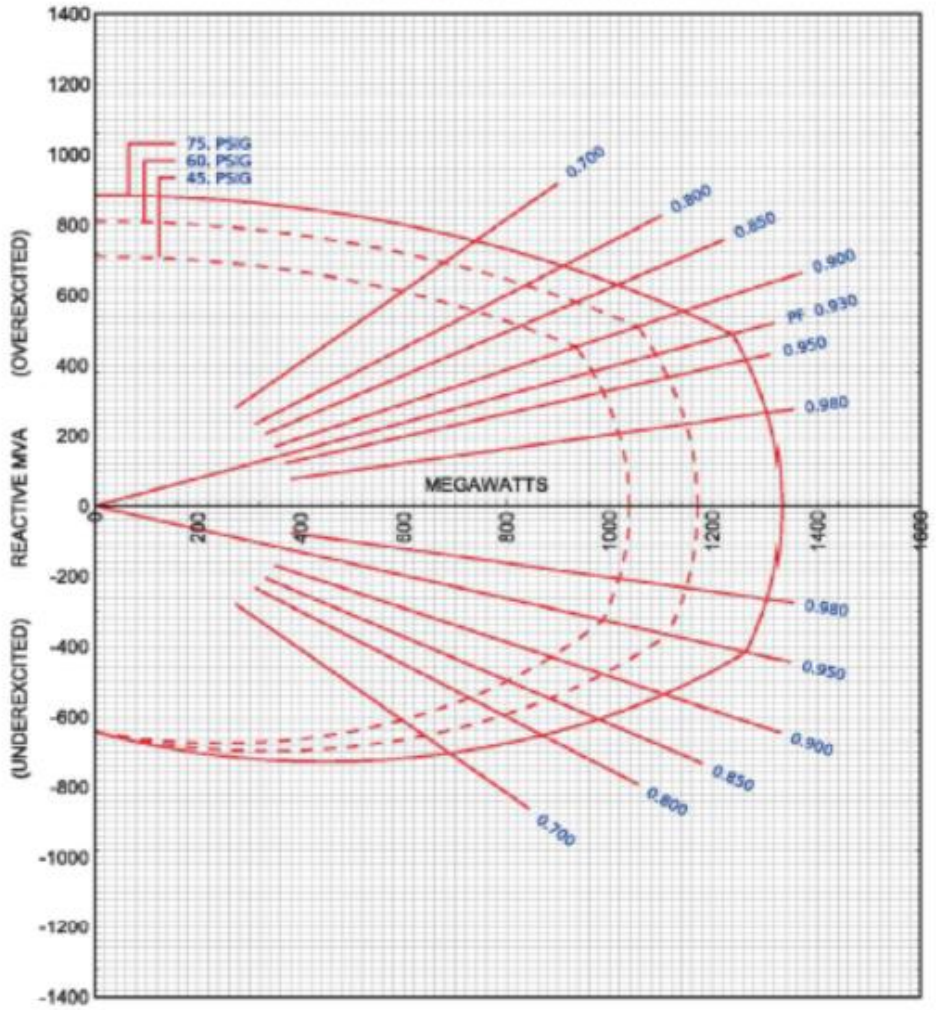
_____
_____
_____
_____
_____
_____

ONLY AFTER REQUESTED - Provide to Applicant

Attachment 7
(Page 2 of 2)

SIEMENS ENERGY, INC
CALCULATED CAPABILITY CURVES
AT 100 PERCENT VOLTAGE (22.0 kV)
WITH WATER COOLED STATOR
HYDROGEN INNER-COOLED TURBINE GENERATOR

CURVE NO. 090506GLC



1332.0 MVA PF 0.930 22.0 kV 34955 A
3 PHASE 60 Hz 1800 RPM SCR 0.58 75 PSIG
RATED COLD GAS 46 °C RATED STATOR WATER 50°C
DATE - 2015-06-26 CURVE NO. - 090506GLC

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Radiation Exposure Limits Under Emergency Conditions	
JPM NUMBER:	645	REVISION:	5	

TASK APPLICABILITY:	☐ SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	RO 3.8		
K/A No. &STATEMENT:	2.3.11 Ability to control radiation releases		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	RADIATION CONTROL-ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☒ YES	☒ NO	
TIME FOR COMPLETION:	8 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number 645

RO ____ DATE: _____

TASK STANDARD: Given radiation levels in an area, the Examinee is expected to determine the expected dose for an emergency evolution and determine if the administrative dose limit has been exceeded in accordance with EPIP-15, Emergency Exposure.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely
OF-3 Operating the Plant with A Conservative Bias

PRA: N/A

REFERENCES/PROCEDURES NEEDED: EPIP-15
NPG-SPP-05.1

VALIDATION TIME: 8 minutes

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Rev No.	Effective Date	Pages Affected	Description
2	09/10/18	All	Updated format
3	10/9/19	All	Incorporated feedback/comments.
4	10/7/19	All	Adjusted to allow dose to be less than 25 REM
5	02/08/23	All	Adjusted to RO JPM to allow dose to be greater than 25 REM

Procedure Revisions

Procedure	Revision
EPIP-15	13
NPG-SPP-05.1	16

****

Classroom

****

INITIAL CONDITIONS:

A General Emergency has been declared on Unit 2. The Technical Support Center (TSC) is **NOT** currently activated and Site Emergency Director duties remain in the Control Room.

You have volunteered as an Operator to stop a large off-site release by manually closing 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE. Radiation Protection Supervision has briefed you on the following information:

- The travel path dose rate is 8 REM/hr to the valve in question
- Dose rate is 50 REM/hr at 2-FCV-73-3
- It is estimated that you will take 7 minutes travel time **(each way)** to and from the valve
- It is estimated that it will take 32 minutes to close the valve
- You have zero dose to date for the year and NO previous emergency dose has been received

INITIATING CUE:

As the Operator who has volunteered to close 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE, perform the following:

1. Determine the expected dose to perform this task due to the radiation dose levels.
2. In accordance with applicable procedures, determine if any emergency dose limit(s) will be exceeded as a result of the emergency task.

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: Determine expected exposure received for this task. Expected Action(s): Examinee determines the total expected dose of 28.37 REM: Travel to/from: (8 REM/HR)(1 HR/60MIN)(14 MIN) = 1.87 REM (Acceptable band is 1.8666 - 1.9 REM) Time at valve: (50 REM/HR)(0.53 HR) = 26.5 REM (Acceptable band is 26.665 - 26.7 REM) Total expected dose 1.87 REM + 26.5 REM = 28.37 REM (Acceptable band is 28.37 – 28.6 REM)	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: EPIP-15, Emergency Exposure 3.4 Dose Limits for Workers During Emergencies D. Dose Limits for Lifesaving Activities and the Protection of Large Populations. Dose Limit (REM) — Receptor 25 — Whole Body (TEDE) 75 — Lens of the Eye 250 — All other Organs Expected Action(s): Examinee determines that the Total expected dose from JPM Step 1 will exceed the Dose Limits for Lifesaving Activities and the Protection of Large Populations (25 REM Whole Body – TEDE) in accordance with EPIP-15, Emergency Exposure.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Examinee may elect to list the TVA Annual Dose Admin Limit of 2 REM TEDE in accordance with NPG-SPP-05-.1, Table 1 (see below), however given that an emergency condition exist, this is not a Critical Step.	
END OF TASK	

STOP TIME _____

KEY

BFN Unit 0	EMERGENCY EXPOSURE	EPIP-15 Rev. 0013 Page 7 of 11

3.4 Dose Limits for Workers During Emergencies (continued)

D. Dose Limits for Lifesaving Activities and the Protection of Large Populations

Dose Limit (rem)	Receptor
25	Whole Body (TEDE)
75	Lens of the Eye
250	All Other Organs

E. Dose Limits Greater than 25 Rem for Lifesaving Activities or to Avoid Extensive Exposures to Large Populations

1. Situations may occur in which a dose in excess of 25 rem would be required in order to carry out lifesaving operations or to avoid extensive exposures to large populations. It is not possible to prejudge the risk that one person should be allowed to take to save the life of others.

2. Personnel made knowledgeable of the risks involved with radiation exposures through training or briefings utilizing the information contained within Appendix A and selected on a voluntary basis may be allowed to exceed the 25 rem emergency dose exposure limit for the purpose of lifesaving activities or to avoid extensive exposures to large populations.

3. Acknowledgment of this decision shall be documented on Appendix B of this procedure by the individual involved in this activity.

3.2.4 Exposure Control (continued)

5. The TVA Nuclear Administrative Dose Level Program is summarized in Table 1 below:

TABLE 1

ADMINISTRATIVE DOSE LEVEL PROGRAM

Dose Equivalent (Rem)	Requirement	Authorization to Exceed (signatures)
Up to **0.5 TEDE** (or 1.5 LDE, 5.0 SDE and 5.0 SDE,ME)	Statement of current year dose and previous years dose signed by individual	Not applicable
Up to **2.0 TEDE** (or 12 LDE, 40 SDE and 40 SDE ME) all sources	NRC FORM-4 or equivalent to document current year and previous years dose equivalent	Not applicable
To exceed **2.0 TEDE** all sources	Same as above	RPM/RSO
To exceed **3.0 TEDE** all sources	Same as above	RPM/RSO, and Plant Manager[1]
To exceed **4.0 TEDE** (or 12 LDE, 40 SDE and 40 SDE ME) all sources	Same as above	RPM/RSO, Plant Manager[1], and Site VP[2]
To exceed **5.0 TEDE[3] all sources**	Form-4 information must be verified and a Planned Special Exposure initiated in Accordance with RCTP-114	RPM/RSO, Plant Manager[1], and Site VP[2]

[1] At non-nuclear plant sites, this will be the RSO's immediate supervisor.

[2] At non-nuclear plant sites, this will be the applicable TVA VP.

[3] Authorizations for a planned special exposure will only be considered in an exceptional situation when alternatives that might avoid the dose estimated to result from the planned special exposure are unavailable or impractical.

6. ADLs are based on dosimeter(s) used in determining the reported dose. Results which exceed an administrative level, based on other dosimeter data, do not violate the ADL.

NOTE

Exceeding an ADL as a result of an incorrect signed documentation of current year or previous years dose does not constitute a violation of the intent of the ADL.

7. When visitor or contract personnel have more restrictive dose limits than TVA - the more restrictive limits will be used. It is the responsibility of the contractor to provide written notification to RP (or RSO) of any company administrative limit.

8. To ensure that ADLs are not exceeded, an administrative control system shall be maintained.

9. Individuals under the age of 18 shall not enter radiologically controlled areas.

Job Performance Measure (JPM)

Provide to Applicant

****

Classroom

****

INITIAL CONDITIONS:

A General Emergency has been declared on Unit 2. The Technical Support Center (TSC) is **NOT** currently activated and Site Emergency Director duties remain in the Control Room.

You have volunteered as an Operator to stop a large off-site release by manually closing 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE. Radiation Protection Supervision has briefed you on the following information:

- The travel path dose rate is 8 REM/hr to the valve in question
- Dose rate is 50 REM/hr at 2-FCV-73-3
- It is estimated that you will take 7 minutes travel time **(each way)** to and from the valve
- It is estimated that it will take 32 minutes to close the valve
- You have zero dose to date for the year and NO previous emergency dose has been received

INITIATING CUE:

As the Operator who has volunteered to close 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE, perform the following:

1. Determine the expected dose to perform this task due to the radiation dose levels.
2. In accordance with applicable procedures, determine if any emergency dose limit(s) will be exceeded as a result of the emergency task.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Fitness For Duty Determination
JPM NUMBER:	507-SRO	REVISION:	3

TASK APPLICABILITY:	☒ SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	SRO 4.3		
K/A No. &STATEMENT:	2.1.39 Knowledge of conservative decision-making practices.		
RELATED PRA INFORMATION:	N/A		
SAFETY FUNCTION:	CONDUCT OF OPERATIONS - ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	10 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: __507-SRO__

RO ____ SRO ____ DATE: _____

TASK STANDARD: Using NPG-SPP-14.1, Fitness for Duty and Fatigue Management, and given Fitness for Duty information on three employees, the Examinee is expected to determine which individuals can be called in to work and any action that must be taken upon arrival at the plant.

Operator Fundamental evaluated:
OF-2 Controlling Plant Evolutions Precisely

PRA: NA

REFERENCES/PROCEDURES NEEDED: NPG-SPP-14.1

VERIFICATION TIME: __10 min__

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
2	09/12/2017	ALL	Updated JPM, converted to new format
3	02/10/2023	ALL	Update JPM

Procedure Revisions

Procedure	Revision
NPG-SPP-14.1	21

****

Classroom

****

INITIAL CONDITIONS:

You are a Nuclear Unit Senior Operator (NUSO). Due to conditions that require actions from a Site Area Emergency Declaration on Unit 1, several personnel have been called in as requested by supervision for **UNSCHEDULED** work to assist with repairs as listed below.

	Answer to "have you consumed alcohol in the last 5 hours?"	Answer to "are you fit for duty?"
TVA Contract Employee	YES	YES
TVA Employee #1	YES	NO
TVA Employee #2	YES	YES
Additionally, the NRC Resident Inspector has been informed about plant status and has arrived at the plant.		

Security informed you that the NRC Resident Inspector is under the influence of a medication or alcohol.

INITIATING CUE:

In reference to the above personnel being called in during the emergency and allowed **UNESCORTED** access to the plant, complete the attached table in accordance with applicable procedure(s) to list the action(s) required for each respective person.

_____

_____

_____

_____

_____

_____

Guide

	Can be called in: Yes/No (Why)	1. Action(s) required if any (N/A if none)	2. Action(s) required if any (N/A if none)
TVA Contract Employee	Yes	NPG-SPP-14.1, Section 3.2.23 NOTE – Considered true emergency situation due to Site Area Emergency Declaration	NPG-SPP-14.1, Section 3.2.23.B.4 - Requires a saliva test that indicates <0.040 before being allowed inside the protected area and permitted to work.
TVA Employee #1	No	NPG-SPP-14.1, Section 3.2.23.B.3-4 - Should **NOT** be called in, since they answered "NO" when asked if he/she was fit for duty.	NPG-SPP-14.1, Section 3.2.23.B.6 - Regardless of reasons for stating not fit for duty, but still called in, an appropriate determination of fitness must be completed PRIOR to allowing the individual to begin work.
TVA Employee #2	Yes	NPG-SPP-14.1, Section 3.2.23.B.4-5 - Notify Security to immediately deactivate individual's site badge access.	NPG-SPP-14.1, Section 3.2.23.B.4 - Requires a saliva test that indicates <0.040 before being allowed inside the protected area and permitted to work.
NRC Resident Inspector	**N/A**	NPG-SPP-14.1, Section 2.2.E – TVA may not deny access, will escort.	The NRC Region II Administrator (or NRC Operations Center) shall be immediately notified by telephone followed by written notification.

KEY-Step 1

2.1 Applicability (continued)

2. All persons required to physically report to the Technical Support Center (TSC) or an Emergency Operations Facility.

3. All FFD program personnel who are involved in the day-to-day operations of the program and whose duties require them to have the following types of access or perform the following activities:

   a. All persons who can link test results with the individual who was tested before an FFD Program or policy violation determination is made, including, but not limited to the Medical Review Officer (MRO);

   b. All persons who make determination of fitness;

   c. All persons who make Authorization decisions (Reviewing Officials);

   d. All persons involved in selecting or notifying individuals for testing; and

   e. All persons involved in the collection or onsite testing of specimens.

2.2 Exemptions

NOTE

While the following individuals are exempt from specific TVA NPG FFD Program elements, they must still adhere to the overall FFD Program requirements to perform their tasks in a reliable and trustworthy manner AND to not be under the influence of any legal or illegal substance or mentally or physically impaired from any cause which in any way could adversely affect their ability to safely and competently perform their duties. Additionally, these individuals must still adhere to any associated TVA NPG FFD Program-related contract agreements/requirements.

A. This SPP does not apply to NRC employees, NRC contractors who are certified by the NRC as being subject to a Federally accepted program which meets or exceeds the requirements of 10 CFR 26 and the NRC program for drug testing, law enforcement personnel, or non-TVA offsite emergency response personnel while responding onsite.

B. This SPP does not apply to individuals who are not employed by TVA, who do not routinely provide FFD Program services to TVA and whose normal workplace is not at a TVA facility, but who may be called on to provide a FFD Program service, including but not limited to: collecting specimens for drug and alcohol testing; performing behavioral observation; or, providing input to a Determination of Fitness. Such individuals may include, but are not limited to: hospital, Employee Assistance Program (EAP) or substance abuse treatment facility personnel; or, other medical professionals.

C. This SPP does not apply to any escorted individuals who have NOT applied for UA/UAA.

KEY-Step 1

2.2 Exemptions (continued)

D. Short-term and offsite TVA NPG contractors and vendors who do not require entry to the protected area may be exempted from the requirements for FFD training. Such exemptions will be determined on a case-by-case basis by the applicable Contract Technical Steward (CTS).

E. If TVA has a reasonable belief that a NRC employee or NRC contractor may be under the influence of any substance, or is otherwise unfit for duty, TVA may not deny access but will escort the individual. In any instance of this occurrence, the NRC Region II Administrator will be notified immediately by telephone followed by written notification (for example, e-mail or fax) to document the oral notification. If the Region II Administrator cannot be reached, the NRC Operations Center will be notified.

F. Additional exemptions which do not affect regulatory requirements and commitments may be granted on a case-by-case basis by the Manager, Nuclear Access Services (NAS), or designee.

KEY-Steps 2-4

3.2.22 Contractor/Vendor Programs (continued)

B. TVA NPG may, at its discretion, accept, either whole or in part, FFD programs administered by contractor, vendors, or industry groups. All such programs must meet the minimum requirements of 10 CFR 26.

C. Corporate NAS will be responsible for review and approval of any such programs.

3.2.23 Call-in for Unscheduled Work

NOTE

Contractor employees will not be requested to report if alcohol has been consumed within the past five (5) hours except in true emergency situations.

A. All individuals are expected to not consume alcohol for at least five (5) hours prior to reporting for SCHEDULED work and to report fit and within FFD guidelines. If called for unscheduled work, including unscheduled work to respond to an emergency, the individual's fitness for duty work must be determined.

B. The following must be done whenever an individual is being called in for unscheduled work:

1. 1. The caller will ask and will document (on Form NPG-SPP-14.1-1 or similar-type form) the individual's responses to BOTH the following questions.

   a. Are you fit for duty?

   b. Have you consumed alcohol within the past five (5) hours?

2. The individual must advise the caller and the supervisor if he or she believes that he or she is unfit for duty for reasons such as illness, fatigue or other potentially impairing conditions AND if he or she has consumed alcohol within the past five (5) hours.

3. If the answer to the "Are you fit for duty?" question is "no", then the individual should not be requested to report for unscheduled work. The caller will document the individual's stated reason(s) for being unfit to report.

4. Consumption of alcohol during the 5-hour abstinence period may not by itself preclude the use of individuals who are needed to respond to an emergency. If the answer to alcohol consumption question is "yes", and the supervisor request that the individual report for work, then Nuclear Security on site will be notified immediately and be requested to immediately deactivate the individual's site badge in the security system, then administer a saliva test to the individual. **The individual will not be allowed inside the protected area before testing.** This test will be administered as soon as the individual arrives on site and only by personnel trained to administer testing. **The test will be administered outside the Protected Area.**

KEY-Steps 2-4

NPG Standard Programs and Processes	Fitness-For-Duty	NPG-SPP-14.1 Rev. 0021 Page 40 of 65

3.2.23 Call-in for Unscheduled Work (continued)

5. If the results are 0.040 or above, the individual will NOT be permitted to work. This will NOT be considered a positive test result for FFD purposes. No sanction will be imposed on an individual who is called in to perform any unscheduled working tour for having consumed alcohol within the pre-duty abstinence period.

6. If an individual reports that he or she considers him or herself to be unfit for other reasons, including illness, fatigue or other potentially impairing conditions, and the individual is called in, an appropriate determination of fitness must be completed **PRIOR** to allowing the individual to begin work.

7. If the results of the determination of fitness indicate the individual may be impaired and the individual is not required to respond to an emergency, he or she will **NOT** be assigned duties within the scope of 10 CFR Part 26.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: (*See above Step 1 key*) Refers to NPG-SPP-14.1, Fitness-For-Duty and Fatigue Management, Section 2.2.E – Exemptions, to determine what actions are required with respect to the NRC Resident Inspector. Expected Action(s): Examinee determines that the NRC Resident Inspector may not be denied access, but shall be escorted. Also states that the NRC Region II Administrator (or NRC Operations Center) shall be immediately notified by telephone followed by written notification.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: (*See above Steps 2 - 4 key*) Refers to NPG-SPP-14.1, Section 3.2.23 B.4, B.5 – Call-in for Unscheduled Work, to determine what actions are required with respect to the TVA Contract Employee: Expected Action(s): Examinee determines that the TVA Contractor can be called in due to a declared emergency situation, Notify Security to immediately deactivate individual's badge until a saliva test that indicates <0.040 is completed to allow them inside the protected area and permitted to work.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: (*See above Steps 2 - 4 key*) Refers to NPG-SPP-14.1, Section 3.2.23 B.3 – Call-in for Unscheduled Work, to determine what actions are required with respect to TVA Employee #1: Expected Action(s): Examinee determines that TVA Employee #1 should **NOT** be called in, because he/she answered "NO" when asked if he/she was fit for duty.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: For JPM Step 3, the Examinee may elect to state that NPG-SPP-14.1, Section 3.2.23 B.6 states that if an individual reports that he or she considers him or herself to be unfit for other reasons, including illness, fatigue or other potentially impairing conditions, and the individual is still called in, an appropriate determination of fitness must be completed PRIOR to allowing the individual to begin work. However, this is not required as a Critical Step.	
Step 4: (*See above Steps 2 - 4 key*) Refers to NPG-SPP-14.1, Section 3.2.23 B.4, B.5 – Call-in for Unscheduled Work, to determine what actions are required with respect to TVA Employee #2: Expected Action(s): Examinee determines that TVA Employee #2 can be called in. Notify Security to immediately deactivate individual's badge until a saliva test that indicates <0.040 is completed to allow them inside the protected area and permitted to work.	**Critical Step** _____ SAT _____UNSAT _____N/A
END OF TASK	

STOP TIME:_____

Job Performance Measure (JPM)

Provide to Applicant

****

Classroom

****

INITIAL CONDITIONS:

You are a Nuclear Unit Senior Operator (NUSO). Due to conditions that require actions from a Site Area Emergency Declaration on Unit 1, several personnel have been called in as requested by supervision for **UNSCHEDULED** work to assist with repairs as listed below.

	Answer to "have you consumed alcohol in the last 5 hours?"	Answer to "are you fit for duty?"
TVA Contract Employee	YES	YES
TVA Employee #1	YES	NO
TVA Employee #2	YES	YES
Additionally, the NRC Resident Inspector has been informed about plant status and has arrived at the plant.		

Security informed you that the NRC Resident Inspector is under the influence of a medication or alcohol.

INITIATING CUE:

In reference to the above personnel being called in during the emergency and allowed **UNESCORTED** access to the plant, complete the attached table in accordance with applicable procedure(s) to list the action(s) required for each respective person.

	Can be called in: Yes/No (Why)	1. Action(s) required if any (N/A if none)	2. Action(s) required if any (N/A if none)
TVA Contract Employee			
TVA Employee #1			
TVA Employee #2			
NRC Resident Inspector	**N/A**		

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	NRC Event Notification
JPM NUMBER:	665-SRO	REVISION:	3

TASK APPLICABILITY:	☒ SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	SRO 3.8		
K/A No. &STATEMENT:	2.1.18 Ability to make accurate, clear, and concise logs, records, status boards, and reports.		
RELATED PRA INFORMATION:	None		
SAFETY FUNCTION:	CONDUCT OF OPERATIONS - ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	10 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 665-SRO

RO _____ SRO _____ DATE: _____

TASK STANDARD: Given a failure of 2-PCV-1-22, MAIN STEAM LINE 'B' RELIEF VALVE, with HPCI already tagged out for scheduled maintenance, the Examinee is expected to determine Technical Specification 3.5.1 Action(s) and Nuclear Regulatory Reporting requirements in accordance with NPG-SPP-3.5, Regulatory Reporting Requirements.

Operator Fundamental evaluated:
OF-3 Operating the Plant With A Conservative Bias

PRA: N/A

REFERENCES/PROCEDURES NEEDED: NPG-SPP-3.5
Technical Specification 3.0
Technical Specification 3.5.1
2-AOI-1-1

VALIDATION TIME: 10 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Rev No.	Effective Date	Pages Affected	Description
1	10/01/2018	ALL	Revised JPM; updated format
2	08/12/2019	ALL	Revised JPM
3	12/06/2022	ALL	Revised JPM

Procedure Revisions

Procedure	Revision
Unit 2 Tech Spec 3.0	266
Unit 2 Tech Spec 3.5.1	269
NPG-SPP-3.5	18

****

Classroom

****

INITIAL CONDITIONS:

Unit 2 is operating at 100% Reactor Power with the following plant conditions:

- At 0900, HPCI is tagged out to support scheduled maintenance and the appropriate Technical Specification has been entered
- At 0930, 2-PCV-1-22, MAIN STEAM LINE 'B' RELIEF VALVE, is declared INOPERABLE due to breaker issues

INITIATING CUES:

As the Nuclear Unit Senior Operator (NUSO), evaluate the given conditions and complete the following as it relates to the most **RESTRICTIVE** conditions:

- Determine what Technical Specification Action(s) must be taken (if any)?
- Determine the Regulatory Reporting Requirement(s) (if any)?

START TIME _____

STEP / STANDARD	SAT / UNSAT
Step 1: Technical Specification 3.5.1, ECCS - Operating	**Critical Step** _____ SAT _____UNSAT _____N/A

ECCS - Operating
3.5.1

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable to HPCI.

H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. OR HPCI System and one or more ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

Expected Action(s):

Examinee determines that LCO 3.5.1, CONDITION H, is applicable since HPCI and an ADS valve are both INOPERABLE requiring entry into LCO 3.0.3 Immediately (*See LCO 3.0.3 on next page*).

STEP / STANDARD	SAT / UNSAT

EXAMINER NOTE: JPM Step 1, neither of the following are required. The Examinee may possibly elect to refer to a variety of references to verify if the given 2-PCV-1-22, MAIN STEAM LINE 'B' RELIEF VALVE, is one of the six ADS valves. Additionally, they may also elect to reference Technical Specification LCO 3.0.3 which states that the Unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Tech Spec 3.5.1 applicability is MODE 1, 2, and 3.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 10 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

(continued)

STEP / STANDARD	SAT / UNSAT
Step 2: Evaluates NPG-SPP-3.5, Regulatory Reporting Requirements, Attachment 1, Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants: Section 3.1, Immediate Notification – NRC C. The following criteria require 4-hour notification: 1. §50.72(b)(2)(i) – The initiation of any Nuclear plant shutdown required by the plant's Technical Specifications. Expected Action(s): Examinee determines that a 4-hour NRC Notification is required given that LCO 3.0.3 entry is required immediately, placing Unit 2 in a MODE or other specified condition in which the LCO is not applicable.	**Critical Step** _____ SAT _____UNSAT _____N/A
Examiner Note: If the candidate requests NRC Form 361, Reactor Plant Event Notification Worksheet, state that another Operator will complete the required form.	
END OF TASK	

STOP TIME _____

Job Performance Measure (JPM)

Provide to Applicant

****

Classroom

****

INITIAL CONDITIONS:

Unit 2 is operating at 100% Reactor Power with the following plant conditions:

- At 0900, HPCI is tagged out to support scheduled maintenance and the appropriate Technical Specification has been entered
- At 0930, 2-PCV-1-22, MAIN STEAM LINE ‘B’ RELIEF VALVE, is declared INOPERABLE

INITIATING CUES:

As the Nuclear Unit Senior Operator (NUSO), evaluate the given conditions and complete the following:

- Determine what Technical Specification Action(s) must be taken (if any)?
- Determine the most restrictive Regulatory Reporting Requirement(s) (if any)?

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Review Jet Pump Operability Surveillance (SR)
JPM NUMBER:	676-SRO	REVISION:	2

TASK APPLICABILITY:	☒ SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	S-000-AD-27, Assess LCO/TRM Actions to determine OPERABILITY		
K/A RATINGS:	SRO 4.6		
K/A No. &STATEMENT:	2.2.37 Ability to determine operability or availability of safety-related equipment (SRO Only)		
RELATED PRA INFORMATION:	None		
SAFETY FUNCTION:	EQUIPMENT CONTROL - ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	15 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 676-SRO

SRO _____ DATE: _____

TASK STANDARD: The Examinee is expected to determine that the SR for Jet Pump Operability has **NOT** been filled out correctly and that Jet Pump 9 is actually INOPERABLE. As a result, enters LCO 3.4.2, Jet Pumps Condition A, Required Action A.1 - Be in MODE 3 in 12 hours.

Operator Fundamental evaluated:
OF-1 Monitoring Plant Indications and Conditions Closely

PRA: N/A

REFERENCES/PROCEDURES NEEDED: (1) Completed 2-SR-3.4.2.1, Jet Pump Mismatch & Operability, with Jet Pump 9 INOPERABLE, but **NOT** identified by the Unit Operator.
(2) Partially completed TVA 40753, STS
(3) Unit 2 Tech Spec 3.4.2 and Bases.

VALIDATION TIME: 15 minutes

PERFORMANCE TIME: _____

COMMENTS: _____
_____
_____
_____
_____
_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Rev No.	Effective Date	Pages Affected	Description
1	09/13/2018	ALL	Updated JPM
2	08/13/2019	ALL	Updated JPM
3	10/9/2019	ALL	Updated JPM
4	2/22/23	ALL	Updated JPM

Procedure Revisions

Procedure	Revision
2-SR-3.4.2.1	53
U2 TS 3.4.2	253
U2 TSB 3.4.2	0

****

Classroom

****

INITIAL CONDITIONS:

Unit 2 is operating at 100% RTP, online for 103 days. You are the Unit 2 Nuclear Unit Senior Operator (NUSO). The Balance of Plant Operator has just completed 2-SR-3.4.2.1, Jet Pump Mismatch and Operability, and has given it to you for review.

INITIATING CUES:

Conduct a review of the completed 2-SR-3.4.2.1, Jet Pump Mismatch and Operability.

Determine OPERABILITY in accordance with the appropriate procedure(s).

START TIME _____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: A Surveillance Task Sheet (STS) is provided with the SR, however ONLY the SR is required to be reviewed for JPM grading purposes. If the Examinee has questions/comments related to the STS, state another NUSO will address them and continue with task.	
Step 1: Nuclear Unit Senior Operator (NUSO) checks that the Unit Operator has checked and initialed each step. Expected Action(s): Examinee notes that all initials are present and all calculations are completed.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: See the attached 2-SR-3.4.2.1, page 20 and 23 KEY that follows JPM Steps 2-3 and 5, respectively.	
Step 2: 7.4.2 Loop 2B Recirc. Pump, Jet Pump Flow – Recirc. Pump Speed Nuclear Unit Senior Operator (NUSO) checks that the Unit Operator has identified any anomalies. Expected Action(s): Examinee notes that the Unit Operator **incorrectly** identified that Loop 2B Recirculation Pump Speed and Recirculation Pump Flow (Step 7.4.2[1]) was marked as YES (Plot falls between the bold lines). However, it fell outside the Attachment 3 curve, 7.4.2 [1] should be correctly marked as **NO.**	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: 7.4.2 Loop 2B Recirc. Pump, Jet Pump Flow to Recirc. Pump Speed Nuclear Unit Senior Operator (NUSO) checks that the Unit Operator has identified any anomalies. Expected Action(s): 7.4.2[3] Using Steps 7.4.[1] **AND** 7.4.2[2] from above requires both be marked correctly. Subsequently, Examinee notes that the Unit Operator **incorrectly** marked Step 7.4.2[3], as YES (Jet Pump Loop 2B criteria is satisfied). However, since 7.4.2 [1] should be marked (from JPM Step 2 above) as **NO**, 7.4.2[3] should also be correctly marked as **NO.**	**Critical Step** _____ SAT _____UNSAT _____N/A

KEY

BFN Unit 2	Jet Pump Mismatch and Operability	2-SR-3.4.2.1 Rev. 0053 Page 20 of 32

Date: Today

NOTES

1) If a Recirculation Pump is not in service, then the associated steps can be marked as No.
2) If Data falls on or outside the bold lines on the graph, then the step can be marked as No.

7.4.2 Loop 2B Recirculation Pump and Jet Pump Flow to Recirculation Pump Speed

[1] Using the 2B Pump Speed recorded in Step 7.2.2[1] and 2B Pump Flow recorded in Step 7.2.2[2]:

CHECK the plot falls between the two bold lines on Attachment 3 and **RECORD** below.

Plot falls between the bold lines Yes ☐ No ☑

[2] Using the 2B Pump Speed recorded in Step 7.2.2[1] and 2B Jet Pump Flow recorded in Step 7.2.2[3]:

CHECK the plot falls between the two bold lines on Attachment 4 and **RECORD** below.

Plot falls between the bold lines Yes ☑ No ☐

[3] Using Steps 7.4.2[1] and 7.4.2[2] from above:

DETERMINE if the Jet Pump Loop 2B criteria is satisfied by marking below if both steps are marked as Yes.

Jet Pump Loop 2B criteria is satisfied Yes ☐ No ☑

STEP / STANDARD	SAT / UNSAT

EXAMINER NOTE: For JPM Step 4 below, the SRO candidate may question as to why Attachment 7 (Page 1 of 1) Engineering Judgement/Review is marked as N/A and possibly believe that it may be required to be conducted as part of their review to determine OPERABILITY.

If so, inform candidate that an Engineering Judgement/Review has been requested, continue with task. This is explained more in detail in Tech Spec Bases for SR 3.4.2.1 attached on the next page.

Step 4: 7.4.4 Engineering Judgement/Review Criteria

CAUTION

An Engineering Judgment/Review may only be utilized until relationships between Core Flow, Jet Pump Flow, and Recirculation Loop Flow have been established following a Refueling Outage or during the initial weeks of extended Single Loop operation. Engineering Judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a Jet Pump failure. Refer to SR 3.4.2.1 bases.

Nuclear Unit Senior Operator (NUSO) checks that the Unit Operator has identified any anomalies.

[1] **IF** any of the following conditions apply:

- Following Refueling Outage. (See Caution above)
  OR
- The Reactor is in Single Loop Operation (See Caution above)
  OR
- If point(s) plotted in Sections 7.4.1 or 7.4.2, AND 7.4.3 fall on or outside the bolded lines, to determine if the graph(s) need updating, THEN PERFORM Attachment 7, Engineering Judgement/Review: (Otherwise N/A this step and Mark all of Attachment 7 as N/A)

_____ SAT

_____UNSAT

_____N/A

Expected Action(s):

Examinee may take time to analyze this section since a CAUTION exists above the section and 3 separate condition bullets exist. Ultimately, the INITIAL CONDITIONS state that the Unit has been online for 103 days, therefore 7.4.4 [1] and Attachment 7, Engineering Judgement/Review, are both correctly marked as N/A.

STEP / STANDARD	SAT / UNSAT
Jet Pumps B 3.4.2 BASES (continued) SURVEILLANCE REQUIREMENTS SR 3.4.2.1 This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Each recirculation loop must satisfy one of the performance criteria provided. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while baselining new "established patterns," engineering judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.	
Step 5: Nuclear Unit Senior Operator (NUSO) checks that the Unit Operator has identified any anomalies. Expected Action(s): 7.4.5[1] **MARK** the appropriate criteria results for the following: From the table, 7.4.1[3] (correctly marked **YES**) **AND** 7.4.2[3] (**NO** - from JPM Step 3 above) will make the Criteria Results (Both Jet Pump Loops steps are marked as YES) correctly marked as **NO**. However, NUSO notes that the Unit Operator **incorrectly** marked this specific Step as **YES**.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT

KEY

BFN Unit 2	Jet Pump Mismatch and Operability	2-SR-3.4.2.1 Rev. 0053 Page 23 of 32

Date: TODAY

7.4.5 Operability Determination

MARK the appropriate criteria results for the following:
(N/A any criteria not performed)

Steps	Criteria Results	Yes	No	N/A
7.4.1[3] and 7.4.2[3]	Both Jet Pump Loops steps are marked as YES	☐	☑	☐
7.4.3[3]	Jet Pump DP to criteria is marked as YES.	☐	☑	☐
Attachment 7	Engineering Evaluation is marked as YES.	☐	☐	☑

[2] Using the Criteria Results in Step 7.4.5[1]

ENSURE at least one Criteria Results is satisfied and marked as YES.

_____(AC)

Step 6: (See Examiner Note and Step 4 above)

Nuclear Unit Senior Operator (NUSO) checks that the Unit Operator has identified any anomalies.

Expected Action(s):

7.4.5[1] **MARK** the appropriate criteria results for the following:

From the table, Attachment 7, Engineering Evaluation

AND

7.5 Completion and Notifications

[1] IF and Engineering Judgement/Review was performed…

Both Steps above are correctly marked as N/A.

_____ SAT

_____UNSAT

_____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: See the attached 2-SR-3.4.2.1, page 23 KEY above for JPM Step 6.	
Step 7: Nuclear Unit Senior Operator (NUSO) checks that the Unit Operator has identified any anomalies. Expected Action(s): 7.4.5[2] Using the Criteria Results in Step 7.4.5[1]: ENSURE at least one Criteria Results is satisfied and marked as YES. However, NUSO notes that the Unit Operator **incorrectly** initialed this Acceptance Criteria (AC) Step when it should **NOT** be initialed. In conjunction with Loop 2B Jet Pump DP being incorrectly marked as YES in 7.4.2[3] and falling outside the Attachment 3 curve, results in an AC step failure (7.4.5[2]) and an UNSAT performance of the surveillance.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT

Step 8:

Nuclear Unit Senior Operator (NUSO) declares Jet Pump 9 INOPERABLE.

Expected Action(s):

Based on the way 2-SR-3.4.2.1 was incorrectly filled out by the Unit Operator in the above steps, resulting in 7.4.5[2] being initialed as meeting Acceptance Criteria (AC), the NUSO may conduct this review to **incorrectly** conclude that Tech Spec 3.4.2 is met and Jet Pumps are OPERABLE.

However, the NUSO review of 2-SR-3.4.2.1, Tech Spec 3.4.2.1 and Bases should conclude that Jet Pump 9 is **INOPERABLE**.

Critical Step

_____ SAT

_____UNSAT

_____N/A

Jet Pumps
3.4.2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Jet Pumps

LCO 3.4.2 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

END OF TASK

STOP TIME _____

Job Performance Measure (JPM)

Provide to Applicant

****

Classroom

****

INITIAL CONDITIONS:

Unit 2 is operating at 100% RTP, online for 103 days. You are the Unit 2 Nuclear Unit Senior Operator (NUSO). The Balance of Plant Operator has just completed 2-SR-3.4.2.1, Jet Pump Mismatch and Operability, and has given it to you for review.

INITIATING CUES:

Conduct a review of the completed 2-SR-3.4.2.1, Jet Pump Mismatch and Operability.

Determine OPERABILITY in accordance with the appropriate procedure(s).

Surveillance Task Sheet (STS)

Work Order# 122967615

PM#: P2383
Procedure: 2-SR-3.4.2.1
Title:
2-SR-3.4.2.1 - JET PUMP MISMATCH & OPERABILITY

Data Sheets Attached:

Perf Grp:	OPS	**Unit:**	2
Test Reason:	Periodic	**Loop/Div:**	/NA
Due Date:	6/14/23	**Tech Spec:**	Y
Reg Date:		**Late Date:**	
Reg Freq:	1 DAYS	**Freqency:**	1 DAYS
EQ:		**ASME OM/XI:**	
Applicable Mode:	1,2	**Perf Modes:**	
Clearance Required:	NO	**Digital:**	N
Dry-Cask Storage:	N	**LCO Entered:**	N

Performed By:

Print Name	Signature	Initial	Section
Mark Mann		M	OPS

Subsequent Reviews:

Group	Signature	Date
NSSS		

Permanent Comments:

Remarks: NOT REQD UNTIL 24 HRS AFTER BOTH RECIRC LOOPS ARE IN OPERATION;App Notes:MODES 1 & 2;Freq Notes: NOT REQ'D UNTIL 24 HRS AFTER BOTH RECIRC LOOPS IN OPERATION.NOT REQD UNTIL 4 HRS AFTER ASSOC RECIRC LOOP IN OPERATION OR 24 HRS AFTER >23% RTP

M C Mullay — Today 0700
Authorization to Begin: SRO — Date & Time

Today 0730 — Today 0800
Start Date & Time — Completion Date & Time

Note: Ensure these dates and times are reflected in Maximo

Was this a complete or partial performance?
(Explain Partial in Remarks below) Complete [✓] Partial [ ]

Were all Tech Spec / Tech Req / ISFSI CoC / ODCM / Fire Protection Req / AMSAC* acceptance criteria satisfied?
Yes [✓] No [ ] N/A [ ]

Were all other acceptance criteria satisifed?
Yes [✓] No [ ] N/A [ ]

If all Tech Spec / Tech Req / ISFSI CoC / ODCM / Fire Protection Req / AMSAC* criteria were not satisfied, was an LCO / TR / ODCM / OR action required? (Explain in Remarks)
Yes [ ] No [ ] N/A [✓]

CR #

Alert Scheduling Required? Yes [ ] No [ ] N/A [✓]

Notify SI Coordinator? Yes [ ] No [ ] N/A [✓]

Copy of STS sent to Work Control (next business day)

Initials — Date

M. Mann — Today
Test Coordinator / Lead — Date

Acceptance Criteria Review: SRO — Date & Time

Independent Reviewer — Date & Time

Remarks:

CONTROLLED

Browns Ferry Nuclear Plant

Unit 2

Surveillance Procedure

2-SR-3.4.2.1

Jet Pump Mismatch and Operability

Revision 0053

Quality Related

Level of Use: Continuous Use

Level of Use or Other Information: Key Number P2383

Effective Date: 01-12-2023

Responsible Organization: OPS, Operations

Prepared By: Jonathan Duke

Approved By: Chris Winton

Current Revision Description

Type of Change: Enhancement Tracking Number: 054

PCR's N/A

Documentation: CR 1828413

CR 1828413:

Replaced curves on Attachment 1 through Attachment 6 (expiration date February 1, 2023) with new curves (expiration date October 1, 2023).

Table of Contents

1.0 INTRODUCTION 5
1.1 Purpose 5
1.2 Scope 5
1.3 Frequency 6
1.4 Applicability 6
2.0 REFERENCES 7
2.1 Technical Specifications 7
2.2 Final Safety Analysis Report 7
2.3 Plant Instructions 7
2.4 TVA Drawings 7
2.5 Miscellaneous Documents 7
3.0 PRECAUTIONS AND LIMITATIONS 8
3.1 General Precautions 8
3.2 Operability and LCO's 8
3.3 Equipment 8
3.4 Initiation/Isolation/Trips 8
3.5 Interlocks 8
3.6 Performance Testing 9
4.0 PREREQUISITES 10
5.0 SPECIAL TOOLS AND EQUIPMENT 12
6.0 ACCEPTANCE CRITERIA 12
7.0 PROCEDURE STEPS 14
7.1 Initial Requirements and Notifications 14
7.2 Data Collections 15
7.2.1 Core Power and Flow Readings 15
7.2.2 Recirculation Pump Loops 16
7.2.3 Jet Pump Loops 17
7.3 Tech Spec 3.4.1.1 - Recirculation Loop Mismatch Verification With Both Recirculation Loops In Operation Checks 18
7.4 Tech Spec 3.4.2.1 - Part A -Jet Pump Performance Checks 19

Table of Contents (continued)

7.4.1 Loop 2A Recirculation Pump and Jet Pump Flow to Recirculation Pump Speed 19

7.4.2 Loop 2B Recirculation Pump and Jet Pump Flow to Recirculation Pump Speed 20

7.4.3 Recirculation Jet Pump Diffuser to Lower Plenum Differential Pressure Verification: 21

7.4.4 Engineering Judgment/Review Criteria 22

7.4.5 Operability Determination 23

7.5 Completion and Notifications 24

8.0 RECORDS 25

Attachment 1: 2A Recirculation Loop Pump Speed VS Recirculation Pump Flow Good Thru October 1, 2023 26

Attachment 2: 2A Recirculation Pump Speed VS Jet Pump Flow Good Thru October 1, 2023 27

Attachment 3: 2B Recirculation Loop Pump Speed VS Recirculation Pump Flow Good Thru October 1, 2023 28

Attachment 4: 2B Recirculation Pump Speed VS Jet Pump Flow Good Thru October 1, 2023 29

Attachment 5: Total Core Flow VS 2A Average Jet Pump DP Good Thru October 1, 2023 30

Attachment 6: Total Core Flow VS 2B Average Jet Pump DP Good Thru October 1, 2023 31

Attachment 7: Engineering Judgment/Review 32

1.0 INTRODUCTION

1.1 Purpose

A. This pump operability test is performed to verify Recirculation Loop Jet Pump Flow mismatch for the Recirculation Loops which are in service in conformance with the requirements specified in Technical Specification 3.4.1.1 for dual Recirculation loop operation.

B. Also this test will determine the integrity of the jet pumps of the Reactor Recirculation System in conformance with the requirements specified in Technical Specifications 3.4.2.1.

1.2 Scope

A. This SR is designed to verify jet pump mismatch and to detect significant degradation in jet pump performance that precedes jet pump failure.

B. This SR is required to be performed only when the loop has forced Recirculation flow.

C. The jet pump failure of concern is complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop.

D. This procedure satisfies both SR 3.4.1.1 and SR 3.4.2.1 for dual Recirculation loop operation.

E. This procedure satisfies ONLY SR 3.4.2.1 for single Recirculation loop operation.

F. The procedure 2-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation, satisfies Technical Specification 3.4.1 during single Recirculation loop operation.

NOTES

1) SR 3.4.1.1 is not required to be performed until 24 hours after both Recirculation Loops are in operation.
2) SR 3.4.2.1 is not required to be performed until:
   - 4 hours after associated Recirculation Loop is in operation

   OR

   - 24 hours after > 23% RTP.

1.3 Frequency

A. Recirculation Loop Jet Pump Flow Mismatch (SR 3.4.1.1) if both Recirculation Loops are in operation.

   1. Once per 24 hours

B. Jet Pump Operability (SR 3.4.2.1)

   1. Once per 24 hours

1.4 Applicability

Modes 1 and 2

2.0 REFERENCES

2.1 Technical Specifications

Sections 3.4.1, Recirculation Loops Operating

Sections 3.4.2, Jet Pumps

2.2 Final Safety Analysis Report

Sections 3.3.4, Description (Reactor Vessel Internals Mechanical Design)

Sections 4.3, Reactor Recirculation System

Sections 7.8.5, Description (Reactor Vessel Instrumentation)

Sections 14.6.3, Loss of Coolant Accident (LOCA)

2.3 Plant Instructions

2-OI-68, Reactor Recirculation System

2-GOI-100-1A, Rx Startup from Cold Shutdown to Power Operations (Unit Startup and Power Operation)

2-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation.

2-SR-3.4.1(DLO), Reactor Recirculation System Dual Loop Operation.

NPG-SPP-06.9.2, Surveillance Test Program

2.4 TVA Drawings

2-47E610-68 Series, Mechanical Control Diagram, Reactor Water Recircn System

2-47E817-1 & 2, Flow Diagram, Reactor Water Recirculation

2-47E600-60, Mechanical Instruments and Controls

2.5 Miscellaneous Documents

General Electric SIL 330 and SIL 330 Addenda - Jet Pump Beam Cracks

NUREG/CR - 3052, Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure

Technical Specification Change No. 387, Single Loop Operation (SLO)

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. No Recirc Pump speed changes or Control Rod manipulations are to be made during the performance of this procedure.

B. Improper instrument calibration can severely affect the data and cause unnecessary failures of the test in this SR.

C. Refueling activities such as Fuel Assembly replacement or shuffle, modifications to Fuel support, orifice size or Core Plate bypass flow can affect the relationship between Core Flow and Recirculation Loop Flow. These relationships may need to be re-established each cycle. During initial weeks of operation under such conditions, while baselining new "established patterns," Engineering Judgment of daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

3.2 Operability and LCO's

A. Technical Specification SR 3.4.2.1 criteria c will not be used in this instruction (Each jet pump flow differs by $\leq$ 10% from established patterns). This criteria use individual jet pump flows which is not available at BFN. Criteria b is used for plants with differential pressure instrumentation.

3.3 Equipment

A. The Robicon VFD for each pump and displayed on 2-SI-96-61 (Pump 2A) and 2-SI-96-73 (Pump 2B) or ICS points 96-61 and 96-73. The VFD Control System calculates speed indications using the VFD output frequency and motor (pump) load. Based on these parameters the actual motor speed for any output frequency and load can be calculated. Since the speeds are calculated they should be used only if the two actual speed indications provide by the Bentley-Nevada System cannot be obtained.

3.4 Initiation/Isolation/Trips

None

3.5 Interlocks

None

3.6 Performance Testing

A. Turbulence in the Jet Pump diffuser causes the differential pressure signal to be noisy when the pump is in operation. The proper method for recording differential pressure is to take the average of the high and low readings.

B. Browns Ferry has not operated in Single Loop for a significant period of time. Therefore, not enough Single Loop Operating data has been obtained. Until operation under such conditions and a baseline data has been obtained the Engineering Judgment of daily surveillance results is used to detect significant abnormalities which could indicate a Jet Pump failure.

C. System Engineering should be notified to create new graphs when conditions are such that the end of the plotted area is reached.

D. The Graphs in this procedure are created using the data from the computer. During the Operation Cycle, the graphs used in the Attachments of this procedure changes based upon the core's life.

1. As this occurs an Engineering Judgment/Review should be performed when the graphs fall outside the Attachments to meet the Acceptance Criteria.

2. The Engineering Judgment/Review should establish new graphs to be incorporated into the procedure as time permits.

E. Step 4.0[5] is used to ensure the current graphs are updated on a regular basis.

F. System Engineering should be notified prior to the "Good Thru Date" on any Attachment being exceeded. This will allow System Engineering time to generate new graphs. This date represents 8 months from the date the graphs were created. The graphs can still be used if the "Good Thru Date" is exceeded. The Eight (8) months is used as a guideline and the graphs can be updated more frequently if desired.

G. During startup following a Refueling Outage, the Attachments are used as a guideline and when enough data is obtained System Engineering will create new graphs. If no work has been performed on the Recirc Pumps or Jet Pumps then the graphs may be provided prior to Startup and verified by System Engineering following the Refuel Outage. The graphs should be used in conjunction with the Engineering Judgment/Review processes until new graphs are provided or the graphs provided prior to Startup are validated by Engineering.

Date: Today

4.0 PREREQUISITES

[1] **ENSURE** this copy of 2-SR-3.4.2.1 is the most current revision.

[2] **OBTAIN** a Surveillance Task Sheet (STS) for this procedure and Work Activity. (Key Number P2383)

[3] **ENSURE** Reactor Recirculation System is in operation in accordance with 2-OI-68, Reactor Recirculation System.

[4] **CHECK** the qualified personnel listed below are available to perform this Surveillance procedure.

UO 1

Date: Today

4.0 PREREQUISITES (continued)

[5] Using the following Attachment graphs:

- Attachment 1 - 2A Recirculation Loop Pump Speed VS Recirculation Pump Flow
- Attachment 2 - 2A Recirculation Pump Speed VS Jet Pump Flow
- Attachment 3 - 2B Recirculation Loop Pump Speed VS Recirculation Pump Flow
- Attachment 4 - 2B Recirculation Pump Speed VS Jet Pump Flow
- Attachment 5 - Total Core Flow VS 2A Average Jet Pump DP
- Attachment 6 - Total Core Flow VS 2B Average Jet Pump DP

PERFORM the following:

[5.1] **IF** RTP is > 23% or Section 7.4 is required to be performed, **THEN**

ENSURE the graphs on the Attachments are good as follows: (Otherwise N/A)

CHECK the "Good Thru Date" on all Attachments has NOT been exceeded.

OR

- **IF** the "Good Thru Date" is exceeded on any Attachment, **THEN**

  NOTIFY System Engineering to provide updated Attachment graphs to Operations Procedure.

5.0 SPECIAL TOOLS AND EQUIPMENT

None

6.0 ACCEPTANCE CRITERIA

A. Responses which fail to meet the following Acceptance Criteria constitute unsatisfactory Surveillance procedure results and require immediate notification of the Unit SRO at the time of failure.

B. Recirculation Loop Jet Pump flow mismatch with both Recirculation Loops in operation shall be verified by the following criteria [2-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation, satisfies Technical Specification 3.4.1 during single Recirculation Loop operation]:

1. $\leq$10% of rated core flow when operating at < 70% of rated core flow ($\leq$ 10.25 Mlb/hr).

2. $\leq$ 5% of rated core flow when operating at $\geq$ 70% of rated core flow ($\leq$ 5.12 Mlb/hr).

6.0 ACCEPTANCE CRITERIA (continued)

NOTE

If either condition in Step 6.0C.1, 6.06.0C.2 or 6.0C.3 exists, the Tech Spec Acceptance Criteria would be satisfied. However, failure of either Step 6.0C.1 or 6.0C.2 of the criteria may be an indication of Jet Pump degradation and shall be immediately reported to the Unit SRO.

C. Jet Pump operation shall be checked by verifying that at least one of the following criteria (Step 6.0C.1 or 6.0C.2) is satisfied for each of the operating Recirculation Loops:

1. Recirculation Pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.

2. Each Jet Pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.

3. Since Refueling activities such as fuel assembly replacement or shuffle, modifications to fuel support, orifice size or core plate bypass flow can affect the relationship between core flow, and recirculation loop flow, these relationships may need to be re-established each cycle.

   a. During initial weeks of operation under such conditions, while baselining new "established patterns," an Engineering Evaluation of daily Surveillance results may be used to meet the Acceptance Criteria for conditions Steps 6.0C.1 and 6.0C.2 above.

   b. This evaluation is to conclude that daily surveillance results do not indicate significant abnormalities or Jet Pump failure.

4. After the new baselining has been completed and new “Established Patterns” have been set, methodology for determining the Acceptance Criteria as being Completed Satisfactorily, as stipulated in Step 6.0C.3 will not be allowed.

D. Steps which determine the above criteria are designated by (AC) next to the initials blank.

Date: Today

7.0 PROCEDURE STEPS

7.1 Initial Requirements and Notifications

[1] **ENSURE** Precautions and Limitations in Section 3.0 have been reviewed.

[2] **ENSURE** Prerequisites listed in Section 4.0 are satisfied.

[3] On the Surveillance Task Sheet (STS)

OBTAIN Authorization Signature and Date/Time from the Unit SRO to perform this surveillance.

[4] **IF** RTP is less than or equal to 23%, **THEN**

MARK Section 7.4 as N/A. (Otherwise N/A this step) N/A

[5] [NRC/C] **NOTIFY** the Unit Operator (UO) that this test is commencing. [RPT 82-16, LER 259/8232]

[6] On the Surveillance Task Sheet (STS)

RECORD the Start Date & Time.

Date: TODAY

7.2 Data Collections

7.2.1 Core Power and Flow Readings

[1] **RECORD** the Core Thermal Power from Core Power and Flow Log. (N/A if ICS is NOT available)

Point CALC002 39.51 CMWT

[2] **RECORD** the Core Plate Differential Pressure from ICS point 68-52 or 2-XR-68-50 (Green Pen). (N/A if not available)

Core Press Drop 68-52 18.4 PSID

[3] **RECORD** the Total Core Flow.

Total Core Flow (Red Pen) 2-XR-68-50
100.6 Mlb/hr

Date: Today

NOTES

1. If 2-SIT-68-59/71 (RB. El 565 R-9 S-Line - Local Panel) is used, log the reason in the Remarks section of the Surveillance Task Sheet (STS).
2. Use the 2-SI-96-61 (or ICS PT 96-61) if both 2-SI-68-59 and 2-SIT-68-59 are not available for the2A Pump Motor and log the reason in the Remarks section of the Surveillance Task Sheet (STS).
3. Use the 2-SI-96-73 (or ICS PT 96-73) if both 2-SI-68-71 and 2-SIT-68-71 are not available for the 2B Pump Motor and log the reason in the Remarks section of the Surveillance Task Sheet (STS).
4. If a Recirculation Pump is not in service then the associated instrumentations can be marked as N/A.

7.2.2 Recirculation Pump Loops

11. **RECORD** the Recirc Pump 2A and 2B Mtr Speeds for operating Recirc Pumps and **MARK** instrumentation used.

Pump Motor 2A	√	Pump Motor 2B	√
2-SI-68-59	☐	2-SI-68-71	☐
2-SIT-068-0059	☐	2-SIT-068-0071	☐
2-SI-96-61	☐	2-SI-96-73	☐
1505 RPM		1520 RPM	

12. **RECORD** the Recirc Pump Discharge flows.

Loop 2A 2-FI-68-5	Loop 2B 2-FI-68-81
46 gpm X 1000	43 gpm X 1000

13. **RECORD** the Recirc Loop 2A and 2B Jet Pump Flow.

Loop 2A 2-FI-68-46	Loop 2B 2-FI-68-48
49 Mlb/hr	50 Mlb/hr

Date: Today

NOTE

If a Recirculation Pump is not in service, then the associated instrumentations can be marked as N/A.

7.2.3 Jet Pump Loops

(11) **RECORD** the following Differential Pressure readings below:

Loop 2A			Loop 2B		
INSTRUMENT	JET PUMP	PSID	INSTRUMENT	JET PUMP	PSID
2-PDI-68-38	11	11.0	2-PDI-68-15	1	11.0
2-PDI-68-39	12	11.5	2-PDI-68-18	2	11.0
2-PDI-68-40	13	12.0	2-PDI-68-19	3	11.5
2-PDI-68-42	14	11.0	2-PDI-68-21	4	11.0
2-PDI-68-43	15	11.0	2-PDI-68-22	5	13.0
2-PDI-68-07	16	11.5	2-PDI-68-25	6	11.5
2-PDI-68-08	17	12.0	2-PDI-68-26	7	12.0
2-PDI-68-10	18	12.0	2-PDI-68-28	8	11.5
2-PDI-68-11	19	11.5	2-PDI-68-29	9	14.5
2-PDI-68-13	20	12.5	2-PDI-68-30	10	11.5

Date: Today

NOTES

1. Section 7.3 is performed when both Recirculation Pumps are in service. This section should be marked N/A when in Single Loop Operation.
2. To satisfy procedure Acceptance Criteria, either Step 7.3[3] or Step 7.3[4] must be satisfied.

7.3 Tech Spec 3.4.1.1 - Recirculation Loop Mismatch Verification With Both Recirculation Loops In Operation Checks

[1] **CALCULATE** percent of Rated Core Flow (%WT) using data obtained in Section 7.2.1[3] as follows.

(Step 7.2.1[3] ÷ 102.5) X 100 =	% Core Flow
(100.6 ÷ 102.5) X 100 =	98.1

[2] **CALCULATE** the absolute value for Recirculation Loop Jet Pump Flow Mismatch using data obtained in Section 7.2.2[3] as follows.

2-FI-68-46	-	2-FI-68-48	=	Mismatch

49 Mlb/hr - 50.2 Mlb/hr = 1.2 Mlb/hr

[3] **IF** %WT is < 70% as recorded in Step 7.3[1], **THEN**

CHECK Recirculation Loop Jet Pump Flow Mismatch recorded in Step 7.3[2] is ≤ 10.25 Mlb/hr. (Otherwise N/A) N/A (AC)

[4] **IF** %WT is ≥ 70% as recorded in Step 7.3[1], **THEN**

CHECK Recirculation Loop Jet Pump Flow Mismatch recorded in Step 7.3[2] is ≤ 5.12 Mlb/hr. (Otherwise N/A) (AC)

Date: Today

NOTES

1. Jet Pump Operability is not required to be performed until 4 hours after associated Recirculation Loop is in operation and then only within 24 hours after RTP is > 23%.
2. If a Recirculation Pump is not in service, then the associated steps can be marked as No.
3. If Data falls on or outside the bold lines on the graph, then the step can be marked as No.

7.4 Tech Spec 3.4.2.1 - Part A -Jet Pump Performance Checks

7.4.1 Loop 2A Recirculation Pump and Jet Pump Flow to Recirculation Pump Speed

[1] Using the 2A Pump Speed recorded in Step 7.2.2[1] and the 2A Pump Flow recorded in Step 7.2.2[2]:

CHECK the plot falls between the two bold lines on Attachment 1 and **RECORD** below.

Plot falls between the bold lines	Yes ☑	No ☐

[2] Using the 2A Pump Speed recorded in Step 7.2.2[1] and the 2A Jet Pump Flow in Step 7.2.2[3]:

CHECK the plot falls between the two bold lines on Attachment 2 and **RECORD** below.

Plot falls between the bold lines	Yes ☑	No ☐

[3] Using Steps 7.4.1[1] and 7.4.1[2] from above:

DETERMINE if the Jet Pump Loop 2A criteria is satisfied by marking below if both steps are marked as Yes.

Jet Pump Loop 2A criteria is satisfied	Yes ☑	No ☐

Date: Today

NOTES

If a Recirculation Pump is not in service, then the associated steps can be marked as No.

If Data falls on or outside the bold lines on the graph, then the step can be marked as No.

7.4.2 Loop 2B Recirculation Pump and Jet Pump Flow to Recirculation Pump Speed

[1] Using the 2B Pump Speed recorded in Step 7.2.2[1] and 2B Pump Flow recorded in Step 7.2.2[2]:

CHECK the plot falls between the two bold lines on Attachment 3 and **RECORD** below.

Plot falls between the bold lines	Yes ☑	No ☐

[2] Using the 2B Pump Speed recorded in Step 7.2.2[1] and 2B Jet Pump Flow recorded in Step 7.2.2[3]:

CHECK the plot falls between the two bold lines on Attachment 4 and **RECORD** below.

Plot falls between the bold lines	Yes ☑	No ☐

[3] Using Steps 7.4.2[1] and 7.4.2[2] from above:

DETERMINE if the Jet Pump Loop 2B criteria is satisfied by marking below if both steps are marked as Yes.

Jet Pump Loop 2B criteria is satisfied	Yes ☑	No ☐

Date: Today

NOTES

1. If a Recirculation Pump is not in service, then the associated steps can be marked as No.
2. If Data falls on or outside the bold lines on the graph, then the step can be marked as No.

7.4.3 Recirculation Jet Pump Diffuser to Lower Plenum Differential Pressure Verification:

[1] Using the individual 2A Jet Pump DP's recorded in Step 7.2.3[1]

CHECK each individual Jet Pump DP recorded fall between the two bold lines on Attachment 5 for the recorded Total Flow in Step 7.2.1[3] and **RECORD** results below.

2A Individual DP's are between the bold lines.	Yes ☑	No ☐

[2] Using the individual 2B Jet Pump DP's recorded in Step 7.2.3[1]

CHECK each individual Jet Pump DP recorded fall between the two bold lines on Attachment 6 for the recorded Total Flow in Step 7.2.1[3] and **RECORD** results below.

2B Individual DP's are between the bold lines.	Yes ☐	No ☑

[3] Using Steps 7.4.3[1] and 7.4.3[2]

DETERMINE whether the Recirculation Jet Pump Diffuser to Lower Plenum Differential Pressure Verification criteria is satisfied by marking below if both steps are marked as Yes.

Jet Pump Diffuser to Lower Plenum Differential Pressure Verification criteria is satisfied	Yes ☐	No ☑

Date: Today

CAUTION

An Engineering Judgment/Review may only be utilized until relationships between Core Flow, Jet Pump Flow, and Recirculation Loop Flow have been established following a Refueling Outage or during the initial weeks of extended Single Loop operation. Engineering Judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a Jet Pump failure. Refer to SR 3.4.2.1 bases.

7.4.4 Engineering Judgment/Review Criteria

[1] **IF** any of the following conditions apply:

- Following Refueling Outage. (See Caution above)

  OR

- The Reactor is in Single Loop Operation (See Caution above)

  OR

- If point(s) plotted in Sections 7.4.1 or 7.4.2, AND 7.4.3 fall on or outside the bolded lines, to determine if the graph(s) need updating, **THEN**

PERFORM Attachment 7, Engineering Judgment/Review: (Otherwise N/A this step and Mark all of Attachment 7 as N/A) N/A

Date: Today

7.4.5 Operability Determination

[1] **MARK** the appropriate criteria results for the following:
(N/A any criteria not performed)

Steps	Criteria Results	Yes	No	N/A
7.4.1[3] and 7.4.2[3]	Both Jet Pump Loops steps are marked as YES	☑	☐	☐
7.4.3[3]	Jet Pump DP to criteria is marked as YES.	☐	☑	☐
Attachment 7	Engineering Evaluation is marked as YES.	☐	☐	☑

[2] Using the Criteria Results in Step 7.4.5[1]

ENSURE at least one Criteria Results is satisfied and marked as YES.

(AC)

Date: Today

7.5 Completion and Notifications

[1] **IF** an Engineering Judgment/Review was performed, **THEN**

ENSURE the Engineering Judgment/Review documentation is attached to this SR. (Otherwise N/A) N/A

[2] On the Surveillance Task Sheet (STS)

[2.1] **RECORD** the Completion Date & Time. [illegible]

[2.2] **REVIEW** and **COMPLETE** through the Test Director/Lead Performer & Date fields. [illegible]

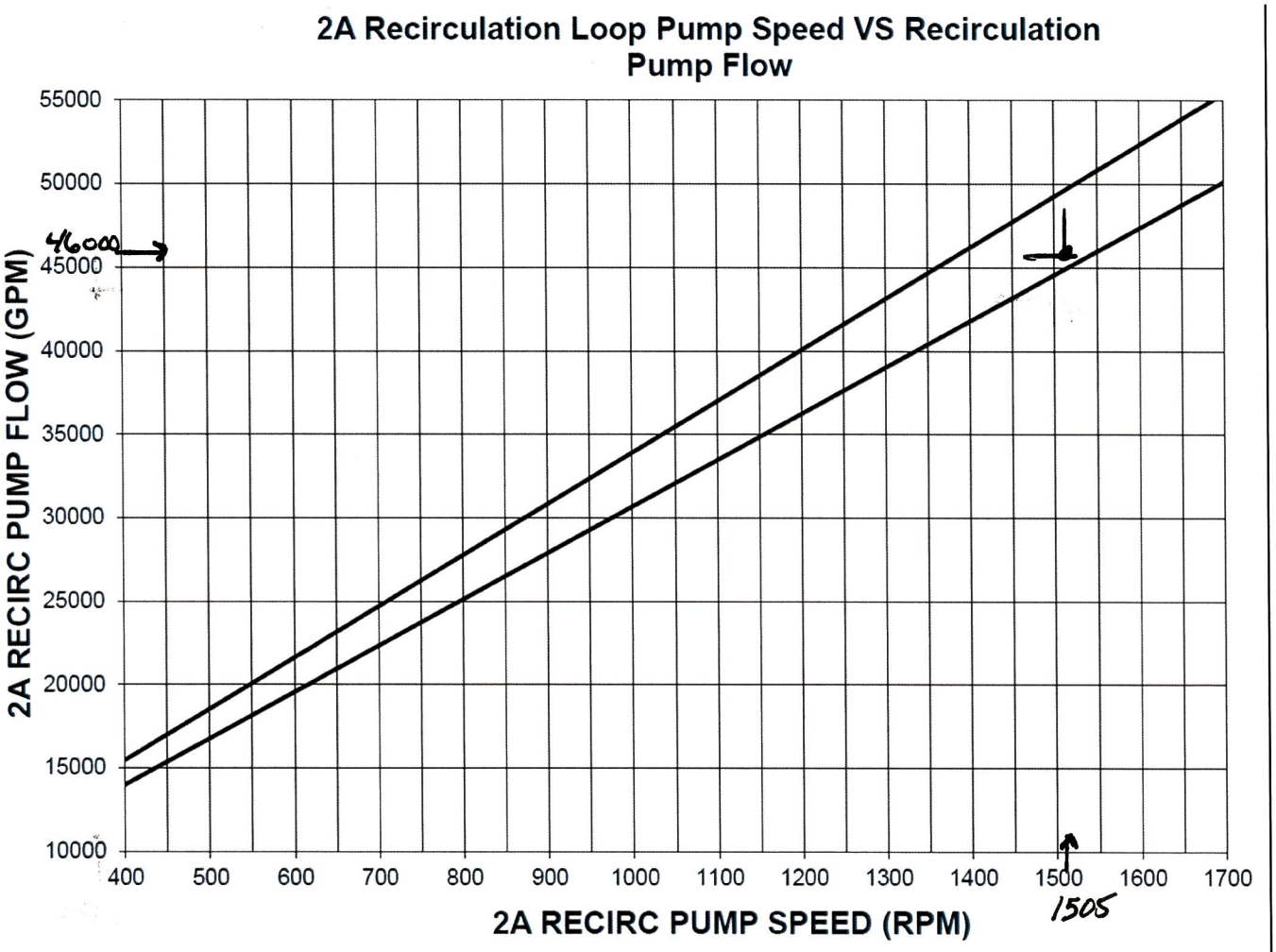
[3] **NOTIFY** the Unit SRO that this test is complete. [illegible]

8.0 RECORDS

All completed portions of this procedure and STS are QA Records and are to be maintained per NPG-SPP-31.2, Records Management.

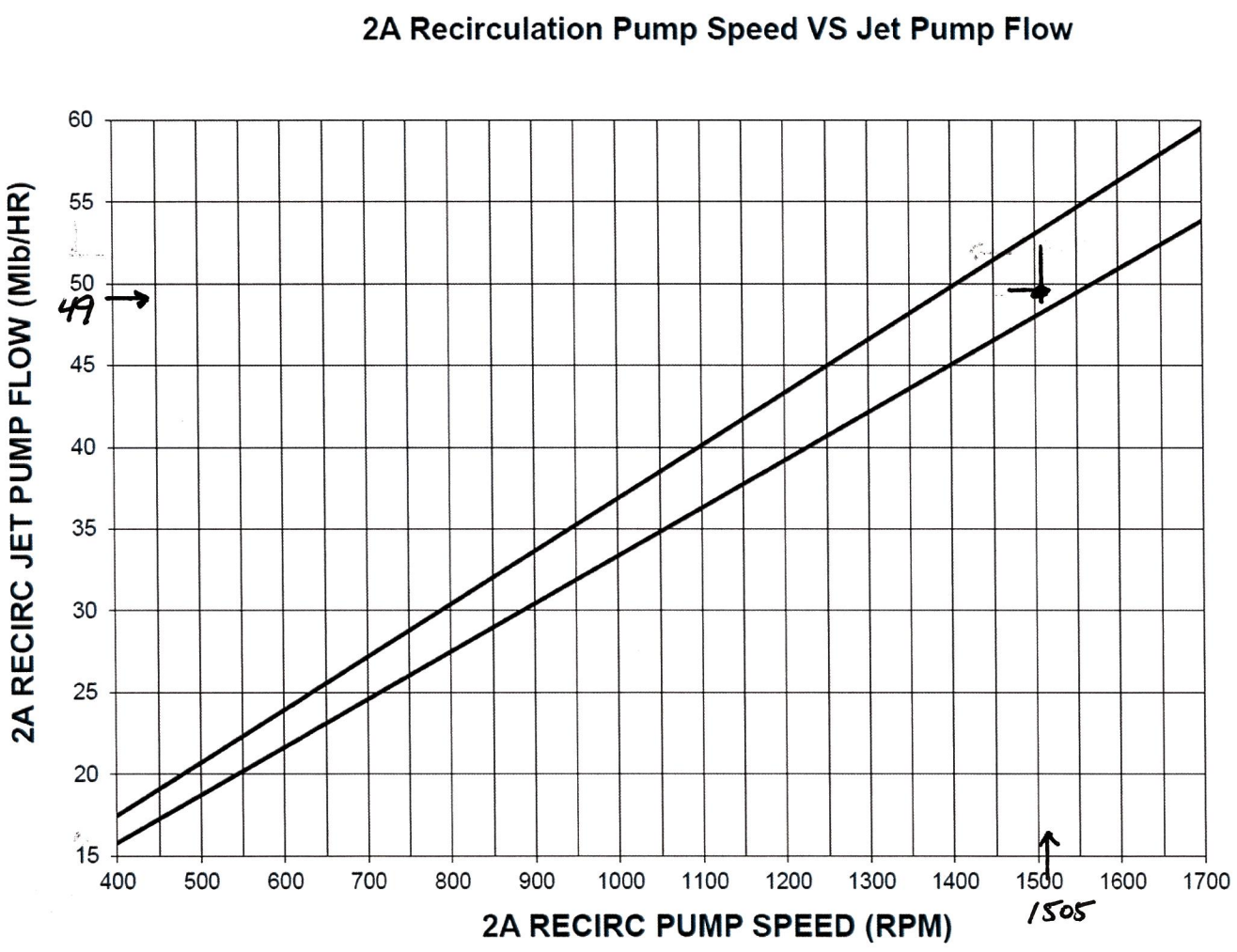
Attachment 1
(Page 1 of 1)

2A Recirculation Loop Pump Speed VS Recirculation Pump Flow
Good Thru October 1, 2023



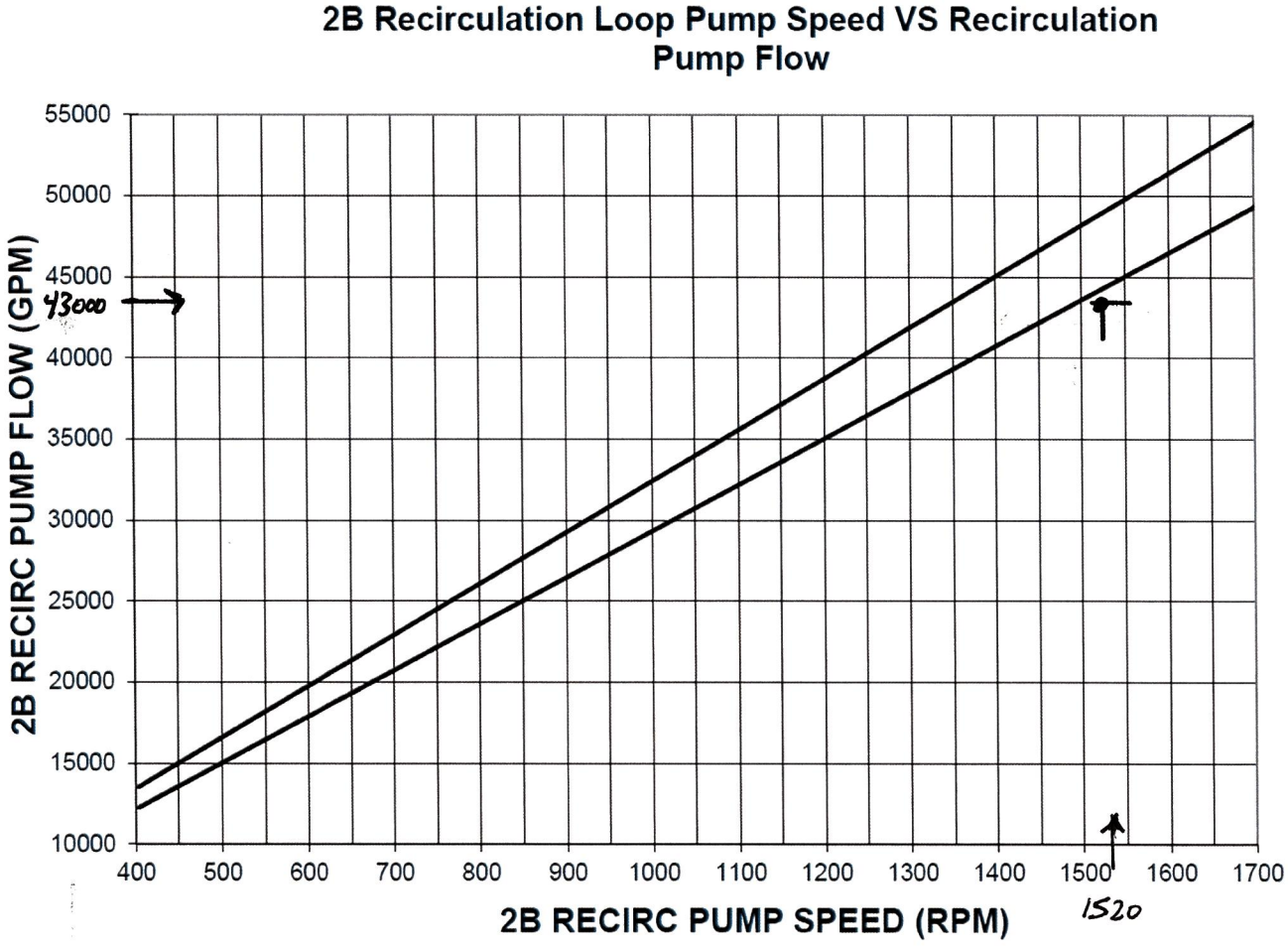
Attachment 2
(Page 1 of 1)

2A Recirculation Pump Speed VS Jet Pump Flow
Good Thru October 1, 2023



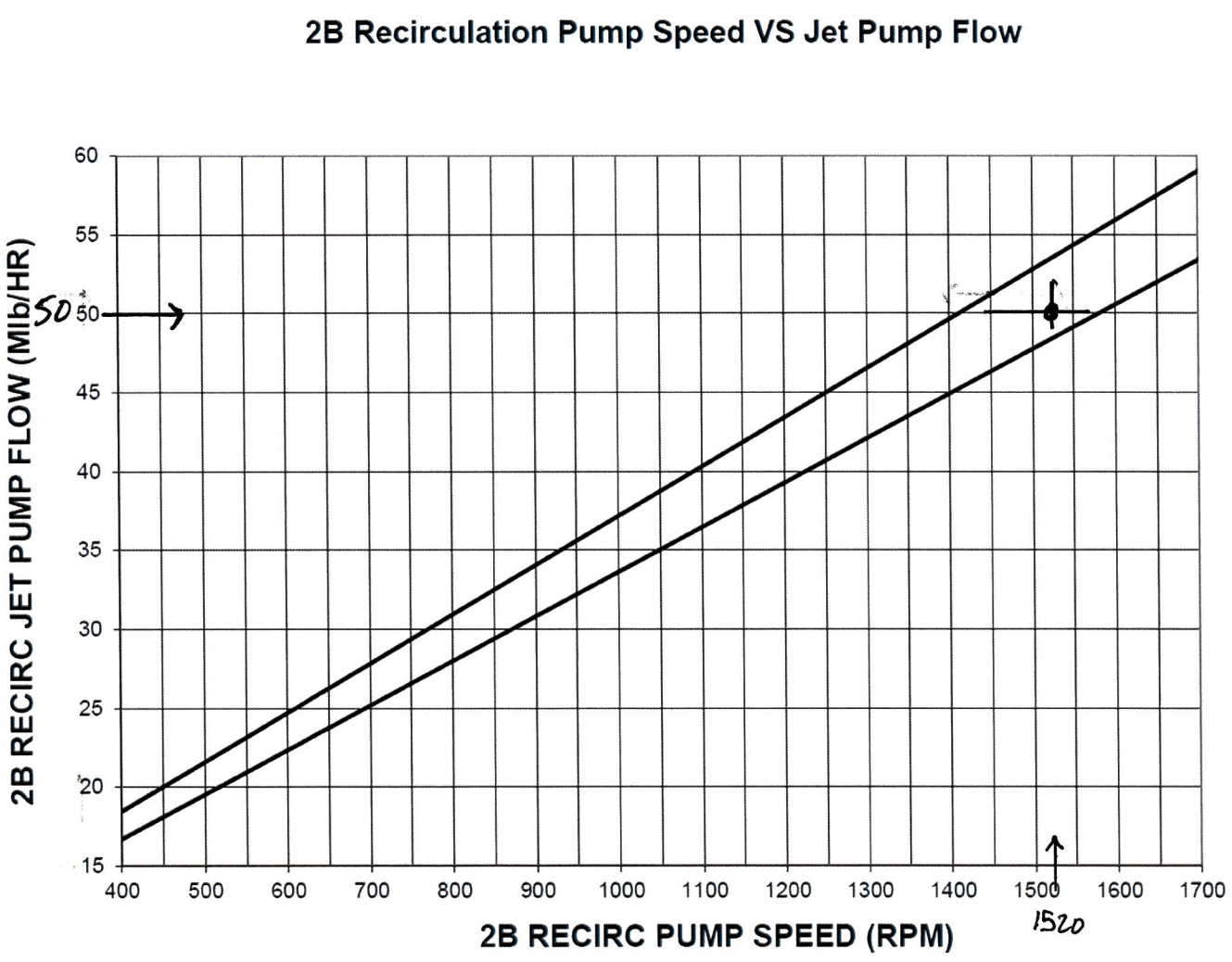
Attachment 3
(Page 1 of 1)

2B Recirculation Loop Pump Speed VS Recirculation Pump Flow
Good Thru October 1, 2023



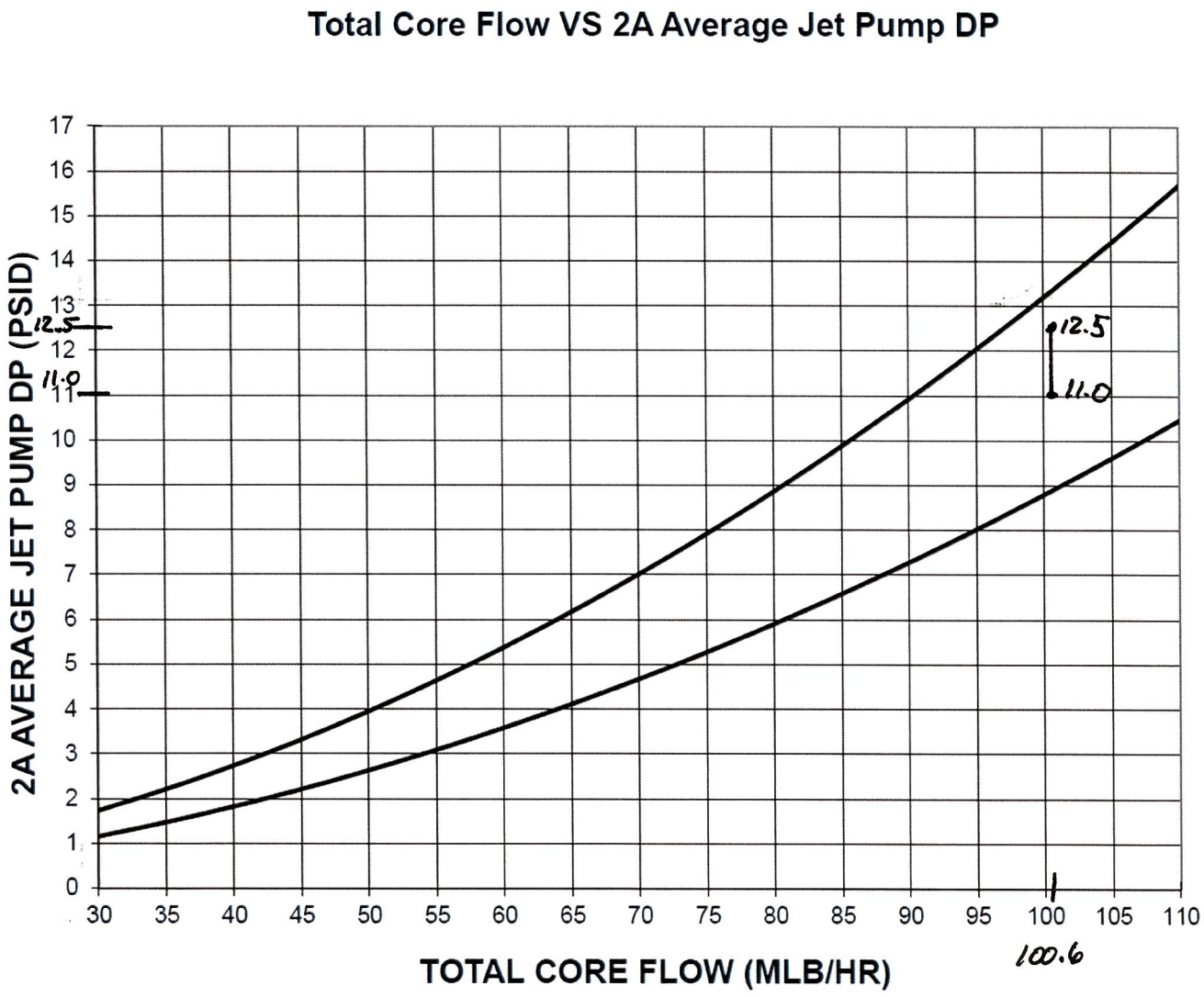
Attachment 4
(Page 1 of 1)

2B Recirculation Pump Speed VS Jet Pump Flow
Good Thru October 1, 2023



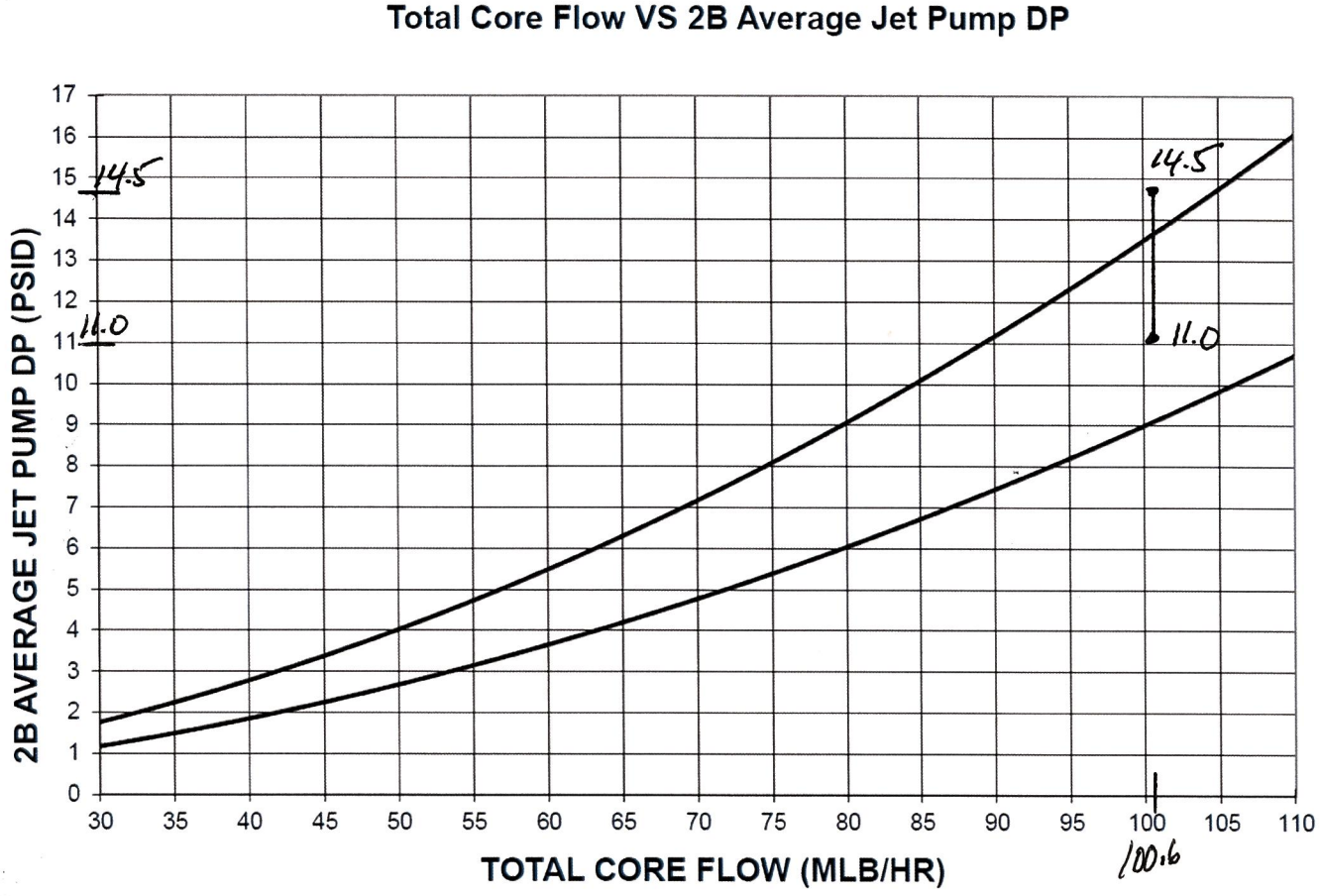
Attachment 5
(Page 1 of 1)

Total Core Flow VS 2A Average Jet Pump DP
Good Thru October 1, 2023



Attachment 6
(Page 1 of 1)

Total Core Flow VS 2B Average Jet Pump DP
Good Thru October 1, 2023



Attachment 7
(Page 1 of 1)
Engineering Judgment/Review

Date: TODAY

CAUTIONS

1) Engineering Judgment Evaluation may only be utilized until relationships between Core Flow, Jet Pump Flow, and Recirculation Loop Flow have been established:
   - During the initial weeks of extended Single Loop Operation
   - Following a Refueling Outage until Engineering has provided or determined that the current graphs are good. (At least one evaluation must be performed following a Refueling Outage)
2) Engineering Judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a Jet Pump failure. (Reference SR 3.4.2.1 bases)

[1] **MARK** the condition that applies:

• Following Refueling Outage.	☐
• The Reactor is in Single Loop Operation	☐
• Point(s) plotted in Sections 7.4.1 or 7.4.2, AND 7.4.3 fall on OR outside the bolded lines	☐

N/A

[2] **REQUEST** System Engineering to perform an Engineering Judgment/Review. N/A

[3] **IF** the Engineering Judgment/Review was performed following a Refueling Outage or during Single Loop Operation, **THEN**

DETERMINE if the Jet Pump Criteria is satisfied and NO significant abnormalities which could indicate a Jet Pump failure are indicated and **RECORD** the results below. (Otherwise N/A)

Jet Pump Criteria is satisfied. Yes ☐ No ☐ N/A

[4] **IF** the Engineering Judgment/Review was performed to determine if the graphs needs updated, **THEN**

REQUEST System Engineering to: (Otherwise N/A)

A. **SUPPLY** new graphs to Operations Procedures. N/A

B. **RECORD** below if Jet Pump Criteria is satisfied.

Jet Pump Criteria is satisfied. Yes ☐ No ☐ N/A

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Emergency Action Level Classification
JPM NUMBER:	762-SRO	REVISION:	0

TASK APPLICABILITY:	☒ SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	S-000-EM-21 / Classify and Declare an Abnormal/Emergency Event		
K/A RATINGS:	SRO 4.6		
K/A No. &STATEMENT:	2.4.41 Knowledge of the Emergency Action Level thresholds and classifications.		
RELATED PRA INFORMATION:	None		
SAFETY FUNCTION:	EMERGENCY PLAN - ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☒ YES	☐ NO	
TIME FOR COMPLETION:	30 minutes		

Validated by: _____ _____
Validator Date

Approved by: _____ _____
Facility Representative Date

NRC Approved: _____ _____
Facility Author Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 762-SRO

SRO _____ DATE: _____

TASK STANDARD: Given plant conditions, the Examinee is expected to use Emergency Preparedness procedures to classify an event as a SITE AREA EMERGENCY within 15 minutes then fill out the required notification form also within 15 minutes.

Operator Fundamental evaluated:
OF-1 Monitoring Plant Indications and Conditions Closely

PRA: N/A

REFERENCES/PROCEDURES NEEDED: EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3
EPIPs 1, 2, 3, 4, and 5

VALIDATION TIME: 30 minutes

PERFORMANCE TIME: _____

COMMENTS: _____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____

IF UNSAT results are obtained

THEN Retain entire JPM for records. (Otherwise just retain this page.)

SIGNATURE: _____ DATE: _____
EXAMINER

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	12012022	ALL	Initial issue

Procedure Revisions

Procedure	Revision
EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3	63
EPIP-1	63
EPIP-2	44
EPIP-3	47
EPIP-4	47
EPIP-5	61

EXAMINER NOTE:
Initially handout each Att. 1 from EPIP 2, 3, 4, 5 and EPIP-1, Att. 1, HOT INITIATING CONDTIONS-MODES 1-2-3.

****

Classroom

****

I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

You are the Shift Manager. Unit 1 and 3 are operating at 100% Reactor Power.

Unit 2 is operating at 95% RTP with the following conditions:

- A fuel leak is suspected
- Reactor Engineering is developing a leak suppression RCP

Continue the INITIAL CONDITIONS on the following page:

Subsequently, the following occurs on Unit 2:

At 0720: Spurious Low Reactor Water Level causes an Automatic Reactor SCRAM

At 0721: ATWS occurs, initial Reactor Power after the SCRAM is 45%

At 0723: ATWS actions complete, Reactor Power is 23%

- SLC is injecting
- Reactor Water Level is (-) 65 inches (band is (-) 50 to (-) 100 inches)
- Main Turbine remains online

At 0729: Stack Gas Radiation High alarm received

– WRGERMS indicates 5.4 x 10^7 µCi/sec and rising

At 0730: CECC Dose Assessment results indicate 55 mrem TEDE at the site boundary

At 0737: Second Dose Assessment results now indicate 110 mrem TEDE at the site boundary. WRGERMS indicates 3.2 x 10^9 µCi/sec and rising.

- 91 meter Met Tower Data:
  - 15 minute average Wind Direction is from 285 degrees
  - 15 minute average Wind Speed is 22 mph

NOTE:

- **NO** previous Events have been classified; therefore, Emergency Facilities (CECC, TSC. OSC) have **NOT** been staffed
- Emergency Director Judgement shall **NOT** be used as a basis for Classification
- Vaporstream is **NOT** available

INITIATING CUE:

Given the conditions above, declare the **HIGHEST** Event Classification, then **IMMEDIATELY** raise your hand to inform the evaluator of your Emergency Action Level (EAL) call.

Then, complete the appropriate Initial Notification and **IMMEDIATELY** raise your hand.

This JPM is TIME CRITICAL

KEY

RS1:

SITE AREA EMERGENCY

RS1 - Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

- **The SED should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.**
- **If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.**
- **If the effluent flow past an effluent monitor is known to have stopped, due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.**
- **The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.**

(1) Reading on Stack Noble Gas (WRGERMS) greater than 5.9 X 10^9 µCi/sec for 15 minutes or longer.

OR

(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary.

OR

(3) Field survey results indicate EITHER of the following at or beyond the site boundary:

- Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

RS2 - Spent fuel pool level at 641' 4".

(1) Lowering of spent fuel pool level to 641' 4".

KEY

KEY

BFN Unit 0	SITE AREA EMERGENCY	EPIP-4 Rev. 0047 Page 13 of 37

Attachment 1
(Page 1 of 3)

Site Area Emergency Initial Notification Form

1.	■ This is a Drill ☐ This is an Actual Event-Repeat-This is an Actual Event
2.	The Site Emergency Director at **Browns Ferry** has declared an **SITE AREA EMERGENCY**.
3.	**Initiating Condition (IC) Designator:** RS1 (USE ONLY ONE IC DESIGNATOR)
4.	**Has a Radiological release occurred or is in progress attributable to the event?** ☐ NO ☐ YES, AIRBORNE, Radiological Release ☐ YES, LIQUID Radiological Release
5.	**Event Declared:** Time: Enters Time Central Time Date: Enters Date
6.	**Protective Action Recommendation:** ☒ None
7.	Completed By: _____ Peer Reviewed By: _____

Note: The Yellow Highlighted steps above are designated as Critical Steps in accordance with Licensed Operator Requalification Performace Indicator Standards.

START TIME: _____

STEP / STANDARD	SAT / UNSAT
EXAMINER NOTE: This JPM has two Time Critical sections. The candidate will have 15 minutes to CLASSIFY the event once they understand their task, then 15 minutes to complete any required paperwork for the INITIAL NOTIFICATION after they complete the DECLARED CLASSIFICATION. **When the candidate informs the Examiner that they have made an Event CLASSIFICATION, record TIME CLASSIFICATION COMPLETE below in JPM Step 1.**	
EXAMINER NOTE: From the Initial Conditions, the following Emergency Action Levels (EALs) are all plausible and correct for the given times, however the Initiating Cue states to declare the HIGHEST Event Classification: • **0723: SA5 based on ATWS** • **0729: RU1 based on WRGERMS greater than 2.88 x 10^7 µCi/sec** • **0730: RA1 based on greater than 10 mrem TEDE**	
Step 1: Classifies the Event using EPIP-1, Emergency Classification Procedure **3.1 Precautions/Limitations** (page 6 of 140) C. The SM/SED shall assess, classify, and declare an emergency condition within 15 minutes after information is first available to plant operators to recognize that an EAL has been exceeded and to make the declaration promptly upon identification of the appropriate Emergency Classification Level (ECL). Expected Action(s): Examinee refers to EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3. Given the plant conditions, within 15 minutes, declares the CLASSIFICATION AS **SITE AREA EMERGENCY - RS1** – Release of gaseous radioactivity in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE: (2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary. **TIME CLASSIFICATION COMPLETE:** _____	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: **Next, direct Examinee to complete the next Time Critical portion of the JPM for the appropriate INITIAL NOTIFICATION using Attachment 1, EPIP 2 or EPIP 3 or EPIP 4 or EPIP 5 respectively (handouts already provided), then IMMEDIATELY bring completed EPIP-4, Attachment 1, to the Examiner.** **Record the 2nd TIME START for the Examinee's last 15 minute required time in JPM Step 2 below.**	
Step 2: Implements EPIP-4, SITE AREA EMERGENCY. **TIME START** _____ Expected Action(s): Examinee implements EPIP-4, SITE AREA EMERGENCY.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3:	

BFN Unit 0	SITE AREA EMERGENCY	EPIP-4 Rev. 0047 Page 6 of 37

3.0 EMERGENCY CLASSIFICATION ACTIONS

NOTES

- Procedure steps can be performed concurrently.
- All procedure steps must be completed.
- All procedure attachments must be returned to the SED.
- Section 3.1 (as soon as possible, within 15 Minutes from classification declaration) and Section 3.4 (as soon as possible, not to exceed 60 minutes from classification declaration) are time critical.
- An Shift Technical Advisor or Senior Reactor Operator should peer review Vaporstream Notification and Attachment 1 completion.

CAUTION

- Ongoing or anticipated security events or severe weather may present a danger to normal staffing and other Emergency Plan implementation processes. Observe all procedural steps carefully during severe weather and security related events.
- Steps 3.1[1.1] of the Main Body and Attachment 7, Steps 1.1[1] and 1.1[4.1.1]CANNOT be delegated.

[1] **WHEN** the TSC SED has assumed the responsibilities from the SM SED, **THEN**

CONTINUE in this procedure at Attachment 7 .

Otherwise continue in this procedure.

_____ SAT

_____UNSAT

_____N/A

Expected Action(s):

Examinee continues in EPIP-4, Site Area Emergency, as the TSC has not yet been staffed as given in the Initial Conditions.

STEP / STANDARD	SAT / UNSAT
Step 4: **3.1 State of Alabama Notification** **NOTE** Notification of the State of Alabama is required to be completed as soon as possible within 15 minutes from the time of emergency classification declaration. [1] **PERFORM** the following: [1.1] **RECORD** the following information: • Time of SITE AREA EMERGENCY Event Classification: _____ Expected Action(s): Examinee enters the time of Site Area Emergency Event Classification.	_____ SAT _____UNSAT _____N/A
Step 5: [1.2] **IF** the Central Emergency Control Center (CECC) is NOT activated, **THEN CONTINUE** in this procedure at step 3.1[2]. Otherwise continue in this section. Expected Action(s): Examinee proceeds to Step 3.1[2], as the CECC has not been activated as given in the Initial Conditions.	_____ SAT _____UNSAT _____N/A
Step 6: [2] **COMPLETE** Initial Notification to the State using Vaporstream…. Expected Action(s): None, [2] Is N/A since Vaporstream is **NOT** available as given in the Initial Conditions.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [2.1] **IF** notification cannot be completed using Vaporstream, **THEN** [2.1.1] **COMPLETE** Attachment 1, "Site Area Emergency Initial Notification Form." Expected Action(s): Examinee completes Attachment 1, and simulates making the initial notification the State within 15 minutes by bringing completed EPIP-4, Attachment 1 to the Examiner. The following items are **Critical Steps** on Attachment 1: • (Step 3) Emergency Action Level (EAL) or Initiating Condition (IC) Designator – RS1 • (Step 4) Has a Radiological release occurred or is in progress attributable to the event? ☐ - YES (see EPIP-4, Attachment 1, Section 2.0, on next page) • (Step 5) Time and Date Event Declared	**Critical Step** _____ SAT _____UNSAT _____N/A
Record the 2nd STOP TIME for the Examinee's last 15 minute required time below.	
EXAMINER CUE: After Examinee hands the examiner their documented INITIATING CUE sheet and EPIP-4, Attachment 1, inform the candidate "Your task is complete."	
END OF TASK	

STOP TIME:_____

BFN Unit 0	SITE AREA EMERGENCY	EPIP-4 Rev. 0047 Page 15 of 37

Attachment 1
(Page 3 of 3)

Site Area Emergency Initial Notification Form

2.0 RADIOLOGICAL RELEASE DETERMINATION GUIDANCE

Release in progress is defined as conditions exist indicating that a radiological release is in progress or has occurred that is attributable to the emergency event. Additional guidance is provided in the table below.

<table>
<tr><th></th><th>IF:</th><th>THEN:</th></tr>
<tr><td>1</td><td>Any of the following Airborne effluent monitors are above the value below attributable to the event.
<table>
<tr><th>Gaseous Monitors
(ICS/PEDS screen XXXX)</th><th>Value</th><th>Units</th></tr>
<tr><td>Stack Noble Gas (WRGERMS)
0-RE-90-306</td><td>1.44×10^7</td><td>µCi/s</td></tr>
<tr><td>Eberline CAM
X-RM-90-249, 250, 251, 252</td><td>1.5×10^5</td><td>µCi/s</td></tr>
<tr><td>GE Stack Gas Radiation Monitor
X-RM-90-147/148</td><td>41,332</td><td>CPS</td></tr>
<tr><td>Offgas Pretreatment Radiation Monitor
1, 2, 3-RM-90-157</td><td>2.1×10^5</td><td>mR/hr</td></tr>
<tr><td>0-SI-4.8.B.1.a.1</td><td>>1</td><td>Release fraction</td></tr>
</table></td><td>Select “YES, AIRBORNE Radiological release”</td></tr>
<tr><td>2</td><td>Sample analysis for a liquid release indicates a concentration or release rate greater than 10 times the Effluent Concentration Limit, as determined by Chemistry, that is attributable to the event.</td><td>Select “YES, LIQUID Radiological release”</td></tr>
<tr><td>3</td><td>Other conditions exist indicating that an unmonitored radiological release attributable to the event is occurring to the environment. Examples may include
Intentional primary containment venting per EOIs/SAMGs
The associated release flowpath radiation monitor is not functioning
Other radiation monitor readings
Internal radiological survey results
A verified report of release path
Field Monitoring Team readings at the site boundary ≥0.05 mR/hr close window or dose assessment results (automated or manual) ≥0.05 mRem TEDE</td><td>Select “YES, AIRBORNE Radiological release”
AND/OR
“YES, LIQUID Radiological release”</td></tr>
<tr><td>4</td><td>None of the above conditions indicative of a radiological release attributable to the event.</td><td>Select “NO”</td></tr>
</table>

Job Performance Measure (JPM)

Provide to Applicant

****

Classroom

****

I will explain the initial conditions and state the task to be performed. When your task is given, I will ask if there are any questions before you begin. When you have completed your assigned task, inform the Examiner that your task is complete and you will be escorted to another room to discuss the JPM. When you complete the task successfully, the objective for this Job Performance Measure (JPM) will be satisfied.

****

INITIAL CONDITIONS:

You are the Shift Manager. Unit 1 and 3 are operating at 100% Reactor Power.

Unit 2 is operating at 95% RTP with the following conditions:

- A fuel leak is suspected
- Reactor Engineering is developing a leak suppression RCP

Continue the INITIAL CONDITIONS on the following page:

This JPM is TIME CRITICAL

Subsequently, the following occurs on Unit 2:

At 0720: Spurious Low Reactor Water Level causes an Automatic Reactor SCRAM

At 0721: ATWS occurs, initial Reactor Power after the SCRAM is 45%

At 0723: ATWS actions complete, Reactor Power is 23%

- SLC is injecting
- Reactor Water Level is (-) 65 inches (band is (-) 50 to (-) 100 inches)
- Main Turbine remains online

At 0729: Stack Gas Radiation High alarm received

– WRGERMS indicates 5.4 x 10^7 µCi/sec and rising

At 0730: CECC Dose Assessment results indicate 55 mrem TEDE at the site boundary

At 0737: Second Dose Assessment results now indicate 110 mrem TEDE at the site boundary. WRGERMS indicates 3.2 x 10^9 µCi/sec and rising.

- 91 meter Met Tower Data:
  - 15 minute average Wind Direction is from 285 degrees
  - 15 minute average Wind Speed is 22 mph

NOTE:

- **NO** previous Events have been classified; therefore, Emergency Facilities (CECC, TSC. OSC) have **NOT** been staffed
- Emergency Director Judgement shall **NOT** be used as a basis for Classification
- Vaporstream is **NOT** available

INITIATING CUE:

Given the conditions above, declare the **HIGHEST** Event Classification, then **IMMEDIATELY** raise your hand to inform the evaluator of your Emergency Action Level (EAL) call.

Then, complete the appropriate Initial Notification and **IMMEDIATELY** raise your hand.

This JPM is TIME CRITICAL

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine ALARA Path for Emergency Radiation Exposure	
JPM NUMBER:	688-SRO	REVISION:	3	

TASK APPLICABILITY:	☒ SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	S-000-EM-28; Authorize Emergency Radiation Exposure Limits		
K/A RATINGS:	SRO 3.7		
K/A No. &STATEMENT:	K/A 2.3.12: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters		
RELATED PRA INFORMATION:	None		
SAFETY FUNCTION:	RADIATION CONTROL - ADMIN		

EVALUATION LOCATION:	☐ In-Plant	☐ Simulator	☒ Classroom
METHOD OF TESTING:	☐ Simulated Performance	☒ Actual Performance	
ALTERNATE PATH (Y/N)	☐ YES	☒ NO	
TIME CRITICAL (Y/N)	☐ YES	☒ NO	
TIME FOR COMPLETION:	25 minutes		

Validated by: _____ Validator _____ Date

Approved by: _____ Facility Representative _____ Date

NRC Approved: _____ Facility Author _____ Date

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 688-SRO

SRO _____ DATE: _____

TASK STANDARD: Given three paths with dose rates to complete a job in the Drywell, the Examinee is expected to determine total dose for each path and decide which path would be the best with respect to maintaining the dose as low as possible. Using EPIP-15, Emergency Exposure, determines if an individual could be allowed to perform the job and fills out required paperwork.

PRA: NA

REFERENCES/PROCEDURES NEEDED: EPIP-15, Emergency Exposure

VALIDATION TIME: 25 Minutes

PERFORMANCE TIME:_____

COMMENTS: _____

_____

_____

_____

_____

_____

Additional comment sheets attached? YES ___ NO ___

RESULTS: SATISFACTORY ____ UNSATISFACTORY ____ (Retain entire JPM for records)

SIGNATURE: _____ DATE: _____
EXAMINER

Revision Summary

Rev No.	Effective Date	Pages Affected	Description
3	02/22/23	ALL	Revised JPM to be SRO only

Procedure Revisions

Procedure	Revision
EPIP-15	13

****

Classroom

****

INITIAL CONDITIONS: A General Emergency has been declared on Unit 2 by you as the Shift Manager and the Technical Support Center has **NOT** yet been staffed. Drywell entry is required to perform a lifesaving event with the following conditions:

- The dose rate in the Drywell is 18.2 R/hr, and the evolution will take 32 minutes
- Lucky Joe is making the entry and has received 1.25R annual dose to date
- Dose rates for this evolution are **NOT** due to airborne activity

Three different paths to arrive at the work location in the Drywell are detailed below:

Dose Rate	Path 1 Transit time	Dose Rate	Path 2 Transit time	Dose Rate	Path 3 Transit time
300 mr/hr	From staging area to point A requires: 10 minutes	300 mr/hr	From staging area to point E requires: 10 minutes	300 mr/hr	From staging area to point H requires: 10 minutes
300 mr/hr	From point A to point B requires: 3 minutes	300 mr/hr	From point E to point F requires: 3 minutes	150 mr/hr	From point H to point I requires: 5 minutes
100 mr/hr	From point B to point C requires: 10 minutes	150 mr/hr	From point F to point G requires: 15 minutes	1.5 R/hr	From point I to the work location requires: 10 minutes
150 mr/hr	From point C to point D requires: 10 minutes	2 R/hr	From point G to the work location requires: 5 minutes		
3 R/hr	From point D to the work location requires: 7 minutes				

INITIATING CUES:

Given the above conditions, determine the **ROUND TRIP** TEDE for each path and Total Dose using ALARA received to perform the required evolution by completing the attached table. **(show all work on the attached pages)**

If an Emergency Exposure is required, determine authorization requirements and complete the necessary documentation as applicable.

Dose Rate	Path 1 Transit time	Dose Rate	Path 2 Transit time	Dose Rate	Path 3 Transit time
300 mr/hr	From staging area to point A requires: 10 minutes =_____	300 mr/hr	From staging area to point E requires: 10 minutes =_____	300 mr/hr	From staging area to point H requires: 10 minutes =_____
300 mr/hr	From point A to point B requires: 3 minutes =_____	300 mr/hr	From point E to point F requires: 3 minutes =_____	150 mr/hr	From point H to point I requires: 5 minutes =_____
100 mr/hr	From point B to point C requires: 10 minutes =_____	150 mr/hr	From point F to point G requires: 15 minutes =_____	1.5 R/hr	From point I to the work location requires: 10 minutes =_____
150 mr/hr	From point C to point D requires: 10 minutes =_____	2 R/hr	From point G to the work location requires: 5 minutes =_____		
3 R/hr	From point D to the work location requires: 7 minutes =_____				
Arrives at work location with: =_____		Arrives at work location with =_____		Arrives at work location with =_____	
Round Trip TEDE	=_____	**Round Trip TEDE**	=_____	**Round Trip TEDE**	=_____

Work Dose =_____

Total Dose using ALARA Path 1 or 2 or 3 (circle one) =_____

KEY

Dose Rate	Path 1 Transit time	Dose Rate	Path 2 Transit time	Dose Rate	Path 3 Transit time
300 mr/hr	From staging area to point A 10 minutes 50mr	300 mr/hr	From staging area to point E 10 minutes 50mr	300 mr/hr	From staging area to point H 10 minutes 50mr
300 mr/hr	From point A to point B 3 minutes 15mr	300 mr/hr	From point E to point F 3 minutes 15mr	150 mr/hr	From point H to point I 5 minutes 12.5mr
100 mr/hr	From point B to point C 10 minutes 16.67mr	150 mr/hr	From point F to point G 15 minutes 37.5mr	1.5 R/hr	From point I to the work location 10 minutes 250mr
150 mr/hr	From point C to point D 10 minutes 25mr	2 R/hr	From point G to the work location 5 minutes 166.67mr		
3 R/hr	From point D to the work location 7 minutes 350mr				
Arrives at work location with 456.67mr		Arrives at work location with 269.17mr		Arrives at work location with 312.5mr	
Round Trip Dose	**913.3 mr (910 to 915 mr)**	**Round Trip Dose**	**538.3 mr (535 to 540 mr)**	**Round Trip Dose**	**625.0 mr (622 to 627 mr)**
Work Dose	**9.7R**				
Total Dose using Path 2	**9.7R + .538R = 10.238 R**				

INITIATING CUES:

Given the above conditions, calculate the **ROUND TRIP** TEDE for each path and determine which path maintains dose ALARA.

If an Emergency Exposure is required, determine who authorizes the evolution and complete the necessary documentation as applicable.

Path 2 maintains dose ALARA and the Shift Manager as the Site Emergency Director would need to authorize the Emergency Exposure using EPIP-15, Appendix B, since TEDE Dose would exceed 5 R.
Total Dose of approximately 10.238 R is estimated for Emergency Exposure for round trip transit and on scene Drywell work which is well below the listed limit of 23.75R on the handout, Appendix B.

KEY

3.0 INSTRUCTION

3.1 Radiation Protection (RP)

The RP section is responsible for performing radiological surveys or other assessments to estimate the radiation doses. The RP section is responsible for establishing dose limits for individuals and recording the limits on Appendix B. The RP section is responsible for conducting the briefing to individuals volunteering for emergency exposures utilizing Appendix A of this procedure.

3.2 Site Emergency Director

A. The Site Emergency Director (SED) shall provide authorization for all emergency radiation doses that may exceed limits found in 10 CFR 20.1201.

B. The documentation of authorization shall be accomplished by SED signature on Appendix B of this procedure.

3.3 Guidance for All Emergency Dose Limits

A. The exposure of personnel during emergency operations shall be maintained As Low As Reasonably Achievable (ALARA).

B. Internal exposure should be minimized by the use of respiratory protection equipment. Respirator Protection Factors are provided in NPG-SPP-05.10, "Radiological Respiratory Protection Program." If a projected dose to the thyroid is expected to exceed 10 rem during emergency conditions, Potassium Iodide (KI) should be issued. EPIP-14 contains information regarding issuance and precautions for the use of KI.

C. Personnel undertaking an emergency operation in which the dose will exceed 10 CFR 20.1201 limits shall do so on a voluntary basis and with full awareness of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects as depicted on Appendix A. Acknowledgment of this decision shall be documented on Appendix B of this procedure by the individual involved in this activity.

D. Other factors being equal, older volunteers should be selected first.

E. Other factors being equal, selection of female volunteers capable of reproduction should be avoided.

F. Exposures under these conditions shall be limited to a once in a lifetime. Personnel who have received previous accident or emergency exposures in excess of 25 rem Total Effective Dose Equivalent (TEDE) shall not participate in further emergency exposure assignments.

KEY

3.3 Guidance for All Emergency Dose Limits (continued)

G. Personnel shall not enter any area where dose rates are unknown or not measurable with instruments and dosimetry immediately available.

NOTE

The value below corresponds to the ratio of external (measured) dose rate to estimate TEDE dose, in accordance with default values in TVA's Dose Assessment model. When accident specific nuclide assessment are available, more definitive dose assessments should be performed to adjust the correction factors.

H. Until isotopic assessments of airborne radioactivity are available, an administrative correction factor of 2 should be used to estimate TEDE exposures in airborne activity areas:

Estimated TEDE = Dosimeter Reading × 2

3.4 Dose Limits for Workers During Emergencies

A. Doses to all workers during emergencies should, to the extent practicable be limited to 10 CFR 20.1201 limits. There are, however, some emergency situations for which higher emergency exposures may be justified. Whenever these situations are justified and ALARA considerations have been evaluated the following limits can be administered.

B. RP considers the to-date annual accrued dose to individuals when establishing the maximum dose limits for workers during emergencies. The to-date annual accrued dose would be subtracted from the applicable emergency dose limit to determine the authorized allowable dose for the emergency.

C. Dose Limits for the Protection of Valuable Property

Dose Limit (rem)	Receptor
10	Whole Body (TEDE)
30	Lens of the Eye
100	All Other Organs

KEY

BFN Unit 0	EMERGENCY EXPOSURE	EPIP-15 Rev. 0013 Page 7 of 11

3.4 Dose Limits for Workers During Emergencies (continued)

D. Dose Limits for Lifesaving Activities and the Protection of Large Populations

Dose Limit (rem)	Receptor
25	Whole Body (TEDE)
75	Lens of the Eye
250	All Other Organs

E. Dose Limits Greater than 25 Rem for Lifesaving Activities or to Avoid Extensive Exposures to Large Populations

1. Situations may occur in which a dose in excess of 25 rem would be required in order to carry out lifesaving operations or to avoid extensive exposures to large populations. It is not possible to prejudge the risk that one person should be allowed to take to save the life of others.

2. Personnel made knowledgeable of the risks involved with radiation exposures through training or briefings utilizing the information contained within Appendix A and selected on a voluntary basis may be allowed to exceed the 25 rem emergency dose exposure limit for the purpose of lifesaving activities or to avoid extensive exposures to large populations.

3. Acknowledgment of this decision shall be documented on Appendix B of this procedure by the individual involved in this activity.

KEY

BFN Unit 0	EMERGENCY EXPOSURE	EPIP-15 Rev. 0013 Page 10 of 11

Appendix B
(Page 1 of 2)

Acknowledgement and Authorization to Exceed Occupational Dose Limits

READ THE FOLLOWING STATEMENT BEFORE SIGNING THIS FORM:

I acknowledge by signature on this form that I am volunteering for exposures in excess of 10 CFR 20.1201 limits and that I have been made aware through training or a briefing of the risks involved. Briefing material was presented from Appendix A of this procedure.

The persons listed below have acknowledged and volunteered to receive dose limits in excess of 10 CFR 20.1201 limits. Authorization is required by the Site Emergency Director (SED) to administer any emergency exposure limit. Authorization is acknowledged by SED signature on the bottom of this form.

Name (Please Print Last, First, MI)	Employee Identification Number (EIN)	Signature	Dose Limit[1] (Rem)
JOE, Lucky D.	86753099		23.75

[1]- Use "Emergency Exposure Reference Guide" on page 2 of this Appendix.

Brief Description of Task: _____

_____

Authorized by: SIGNATURE / TIME/DATE

Site Emergency Director — Time/Date

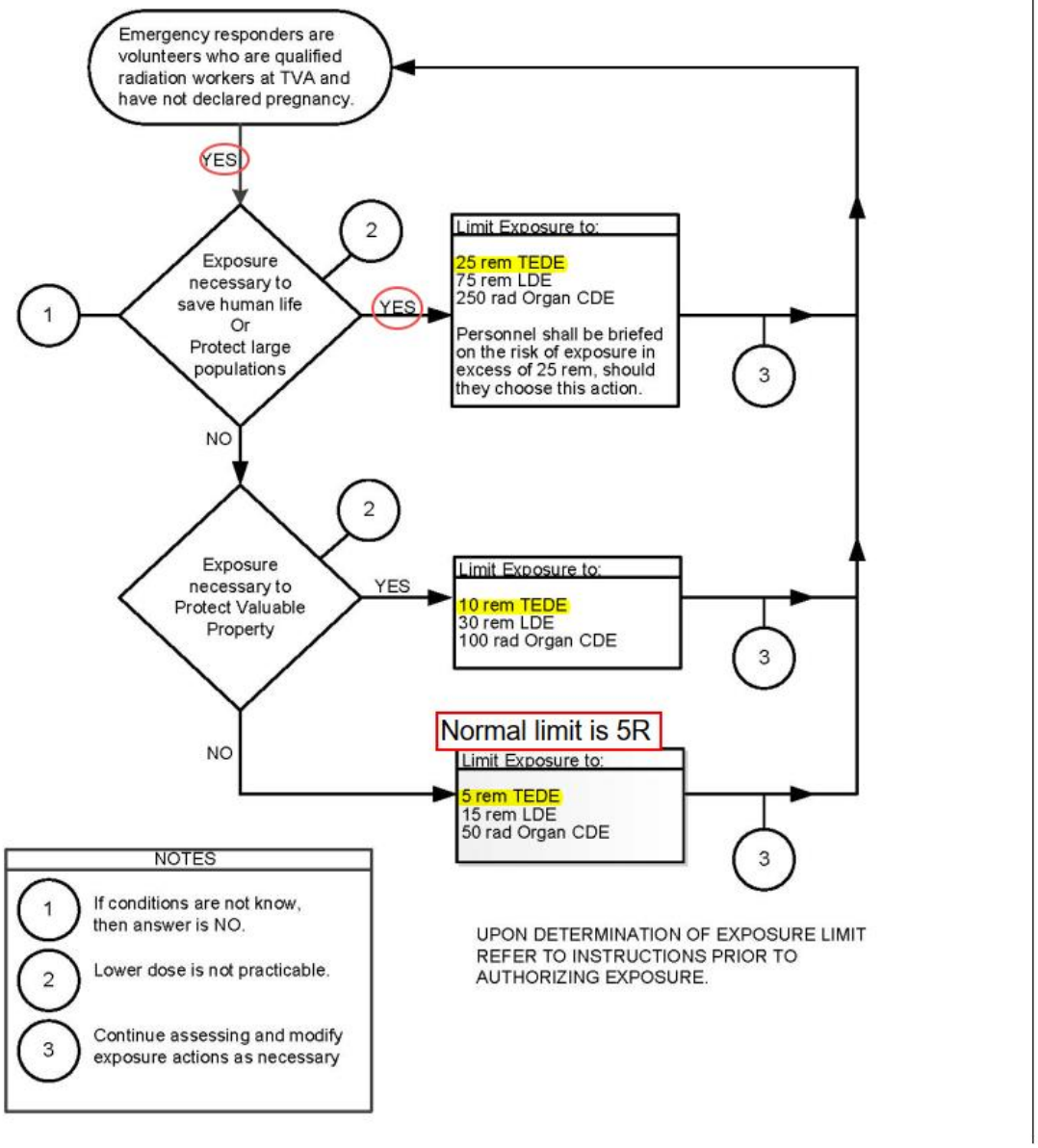
KEY

BFN Unit 0	EMERGENCY EXPOSURE	EPIP-15 Rev. 0013 Page 11 of 11

Appendix B
(Page 2 of 2)

Acknowledgement and Authorization to Exceed Occupational Dose Limits

Emergency Exposure Reference Guide



START TIME: _____

STEP / STANDARD	SAT / UNSAT
Step 1: Determines Path 1 transit dose. Expected Action(s): Examinee calculates Path 1 total dose for one way = 456.67mr Determines Path 1 Round Trip Dose to be: 456.67 X 2 = **913.34 mr (910 to 915 mr)**	**Critical Step** ___SAT ___UNSAT ___N/A
Step 2: Determine Path 2 transit dose. Expected Action(s): Examinee calculates Path 2 total dose for one way = 269.17 mr Determines Path 2 Round Trip Dose to be: 269.17 X 2 = **538.34mr (535 to 540 mr)**	**Critical Step** ___SAT ___UNSAT ___N/A
Step 3: Determine Path 3 transit dose. Expected Action(s): Examinee calculates Path 3 total dose for one way = 312.5 mr Determines Path 3 Round Trip Dose to be: 312.5 X 2 = **625 mr (622 to 627 mr)**	**Critical Step** ___SAT ___UNSAT ___N/A
Step 4: Determines the transit Path with the lowest dose option. Expected Action(s): Examinee determines Path 2 Round Trip Dose of **538.34mr (535 to 540 mr)** is the lowest dose option between Path 1, 2, or 3.	**Critical Step** ___SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 5: Determines Drywell work dose. Expected Action(s): Examinee determines the Drywell on scene work dose: Given the current dose rate of 18.2 R/hr and the expected evolution time of 32 minutes, the Drywell on scene work dose = **9.7 R. (9.6 to 9.8 R)**	**Critical Step** ___SAT ___UNSAT ___N/A
Step 6: Determines Total Dose Lucky Joe will receive. Expected Action(s): Examinee determines Total Dose Lucky Joe will receive using JPM Step 4 and 5 from above: **538.3 mr (Path 2 transit) + 9.7 R (Drywell on scene work) = 10.238 R (10.1 - 10.4 R)**	**Critical Step** ___SAT ___UNSAT ___N/A
Examiner Note: Total Dose of approximately 10.238 R is estimated Emergency Exposure for round trip and on scene Drywell work which is well below the limit of 23.75R listed on Appendix B for JPM Step 7 below. Radiation Protection will list the Dose Limit of 23.75R (25R (Lifesaving) - 1.25 (Annual dose from Cue)) in accordance with EPIP-15, Section 3.4.B.	

Examiner Note: The candidate must raise their hand to request a copy of Appendix B once they determine it is required to be completed. The Critical Step for Step 7 below is the Examinee authorizes the emergency exposure with a signature, time and date given that they are the Shift Manager/Site Emergency Director (SED).	
Step 7: Completes EPIP 15, Emergency Exposure, Appendix B, Acknowledgment and Authorization to Exceed Occupational Dose Limits. Expected Action(s): Examinee determines that as the Shift Manager and acting Site Emergency Director, they will authorize the Emergency Dose using EPIP-15, Emergency Exposure, Appendix B, Acknowledgment and Authorization to Exceed Occupational Dose Limit. The Total Dose is greater than the normal 10 CFR20.1201 Occupational Dose Limit of 5R.	**Critical Step** ___SAT ___UNSAT ___N/A
END OF TASK	

STOP TIME:_____

Job Performance Measure (JPM)

****

Provide to Applicant - Classroom

****

INITIAL CONDITIONS: A General Emergency has been declared on Unit 2 by you as the Shift Manager and the Technical Support Center has **NOT** yet been staffed. Drywell entry is required to perform a lifesaving event with the following conditions:

- The dose rate in the Drywell is 18.2 R/hr, and the evolution will take 32 minutes
- Lucky Joe is making the entry and has received 1.25R annual dose to date
- Dose rates for this evolution are **NOT** due to airborne activity

Three different paths to arrive at the work location in the Drywell are detailed below:

Dose Rate	Path 1 Transit time	Dose Rate	Path 2 Transit time	Dose Rate	Path 3 Transit time
300 mr/hr	From staging area to point A requires: 10 minutes	300 mr/hr	From staging area to point E requires: 10 minutes	300 mr/hr	From staging area to point H requires: 10 minutes
300 mr/hr	From point A to point B requires: 3 minutes	300 mr/hr	From point E to point F requires: 3 minutes	150 mr/hr	From point H to point I requires: 5 minutes
100 mr/hr	From point B to point C requires: 10 minutes	150 mr/hr	From point F to point G requires: 15 minutes	1.5 R/hr	From point I to the work location requires: 10 minutes
150 mr/hr	From point C to point D requires: 10 minutes	2 R/hr	From point G to the work location requires: 5 minutes		
3 R/hr	From point D to the work location requires: 7 minutes				

INITIATING CUES:

Given the above conditions, determine the **ROUND TRIP** TEDE for each path and Total Dose using ALARA received to perform the required evolution by completing the attached table. **(show all work on the attached pages)**

If an Emergency Exposure is required, determine authorization requirements and complete the necessary documentation as applicable.

SRO – RC

Dose Rate	Path 1 Transit time	Dose Rate	Path 2 Transit time	Dose Rate	Path 3 Transit time
300 mr/hr	From staging area to point A requires: 10 minutes =_____	300 mr/hr	From staging area to point E requires: 10 minutes =_____	300 mr/hr	From staging area to point H requires: 10 minutes =_____
300 mr/hr	From point A to point B requires: 3 minutes =_____	300 mr/hr	From point E to point F requires: 3 minutes =_____	150 mr/hr	From point H to point I requires: 5 minutes =_____
100 mr/hr	From point B to point C requires: 10 minutes =_____	150 mr/hr	From point F to point G requires: 15 minutes =_____	1.5 R/hr	From point I to the work location requires: 10 minutes =_____
150 mr/hr	From point C to point D requires: 10 minutes =_____	2 R/hr	From point G to the work location requires: 5 minutes =_____		
3 R/hr	From point D to the work location requires: 7 minutes =_____				
Arrives at work location with: =_____		Arrives at work location with =_____		Arrives at work location with =_____	
Round Trip TEDE	=_____	**Round Trip TEDE**	=_____	**Round Trip TEDE**	=_____

Work Dose =_____

Total Dose using ALARA Path 1 or 2 or 3 (circle one) =_____

Job Performance Measure (JPM)

Name: _____ Date: _____