Form 4.1-BWR Boiling-Water Reactor Examination Outline

Facility: Browns Ferry														Date of Exam: May 2024													
Tier	Group	RO K/A Category Points												SRO-Only Points													
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Total	A2				G			Total						
1. Emergency and Abnormal Plant Evolutions	1	3	3	4				4	3			3	20	3				4			7						
	2	1	1	1				1	1			1	6	2				1			3						
	Tier Totals	4	4	5				5	4			4	26	5				5			10						
2. Plant Systems	1	2	2	2	3	3	2	2	2	3	3	2	26	2				3			5						
	2	1	1	1	1	1	1	1	2	1	0	1	11	0	2			1			3						
	Tier Totals	3	3	3	4	4	3	3	4	4	3	3	37	4				4			8						
3. Generic Knowledge and Abilities Categories	CO		EC				RC		EM				6	CO		EC		RC		EM	7						
	2		2				1		1					2		2		1		2							
4. Theory	Reactor Theory						Thermodynamics						6														
	3						3																				

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

* These systems/evolutions may be eliminated from the sample when Revision 2 of the K/A catalog is used to develop the sample plan

** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan

Form 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)							Page 2	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
(295001) (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION	X						(295001AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.8 to 41.10) Thermal limits	4.1	1
(295001) (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION						X	(295001) (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION (G2.2.23) EQUIPMENT CONTROL Ability to track technical specification limiting conditions for operation (CFR: 41.10 / 43.2 / 45.13)	4.6	76
(295003) (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER	X						(295003AK1.07) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 3) PARTIAL OR COMPLETE LOSS OF AC POWER:(CFR: 41.8 to 41.10) Load sequencing	3.8	2
(295004) (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER						X	(295004) (APE 4) PARTIAL OR COMPLETE LOSS OF DC POWER (G2.1.20) CONDUCT OF OPERATIONS Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12)	4.6	3
(295005) (APE 5) MAIN TURBINE GENERATOR TRIP			X				(295005AK3.05) Knowledge of the reasons for the following responses or actions as they apply to (APE 5) MAIN TURBINE GENERATOR TRIP: (CFR: 41.5 / 45.6) Extraction steam/moisture separator isolations	2.8	4
(295006) (APE 6) SCRAM		X					(295006AK2.01) Knowledge of the relationship between the (APE 6) SCRAM and the following systems or components: (CFR: 41.7 / 45.8) RPS	4.5	5
(295016) (APE 16) CONTROL ROOM ABANDONMENT					X		(295016AA2.03) Ability to determine and/or interpret the following as they apply to (APE 16) CONTROL ROOM ABANDONMENT: (CFR: 41.10 / 43.5 / 45.13) Reactor pressure	4.5	6
(295016) (APE 16) CONTROL ROOM ABANDONMENT						X	(295016) (APE 16) CONTROL ROOM ABANDONMENT (G2.2.37) EQUIPMENT CONTROL Ability to determine operability or availability of safety-related equipment (SRO Only) (CFR: 43.2 / 43.5 / 45.12)	4.6	77
(295018) (APE 18) PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)			X				(295018AK3.04) Knowledge of the reasons for the following responses or actions as they apply to (APE 18) PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): (CFR: 41.5 / 45.6) Starting the standby pump	3.5	7

(295019) (APE 19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR					X		(295019AA2.01) Ability to determine and/or interpret the following as they apply to (APE 19) PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.10 / 43.5 / 45.13) Instrument air pressure	4.1	8
(295021) (APE 21) LOSS OF SHUTDOWN COOLING				X			(295021AA1.01) Ability to operate and/or monitor the following as they apply to (APE 21) LOSS OF SHUTDOWN COOLING: (CFR: 41.7 / 45.6) Reactor water cleanup system	3.4	9
(295023) (APE 23) REFUELING ACCIDENTS			X				(295023AK3.03) Knowledge of the reasons for the following responses or actions as they apply to (APE 23) REFUELING ACCIDENTS: (CFR: 41.5 / 45.6) Ventilation isolation	3.7	10
(295023) (APE 23) REFUELING ACCIDENTS					X		(295023AA2.01) Ability to determine and/or interpret the following as they apply to (APE 23) REFUELING ACCIDENTS: (CFR: 41.10 / 43.5 / 45.13) Radiation levels	3.9	78
(295024) (EPE 1) HIGH DRYWELL PRESSURE						X	(295024) (EPE 1) HIGH DRYWELL PRESSURE (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 (CFR: 41.10 / 45.12)	4.6	11
(295024) (EPE 1) HIGH DRYWELL PRESSURE						X	(295024) (EPE 1) HIGH DRYWELL PRESSURE (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 (CFR: 41.10 / 43.5 / 45.13)	4.4	79
(295025) (EPE 2) HIGH REACTOR PRESSURE	X						(295025EK1.07) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (EPE 2) HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) Pressure control strategies	4.2	12
(295026) (EPE 3) SUPPRESSION POOL HIGH WATER TEMPERATURE				X			(295026EA1.08) Ability to operate and/or monitor the following as they apply to (EPE 3) SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.7 / 45.6) LPCS	3.8	13
(295028) (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY)		X					(295028EK2.01) Knowledge of the relationship between the (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY) and the following systems or components: (CFR: 41.7 / 45.8) RHR/LPCI: containment spray system mode	3.9	14

(295028) (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY)						X	(295028) (EPE 5) HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY) (G2.4.18) EMERGENCY PROCEDURES / PLAN Knowledge of the specific bases for emergency and abnormal operating procedures (CFR: 41.10 / 43.1 / 45.13)	4.0	80
(295030) (EPE 7) LOW SUPPRESSION POOL WATER LEVEL				X			(295030EA1.04) Ability to operate and/or monitor the following as they apply to (EPE 7) LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.7 / 45.6) Suppression pool makeup system(s)	3.5	15
(295031) (EPE 8) REACTOR LOW WATER LEVEL					X		(295031EA2.01) Ability to determine and/or interpret the following as they apply to (EPE 8) REACTOR LOW WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Reactor water level	4.7	16
(295031) (EPE 8) REACTOR LOW WATER LEVEL					X		(295031EA2.04) Ability to determine and/or interpret the following as they apply to (EPE 8) REACTOR LOW WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Adequate core cooling	4.7	81
(295037) (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN				X			(295037EA1.02) Ability to operate and/or monitor the following as they apply to (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.7 / 45.5 to 45.8) Redundant reactivity control system	4.0	17
(295037) (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN					X		(295037EA2.04) Ability to determine and/or interpret the following as they apply to (EPE 14) SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.10 / 43.5 / 45.13) Suppression pool temperature	4.4	82
(295038) (EPE 15) HIGH OFFSITE RADIOACTIVITY RELEASE RATE		X					(295038EK2.15) Knowledge of the relationship between the (EPE 15) HIGH OFFSITE RADIOACTIVITY RELEASE RATE and the following systems or components: (CFR: 41.7 / 45.8) Primary containment system and auxiliaries	3.7	18
(600000) (APE 24) PLANT FIRE ON SITE			X				(600000AK3.02) Knowledge of the reasons for the following responses or actions as they apply to (APE 24) PLANT FIRE ON SITE: (CFR: 41.5 / 41.10 / 45.6 / 45.13) Steps called out in the site fire protection plant, fire protection system manual, and fire zone manual	3.3	19
(700000) (APE 25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES						X	(700000) (APE 25) GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES (G2.4.46) EMERGENCY PROCEDURES / PLAN Ability to verify that the alarms are consistent with the plant conditions (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.2	20
K/A Category Totals:	3	3	4	4	3/3	3/4	Group Point Total:		20/7

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
(295007) (APE 7) HIGH REACTOR PRESSURE					X		(295007AA2.04) Ability to determine and/or interpret the following as they apply to (APE 7) HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) Bypass valve capacity	4.0	83
(295008) (APE 8) HIGH REACTOR WATER LEVEL	X						(295008AK1.02) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 8) HIGH REACTOR WATER LEVEL: (CFR: 41.8 to 41.10) Component erosion/damage	3.1	21
(295009) (APE 9) LOW REACTOR WATER LEVEL					X		(295009AA2.02) Ability to determine and/or interpret the following as they apply to (APE 9) LOW REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Steam flow/feed flow mismatch	3.9	22
(295010) (APE 10) HIGH DRYWELL PRESSURE				X			(295010AA1.11) Ability to operate and/or monitor the following as they apply to (APE 10) HIGH DRYWELL PRESSURE: (CFR: 41.7 / 45.6) Drywell/suppression chamber differential pressure (Mark I, II)	3.6	23
(295013) (APE 13) HIGH SUPPRESSION POOL TEMPERATURE.		X					(295013AK2.01) Knowledge of the relationship between the (APE 13) HIGH SUPPRESSION POOL TEMPERATURE. and the following systems or components: (CFR: 41.7 / 45.8) Suppression pool cooling	3.9	24
(295022) (APE 22) LOSS OF CONTROL ROD DRIVE PUMPS			X				(295022AK3.01) Knowledge of the reasons for the following responses or actions as they apply to (APE 22) LOSS OF CONTROL ROD DRIVE PUMPS: (CFR: 41.5 / 45.6) Reactor SCRAM	4.0	25
(295029) (EPE 6) HIGH SUPPRESSION POOL WATER LEVEL						X	(295029) (EPE 6) HIGH SUPPRESSION POOL WATER LEVEL (G2.1.7) CONDUCT OF OPERATIONS Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 43.5 / 45.12 / 45.13)	4.4	26
(295032) (EPE 9) HIGH SECONDARY CONTAINMENT AREA TEMPERATURE					X		(295032EA2.02) Ability to determine and/or interpret the following as they apply to (EPE 9) HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Equipment operability	3.5	84

(295036) (EPE 13) SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL						X	(295036) (EPE 13) SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL (G2.4.20) EMERGENCY PROCEDURES / PLAN Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions and notes (CFR: 41.10 / 43.5 / 45.13)	4.3	85
K/A Category Point Totals:	1	1	1	1	1/2	1/1	Group Point Total:		6/3

Form 4.1-BWR	BWR Examination Outline Plant Systems—Tier 2/Group 1 (RO/SRO)											Page 4	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
(203000) (SF2, SF4 RHR/LPCI) RHR/LPCI: INJECTION MODE									X			(203000A3.01) Ability to monitor automatic operation of the (SF2, SF4 RHR/LPCI) RHR/LPCI: INJECTION MODE (BWR 3, 4, 5, 6) including: (CFR: 41.7 / 45.7) Valve operation	4.0	27
(205000) (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE)						X						(205000K6.08) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF4 SCS) SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.7 / 45.7) Service water	3.3	28
(206000) (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM					X							(206000K5.10) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM: (CFR: 41.5 / 45.3) Reactor pressure control	3.9	29
(206000) (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM								X				(206000A2.05) Ability to (a) predict the impacts of the following on the (SF2, SF4 HPCI) HIGH PRESSURE COOLANT INJECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) DC electrical distribution system failures	3.9	86
(209001) (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM	X											(209001K1.02) Knowledge of the physical connections and/or cause and effect relationships between the (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM and the following systems: (CFR: 41.2 to 41.9 / 45.7 / 45.8) Primary containment	3.7	30
(209001) (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM											X	(209001) (SF2, SF4 LPCS) LOW PRESSURE CORE SPRAY SYSTEM (G2.4.45) EMERGENCY PROCEDURES / PLAN Ability to prioritize and interpret the significance of each annunciator or alarm (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.3	87
(211000) (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM							X					(211000A1.01) Ability to predict and/or monitor changes in parameters associated with operation of the (SF1 SLCS) STANDBY LIQUID CONTROL SYSTEM including: (CFR: 41.5 / 45.5) Tank level	3.8	31
(212000) (SF7 RPS) REACTOR PROTECTION SYSTEM										X		(212000A4.18) Ability to manually operate and/or monitor the (SF7 RPS) REACTOR PROTECTION SYSTEM in the control room: (CFR: 41.7 / 45.5 to 45.8) Transferring RPS power supplies	3.6	32

(215003) (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM									X		(215003A4.04) Ability to manually operate and/or monitor the (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM in the control room: (CFR: 41.7 / 45.5 to 45.8) IRM back panel switches, meters, and indicating lights	3.2	33
(215004) (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM								X			(215004A3.04) Ability to monitor automatic operation of the (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM including: (CFR: 41.7 / 45.7) Control rod block status	3.7	34
(215004) (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM				X							(215004K4.03) Knowledge of (SF7 SRMS) SOURCE RANGE MONITOR SYSTEM design features and/or interlocks that provide for the following: (CFR: 41.7) Gamma compensation	2.7	35
(215005) (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR								X			(215005A3.05) Ability to monitor automatic operation of the (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR including: (CFR: 41.7 / 45.7) Flow converter/ comparator signals	3.2	36
(215005) (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR			X								(215005K3.10) Knowledge of the effect that a loss or malfunction of the (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters: (CFR: 41.7 / 45.4) OPRM	3.7	37
(217000) (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM					X						(217000K5.02) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF2, SF4 RCIC) REACTOR CORE ISOLATION COOLING SYSTEM (BWR 3, 4, 5, 6): (CFR: 41.5 / 45.3) Flow control	3.8	38
(218000) (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM										X	(218000) (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM (G2.2.36) EQUIPMENT CONTROL Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation (CFR: 41.10 / 43.2 / 45.13)	3.1	39
(218000) (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM							X				(218000A2.02) Ability to (a) predict the impacts of the following on the (SF3 ADS) AUTOMATIC DEPRESSURIZATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) Loss of coolant accident	4.1	88

(223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF						X					(223002K6.04) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF: (CFR: 41.7 / 45.7) Nuclear boiler instrumentation	3.7	40
(223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF										X	(223002) (SF5 PCIS) PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF (G2.1.6) CONDUCT OF OPERATIONS Ability to manage the control room crew during plant transients (SRO Only) (CFR: 43.5 / 45.12 / 45.13)	4.8	89
(239002) (SF3 SRV) SAFETY RELIEF VALVES	X										(239002K1.02) Knowledge of the physical connections and/or cause and effect relationships between the (SF3 SRV) SAFETY RELIEF VALVES and the following systems: (CFR: 41.2 to 41.9 / 45.7 / 45.8) Plant process computer/parameter display systems	3.0	41
(259002) (SF2 RWLCS) REACTOR WATER LEVEL CONTROL SYSTEM								X			(259002A2.03) 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: (CFR: 41.5 / 45.6) Loss of reactor water level input	3.8	42
(261000) (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM				X							(261000K4.02) Knowledge of (SF9 SGTS) STANDBY GAS TREATMENT SYSTEM design features and/or interlocks that provide for the following: (CFR: 41.7) Charcoal bed decay heat removal	3.0	43
(262001) (SF6 AC) AC ELECTRICAL DISTRIBUTION		X									(262001K2.02) (SF6 AC) AC ELECTRICAL DISTRIBUTION Knowledge of electrical power supplies to the following: (CFR: 41.7) AC breaker control power	3.7	44
(262001) (SF6 AC) AC ELECTRICAL DISTRIBUTION				X							(262001K4.08) Knowledge of (SF6 AC) AC ELECTRICAL DISTRIBUTION design features and/or interlocks that provide for the following: (CFR: 41.7) Alternate breaker control methods	3.2	45
(262002) (SF6 UPS) UNINTERRUPTABLE POWER SUPPLY (AC/DC)								X			(262002A2.05) 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: (CFR: 41.5 / 43.5 / 45.6) Loss of UPS	3.8	46

(263000) (SF6 DC) DC ELECTRICAL DISTRIBUTION										X		(263000A4.04) Ability to manually operate and/or monitor the (SF6 DC) DC ELECTRICAL DISTRIBUTION in the control room: (CFR: 41.7 / 45.5 to 45.8) Ground detection circuit	2.8	47
(264000) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET)		X										(264000K2.02) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET) Knowledge of electrical power supplies to the following: (CFR: 41.7) Fuel oil pumps	2.9	48
(264000) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET)											X	(264000) (SF6 EGE) EMERGENCY GENERATORS (DIESEL/JET) (G2.2.21) EQUIPMENT CONTROL Knowledge of pre- and post-maintenance operability requirements (CFR: 41.10 / 43.2)	4.1	90
(300000) (SF8 IA) INSTRUMENT AIR SYSTEM			X									(300000K3.10) 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: (CFR: 41.7 / 45.6) Control rod drive hydraulic system	3.6	49
(400000) (SF8 CCS) COMPONENT COOLING WATER SYSTEM											X	(400000) (SF8 CCS) COMPONENT COOLING WATER SYSTEM (291004K1.05) PUMPS (CFR: 41.3) (CENTRIFUGAL) Discuss relationships among head, flow, and power, as related to pump speed	2.9	50
(510000) (SF4 SWS*) SERVICE WATER SYSTEM							X					(510000A1.02) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4 SWS*) SERVICE WATER SYSTEM including: (CFR: 41.5 / 45.5) Temperature control valve position	3.3	51
(510000) (SF4 SWS*) SERVICE WATER SYSTEM					X							(510000K5.03) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4 SWS*) SERVICE WATER SYSTEM: (CFR: 41.4 / 41.7 / 45.5) Flood prevention	3.1	52
K/A Category Point Totals:	2	2	2	3	3	2	2	2/2	3	3	2/3	Group Point Total:		26/5

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
(201001) (SF1 CRDH) CRD HYDRAULIC SYSTEM		X										(201001K2.05) (SF1 CRDH) CRD HYDRAULIC SYSTEM Knowledge of electrical power supplies to the following: (CFR: 41.7) Alternate rod insertion valve solenoids	3.9	53
(202002) (SF1 RSCTL) RECIRCULATION FLOW CONTROL SYSTEM											X	(202002) (SF1 RSCTL) RECIRCULATION FLOW CONTROL SYSTEM (G2.2.1) EQUIPMENT CONTROL Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity (CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1)	4.4	91
(214000) (SF7 RPIS) ROD POSITION INFORMATION SYSTEM					X							(214000K5.01) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF7 RPIS) ROD POSITION INFORMATION SYSTEM (BWR 2, 3, 4, 5): (CFR: 41.5 / 45.3) Rod position indication failures	3.5	54
(216000) (SF7 NBI) NUCLEAR BOILER INSTRUMENTATION	X											(216000K1.07) Knowledge of the physical connections and/or cause and effect relationships between the (SF7 NBI) NUCLEAR BOILER INSTRUMENTATION and the following systems: (CFR: 41.2 to 41.9 / 45.7 / 45.8) ADS	4.1	55
(219000) (SF5 RHR SPC) RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE									X			(219000A3.01) Ability to monitor automatic operation of the (SF5 RHR SPC) RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE including: (CFR: 41.7 / 45.7) Valve operation	3.9	56
(233000) (SF9 FPCCU) FUEL POOL COOLING/CLEANUP											X	(233000) (SF9 FPCCU) FUEL POOL COOLING/CLEANUP (G2.4.4) EMERGENCY PROCEDURES / PLAN Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures (CFR: 41.10 / 43.2 / 45.6)	4.5	57
(241000) (SF3 RTPRS) REACTOR/TURBINE PRESSURE REGULATING SYSTEM								X				(241000A2.07) Ability to (a) predict the impacts of the following on the (SF3 RTPRS) REACTOR/ TURBINE PRESSURE REGULATING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) Loss of condenser vacuum	3.8	92

(245000) (SF4 MTGEN) MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS				X							(245000K4.06) Knowledge of (SF4 MTGEN) MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design features and/or interlocks that provide for the following: (CFR: 41.7) Generator protection	3.5	58
(256000) (SF2 CDS) CONDENSATE SYSTEM							X				(256000A1.08) Ability to predict and/or monitor changes in parameters associated with operation of the (SF2 CDS) CONDENSATE SYSTEM including: (CFR: 41.5 / 45.5) System water quality	2.8	59
(259001) (SF2 FWS) FEEDWATER SYSTEM						X					(259001K6.09) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF2 FWS) FEEDWATER SYSTEM: (CFR: 41.7 / 45.7) Reactor feedwater pump lube oil system	3.1	60
(268000) (SF9 RW) RADWASTE SYSTEM			X								(268000K3.04) Knowledge of the effect that a loss or malfunction of the (SF9 RW) RADWASTE SYSTEM will have on the following systems or system parameters: (CFR: 41.5 / 45.3) Primary containment drain sumps	3.0	61
(272000) (SF7, SF9 RMS) RADIATION MONITORING SYSTEM								X			(272000A2.01) 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: (CFR: 41.5 / 45.6) Fuel element failure	4.1	62
(286000) (SF8 FPS) FIRE PROTECTION SYSTEM								X			(286000A2.02) Ability to (a) predict the impacts of the following on the (SF8 FPS) FIRE PROTECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 45.6) DC electrical distribution failure	2.8	93
(290002) (SF4 RVI) REACTOR VESSEL INTERNALS								X			(290002A2.03) Ability to (a) predict the impacts of the following on the (SF4 RVI) REACTOR VESSEL INTERNALS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) Control rod drop accident	4.4	63
K/A Category Point Totals:	1	1	1	1	1	1	1	2/2	1	0	1/1	Group Point Total:	11/3

Form 4.1-BWR	BWR Examination Outline Generic Knowledge and Abilities—Tier 3 (RO/SRO)					Page 6
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	G2.1.37	(G2.1.37) CONDUCT OF OPERATIONS Knowledge of procedures, guidelines, or limitations associated with reactivity management (CFR: 41.1 / 41.5 / 41.10 / 43.6 / 45.6)	4.3	64	-	-
	G2.1.44	(G2.1.44) CONDUCT OF OPERATIONS Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with fuel handling personnel, operating systems from the control room in support of fueling operations, or supporting instrumentation (CFR: 41.10 / 43.7 / 45.12)	3.9	65	-	-
	G2.1.4	(G2.1.4) CONDUCT OF OPERATIONS Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no solo” operation, maintenance of active license status, 10 CFR Part 55 (CFR: 41.10 / 43.2)	-	-	3.8	94
	G2.1.15	(G2.1.15) CONDUCT OF OPERATIONS Knowledge administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda (CFR: 41.10 / 45.12)	-	-	3.4	95
	Subtotal			2		2
2. Equipment Control	G2.2.40	(G2.2.40) EQUIPMENT CONTROL Ability to apply technical specifications with action statements of less than or equal to 1 hour (CFR: 41.10 / 43.2 / 43.5 / 45.3)	3.4	66	-	-
	G2.2.43	(G2.2.43) EQUIPMENT CONTROL Knowledge of the process used to track inoperable alarms (CFR: 41.10 / 43.5 / 45.13)	3.0	67	-	-
	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 (CFR: 41.10 / 43.5 / 45.13)	-	-	3.9	96
	G2.2.45	(G2.2.45) EQUIPMENT CONTROL Ability to determine or interpret technical specifications with action statements of greater than 1 hour (SRO Only) (CFR: 43.2 / 43.5 / 45.3)	-	-	4.7	97
	Subtotal			2		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 (CFR: 41.11 / 41.12 / 43.4 / 45.9)	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 (CFR: 41.12 / 43.4 / 45.9 / 45.10)	-	-	3.7	98
	Subtotal			1		1
4. Emergency Procedures/Plan	G2.4.23	(G2.4.23) EMERGENCY PROCEDURES / PLAN Knowledge of the bases for prioritizing emergency operating procedures implementation (CFR: 41.10 / 43.5 / 45.13)	3.4	69	-	-
	G2.4.41	(G2.4.41) EMERGENCY PROCEDURES / PLAN Knowledge of the emergency action level thresholds and classifications (SRO Only) (CFR: 43.5 / 45.11)	-	-	4.6	99

	G2.4.42	(G2.4.42) EMERGENCY PROCEDURES / PLAN Knowledge of emergency response facilities (CFR: 41.10 / 45.11)	-	-	3.8	100
	Subtotal			1		2
Tier 3 Point Total				6		7

Form 4.1-BWR	BWR Examination Outline Theory—Tier 4 (RO)		Page 7	
Category	K/A #	Topic	RO	
			IR	#
1. Reactor Theory	292001	(292001K1.04) NEUTRONS (CFR: 41.1) Describe neutron moderation	3.2	70
	292006	(292006K1.04) FISSION PRODUCT POISONS (CFR: 41.1) Describe the removal of xenon-135	2.9	71
	292008	(292008K1.30) REACTOR OPERATIONAL PHYSICS (CFR: 41.1) (NORMAL REACTOR SHUTDOWN) Explain the relationship between decay heat generation and (a) power level history, (b) power production, and (c) time since reaction shut down	3.5	72
	Subtotal			3
2. Thermodynamics	293003	(293003K1.22) STEAM (CFR: 41.14) Explain the usefulness of steam tables to the control room operator	3.2	73
	293005	(293005K1.05) THERMODYNAMIC CYCLES (CFR: 41.14) State the advantages of moisture separators/reheaters and feedwater heaters for a typical steam cycle	2.8	74
	293009	(293009K1.10) CORE THERMAL LIMITS (CFR: 41.14) (MAPLHGR) Define APLHGR	3.7	75
	Subtotal			3
Tier 4 Point Total				6

INFO STARTS ON PAGE 2

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 1 (RO / SRO)	MAY 20-31 2024

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#current (previous)
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	1						**AK1.03** (10CFR 55.41.10) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: ♦ Thermal limits	4.1	69 (1)
						1	***G2.2.23 (10CFR55.43.2 – SRO ONLY)*** *Ability to track technical specification limiting conditions for operation*	*4.6*	*76*
295003 (APE 3) Partial or Complete Loss of A.C. Power / 6	1						**AK1.07** (10CFR 55.41.10) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF AC POWER: ♦ Load sequencing	3.8	2
295004 (APE 4) Partial or Complete Loss of D.C. Power / 6						1	**G2.1.20** (10CFR 55.41.10) Ability to interpret and execute procedure steps	4.6	24(3)
295005 (APE 5) Main Turbine Generator Trip / 3			1				**AK3.05** (10CFR 55.41.5) Knowledge of the reasons for the following responses or actions as they apply to MAIN TURBINE GENERATOR TRIP: ♦ Extraction steam/moisture separator isolations	2.8	4
295006 (APE 6) SCRAM / 1		1					**AK2.01** (10CFR 55.41.7) Knowledge of the relationship between SCRAM and the following systems or components: ♦ RPS	4.5	31(5)

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 1 (RO / SRO)	MAY 20-31 2024

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#current (previous)
295016 (APE 16) Control Room Abandonment / 7					1		**AA2.03** (10CFR 55.41.10) Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: ♦ Reactor Pressure	4.5	33(6)
						1	***G2.2.37 (10CFR 55.43.5 – SRO ONLY)*** *Ability to determine operability or availability of safety-related equipment (SRO Only)*	*4.6*	*77*
295018 (APE 18) Partial or Complete Loss of CCW / 8			1				**AK3.04** (10CFR 55.41.5) Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): ♦ Starting the standby pump	3.5	7
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8					1		**AA2.01** (10CFR 55.41.10) Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: ♦ Instrument air pressure	4.1	16(8)
295021 (APE 21) Loss of Shutdown Cooling / 4				1			**AA1.01** (10CFR 55.41.7) Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: ♦ Reactor water cleanup system	3.4	5(9)
295023 (APE 23) Refueling Accidents / 8			1				**AK3.03** (10CFR 55.41.5) Knowledge of the reasons for the following responses or actions as they apply to REFUELING ACCIDENTS: ♦ Ventilation isolation	3.7	41(10)
					1		***AA2.01 (10CFR 55.43.5 – SRO ONLY)*** *Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS:* ♦ *Radiation Levels*	*3.9*	*80*(78)

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 1 (RO / SRO)	MAY 20-31 2024

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#current (previous)
295024 (EPE 1) High Drywell Pressure / 5						1	**G2.1.31** (10CFR 55.41.10) Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup	4.6	34(11)
						1	***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*	*4.4*	*79*
295025 (EPE 2) High Reactor Pressure / 3	1						**EK1.07** (10CFR 55.41.8) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to HIGH REACTOR PRESSURE: ♦ Pressure control strategies	4.2	49(12)
295026 (EPE 3) Suppression Pool High Water Temperature / 5				1			**EA1.08** (10CFR 55.41.7) Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: ♦ LPCS	3.8	58(13)
~~295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5~~	--	--	--	--	--	--	N/A for BFN – K/A for Mark III Containments ONLY----------	----	----
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5		1					**EK2.01** (10CFR 55.41.7) Knowledge of the relationship between HIGH DRYWELL TEMPERATURE and the following systems or components: ♦ RHR/LPCI: containment spray system mode	3.9	14
						1	***G2.4.18 (10CFR 55.43.1 - SRO Only)*** *Knowledge of the specific bases for emergency and abnormal operating procedures*	*4.0*	*78*(80)
295030 (EPE 7) Low Suppression Pool Water Level / 5				1			**EA1.04 (**10CFR 55.41.7) Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: ♦ Suppression pool makeup system(s)	3.5	8(15)

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 1 (RO / SRO)	MAY 20-31 2024

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#current (previous)
295031 (EPE 8) Reactor Low Water Level / 2					1		**EA2.01 (**10CFR 55.41.10) Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: ♦ Reactor water level	4.7	15(16)
					1		***EA2.04 (10CFR 55.43.5 - SRO Only)*** *Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL*: ♦ Adequate core cooling	*4.7*	*81*
295037 (EPE 14) SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1				1			**EA1.02 (**10CFR 55.41.7) Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: ♦ Redundant reactivity control system	4.0	17
					1		***EA2.04 (10CFR 55.43.5 - SRO Only)*** *Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:* ♦ *Suppression pool temperature*	*4.4*	*82*
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9		1					**EK2.15 (**10CFR 55.41.7) Knowledge of the relationship between the HIGH OFFSITE RADIOACTIVITY RELEASE RATE and the following systems or components: ♦ Primary containment system and auxiliaries	3.7	18

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 1 (RO / SRO)	MAY 20-31 2024

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#current (previous)
600000 (APE 24) Plant Fire On Site / 8			1				**AK3.02 (**10CFR 55.41.10) Knowledge of the reasons for the following responses or actions as they apply to PLANT FIRE ON SITE: ♦ Steps called out in the site fire protection plant, fire protection system manual, and fire zone manual	3.3	19
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6						1	**G2.4.46 (**10CFR 55.41.10) Ability to verify that the alarms are consistent with the plant conditions	4.2	38(20)
K/A Category Totals: **(RO)**	3	3	4	4	3	3		20	
K/A Category Totals: ***(SRO)***					3	4	*Group Point Total:* ***(SRO)***	*7*	

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 2 (RO / SRO)	MAY 20-31 2024

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#current (previous)
~~295002 (APE 2) Loss of Main Condenser Vacuum / 3~~	--	--	--	--	--	--	NOT RANDOMLY SELECTED----------	----	----
295007 (APE 7) High Reactor Pressure / 3					*1*		***AA2.04 (10CFR 55.43.5 – SRO Only)*** *Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:* ♦ *Bypass valve capacity*	*4.0*	*83*
295008 (APE 8) High Reactor Water Level / 2	1						**AK1.02** (10CFR 55.41.10) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to HIGH REACTOR WATER LEVEL: ♦ Component erosion/damage	3.1	21
295009 (APE 9) Low Reactor Water Level / 2					1		**AA2.02** (10CFR 55.41.10) Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: ♦ Steam flow/feed flow mismatch	3.9	22
295010 (APE 10) High Drywell Pressure / 5				1			**AA1.11** (10CFR 55.41.7) Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: ♦ Drywell/suppression chamber differential pressure (Mark I, II)	3.6	23
~~295011 (APE 11) High Containment Temperature (Mark III Containment Only) / 5~~	--	--	--	--	--	--	N/A for BFN – K/A for Mark III Containments ONLY----------	----	----
~~295012 (APE 12) High Drywell Temperature / 5~~	--	--	--	--	--	--	NOT RANDOMLY SELECTED----------	----	----
295013 (APE 13) High Suppression Pool Temperature / 5		1					**AK2.01** (10CFR 55.41.7) Knowledge of the relationship between HIGH SUPPRESSION POOL WATER TEMPERATURE and the following systems or components: ♦ Suppression pool cooling	3.9	3(24)
~~295014 (APE 14) Inadvertent Reactivity Addition / 1~~	--	--	--	--	--	--	NOT RANDOMLY SELECTED----------	----	----

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 2 (RO / SRO)	MAY 20-31 2024

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#current (previous)
~~295017 (APE 17) High Off-Site Radioactive Release Rate / 9~~	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	----
~~295020 (APE 20) Inadvertent Containment Isolation / 5 & 7~~	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	----
295022 (APE 22) Loss of Control Rod Drive Pumps / 1			1				**AK3.01** (10CFR 55.41.5) Knowledge of the reasons for the following responses or actions as they apply to LOSS OF CONTROL ROD DRIVE PUMPS: ♦ Reactor SCRAM	4.0	25
295029 (EPE 6) High Suppression Pool Water Level / 5						1	**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	4.4	26
295032 (EPE 9) High Secondary Containment Area Temperature / 5					*1*		***EA2.02 (10CFR 55.43.5 – SRO Only)*** *Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE:* ♦ *Equipment operability*	*3.5*	*84*
~~295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9~~	--	--	--	--	--	--	NOT RANDOMLY SELECTED----------	----	----
~~295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9~~	--	--	--	--	--	--	NOT RANDOMLY SELECTED----------	----	----
~~295035 (EPE 12) Secondary Containment High Differential Pressure / 5~~	--	--	--	--	--	--	NOT RANDOMLY SELECTED----------	----	----
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5						*1*	**G2.4.20** (***(10CFR 55.43.5 – SRO Only)*** Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions and notes	4.3	85
~~500000 (EPE 16) High Containment Hydrogen Concentration / 5~~	--	--	--	--	--	--	NOT RANDOMLY SELECTED----------	----	----
K/A Category Totals: **(RO)**	1	1	1	1	1	1	Group Point Total: **(RO)**	6	
K/A Category Totals: ***(SRO)***					2	1	*Group Point Total:* ***(SRO)***	3	

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 1 (RO / SRO)	MAY 20-31 2024

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	K/A Topic(s)	IR	#current (previous)
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode									1			**A3.01** (10CFR 55.41.7) Ability to monitor automatic operation of RHR/LPCI: INJECTION MODE including: ♦ Valve operation	4.0	27
205000 (SF4 SCS) Shutdown Cooling (RHR Shutdown Cooling Mode)						1						**K6.08** (10CFR 55.41.7) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): ♦ Service water	3.3	28
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection					1			*1*				**K5.10** (10CFR 55.41.5) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the HIGH PRESSURE COOLANT INJECTION SYSTEM: ♦ Reactor pressure control	3.9	29
												A2.05 (10CFR 55.43.5 – SRO Only) *Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:* ♦ *DC electrical distribution system failures*	*3.9*	*86*
~~207000 (SF4 IC) Isolation (Emergency) Condenser~~	--	--	--	--	--	--	--	--	--	--	--	N/A for BFN – No Isolation Condenser(s)----------	----	------

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 1 (RO / SRO)	MAY 20-31 2024

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	K/A Topic(s)	IR	#current (previous)
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray	1											**K1.02** (10CFR 55.41.8) Knowledge of the physical connections and/or cause and effect relationships between the LOW PRESSURE CORE SPRAY SYSTEM and the following systems: ♦ Primary containment	3.7	30
											1	*G2.4.45* ***(10CFR 55.43.5 – SRO Only)*** *Ability to prioritize and interpret the significance of each annunciator or alarm*	*4.3*	*87*
~~209002 (SF2, SF4 HPCS) High-Pressure Core Spray~~	--	--	--	--	--	--	--	--	--	--	--	N/A for BFN – No HPCS System ----------	----	------
211000 (SF1 SLCS) Standby Liquid Control							1					**A1.01** (10CFR 55.41.5) Ability to predict and/or monitor changes in parameters associated with operation of the STANDBY LIQUID CONTROL SYSTEM including: ♦ Tank level	3.8	9(31)
212000 (SF7 RPS) Reactor Protection										1		**A4.18** (10CFR 55.41.7) Ability to manually operate and/or monitor in the control room: ♦ Transferring RPS power supplies	3.6	32
215003 (SF7 IRM) Intermediate-Range Monitor System										1		**A4.04** (10CFR 55.41.7) Ability to manually operate and/or monitor in the control room: ♦ IRM back panel switches, meters, and indicating lights	3.2	6(33)

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 1 (RO / SRO)	MAY 20-31 2024

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	K/A Topic(s)	IR	#current (previous)
215004 (SF7 SRMS) Source-Range Monitor System									1			**A3.04** (10CFR 55.41.7) Ability to monitor automatic operation of the SOURCE RANGE MONITOR SYSTEM including: ♦ Control rod block system	3.7	11(34)
				1								**K4.03** (10CFR 55.41.7) Knowledge of SOURCE RANGE MONITOR SYSTEM design features and/or interlocks that provide for the following: ♦ Gamma compensation	2.7	35
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor									1			**A3.05 (**10CFR 55.41.7) Ability to monitor automatic operation of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR including ♦ Flow converter/comparator signals	3.2	36
			1									**K3.10** (10CFR 55.41.7) Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters: ♦ OPRM	3.7	37
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling System					1							**K5.02 (**10CFR 55.41.5) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the REACTOR CORE ISOLATION COOLING SYSTEM: ♦ Flow control	3.8	20(38)

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 1 (RO / SRO)	MAY 20-31 2024

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	K/A Topic(s)	IR	#current (previous)
218000 (SF3 ADS) Automatic Depressurization											1	**G2.2.36** (10CFR 55.41.10) Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation	3.1	39
								1				***A2.02 (10CFR 55.43.5 – SRO Only)*** *Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:* ♦ *Loss of coolant accident*	*4.1*	*96*(88)
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff						1						**K6.04** (10CFR 55.41.7) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF: ♦ Nuclear boiler instrumentation	3.7	40
											1	***G2.1.6 (10CFR 55.43.5 – SRO Only)*** *Ability to manage the control room crew during plant transients (SRO Only)*	*4.8*	*89*
239002 (SF3 SRV) Safety Relief Valves	1											**K1.02 (**10CFR 55.41.7/8) Knowledge of the physical connections and/or cause and effect relationships between the SAFETY RELIEF VALVES and the following systems ♦ Plant process computer/parameter display systems	3.0	10(41)
259002 (SF2 RWLCS) Reactor Water Level Control								1				**A2.03 (**10CFR 55.41.5) 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	3.8	45(42)

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 1 (RO / SRO)	MAY 20-31 2024

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	K/A Topic(s)	IR	#current (previous)
261000 (SF9 SGTS) Standby Gas Treatment				1								**K4.02 (**10CFR 55.41.7) Knowledge of STANDBY GAS TREATMENT SYSTEM design features and/or interlocks that provide for the following: ♦ Charcoal bed decay heat removal	3.0	43
262001 (SF6 AC) AC Electrical Distribution		1										**K2.02** (10CFR 55.41.7) Knowledge of electrical power supplies to the following: ♦ AC breaker control power	3.7	44
				1								**K4.08** (10CFR 55.41.7) Knowledge of AC ELECTRICAL DISTRIBUTION design features and/or interlocks that provide for the following: ♦ Alternate breaker control methods	3.2	42(45)
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)								1				**A2.05** (10CFR 55.41.5) 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: ♦ Loss of UPS	3.8	46
263000 (SF6 DC) DC Electrical Distribution										1		**A4.04** (10CFR 55.41.7) Ability to manually operate and/or monitor DC ELECTRICAL DISTRIBUTION in the control room: ♦ Ground detection circuit	2.8	47
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG		1										**K2.02** (10CFR 55.41.7) Knowledge of electrical power supplies to the following: ♦ Fuel oil pumps	2.9	48
											1	***G2.2.21 (10CFR 55.43.2 – SRO Only)*** *Knowledge of pre- and post-maintenance operability requirements*	*4.1*	*90*

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 1 (RO / SRO)	MAY 20-31 2024

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	K/A Topic(s)	IR	#current (previous)
300000 (SF8 IA) Instrument Air			1									**K3.10** (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: ♦ Control rod drive hydraulic system	3.6	12(49)
400000 (SF8 CCW) Component Cooling Water											1	**291004 K1.05** (10CFR 55.41.7) Pumps ♦ Discuss relationships among head, flow, and power, as related to pump speed	2.9	50
510000 (SF4 SWS*) Service Water (Normal and Emergency)							1					**A1.02** (10CFR 55.41.5) Ability to predict and/or monitor changes in parameters associated with operation of the SERVICE WATER SYSTEM including: ♦ Temperature control valve position	3.3	51
					1							**K5.03** (10CFR 55.41.4/7) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the SERVICE WATER SYSTEM: ♦ Flood prevention	3.1	52
K/A Category Point Totals: **(RO)**	2	2	2	3	3	2	2	2	3	3	2	Group Point Total: **(RO)**	26	
K/A Category Point Totals: ***(SRO)***								*2*			*3*	*Group Point Total:* ***(SRO)***	*5*	

SRO only K/As shown in italics

FORM 4.1-BWR | BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 2 (RO / SRO) | MAY 20-31 2024

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#current (previous)
201001 (SF1 CRDH) CRD Hydraulic		1										**K2.05** (10CFR 55.41.7) Knowledge of electrical power supplies to the following: ♦ Alternate rod insertion valve solenoids	3.9	53
~~201002 (SF1 RMCS) Reactor Manual Control~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~201003 (SF1 CRDM) Control Rod and Drive Mechanism~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~201004 (SF7 RSCS) Rod Sequence Control~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~201005 (SF1, SF7 RCIS) Rod Control and Information~~	-	-	-	-	-	-	-	-	-	-	-	N/A for BFN – BWR 6 ONLY----------	----	-------
~~201006 (SF7 RWMS) Rod Worth Minimizer~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~202001 (SF1, SF4 RS) Recirculation~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
202002 (SF1 RSCTL) Recirculation Flow Control											*1*	***G2.2.1 (10CFR 55.43.5 – SRO Only)*** *Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity*	4.4	91
~~204000 (SF2 RWCU) Reactor Water Cleanup~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
214000 (SF7 RPIS) Rod Position Information					1							**K5.01** (10CFR 55.41.5) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the ROD POSITION INFORMATION SYSTEM: ♦ Rod position indication failures	3.5	54
~~215001 (SF7 TIP) Traversing In-Core Probe~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~215002 (SF7 RBMS) Rod Block Monitor~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 2 (RO / SRO)												MAY 20-31 2024
System # / Name	**K 1**	**K 2**	**K 3**	**K 4**	**K 5**	**K 6**	**A 1**	**A 2**	**A 3**	**A 4**	**G**	**K/A Topic(s)**	**IR**	**#current (previous)**
216000 (SF7 NBI) Nuclear Boiler Instrumentation	1											**K1.07** (10CFR 55.41.7/8) Knowledge of the physical connections and/or cause and effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following systems: ♦ ADS	4.1	55
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode									1			**A3.01** (10CFR 55.41.7) Ability to monitor automatic operation of the RHR/LPCI TORUS/SUPPRESSION POOL COOLING MODE including: ♦ Valve operation	3.9	56
~~223001 (SF5 PCS) Primary Containment and Auxiliaries~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup											1	**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	4.5	57
~~234000 (SF8 FH) Fuel Handling Equipment~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~239001 (SF3, SF4 MRSS) Main and Reheat Steam~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~239003 (SF9 MSVLCS) Main Steam Isolation Valve Leakage Control~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating								*1*				***A2.07 (10CFR 55.43.5 – SRO Only)*** *Ability to (a) predict the impacts of the following on the REACTOR/ TURBINE PRESSURE REGULATING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:* ♦ *Loss of condenser vacuum*	*3.8*	*92*

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 2 (RO / SRO)											MAY 20-31 2024	
System # / Name	**K 1**	**K 2**	**K 3**	**K 4**	**K 5**	**K 6**	**A 1**	**A 2**	**A 3**	**A 4**	**G**	**K/A Topic(s)**	**IR**	**#current (previous)**
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary				1								**K4.06** (10CFR 55.41.7) Knowledge of MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design features and/or interlocks that provide for the following: ♦ Generator protection	3.5	13(58)
256000 (SF2 CDS) Condensate							1					**A1.08** (10CFR 55.41.5) Ability to predict and/or monitor changes in parameters associated with operation of the CONDENSATE SYSTEM including: ♦ System water quality	2.8	59
259001 (SF2 FWS) Feedwater						1						**K6.09** (10CFR 55.41.7) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the FEEDWATER SYSTEM: ♦ Reactor feedwater pump lube oil system	3.1	60
268000 (SF9 RW) Radwaste			1									**K3.04** (10CFR 55.41.5) Knowledge of the effect that a loss or malfunction of the RADWASTE SYSTEM will have on the following systems or system parameters: ♦ Primary containment drain sumps	3.0	61
~~271000 (SF9 OG) Offgas~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
272000 (SF7, SF9 RMS) Radiation Monitoring								1				**A2.01 (10CFR 55.41.7)** 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: ♦ Fuel element failure	4.1	62

SRO only K/As shown in italics

FORM 4.1-BWR	BWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 2/Group 2 (RO / SRO)											MAY 20-31 2024	
System # / Name	**K 1**	**K 2**	**K 3**	**K 4**	**K 5**	**K 6**	**A 1**	**A 2**	**A 3**	**A 4**	**G**	**K/A Topic(s)**	**IR**	**#current (previous)**
286000 (SF8 FPS) Fire Protection								*1*				***A2.02 (10CFR 55.43.5 – SRO Only)*** Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: ♦ DC electrical distribution failure	2.8	93
~~288000 (SF9 PVS) Plant Ventilation~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~290001 (SF5 SC) Secondary Containment~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
290002 (SF4 RVI) Reactor Vessel Internals								1				**A2.03** (10CFR 55.41.5) Ability to (a) predict the impacts of the following on the (SF4 RVI) REACTOR VESSEL INTERNALS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: ♦ Control rod drop accident	4.4	66(63)
~~290003 (SF9 CRV) Control Room Ventilation~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
~~510001 (SF8 CWS*) Circulating Water~~	-	-	-	-	-	-	-	-	-	-	-	NOT RANDOMLY SELECTED----------	----	-------
K/A Category Point Totals: **(RO)**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**2**	**1**	**0**	**1**	Group Point Total: **(RO)**	11	
K/A Category Point Totals: ***(SRO)***								**2**			**1**	*Group Point Total:* ***(SRO)***	3	

SRO only K/As shown in italics

Facility: Browns Ferry NPP Date of Exam: May 20 – 31, 2024

Category	K/A #	Topic	RO(previous)		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1	**G2.1.37** (10CFR 55.41.10) Knowledge of procedures, guidelines, or limitations associated with reactivity management	4.3	64		
		G2.1.44 (10CFR 55.41.10) Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with fuel handling personnel, operating systems from the control room in support of fueling operations, or supporting instrumentation	3.9	65		
		G2.1.4 (10CFR 55.43.2 – SRO Only) *Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no solo" operation, maintenance of active license status, 10 CFR Part 55*			*3.8*	*94*
		G2.1.15 (10CFR 55.43.10 – SRO Only) *Knowledge administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda*			*3.4*	95
		Subtotal		2		*2*
2. Equipment Control	2.2	**G2.2.40 (**10CFR 55.41.10) Ability to apply technical specifications with action statements of less than or equal to 1 hour	3.4	63(66)		
		G2.2.43 (10CFR 55.41.10) Knowledge of the process used to track inoperable alarms	3.0	67		
		G2.2.18 (10CFR 55.43.5 – SRO Only) *Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments and work prioritization*			*3.9*	88 (96)
		G2.2.45 (10CFR 55.43.2/5 – SRO Only) *Ability to determine or interpret technical specifications with action statements of greater than 1 hour (SRO Only)*			*4.7*	*97*
		Subtotal		2		*2*

SRO only K/As shown in italics

Facility: Browns Ferry NPP Date of Exam: May 20 – 31, 2024

Category	K/A #	Topic	RO(previous)		SRO-Only	
			IR	#	IR	#
3. Radiation Control	2.3	**G2.3.5 (**10CFR 55.41.12) Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms or personnel monitoring equipment	2.9	68		
		G2.3.12 (10CFR 55.43.4 – SRO Only) *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*			*3.7*	*98*
		Subtotal		1		*1*
4. Emergency Procedures / Plan	2.4	**G2.4.23 (**10CFR 55.41.10) Knowledge of the bases for prioritizing emergency operating procedures implementation	3.4	1(69)		
		G2.4.41 (10CFR 55.43.5 – SRO Only) *Knowledge of the emergency action level thresholds and classifications (SRO Only)*			*4.6*	*99*
		G2.4.42 (10CFR 55.43.5 – SRO Only) *Knowledge of emergency response facilities*			*3.8*	*100*
		Subtotal		1		*2*
		Tier 3 Point Total		6		*7*

SRO only K/As shown in italics

Facility: Browns Ferry NPP Date of Exam: May 20 – 31, 2024

Category	K/A #	Topic	RO	
			IR	# current (previous)
1. Reactor Theory	292001	(292001K1.04) NEUTRONS (10CFR55.41.1) Describe neutron moderation	3.2	70
	292006	(292006K1.04) FISSION PRODUCT POISONS (10CFR55.41.1) Describe the removal of xenon-135	2.9	71
	292008	(292008K1.30) REACTOR OPERATIONAL PHYSICS (10CFR55.41.1) (NORMAL REACTOR SHUTDOWN) Explain the relationship between decay heat generation and (a) power level history, (b) power production, and (c) time since reaction shut down	3.5	75(72)
		Subtotal		3
2. Thermodynamics	293003	(293003K1.22) STEAM (10CFR55.41.14) Explain the usefulness of steam tables to the control room operator	3.2	73
	293005	(293005K1.05) THERMODYNAMIC CYCLES (10CFR55.41.14) State the advantages of moisture separators/reheaters and feedwater heaters for a typical steam cycle	2.8	74
	293009	(293009K1.10) CORE THERMAL LIMITS (10CFR55.41.14) (MAPLHGR) Define APLHGR	3.7	72(75)
		Subtotal		3
		Tier 4 Point Total		6

SRO only K/As shown in italics

Form 3.2-1 **Administrative Topics Outline**

Facility Browns Ferry NPP			Date of Examination: 5/20/24
Examination Level: RO ☒ SRO ☐			Operating Test Number: 24-04
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)		Type Code* (Step 3)
Conduct of Operations	JPM 500	0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards	R, M
	K/A 2.2.12 (RO 3.7)	Knowledge of surveillance procedures	
Conduct of Operations	JPM 745	Place RPS Channel in Trip	R, D
	K/A 2.1.25 (RO 3.9)	Ability to interpret reference materials, such as graphs, curves, tables, etc.	
Equipment Control	JPM 639	3-SR-3.4.9.3&4, Reactor Recirculation Pump Start Limitations	R, D
	K/A 2.2.12 (RO 3.7)	Knowledge of surveillance procedures.	
Radiation Control	JPM 688	Determine ALARA Path	R, M
	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** – 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards

   Given 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards, the Examinee is expected to complete 0-SR-3.8.1.A.1 to determine if Acceptance Criteria (AC) is met.

   K/A 2.2.12: Knowledge of surveillance procedures. (RO 3.7)

2. **Conduct of Ops** – Place RPS Channel in Trip

   Given a failed Reactor Protection System (RPS) instrument, the Examinee is expected to determine how to place the applicable instrument channel in trip and identify the correct procedure(s)/print(s).

   K/A 2.1.25: Ability to interpret reference materials, such as graphs, curves, tables, etc. (RO 3.9)

3. **Equipment Control** – 3-SR-3.4.9.3&4 Reactor Recirculation Pump Start Limitations

   Given 3-SR-3.4.9.3&4, Reactor Recirculation Pump Start Limitations, the Examinee is expected to determine that Recirculation Pump 3A can be started and Recirculation Pump 3B can NOT be started.

   K/A 2.2.12: Knowledge of surveillance procedures. (RO 3.7)

4. **Radiation Control** – Determine ALARA Path

   Given plant conditions, the Examinee is expected to determine task total dose for each transit path and determine which path will maintain ALARA.

   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: 5/20/24

Examination Level: RO ☐ SRO ☒ — Operating Test Number: 24-04

Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)		Type Code* (Step 3)
Conduct of Operations	JPM 500	0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards	R, M
	K/A 2.1.12 (SRO 4.1)	Knowledge of surveillance procedures	
Conduct of Operations	JPM 745	Place an RPS Channel in Trip	R, D
	K/A 2.1.25 (SRO 4.2)	Ability to interpret reference materials, such as graphs, curves, tables, etc.	
Equipment Control	JPM 759	Review 3-SR-3.3.8.2.1(A), RPS Circuit Protector Calibration/Functional Test for 3A1 and 3A2	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 732	Determine Radiation Exposure Limits Under Emergency Conditions	R, M
	K/A 2.3.14 (SRO 3.8)	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only).	

Emergency Plan	JPM 760	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** – 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards

   Given 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards, the Examinee is expected to review 0-SR-3.8.1.A.1 to determine if Acceptance Criteria (AC) is met and what Technical Specification Required Action(s) must be taken (if any).

   2.2.12 Knowledge of surveillance procedures. (SRO 4.1)

2. **Conduct of Ops** – Place an RPS Channel in trip and determine REQUIRED ACTIONS in accordance with Technical Specifications

   Given a failed RPS instrument, the Examinee is expected to determine how to place the applicable instrument channel in trip, identify the correct procedure(s)/print(s) and REQUIRED ACTIONS in accordance with Technical Specifications (if any).

   K/A 2.1.25: Ability to interpret reference materials, such as graphs, curves, tables, etc. (SRO 4.2)

3. **Equipment Control** – Review 3-SR-3.3.8.2.1(A) RPS Circuit Protector Calibration/Functional Test for 3A1 and 3A2

   Given a completed 3-SR-3.3.8.2.1(A), RPS Circuit Protector Calibration/Functional Test for 3A1 and 3A2, the Examinee is expected to review and determine if Acceptance Criteria (AC) is met, and REQUIRED ACTIONS in accordance with Technical Specifications (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 plant conditions, the Examinee is expected to determine if an Operator can be allowed to perform an emergency evolution due to radiation levels.

   K/A 2.3.14: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only). (SRO 3.8)

5. Emergency Plan – Emergency Action Level Classification

Given plant conditions, the Examinee is expected to use EPIP-1, Emergency Classification Procedure, to classify an event as a GENERAL EMERGENCY within 15 minutes, then fill out the required notification form within 15 minutes.

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: 5/20/24

Exam Level: ☒ RO Operating Test No.: 24-04

Control Room Systems			
System / JPM Title		**Type Code**	**Safety Function**
a. JPM 727A	Inject to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI	A, D, L, EN	2
b. JPM 48A	Reduce Reactor Pressure in accordance with EOI Appendix-11F, Alternate RPV Pressure Control Systems-RFPT on Minimum Flow	A, D, L	3
c. JPM 756A	Raise Main Turbine Load	A, N	4
d. JPM 281A	Traversing Incore Probe (TIP) Isolation Failure (**U2 ONLY**)	M	5
e. JPM 757	Transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System (**U2 ONLY**)	N	6
f. JPM 758	Place a Second RHRSW Pump in Service in accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW)	N, EN	8
g. JPM 55A	Emergency Vent Primary Containment	A, D, L, P, EN	9
h. JPM 706	Prepare Intermediate Range Monitors (IRMs) for Startup in accordance with OI-92A, Intermediate Range Monitors	D, L	7
In-Plant Systems			
i. JPM 312	Vent and Re-pressurize the SCRAM Pilot Air Header	D, L, E, R	1
j. JPM 314	Defeat Alternate Rod Insertion (ARI) Logic Trips	D, P, E, L	7
k. JPM 211A	Emergency Shutdown Diesel Generator (**U3 ONLY**)	A, D, E, EN	6

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			

Simulator Job Performance Measure

a. JPM 727A

Title: Inject to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI

Description: The Examinee is expected to inject water to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI. A low suction pressure trip condition fails to initiate an automatic trip requiring HPCI to be manually tripped to complete the task.

K/A: 206000 SF2 High Pressure Coolant Injection System A4.02: Ability to manually operate and/or monitor in the Control Room: Flow Controller (4.3)

b. JPM 48A

Title: Reduce Reactor Pressure in accordance with EOI Appendix-11F, Alternate RPV Pressure Control Systems-RFPT on Minimum Flow

Description: The Examinee is expected to perform operations required to place a Reactor Feedwater Pump (RFPT) in Reactor Pressure Control in accordance with EOI Appendix-11F. The first RFPT Discharge Valve will fail open requiring another RFPT to be selected to complete the task.

K/A: 241000 SF3 RTPRS Reactor/Turbine Pressure Regulating System A2.02; Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
High Reactor Pressure (4.4)

c. JPM 756A

Title: Raise Main Turbine Load

Description: The Examinee is expected to raise Main Turbine load and respond to a MTOT leak in accordance with MAIN TURBINE OIL TANK LEVEL LOW LOW (9-7A, Window 18) requiring the Main Turbine to be tripped.

K/A: 245000 SF4 MTGEN Main Turbine Generator and Auxiliary Systems K1.16; Knowledge of the physical connections and/or cause and effect relationships between the Main Turbine Generator and Auxiliary Systems and the following systems:
Lube oil system (3.1)

d. JPM 281A - (**U2 ONLY**)

Title: Traversing Incore Probe (TIP) Isolation Failure (**U2 ONLY**)

Description: Following a Traversing Incore Probe (TIP) Isolation Failure, the Examinee is expected to verify a Primary Containment Isolation System (PCIS) Group 8 Isolation occurred in accordance with 2-AOI-64-2E, TIP Isolation. (**U2 ONLY**)

K/A: 223002 SF5 PCIS Primary Containment Isolation System/Nuclear
Steam Supply Shutoff K1.13; Knowledge of the physical connections and/or cause and effect relationships between the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and the following systems: Traversing in-core probe system (3.1)

e. JPM 757 - (**U2 ONLY**)

Title: Transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System (**U2 ONLY**)

Description: The Examinee is expected to transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System. (**U2 ONLY**)

K/A: 262001 SF6 AC Electrical Distribution A4.01: Ability to manually operate and/or monitor in the control room: Breakers and disconnects (3.7)

f. JPM 758

Title: Place a Second RHRSW Pump in Service in accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW)

Description: The Examinee is expected to perform 0-OI-23, Residual Heat Removal Service Water (RHRSW), Section 6.1.1, Starting Second RHRSW Pump (A1) on 'A' RHRSW Header, with A2 RHRSW Pump already in-service.

K/A: 400000 SF8 CCW Component Cooling Water System K1.01: Knowledge of the physical connections and/or cause and effect relationships between the Component Cooling Water System and the following systems: Service water system (3.8)

g. JPM 55A

Title: Emergency Vent Primary Containment

Description: The Examinee is expected to vent Primary Containment in accordance with 2-EOI Appendix-13, Emergency Venting Primary Containment. The preferred vent path via the Suppression Chamber cannot be aligned requiring Drywell vent path alignment.

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)

h. JPM 706

Title: Prepare Intermediate Range Monitors (IRMs) for Startup in accordance with OI-92A, Intermediate Range Monitors.

Description: The Examinee is expected to complete IRM preparations for Reactor Startup in accordance with OI-92A, Intermediate Range Monitors.

K/A: 215003 SF7 Intermediate Range Monitor System A4.03 Ability to manually operate and/or monitor in the control room: IRM range switches (3.9)

In-Plant Job Performance Measures

i. JPM 312

Title: Vent and Re-pressurize the SCRAM Pilot Air Header

Description: The Examinee is expected to perform EOI Appendix-1B, Venting and Repressurizing the SCRAM Air Pilot Header.

K/A: 201001 SF1 CRDH Control Rod Drive Hydraulic System; K1.09; Knowledge of the physical connections and/or cause and effect relationships between the Control Rod Drive Hydraulic System and the following systems: IA (3.6)

j. JPM 314

Title: Defeat Alternate Rod Insertion (ARI) Logic Trips

Description: The Examinee is expected to perform 2-EOI Appendix-2, Defeating ARI Logic Trips.

K/A: 295037 SF7 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor Protection System. (4.2)

k. JPM 211A - (**U3 ONLY**)

Title: Emergency Shutdown Diesel Generator (**U3 ONLY**)

Description: The Examinee is expected to shut down a Diesel Generator at the Diesel Engine Control Cabinet and emergency shut down the Diesel Engine. The Injection Control Lever fails to shut down the Diesel Generator requiring an alternate local shut down method to complete the task. (**U3 ONLY**)

K/A: 264000 SF6 EGE Emergency Generators (Diesel/Jet) K4.07 Knowledge of Emergency Generators design features and/or interlocks that provide for the following: Local operation and control (3.5)

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

Facility: Browns Ferry NPP Date of Examination: 5/20/24

Exam Level: ☒ SRO-I Operating Test No.: 24-04

Control Room Systems			
System / JPM Title		**Type Code**	**Safety Function**
a. JPM 727A	Inject to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI	A, D, L, EN	2
b. JPM 48A	Reduce Reactor Pressure in accordance with EOI Appendix-11F, Alternate RPV Pressure Control Systems-RFPT on Minimum Flow	A, D, L	3
c. JPM 756A	Raise Main Turbine Load	A, N	4
d. JPM 281A	Traversing Incore Probe (TIP) Isolation Failure (**U2 ONLY**)	M	5
e. JPM 757	Transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System (**U2 ONLY**)	N	6
f. JPM 758	Place a Second RHRSW Pump in Service in accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW)	N, EN	8
g. JPM 55A	Emergency Vent Primary Containment	A, D, L, P, EN	9
h. N/A			
In-Plant Systems			
i. JPM 312	Vent and Re-pressurize the SCRAM Pilot Air Header	D, L, E, R	1
j. JPM 314	Defeat Alternate Rod Insertion (ARI) Logic Trips	D, P, E, L	7
k. JPM 211A	Emergency Shutdown Diesel Generator (**U3 ONLY**)	A, D, E, EN	6

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			

Simulator Job Performance Measure

a. JPM 727A

Title: Inject to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI

Description: The Examinee is expected to inject water to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI. A low suction pressure trip condition fails to initiate an automatic trip requiring HPCI to be manually tripped to complete the task.

K/A: 206000 SF2 High Pressure Coolant Injection System A4.02: Ability to manually operate and/or monitor in the Control Room: Flow Controller (4.3)

b. JPM 48A

Title: Reduce Reactor Pressure in accordance with EOI Appendix-11F, Alternate RPV Pressure Control Systems-RFPT on Minimum Flow

Description: The Examinee is expected to perform operations required to place a Reactor Feedwater Pump (RFPT) in Reactor Pressure Control in accordance with EOI Appendix-11F. The first RFPT Discharge Valve will fail open requiring another RFPT to be selected to complete the task.

K/A: 241000 SF3 RTPRS Reactor/Turbine Pressure Regulating System A2.02; Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
High Reactor Pressure (4.1)

c. JPM 756A

Title: Raise Main Turbine Load

Description: The Examinee is expected to raise Main Turbine load and respond to a MTOT leak in accordance with MAIN TURBINE OIL TANK LEVEL LOW LOW (9-7A, Window 18) requiring the Main Turbine to be tripped.

K/A: 245000 SF4 MTGEN Main Turbine Generator and Auxiliary Systems K1.16; Knowledge of the physical connections and/or cause and effect relationships between the Main Turbine Generator and Auxiliary Systems and the following systems:
Lube oil system (3.1)

d. JPM 281A - (**U2 ONLY**)

Title: Traversing Incore Probe (TIP) Isolation Failure (**U2 ONLY**)

Description: Following a Traversing Incore Probe (TIP) Isolation Failure, the Examinee is expected to verify a Primary Containment Isolation System (PCIS) Group 8 Isolation occurred in accordance with 2-AOI-64-2E, TIP Isolation. (**U2 ONLY**)

K/A: 223002 SF5 PCIS Primary Containment Isolation System/Nuclear
Steam Supply Shutoff K1.13; Knowledge of the physical connections and/or cause and effect relationships between the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and the following systems: Traversing in-core probe system (3.1)

e. JPM 757 - (**U2 ONLY**)

Title: Transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System (**U2 ONLY**)

Description: The Examinee is expected to transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System. (**U2 ONLY**)

K/A: 262001 SF6 AC Electrical Distribution A4.01: Ability to manually operate and/or monitor in the control room: Breakers and disconnects (3.7)

f. JPM 758

Title: Place a Second RHRSW Pump in Service in accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW)

Description: The Examinee is expected to perform 0-OI-23, Residual Heat Removal Service Water (RHRSW), Section 6.1.1, Starting Second RHRSW Pump (A1) on 'A' RHRSW Header, with A2 RHRSW Pump already in-service.

K/A: 400000 SF8 CCW Component Cooling Water System K1.01: Knowledge of the physical connections and/or cause and effect relationships between the Component Cooling Water System and the following systems: Service water system (3.8)

g. JPM 55A

Title: Emergency Vent Primary Containment

Description: The Examinee is expected to vent Primary Containment in accordance with 2-EOI Appendix-13, Emergency Venting Primary Containment. The preferred vent path via the Suppression Chamber cannot be aligned requiring Drywell vent path alignment.

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)

h. N/A

In-Plant Job Performance Measures

i. JPM 312

Title: Vent and Re-pressurize the SCRAM Pilot Air Header

Description: The Examinee is expected to perform EOI Appendix-1B, Venting and Repressurizing the SCRAM Air Pilot Header.

K/A: 201001 SF1 CRDH Control Rod Drive Hydraulic System; K1.09; Knowledge of the physical connections and/or cause and effect relationships between the Control Rod Drive Hydraulic System and the following systems: IA (3.6)

j. JPM 314

Title: Defeat Alternate Rod Insertion (ARI) Logic Trips

Description: The Examinee is expected to perform 2-EOI Appendix-2, Defeating ARI Logic Trips.

K/A: 295037 SF7 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor Protection System. (4.2)

k. JPM 211A - (**U3 ONLY**)

Title: Emergency Shutdown Diesel Generator (**U3 ONLY**)

Description: The Examinee is expected to shut down a Diesel Generator at the Diesel Engine Control Cabinet and emergency shut down the Diesel Engine. The Injection Control Lever fails to shut down the Diesel Generator requiring an alternate local shut down method to complete the task. (**U3 ONLY**)

K/A: 264000 SF6 EGE Emergency Generators (Diesel/Jet) K4.07 Knowledge of Emergency Generators design features and/or interlocks that provide for the following: Local operation and control (3.5)

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

Facility: Browns Ferry NPP Date of Examination: 5/20/24

Exam Level: ☒ SRO-U Operating Test No.: 24-04

Control Room Systems			
System / JPM Title		**Type Code**	**Safety Function**
a. JPM 727A	Inject to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI	A, D, L, EN	2
b. N/A			
c. JPM 756A	Raise Main Turbine Load	A, N	4
d. N/A			
e. JPM 757	Transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System (**U2 ONLY**)	N	6
f. N/A			
g. N/A			
h. N/A			
In-Plant Systems			
i. JPM 312	Vent and Re-pressurize the SCRAM Pilot Air Header	D, L, E, R	1
j. JPM 314	Defeat Alternate Rod Insertion (ARI) Logic Trips	D, P, E, L	7
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			

Simulator Job Performance Measure

a. JPM 727A

Title: Inject to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI

Description: The Examinee is expected to inject water to the Reactor in accordance with EOI Appendix-5D, Injection System Lineup HPCI. A low suction pressure trip condition fails to initiate an automatic trip requiring HPCI to be manually tripped to complete the task.

K/A: 206000 SF2 High Pressure Coolant Injection System A4.02: Ability to manually operate and/or monitor in the Control Room: Flow Controller (4.3)

b. N/A

c. JPM 756A

Title: Raise Main Turbine Load

Description: The Examinee is expected to raise Main Turbine load and respond to a MTOT leak in accordance with MAIN TURBINE OIL TANK LEVEL LOW LOW (9-7A, Window 18) requiring the Main Turbine to be tripped.

K/A: 245000 SF4 MTGEN Main Turbine Generator and Auxiliary Systems K1.16; Knowledge of the physical connections and/or cause and effect relationships between the Main Turbine Generator and Auxiliary Systems and the following systems:
Lube oil system (3.1)

d. N/A

e. JPM 757 - (**U2 ONLY**)

Title: Transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System (**U2 ONLY**)

Description: The Examinee is expected to transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System. (**U2 ONLY**)

K/A: 262001 SF6 AC Electrical Distribution A4.01: Ability to manually operate and/or monitor in the control room: Breakers and disconnects (3.7)

f. N/A

g. N/A

h. N/A

In-Plant Job Performance Measure

i. JPM 312

Title: Vent and Re-pressurize the SCRAM Pilot Air Header

Description: The Examinee is expected to perform EOI Appendix-1B, Venting and Repressurizing the SCRAM Air Pilot Header.

K/A: 201001 SF1 CRDH Control Rod Drive Hydraulic System; K1.09; Knowledge of the physical connections and/or cause and effect relationships between the Control Rod Drive Hydraulic System and the following systems: IA (3.6)

j. JPM 314

Title: Defeat Alternate Rod Insertion (ARI) Logic Trips

Description: The Examinee is expected to perform 2-EOI Appendix-2, Defeating ARI Logic Trips.

K/A: 295037 SF7 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor Protection System. (4.2)

k. N/A

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

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures / Plan	Tier #	3	------
G2.4.23 (10CFR 55.41.10)	Group #	------	------
Knowledge of the bases for prioritizing emergency operating procedures implementation	K/A #	G2.4.23	
	Importance Rating	3.4	------

Proposed Question: **# 1**

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

- Transient conditions exist
- Several annunciators are lit
- Events are in progress requiring Operator actions
- Several procedures are in use

Which **ONE** of the following completes the statement below?

Given the conditions above, _____ take precedence since they are designed to be executed based upon specific plant parameters (symptoms) rather than being designed to direct Operator actions based upon a specific plant transient condition.

A. Operating Instructions (OIs)

B. Alarm Response Procedures (ARPs)

C. Abnormal Operating Instructions (AOIs)

D. Emergency Operating Instructions (EOIs)

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that OIs direct general plant actions that can supplement EOIs.

B INCORRECT: Plausible in that ARPs direct operator actions based on event specific conditions, that can supplement EOIs.

C INCORRECT: Plausible in that AOIs direct operator actions based on event specific conditions, that can supplement EOIs.

D **CORRECT**: The examinee must recognize that EOIs are entered based on conditions that could degrade to, or are symptoms of emergencies, as the reason for prioritizing EOIs over other procedures.

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

RO Level Justification: Tests the candidate's knowledge of bases for prioritizing emergency operating procedures implementation. This question is rated as Memory due to the requirement to strictly recall procedural hierarchy.

Technical Reference(s): EOIPM Section 0-I(J), Rev. 1 (Attach if not previously provided)
OPL171.201, Rev. 10U2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.201, Obj. 7a-e (As available)

Question Source:
Bank # QC NRC 2016 #62
Modified Bank #
New
(Note changes or attach parent)

Question History: Last NRC Exam 2016

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:

EXAMINATION ANSWER KEY

U.S. Nuclear Regulatory Commission 2016 RO/SRO Written Exam (Quad Cities)

62 **ID: 1245371** **Points: 1.00**

Transient conditions exist.
Several annunciators are lit.
Events are in progess requiring operator actions.
Several procedures are in use.

The _____ procedures take precedence because they are designed to be executed based upon plant parameters (symptoms) rather than being designed to direct operator actions based upon a specific plant transient condition.

A. QGA

B. QCOA

C. QCOP

D. QCAN

Answer: A

Answer Explanation

The K/A is the knowledge of the bases for prioritizing emergency procedure implementation during emergency operations. The question requires the examinee to recognize the QGA basis as the reason for prioritizing QGAs over other procedures. The correct answer is directly from QCAP 0200-10, "EMERGENCY OPERATING PROCEDURE (QGA) EXECUTION STANDARDS

Distractor 1 is incorrect: Plausible because QCOA direct operator actions for transients and casualties. However, they are event based procedures.
Distractor 2 is incorrect: Plausible because QCOP direct general plant actions and they can supplement QGAs. However, they do not deal with transients and casualties.
Distractor 3 is incorrect: Plausible because QCOA direct operator actions for transients and casualties. However, they are event based procedures.

Reference: QCAP 0200-10 rev. 49
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 3

K/A: 2.4.23 EMERGENCY PROCEDURES / PLAN: Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

(CFR: 41.10/43.5/45.13)
IMPORTANCE RO 3.4 / SRO 4.4

SRO Justification: N/A

Question Source: New
Question History: N/A

Excerpt from EOIPM Section-0-I(J):

BFN Unit 0	User's Guide for the Emergency Operating Instructions	EOIPM Section 0-I(J) Rev. 0001 Page 55 of 72

3.4 Guidelines for Execution of Emergency Operating Instructions (continued)

H. Action Verbs in the EOIs

All action verbs within the EOIs are highlighted using uppercase and bold print. This helps the operator in determining the specific task(s) directed in the procedure steps. The type of action verbs used in the EOIs and their meaning are covered in detail in EOIPM Section 0-I(C) Glossary of Key Words and Terms.

I. EOI Flowchart Use With Other Plant Procedures

The EOIs are entered, based upon specific conditions symptomatic of emergencies, or conditions that could degrade into emergencies. Therefore, the operator actions provided within the EOIs allow the operator to mitigate the consequences of a broad range of accidents and multiple equipment failures.

Other procedures, such as AOIs, ARPs, EPIPs, FSSs etc., have event specific entry conditions and may be used to supplement EOIs. In some instances the EOIs will direct the operators to other procedures (OIs, GOIs, EDMGs and AOIs) for completion of specific tasks. Usually, the EOIs direct the operators to specific EOI Appendices. The Appendices are specific task-related procedures written to satisfy directives given within the EOIs.

Actions that contradict any direction given by the EOIs or reduce the effectiveness of any directions given by the EOIs WILL NOT be implemented for any reason.

The exception to this rule is the FSS-16-2s and AOI-100-2. In the case where direction conflicts between the FSS and the EOI, the FSS will take precedence. Any time that the operators must leave the control room, as directed by AOI-100-2 or the Unit 1, 2, or 3-FSS-16-2, the EOIs shall be exited and the Main Control Room Abandonment procedures shall be used to shut down and cool down the reactor.

It is recognized that during execution of the EOIs the control room will receive assistance from various support groups. This is especially the case under conditions in the EPIPs that result in the Technical Support Center (TSC) being staffed. For example, the TSC may make recommendations regarding when it is best to vent primary containment based upon present or predicted meteorological conditions. This would not contradict the directions provided by the EOIs but help to meet the intent of minimizing radiological releases to the general public.

Excerpt from OPL171.201 Lesson Plan:

e) Inability to Determine a Requested Parameter or Condition — Objective 6

a) When a specified parameter cannot be determined, or the operator is uncertain of the ability to maintain a specific condition, the most appropriate path to follow is that resulting in the most conservative action for the present condition. — OF-4

b) What the operator must look at and consider in order to determine a specific condition or parameter varies greatly depending upon the item in question. — Conservative Bias – Decision Making

c) In addition, the most conservative path to take under any given situation is not always as simple as yes or no, black and white etc.

d) Example:
If reactor power cannot be determined to be below 5%, then assume it is above 5%. If the operator is not convinced that the reactor will remain subcritical, then the path taken should reflect the possibility that the reactor may become critical. If the operator is not certain that a "CAN" type Decision Step parameter, or condition, can be maintained within that stated then the most conservative path (typically the "NO" path) should be followed.

f) EOI flowchart use with other plant procedures — Objective 7a-e

a) The EOIs are entered, based upon specific conditions symptomatic of emergencies, or conditions that could degrade into emergencies. Therefore, the operator actions provided within the EOIs allow the operator to mitigate the consequences of a broad range of accidents and multiple equipment failures.

b) Other procedures, such as FSSs, AOIs, ARPs, EPIPs, etc., have event specific entry conditions and may be used to supplement EOIs. In some instances the EOIs will direct the operators to other procedures (OIs, GOIs, EDMGs and AOIs) for completion of specific tasks. Usually, the EOIs direct the operators to specific EOI Appendices. The Appendices are specific task-related procedures written to satisfy directives given within the EOIs.

c) Actions that contradict any direction given by the EOIs or reduce the effectiveness of any directions given by the EOIs WILL NOT be implemented for any reason.

d) The exception to this rule is AOI-100-2. Any time that the operators must leave the control room, as directed by AOI-100-2, the EOIs shall be exited and AOI-100-2 shall be used to shut down and cool down the reactor. The EOIs are not designed or written to support their use outside of the main control room.

e) It is recognized that during execution of the EOIs the control room will receive assistance from various support groups. This is especially the case under conditions in the EPIPs that result in the Technical Support Center (TSC) being staffed. For example, the TSC may make recommendations regarding when it is best to vent primary containment based upon present or predicted meteorological conditions. This would not

Examination Outline Cross-reference:	Level	RO	SRO
295003 (APE 3) Partial or Complete Loss of A.C. Power / 6	Tier #	1	------
AK1.07 (10CFR 55.41.10)	Group #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF AC POWER:	K/A #	295003AK1.07	
Load sequencing	Importance Rating	3.8	------

Proposed Question: **# 2**

A LOCA occurred on Unit 3 resulting in the following conditions:

- Reactor Water Level reaches (-) 122 inches, slowly lowering
- '3B' EDG fails to start

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

Core Spray Pump 3A **(1)**.

Core Spray Pump 3D starts in **(2)**.

A. (1) starts automatically
(2) 7 seconds

B. (1) starts automatically
(2) 21 seconds

C. (1) can be manually started **ONLY**
(2) 7 seconds

D. (1) can be manually started **ONLY**
(2) 21 seconds

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that the starting sequence for both RHR and Core Spray Pumps during an accident is complex and easily confused. If EDGs were supplying power to the 4KV Shutdown Boards (Diesel Generator Voltage Available - DGVA), RHR Pumps start immediately and all Core Spray Pumps start 7 seconds later.

B **CORRECT:** (*See attached*) In accordance with 3-OI-75, Core Spray System, Core Spray will auto initiate with an Accident Signal (Reactor Water Level (-) 122 inches) present. Core Spray Pump 3A (loop companion pump to 3C) does have normal power available (NVA) so it will immediately and automatically start at time zero after the initiation of the Accident Signal. Second part, with normal power available (NVA), Core Spray Pump 3A will automatically start immediately at time zero, and 3B, 3C, 3D will then sequentially start at 7-second intervals. Therefore, Core Spray Pump 3D will start in 21 seconds with NVA since it is powered from 4KV Shutdown Board 3ED.

C INCORRECT: First part is incorrect but plausible in that with an Accident Signal present (Reactor Water Level (-) 122 inches), RHR and Core Spray Pumps will automatically start. However, Core Spray Pumps auto start based on loop companion pump logic. Core Spray Pump 3A must also see its loop companion pump, 3C (alternate power comes from '3B' EDG) available to start. Given that '3B' EDG fails to start, adds to the plausibility Core Spray Pump 3A will not automatically start but may be manually started. 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 determine the impact on the Core Spray System and integrated plant operations relative to accident conditions with Normal Power Available as compared to Diesel Generator Voltage Available to the 4KV Shutdown Boards. Additionally, this requires knowledge of the Electrical Distribution System and the effect of a loss of an EDG on the Core Spray System load sequencing. 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.

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; (3) The progression of an event.

Technical Reference(s): 3-OI-75, Rev. 81 (Attach if not previously provided)
3-OI-74, Rev. 147
OPL171.045, Rev. 23
OPL171.038, Rev. 27U1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.045, Obj. 4.e (As available)

Question Source: Bank # BFN NRC 1909 #38
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 **X**
55.43

Copy of Bank Question:

Proposed Question: **# 38**

A LOCA occurred on Unit 3 resulting in the following conditions:

- Reactor Water Level reaches (-) 122 inches and is slowly lowering
- '3B' EDG fails to start

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

Core Spray Pump 3A ___**(1)**___.

Core Spray Pump 3D starts in ___**(2)**___.

A. **(1)** starts automatically
(2) 7 seconds

B. **(1)** starts automatically
(2) 21 seconds

C. **(1)** can be manually started **ONLY**
(2) 7 seconds

D. **(1)** can be manually started **ONLY**
(2) 21 seconds

Proposed Answer: **B**

Excerpts from 3-OI-75:

BFN Unit 3	Core Spray System	3-OI-75 Rev. 0081 Page 20 of 135

5.0 STARTUP

5.1 Automatic Initiation

NOTES

1) ECCS SYS I(II) HI DW PRESS TEST/INHIBIT keylock switch, 3-HS-75-59(60), on Panel 3-9-3 must be in AUTO to allow automatic initiation on High Drywell Pressure concurrent with Low RPV pressure.
2) Core Spray will auto initiate on any of the following signals:
   - RPV vessel Low-Low-Low water level -122 inches
   - High Drywell pressure, at or above 2.45 psig with low RPV vessel pressure, at or below 450 psig
3) If only one pump in a loop is running, the injection valve for that loop must be throttled to achieve 3125 gpm pump flow to avoid pump run-out.
4) Upon an automatic Core Spray initiation with normal power available, Core Spray Pump 3A will start immediately and 3B, 3C, 3D will then sequentially start at 7-second intervals. Otherwise, all Core Spray Pumps will start 7 seconds after diesel power is available.
5) Core Spray Room Coolers auto start on a Core Spray initiation.

[1] **ENSURE** the following automatic actions occur:

- RHRSW Pumps A3 and C3 start.
- **IF** B1 or D1 RHRSW pumps are aligned to EECW, **THEN** the RHRSW Pump(s) that are aligned to EECW start.
- Core Spray Pumps A thru D start.
- **IF** OPEN, **THEN**:
  - CORE SPRAY SYS I and II TEST VALVEs, 3-FCV-075-0022 and 3-FCV-075-0050 CLOSES.
- PSC PUMP SUCTION INBD and OUTBD ISOL VALVEs, 3-FCV-075-0057 and 3-FCV-075-0058 close.

3.7 Interlocks

B. 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)
3A	C3, B1
3B	A3, D1
3C	A3, B1
3D	C3, B1

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 per Tech Spec 3.5.1/3.5.2.

C. After operation of 4160V breakers the charging spring, is required to be verified to have recharged by ensuring locally, the breaker closing spring target indicates charged and the amber breaker spring charged light is on to ensure 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 3-OI-74:

5.0 STARTUP

5.1 Automatic LPCI Initiation

NOTES

1) The following LPCI automatic functions are disabled when both logic channels are in Test:
   - RHR SYS I(II) LPCI INBD INJECT VALVE, 3-FCV-74-53(67), opening.
   - RECIRC PUMP 3A(3B) DISCHARGE VALVE, 3-FCV-68-3(79), closure.
   - RHR SYS I and II SUPPR POOL CLG/TEST VLV, 3-FCV-74-59(73), closure.
2) DG/CS-LPCI SYS I(II) HI DW PRESS TEST / INHIBIT switch, 3-HS-75-59 (3-HS-75-60), on Panel 3-9-3 must be in AUTO to allow automatic LPCI initiation on high drywell pressure concurrent with low reactor pressure.
3) Upon an automatic LPCI initiation with normal power available, RHR Pump 3A starts immediately and 3B, 3C, 3D then sequentially starts at 7-second intervals. Otherwise, all RHR Pumps start immediately once their respective Diesel power is available (and normal power unavailable).
4) All operations are performed at Panel 3-9-3 unless otherwise noted.

[1] **ENSURE** All available RHR pumps start.

[2] RHR SYS I and II SUPPR POOL CLG/TEST VLVs, 3-FCV-74-59 and 3-FCV-74-73, close.

[3] **WHEN** reactor pressure is < 450 psig, **AND** annunciator RX PRESS LOW CORE SPRAY/RHR PERMISSIVE (3-XA-55-3C, Window 35) alarms, **THEN**:

- **ENSURE OPEN** RHR SYS I LPCI OUTBD INJECT VALVEs, 3-FCV-74-52.
- **ENSURE OPEN** RHR SYS I LPCI INBD INJECT VALVE 3-FCV-74-53
- **ENSURE OPEN** RHR SYS II LPCI OUTBD INJECT VALVE, 3-FCV-74-66
- **ENSURE OPEN** RHR SYS II LPCI OUTBD INJECT VALVE, 3-FCV-74-67.

Excerpt from OPL171.045 Lesson Plan:

h. Core spray DWP is developed as follows:	One out of two taken twice
i. 2.45# DWP from the ECCS ATUs 64- 58B(K5A) and 64-58D(K6A)	
AND	
ii. 2.45# DWP from the ECCS ATUs 64- 58A(K5B) or 64-58C(K6B)	
iii. This picks up 14AK36 which picks up 14AK35 which starts the D/Gs (PASA)	2.45# and < 450#
iv. 14AK36 in conjunction 14AK9 or 23 will:	
1. Energizes CASA relays	Note: DGVA is deenergize to function.
2. Inputs to 480V load shed	
i. Core spray initiation signal. CS Pump will not auto start if companion Pump's 4KV SD BD doesn't have power to prevent a single pump auto start and runout in a loop. Both boards must have either NVA logic energized or DGVA logic deenergized to pick up the start time delay relay.	
j. There are two specific conditions (Three for U1 & 2) that will prevent a CS pump from automatically starting	Obj. ILT 2.h Obj. LOR 1.h Obj. LOR 1.g
i. If a shutdown board is deenergized, neither CS pump in that system will automatically start. Systems are designed for only two-pump operation in automatic.	Must have either K29 or K30 contact closed AND

Excerpts from OPL171.038 Lesson Plan:

Outline of Instruction | **Instructor Notes and Methods (optional)**

timers to time out before resetting the trip lever

(10) To reset the overspeed trip rotate the limit switch fully clockwise by hand and pull down on the reset lever. This should latch and keep the limit switch positioned clockwise

(11) A local emergency stop may be required if the normal remote or local electrical stop signals fail to begin the engine stop sequence. This may be performed by pulling the fuel rack injector lever (located between the diesel and the engine control cabinet) outward from the engine, and holding until the engine comes to a stop. If the DG had been started using the local slow start procedure, a "start failure" signal may be sent to the redundant start signal, which would cause the engine to try and restart until the redundant start circuit locks out after 5.5 seconds elapse.

NLOR OBJ. 17
NLO OBJ. 17

(12) The redundant start may be canceled before the start circuit locks out by opening the logic breaker and pushing both engine stop push-buttons. Note that pulling the engine driven fuel pump shutoff plunger will not stop the diesel since the electric fuel pump will still be supplying fuel

3. Accident Operation

a) Accident signal received **CAS**

(1) Signals ALL 8 diesel generators to start

(2) Opens ALL 8 diesel output breakers if shut

b) If normal voltage is available (NVA), load will sequence on as follows

Time After Accident	S/D Board A	S/D Board C	S/D Board B	S/D Board D
0	RHR/CS A			
7		RHR/CS B		
14			RHR/CS C	
21				RHR/CS D
28	RHRSW*	RHRSW*	RHRSW*	RHRSW*
RHRSW pumps assigned for EECW automatic start				

ILT OBJ. 9
LOR OBJ. 6
NLOR OBJ. 15, 16
NLO OBJ. 15, 16

c) If normal voltage is NOT available: i.e. on (DGVA)

(1) After 5-second time delay, all 4kV Shutdown Board loads except 4160/480V transformer breakers are automatically tripped

(2) Diesel generator output breaker closes when diesel is at speed.
Loads sequence as indicated below

OPL171.038 Diesel Generator and Auxiliary Power System Rev 27 U1

Outline of Instruction	Instructor Notes and Methods (optional)

*Time After Accident	S/D Board A	S/D Board B	S/D Board C	S/D Board D
0	RHR A	RHR C	RHR B	RHR D
7	CS A	CS C	CS B	CS D
14	RHRSW *	RHRSW *	RHRSW *	RHRSW *
RHRSW pumps assigned for EECW automatic start				

d) Certain 480V loads are shed whenever an accident signal is received in conjunction with the diesel generator tied to the board. (see OPL171.072)

4. OI-82 precautions for the operator

a) Lube oil modification for Diesel Generators eliminate the vendor recommended prohibition on fast starts of the diesel generators during the 15 minute to 3 hour time period following shutdown.

ILT OBJ. 9
LOR OBJ. 6
NLOR OBJ. 15, 16
NLO OBJ. 15, 16

b) Continuous operation of the Diesel Generators at <50% load should be avoided to prevent oil and soot accumulation in the exhaust system, air box, cylinders, and injection nozzles. If diesel generator idle time exceeds 8 hours, or if diesel operates greater than 4 hours at < 50% load, the diesel should be fully loaded for at least 30 minutes prior to engine shutdown

c) Potentially significant safety problems can arise when one or more emergency diesel generators are operated in parallel with off-site power

(1) Experience has shown that such a practice can lead to a complete loss of AC power to safety buses. Although operators may be tempted to start the DGs when off-site power is threatened or is undergoing disturbances (during periods of severe weather, for example), running them is likely to be more of a hindrance than a help

(2) When a DG is connected to off-site or non-vital loads, it is no longer independent of disturbances that can adversely affect emergency power availability

5. Technical Specifications

a) Review Current Technical Specification Section 3.3.8.1, 3.8.1, 3.8.2, 3.8.3, 3.8.4.d, 3.8.4.e, 3.8.5, 3.8.6.

b) Common cause evaluation by engineering

D. Plant Operating Experience

1. The Probabilistic Safety Assessment (PSA) indicates that failure of emergency electrical power systems, leading to SBO conditions and subsequent inability to cool reactor fuel, is one of the most credible scenarios that can lead to a core damaging event.

2. Despite the fundamental importance of reliable emergency power, demand failures of emergency power systems continue to occur at a high rate. Additionally, even larger numbers of credible potential

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

Examination Outline Cross-reference:	Level	RO	SRO
295013 (APE 13) High Suppression Pool Temperature / 5 **AK2.01** (10CFR 55.41.7) Knowledge of the relationship between HIGH SUPPRESSION POOL WATER TEMPERATURE and the following systems or components: • Suppression pool cooling	Tier #	1	------
	Group #	2	------
	K/A #	295013AK2.01	
	Importance Rating	3.9	------

Proposed Question: # **3**

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

- A RCIC SR is in progress
- NUSO directed RHR Loop II Suppression Pool Cooling in service in accordance with 2-OI-74, RHR System

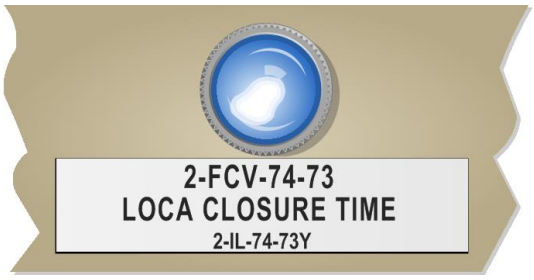
Subsequently,

- Suppression Pool Temperature reaches 100 °F

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

EOI-2, PRIMARY CONTAINMENT CONTROL, entry **(1)** required.

When throttling 2-FCV-74-73, if 2-IL-74-73Y, 2-FCV-74-73 LOCA CLOSURE TIME, blue light is **EXTINGUISHED**, and a LOCA occurs, the required LPCI injection time **(2)** be exceeded.



Note: 2-FCV-74-73, RHR SYS II SUPPRESSION POOL COOLING/TEST VALVE

A. (1) is
(2) will

B. (1) is
(2) will NOT

C. (1) is NOT
(2) will

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

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with EOI-2, Suppression Pool Temperature entry condition is 95 °F. Second part, in accordance with 2-OI-74, RHR System, P&L 3.6, if 2-FCV-74-59 (73) LOCA CLOSURE TIME light 2-IL-74-59Y (73Y) Loop I (II) on Panel 2-9-3 is **extinguished** due to its associated valve being opened, that Loop is INOPERABLE for LPCI. Additionally, this indicates that the 2-FCV-74-59 (73), RHR SYS I (II) SUPPR POOL CLG/TEST VLV is open too far and may NOT close in time to satisfy the minimum LPCI injection assumptions in UFSAR 6.4.4 LPCI.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible if the candidate reverses the logic for FCV-74-73 to conclude the LOCA CLOSURE TIME blue light is normally extinguished and illuminated to indicate the required LPCI injection time will be exceeded.

C INCORRECT: First part is incorrect but plausible in that the Suppression Pool Temperature of 110 °F (SCRAM required) could be confused with EOI-2 entry condition. Second part is correct (*See B).*

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 High Suppression Pool Water Temperature and Suppression Pool 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. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution as it relates to RHR logic.

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. 19 (Attach if not previously provided)
2-OI-74, Rev. 202
OPL171.044, Rev. 23
UFSAR 6.4.4, Rev. 27

Proposed references to be provided to applicants during examination: Blue indicating light for 2-IL-74-73Y, 2-FCV-74-73 LOCA CLOSURE TIME

Learning Objective: OPL171.044 Obj. 9 (As available)

Question Source: Bank #
Modified Bank # BFN NRC 1804 #28 (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 **X**
55.43

Comments:

Copy of Bank Question:

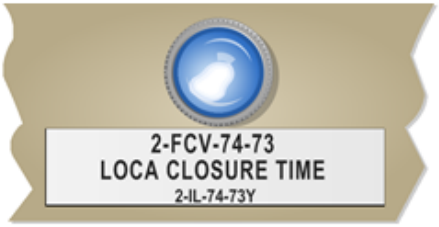
Examination Outline Cross-reference:	Level	RO	SRO
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode	Tier #	2	------
G2.2.37 (10CFR 55.41.7)	Group #	1	------
Ability to determine operability and/or availability of safety related equipment.	K/A #	203000G2.2.37	
	Importance Rating	3.6	------

Proposed Question: **# 28**

Unit 2 is operating at 100% RTP with Suppression Pool Cooling in service to support a surveillance in progress.

Which one of the following completes the statement below regarding the blue indicating light for 2-IL-74-73Y, 2-FCV-74-73, LOCA CLOSURE TIME?

When the light is ___**(1)**___, it means that IF a LOCA occurs, the required ___**(2)**___.



Note:
2-FCV-74-73, RHR SYS II SUPPR POOL CLG/TEST VLV

A. **(1)** illuminated
(2) LPCI injection time will be exceeded

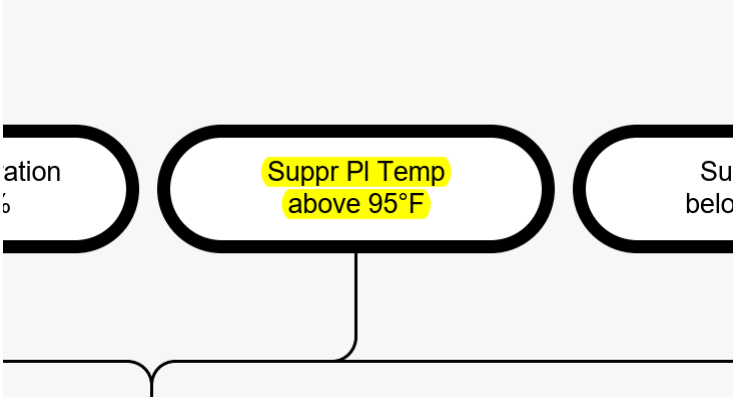
B. **(1)** illuminated
(2) 2-FCV-74-73 will close in time to minimize radioactive release

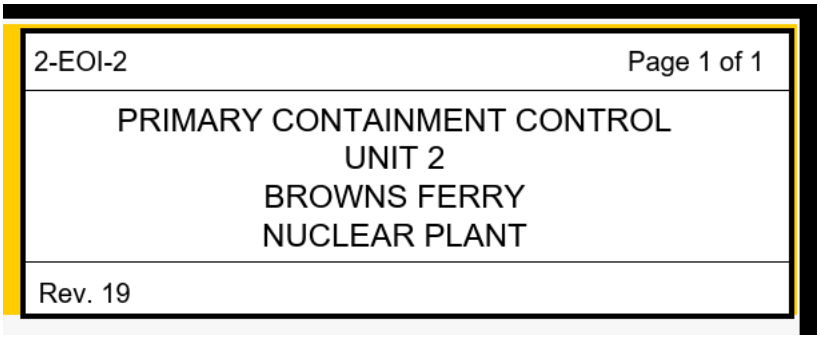
C. **(1)** extinguished
(2) LPCI injection time will be exceeded

D. **(1)** extinguished
(2) 2-FCV-74-73 will close in time to minimize radioactive release

Proposed Answer: C

Excerpts from 2-EOI-2:





Excerpts from 2-OI-74:

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0202 Page 27 of 551

3.6 Interlocks (continued)

11. If primary containment cooling is desired with reactor level at less than2/3 core height, the keylock bypass switch is required to be placed in BYPASS before the select reset switch is placed in SELECT to ensure relay logic is made up.\

12. The RHR torus spray valves, 2-FCV-74-58(72), have the same in-line valve interlocks as those outlined in 3.6A 10 for the torus spray/cooling valves. Additionally these valves have an interlock preventing opening unless drywell pressure is ≥1.96 psig which cannot be bypassed.

13. The RHR torus cooling/test valves, 2-FCV-74-59(73), receive an auto closure signal in the presence of a LPCI initiation signal. Auto closure may be bypassed by the same conditions/actions outlined in step 3.6A.10

14. The RHR containment spray valves, 2-FCV-74-60(74) and 61(75), have in-line valve interlocks similar to these described in step 3.6A.10 through 3.6A.12 for the RHR torus spray valves 2-FCV-74-57(58) and 71(72).

15. If 2-FCV-74-59(73) LOCA CLOSURE TIME light (2-IL-74-59Y Loop I; 2-IL-74-73Y Loop II) on Panel 2-9-3 is extinguished due to its associated valve being opened, that Loop is inoperable for LPCI.

16. If 2-HS-74-148(149) RHR SYSTEM I (II) MIN FLOW INHIBIT switch is in the INHIBIT position, the pumps on that loop do not have automatic minimum flow protection.

17. If RHR pumps 2A(B) and 2C(D) SD COOLING SUCT VLV, 2-FCV-74-2(25) AND 13(36) are not fully closed, the RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57(71) cannot be opened.

    a. 2-HS-074-0057D placed in BYPASS position, defeats the interlocks on 2-FCV-74-57 and the SD COOLING SUCT VLV 2-FCV-74-2 AND 13.

    b. 2-HS-074-0071C placed in BYPASS position, defeats the interlocks on 2-FCV-74-71 and the SD COOLING SUCT VLV 2-FCV-74-25 AND 36.

18. If RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57(71) is not fully closed, the RHR pumps 2A(B) and 2C(D) SD COOLING SUCT VLV, 2-FCV-74-2(25) AND 13(36) cannot be opened.

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0202 Page 111 of 551

8.5 Initiation of Loop I(II) Suppression Pool Cooling (continued)

CAUTIONS

1) RHR Pumps 2B & 2D Motor full load current is 247 amps.
2) The RHR pump shall be stopped immediately if motor current exceeds 247 amps for RHR Pump Motors 2B and 2D.
3) When throttling 2-FCV-74-73, RHR SYS II SUPPR POOL CLG/TEST VLV, maintain 2-IL-74-73Y, 2-FCV-74-73, LOCA CLOSURE TIME, blue indicating light LIT to maintain LPCI operability.

[10.9] **START** RHR PUMP 2B(2D) using 2-HS-74-28A(39A).

[10.10] **THROTTLE** RHR SYS II SUPPR POOL CLG/TEST VLV, 2-FCV-74-73, to maintain RHR flow within limits, as indicated on RHR SYS II CTMT FLOW, 2-FI-74-70.

RHR Pumps in Operation	1	2
Loop Flow	7,000 to 10,000 gpm & Blue light illuminated	<13,000 gpm & Blue light illuminated

[11] **IF** desired to RAISE Suppression Pool Cooling flow and only one Loop II pump is in service, **THEN**

PLACE the second Loop II RHR Pump AND Heat Exchanger in service. **REPERFORM** step 8.5[10] for the second pump.

[12] **CHECK** pump motor breaker charging spring recharged for all 4160 Volt pump motors operated in this section, as follows:

- Amber breaker spring charged light on,
- Closing spring target indicates charged.

[13] **WHEN** Suppression Pool Cooling is no longer required, **THEN**

SHUTDOWN Loop II of Suppression Pool Cooling. **REFER TO** Section 8.6.

Excerpt from UFSAR 6.4.4:

The Core Spray System is assumed filled with water for seismic analysis. It is concluded that safety design basis 10 is satisfied.

Upon signals of reactor low-low-low water level or drywell high pressure plus low reactor vessel pressure, the automatic controls turn on the core spray pumps and restore valves to the spray mode. When reactor pressure decreases, the core spray shutoff valves are signaled to open. Flow to the sparger begins when the core spray discharge pressure is sufficient to open the injection check valve located inside the drywell. Subsection 7.4, "Emergency Core Cooling System Controls and Instrumentation," contains further details and evaluation.

For Units 1 and 2, a Core Spray signal will initiate the ECCS preferred pump logic to trip any running RHR and Core Spray pumps in the opposite unit. In the event of a LOCA in conjunction with a spurious accident signal from the opposite unit, the ECCS preferred pump logic will dedicate the Division I pumps (1A and 1C) to Unit 1 and the Division II pumps (2B and 2D) to Unit 2. This will ensure that the shared Unit 1/2 4KV shutdown boards are not overloaded in the event of a real and spurious accident signal.

6.4.4 Low Pressure Coolant Injection System

In case of low-low-low water level in the reactor or high pressure in the containment drywell plus low reactor vessel pressure, the Low Pressure Coolant Injection System (LPCI) mode of operation of the Residual Heat Removal System pumps water into the reactor vessel in time to flood the core and limit fuel clad temperature. (The Core Spray System starts from the same signals and operates independently to achieve the same objective.)

The LPCI has been modified for Units 1, 2, and 3. This modification has been described in reports entitled "Browns Ferry Nuclear Plant Units 1 and 2 Emergency Core Cooling Systems Low Pressure Coolant Injection Modifications for Performance Improvement" (October, 1977) for Units 1 and 2 and for Unit 3. The reports were submitted to the Nuclear Regulatory Commission.

The LPCI system modifications to BFNP Units 1, 2, and 3 provide a significant reduction in peak fuel clad temperature following a postulated recirculation line break. This, in turn, results in increased MAPLHGR limits. The modifications eliminate the LPCI recirculation loop selection logic and permit simultaneous injection into both recirculation loops. Additional resistance in the form of orifices in the pump discharge line was added to prevent the RHR pumps from going to a full runout flow in the event the RHR pumps are discharging to a broken recirculation loop. The orifices (and resultant increased NPSH) assure the availability of the RHR pumps for long-term cooling. For the modified design, the LPCI injection valves and RHR cross-tie valve are normally closed. (The Unit 1 LPCI loop cross-tie valve is removed; and the corresponding cross connection is removed by a combination of a

Excerpts from OPL171.044 Lesson Plan:

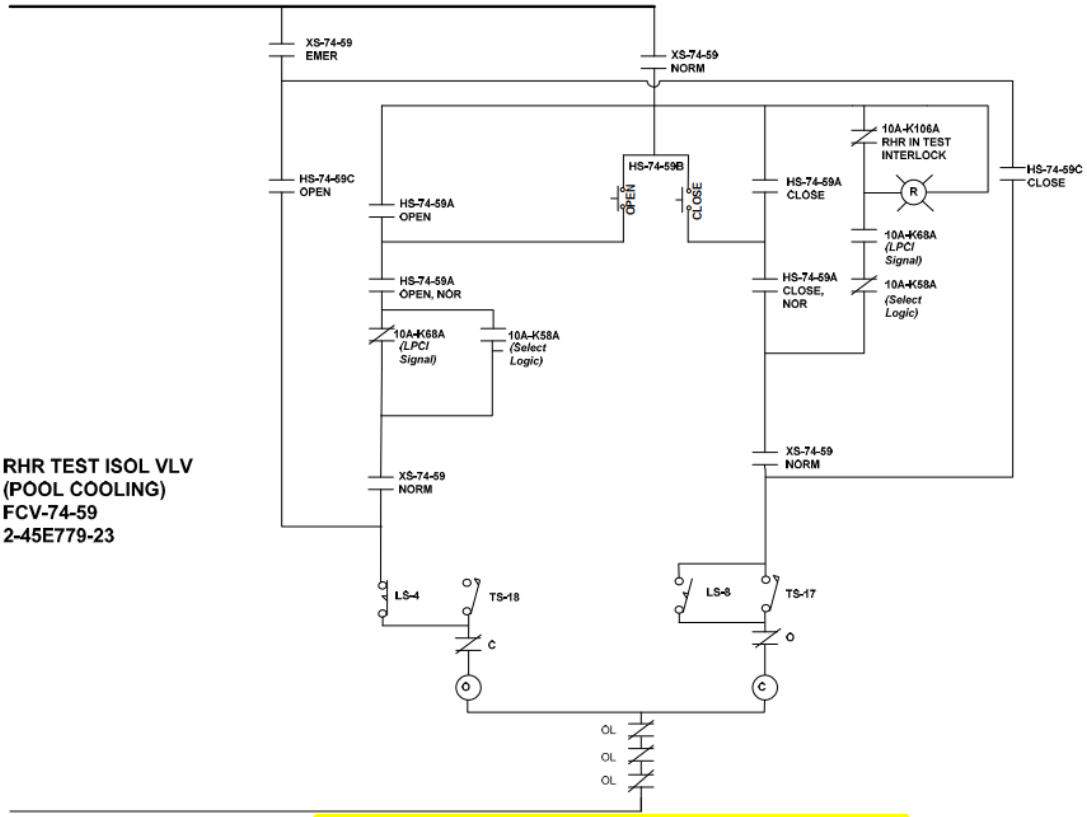


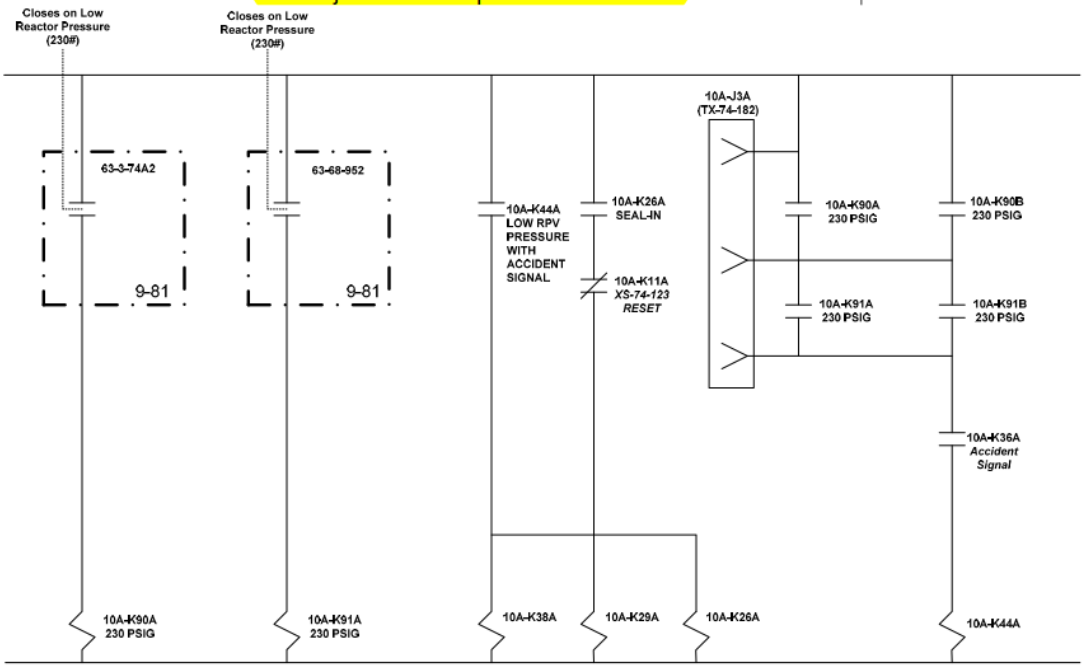
Figure 49 – RHR Suppression Pool Cooling Valve Control Circuit

v) RHR Test Isolation Valves (FCV-74-59/74-73)
   (1) No automatic opening interlock
   (2) No in-line valve interlock
   (3) Automatically closes on a LPCI signal unless select logic is aligned (10A-K68A/B, K58A/B)
   (4) The in-line valve interlock and/or the automatic closure signal (LPCI) can be bypassed if the following exists:
      (a) RPV Level >-183 inches PAL (K14A/B) and
      (b) LPCI initiation signal present (K61A/B) and
      (c) Select/reset switch taken to SELECT (XS-74-121/129)
   (5) Reactor level and LPCI initiation signals may be bypassed using a keylock bypass switch (XS-74-122/130)
   (6) Valve is not a seal-in valve (throttle) AND must be held to operate.

OPL171.044, Residual Heat Removal (RHR) System, Revision 23

(7) Emergency position at breaker bypasses in-line valve interlock and LPCI closure signal
  (a) Valve can only be operated at the breaker under this condition
(8) Blue indicating light (IL-74-59Y/73Y) is adjusted to operate at an intermediate position.
  (a) The closing time from that position shall not exceed 49 seconds
  (b) Light will extinguish when open past that point
  (c) When extinguished, indicates that the valve is open too far and may not close in time to satisfy the minimum LPCI injection assumptions in the FSAR.

There is no NORMAL / EMERGENCY switch for 1-FCV-74-59, 2-FCV-74-73, and 3-FCV-74-73.



2-730E937-6
RECIRCULATION PUMP DISCHARGE VALVE AUTO CLOSURE LOGIC

Figure 50 – Recirculation Pump Discharge Valve Automatic Closure Logic

OPL171.044, Residual Heat Removal (RHR) System, Revision 23

3. Status Indications and Controls
   a) RHR SYS I AUTO-INIT LOCKOUT – amber
   b) RHR SYS II AUTO-INIT LOCKOUT – amber
   c) Unit 1 RHR SYS I INITIATION – amber
      (1) Unit 2 due to shared power supplies
      (2) Alerts operator that availability of RHR/CS could be affected if accident signal is received
      (3) 1-IL-10AK74A
      (4) 1-IL-10AK74B
   d) FCV-74-59 LOCA CLOSURE TIME – blue
      (1) Light is normally energized
      (2) Light de-energized indicates that valve is too far open and may not close in time to satisfy the minimum LPCI injection assumptions in the FSAR.
   e) CTMT SPRAY/CLG VLV INTLK LOGIC A/B – amber
   f) CTMT SPRAY/CLG VLV INTLK LOGIC A/B RESET pushbutton
   g) XS-74-121 in SELECT – amber
   h) XS-74-122 in SELECT – amber
   i) RHR SYS I/II CTMT SPRAY/CLG VLV SELECT switch
      (1) System I – XS-74-121
      (2) System II – XS-74-122
      (3) Used to bypass the CAS closure signal and in-line valve interlock on the Containment Cooling valves
      (4) Used to allow operation of Drywell and/or Suppression Pool spray valves (bypasses in-line valve interlock) when CAS signal is not present (keylock also used).
   j) XS-74-122/XS-74-130 in OVERRIDE
      (1) Amber light ON indicates that the keylock switch is in the MANUAL OVERRIDE position
      (2) Bypasses the 2/3 core height and CAS signals to allow operation of the containment cooling valves
   k) RHR SYS I/II SD CLG INBD INJECT ISOL
      (1) IL-74-126/132
      (2) Indicates that closure signal is present on LPCI Inboard Injection Valve – requires manual reset
   l) LPCI AUTO-INIT LOGIC A/B SEAL-IN – amber
      (1) IL-74-151/152
      (2) Indicates a LPCI initiation signal has been generated and has not cleared
      (3) Seal-in reset using pushbutton
   m) RECIRC DISCH VALVES ISOL LOGIC A/B reset pushbuttons

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

Examination Outline Cross-reference:	Level	RO	SRO
295005 (APE 5) Main Turbine Generator Trip / 3 **AK3.05** (10CFR 55.41.5)	Tier #	1	------
Knowledge of the reasons for the following responses or actions as they apply to MAIN TURBINE GENERATOR TRIP: • Extraction steam/moisture separator isolations	Group #	1	------
	K/A #	295005AK3.05	
	Importance Rating	2.8	------

Proposed Question: **# 4**

Unit 2 is operating at 21% RTP, when the Main Turbine trips.

Which **ONE** of the following completes the statements below?

Following a Main Turbine trip, the Extraction Non-Return Valves close in order to prevent Main Turbine **(1)**. As a result of this transient, the Reactor **(2)** SCRAM.

A. (1) overspeed
(2) will

B. (1) overspeed
(2) will NOT

C. (1) overheating
(2) will

D. (1) overheating
(2) will NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that when the Main Turbine trips while online, a Reactor SCRAM is probable based on Main Turbine loading. However, the given Reactor Power level of 21% is within the Turbine Bypass Valve capacity to sufficiently control Reactor Pressure, which will prevent the Reactor SCRAM under the given conditions.

B **CORRECT:** (*See attached*) In accordance with 2-AOI-47-1, Unplanned Turbine Trip Below 26% Reactor Power (Without Reactor Scram) Automatic Actions, the Extraction Non-Return Valves close when the Main Turbine is tripped. This protects the Main Turbine from an overspeed condition due to the isolation of Extraction Steam. Second part, the Turbine Bypass Valves have the capacity to sufficiently control Reactor Pressure to pass the steam flow for 21% Reactor Power, which will prevent a SCRAM at the given Reactor Power Level.

C INCORRECT: First part is incorrect but plausible in that at low steam loads/flows, turbine blade heating occurs which may possibly cause turbine blade expansion and damage; however the Extraction Non-Return valve is closed to prevent an overspeed condition. 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 Extraction Steam isolates on a Main Turbine Trip and the effect on Reactor Operation. 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.

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-47-1, Rev. 18 (Attach if not previously provided)
2-OI-47, Rev. 209
OPL171.010, Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.095 Obj. 2, 7a (As available)
OPL171.010 Obj. 8

Question Source: Bank # BFN NRC 2104 #17
Modified Bank # (Note changes or attach parent)
Question History: New
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:

Proposed Question: **# 17**

Unit 2 is operating at 21% RTP, when the Main Turbine trips.

Which **ONE** of the following completes the statements below?

Following a Main Turbine trip, the Extraction Non-Return Valves close in order to prevent Main Turbine __**(1)**__. As a result of this transient, the Reactor__**(2)**__ SCRAM.

A. (1) overspeed
(2) will

B. (1) overspeed
(2) will NOT

C. (1) overheating
(2) will

D. (1) overheating
(2) will NOT

Proposed Answer: **B**

Excerpt from 2-OI-47:

6.8 Control Valve Tightness Test

NOTES

1) This test is performed from Panel 2-9-7 unless specifically stated otherwise.
2) Turbine speed must lower to less than 900 RPM for this test to be considered satisfactorily completed. However valves are designed to bring turbine to a standstill.
3) If this test is unsatisfactorily completed, turbine should be shut down until repairs are completed.
4) This test should be performed once per operating cycle.

CAUTION

DO NOT perform this test unless reactor power is less than 21.3% to ensure the capacity of the Bypass Valves is sufficient to control reactor pressure.

[1] **ENSURE** the following initial conditions are satisfied:

- Turbine-Generator is at rated speed off line. (Breaker 224 open)
- Generator field breaker is open.
- CONDENSER A, B OR C VACUUM LOW 2-PA-47-125 (2-XA-55-7B, Window 17) alarm is reset.

NOTE

The intent of the next initial condition is that the Turbine-Generator has not been less than 160 MWe for greater than 90 minutes.

- Turbine-Generator has operated continuously for at least one-half hour at greater than 160 MWe, in the past 2 hours, to properly warm the turbine.

[2] **REVIEW** all Precautions and Limitations Section 3.0.

Excerpt from 2-AOI-47-1:

BFN Unit 2	Unplanned Turbine Trip Below 26% Reactor Power (Without Reactor Scram)	2-AOI-47-1 Rev. 0018 Page 5 of 10

3.0 AUTOMATIC ACTIONS

A. Turbine stop valves close.

B. Turbine control valves close.

C. Turbine combined intermediate valves close.

D. Extraction non-return valves close.

E. Turbine bypass valves open to maintain reactor pressure.

F. The generator air circuit breaker opens.

G. Exciter field breaker opens.

H. Generator load set runs back to minimum.

I. Generator voltage regulator transfers to manual.

Excerpt from OPL171.010 Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes (Optional)
d. When the intermediate stop valve is fully closed, the operator releases the pushbutton. e. The intermediate stop valve fully opens and then the intercept valve fully opens.	
E. Extraction Non-return Valves	
1. Purpose	
To protect the turbine from an over speed condition, this might occur when the turbine is tripped. A subsequent lowering of pressure in the turbine and heaters, due to vacuum in the condenser will cause hot water from the heater to flash to steam. The reverse steam flow back through the extraction steam piping to the Main Turbine could cause blade damage.	Figure-6 ILT-8 LOR-3 NLO-25
2. Construction	
Ordinary check valve with an air-operated plunger is used to cause a positive closing action to the valve. Under normal conditions, the plunger is lifted out of the way.	Why is there none on 4th/5th Heaters?
3 Quantity - 5 Total	Low energy steam will not cause damage
a. One valve on the common extraction line to the #1 heaters (one valve per three heaters)	
b. One valve on the common extraction line to the #2 heaters (one valve per three heaters)	
c. One valve on each individual extraction line to the #3 heaters (one valve per heater)	
4. Operation	
When the turbine trips, air is bled from the air cylinder forcing the disc to fall partially down into the extraction flow path. If reverse steam flow occurs, the disc slams shut.	
F. Turbine Bypass Valves (BPV) (Nos. 1 through 9)	

Lesson Plan Content

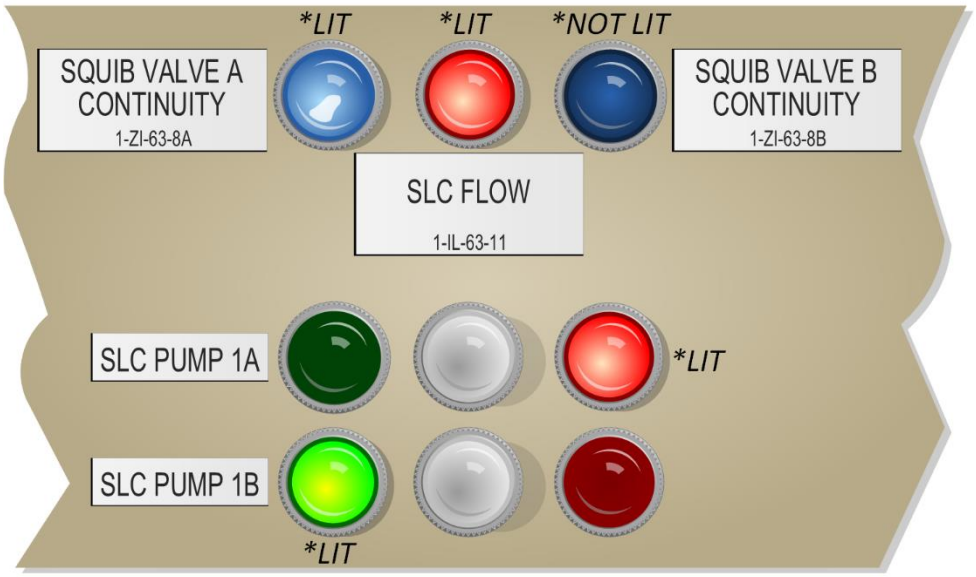
Outline of Instruction	Instructor Notes (Optional)
(3) Six-flow: Number of exhaust flow paths to the main condenser	
e. Non-reheat: Steam is not reheated before returning to the LP turbines.	
3. Steam Flow Path	ILT-2,NLOR-2 NLO-2, ILT-23, LOR-11
a. Four main steam lines (24 inches)	
b. 8-inch lines to Bypass Valves from mixing header (9 valves ~21% design capacity) 6-inch lines tapping off of the mixing header supply going to:	Figure-1
(1) Off-gas pre-heaters	
(2) Reactor feedwater pump turbine high-pressure steam supply	
(3) Seal steam regulators	
(4) Steam jet air ejector regulators	
c. Through four Main Turbine Stop Valves (TSV)	
d. Steam chest (that area below the seats of the TSVs to the TCV seats)	
e. Through the turbine control valves (TCV)	
Four turbine control valves (24 inches)	
g. Through the high-pressure turbine	
h. Extraction steam for the number 1 heaters comes off the exhaust of the high-pressure turbine prior to the moisture separators.	
i. Steam exhausts to six moisture separators. Dried steam from the B1 and C1 (U2 and U3) crossover lines supplies reactor feedwater pump turbine low-pressure steam.	Unit difference U1 (B2 and C2)
j. Dried steam is admitted to the low-pressure turbine through 6 Combined Intermediate Valves (CIVs).	

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

Examination Outline Cross-reference:	Level	RO	SRO
211000 (SF1 SLCS) Standby Liquid Control **A1.01** (10CFR 55.41.5) Ability to predict and/or monitor changes in parameters associated with operation of the STANDBY LIQUID CONTROL SYSTEM including: • Tank level	Tier #	2	------
	Group #	1	------
	K/A #	211000A1.01	
	Importance Rating	3.8	------

Proposed Question: **# 5**

Unit 1 has suffered an Anticipated Transient Without SCRAM (ATWS) with the following indications:



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

SLC Squib Valve 'A' ___**(1)**___ OPEN, and **COLD** Shutdown Boron Weight (CSBW) is reached when SLC injection has lowered the SLC Tank Level by ___**(2)**___.

A. (1) is
(2) 30%

B. (1) is
(2) 12%

C. (1) is not
(2) 30%

D. (1) is not
(2) 12%

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate confuses Squib Valve firing indications, i.e., lit versus not lit. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that Hot Shutdown Boron Weight (HSBW) will be reached first when SLC tank level lowers 12% from initial tank level.

C **CORRECT**: (*See attached*) In accordance with 1-EOI Appendix-3A, SLC INJECTION, Squib Valve B has fired. This is indicated by SQUIB VALVE 'B' CONTINUITY blue light being **extinguished** as shown in the provided drawing. Second part, Cold Shutdown Boron Weight (CSBW) is when SLC has injected long enough so SLC tank level has lowered by 30%.

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 and/or monitor changes in parameters associated with the redundant reactivity control design of the Standby Liquid Control (SLC) System operation as it relates to Anticipated Transient Without SCRAM (ATWS) 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 specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-EOI-1A, Rev. 6 (Attach if not previously provided)
EOIPM Section 0-V(D), Rev. 3

Proposed references to be provided to applicants during examination: 1-HS-63-6A, SLC Pump 1A/1B Pump/light indications and 1-ZI-63-8A/B, SLC Valve A/B Continuity lights and 1-IL-63-11, SLC Flow light indication

Learning Objective: OPL171.039 Obj. 5b,d (As available)

Question Source:
Bank #
Modified Bank # BFN NRC 2204 #6 (Note changes or attach parent)
New

Question History: Last NRC Exam 2022

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

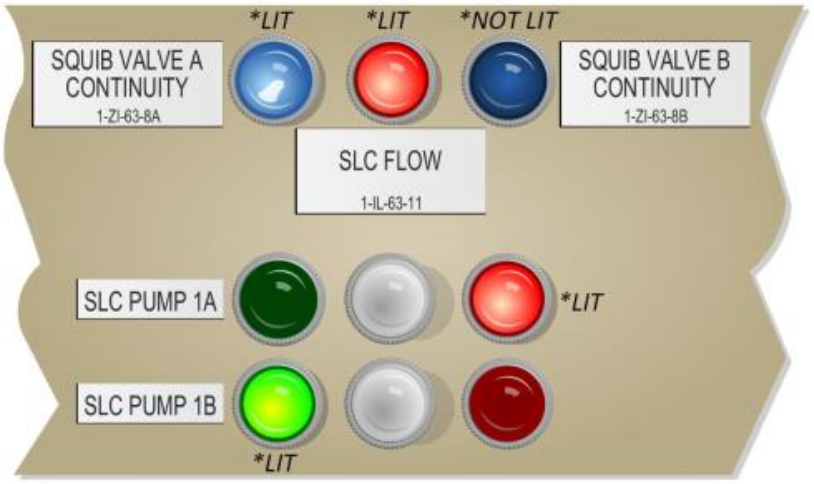
10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 6**

Unit 1 has suffered an Anticipated Transient Without SCRAM (ATWS) with the following indications:



Given the indications above, SLC Squib Valve **(1)** is open and the time to reach Hot Shutdown Boron Weight is **(2)** compared to the time with both squib valves open.

A. (1) A
(2) longer

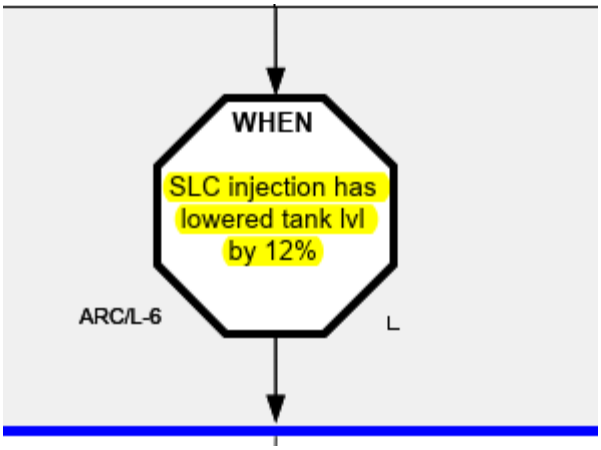
B. (1) A
(2) the same

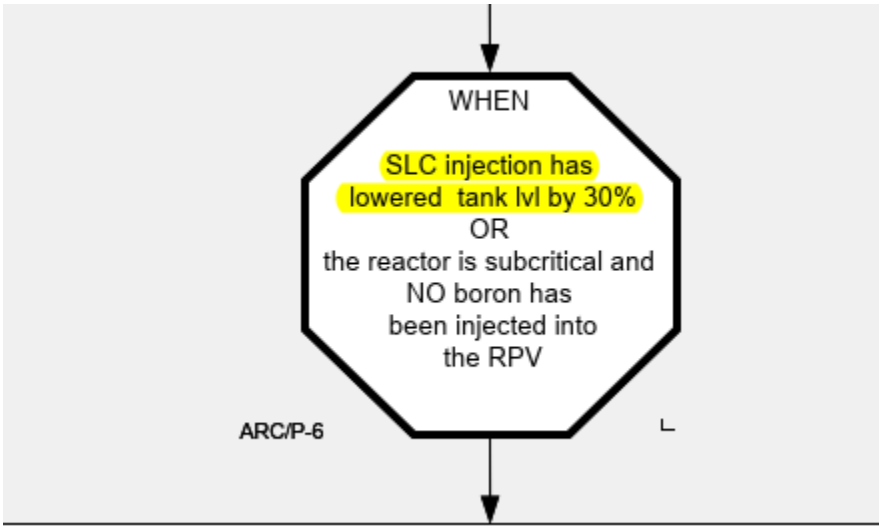
C. (1) B
(2) longer

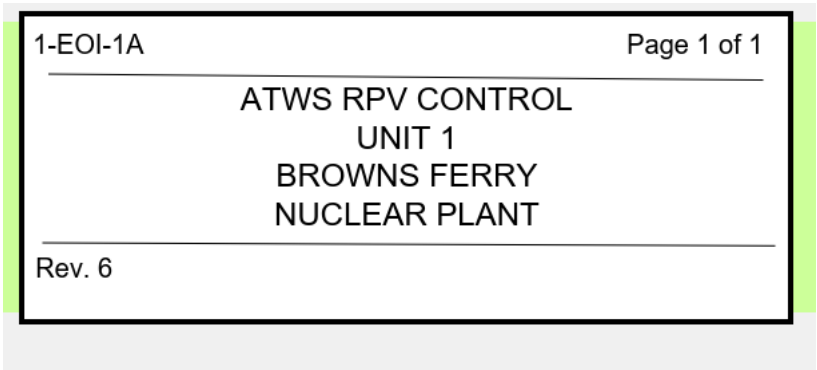
D. (1) B
(2) the same

Proposed Answer: **D**

Excerpts from 1-EOI-1A:







Excerpts from EOIPM Section 0-V(D):

BFN Unit 0	EOI-1A, ATWS RPV Control Bases	EOIPM Section 0-V(D) Rev. 0003 Page 67 of 179

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/L-6

Once the Hot Shutdown Boron Weight has been injected, Step ARC/L-8 will restore RPV water level to reestablish natural circulation flow and distribute boron that has accumulated in the lower plenum throughout the core. This strategy minimizes the integrated primary containment heatup by maintaining reactor power as low as possible during the time of boron injection.

The Hot Shutdown Boron Weight (HSBW) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions.

BFN Unit 0	EOI-1A, ATWS RPV Control Bases	EOIPM Section 0-V(D) Rev. 0003 Page 119 of 179

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/P-6

RPV depressurization and cooldown in subsequent steps may not proceed until at least one of the three conditions listed in this step is satisfied.

- Injection of the Cold Shutdown Boron Weight (CSBW) of boron into the RPV also provides adequate assurance that the reactor is and will remain shutdown. The CSBW is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. This weight is utilized to assure the reactor will remain shutdown irrespective of control rod position or RPV temperature. Refer to EOIPM Section 0-V(B) for discussion of the CSBW. If any amount of boron less than the CSBW has been injected into the RPV, cooldown is not permitted unless it can be determined that control rod insertion alone assures the reactor will remain subcritical under all conditions. The core reactivity response from cooldown in a partially borated core is unpredictable and subsequent steps may not prescribe the correct actions for such conditions if criticality were to occur.

- SLC injects at approximately 1% per minute. Therefore, CSBW should be injected within 30 minutes of initiation.

- If no boron has been injected into the RPV, depressurization and cooldown may proceed as long as control rod insertion is sufficient to keep the reactor subcritical. Such action is permitted even though the existing margin to criticality may be small. A return to criticality under these conditions is acceptable because termination of the cooldown will stop the reactor power increase.

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

Examination Outline Cross-reference:

215003 (SF7 IRM) Intermediate-Range Monitor System
A4.04 (10CFR 55.41.7)
Ability to manually operate and/or monitor in the control room:

- IRM back panel switches, meters, and indicating lights

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	215003A4.04	
Importance Rating	3.2	------

Proposed Question: **# 6**

Unit 2 is in MODE 2 with the following conditions:

- A Reactor startup is in progress
- Reactor Power is being monitored using IRMs in the Main Control Room
- IRM 'B' and 'H' are currently on Range 7 reading as indicated



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

As shown, the **GIVEN** IRM indication is on Panel **(1)**.

In accordance with 2-OI-92A, Intermediate Range Monitors, IRMs 'B' and 'H' are assigned to RPS trip system(s) **(2)**.

A. (1) 2-9-12 **ONLY**
(2) 'A' **AND** 'B'

B. (1) 2-9-12 **ONLY**
(2) 'B' **ONLY**

C. (1) 2-9-5 **AND** 2-9-12
(2) 'A' AND 'B'

D. (1) 2-9-5 **AND** 2-9-12
(2) 'B' **ONLY**

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that IRMs are separated into two sections for RPS 'A' and 'B', respectively. RPS trip system 'A' has IRM A, C, E, G, and RPS trip system 'B' has IRM B, D, F, H.

B **CORRECT**: (*See attached*) In accordance with 2-OI-92A, the IRM drawers are located on Panel 2-9-12 in the Main Control. The given IRM indication is located within each respective IRM drawer on Panel 2-9-12. Second part, in accordance with 2-OI-92A, Intermediate Range Monitors, IRMs are separated into two sections for RPS 'A' and 'B' respectively. IRMs 'B' and 'H' are assigned to RPS trip system 'B'.

C INCORRECT: First part is incorrect but plausible in that IRMs, whether startup, withdrawing, bypass, or inserting, all operations are performed on Panel 2-9-5 with the digital indication, unless specifically stated otherwise. 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 manually operate and/or monitor IRM back panel switches, meters, and indicating lights in the Main Control Room. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to IRM operations.

Technical Reference(s): 2-OI-92A, Rev. 29 (Attach if not previously provided)
OPL171.020, Rev. 13U1

Proposed references to be provided to applicants during examination: Intermediate Range Monitor (IRM) analog indication

Learning Objective: OPL171.020 Obj. 2 (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

Excerpts from 2-OI-92A:

BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 7 of 20

3.0 PRECAUTIONS AND LIMITATIONS (continued)

H. The IRMs produce the following trip outputs to the Reactor Protection System auto-scram circuitry:

1. High-High (> 116.4 on 125 scale).

2. Inop (module unplugged, mode switch not in OPERATE, HV power supply low voltage, loss of 24 VDC power supply to IRM drawer).

3. In addition, by removing the blue shorting links (2 total links), the IRMs are placed in the non-coincident trip logic where any one channel, if tripped, will produce a full reactor scram. The 2/4 Voters are also in this logic such that a trip output from any one Voter yields a full Reactor Scram.

I. The time required to drive a detector from full out to full in is approximately 3 minutes.

J. The INOP TRIP BY-PASS switches located on the IRM drawers on Panel 9-12 by-pass the IRM switch position out-of-operate trip. These switches are to be used only during testing of the IRM channels.

K. [NRC/C] Upon return to service of 24-VDC Neutron Monitoring Battery A or B, Instrument Maintenance is required to perform functional tests on SRMs and IRMs that are powered from the affected battery board. [NRC IE Inspector Follow-up Item 86-40-03]

5.0 STARTUP

5.1 Preparing Intermediate Range Monitors for Reactor Startup

NOTE

All operations are performed on Panel 2-9-5 unless specifically stated otherwise.

[1] **VERIFY** the Intermediate Range Monitor System is in standby readiness. **REFER TO** Section 4.0 ☐

[2] **REVIEW** precautions and limitations in Section 3.0. ☐

[3] **CHECK** that no more than one IRM Channel per trip system is bypassed. ☐

[4] **VERIFY** all operable IRM drives are working properly as follows:

[4.1] **DEPRESS** the SRM/IRM DETECTOR POSITION pushbutton and **OBSERVE** the pushbutton illuminates. ☐

[4.2] **DEPRESS** the IN OUT CH (A) IRM SELECT pushbutton and **OBSERVE** the CH (A) IRM SELECT and IN backlights illuminate. ☐

[4.3] **DEPRESS** and **HOLD** the DRIVE OUT pushbutton and **OBSERVE** the (IRM A) IN backlight extinguishes. ☐

[4.4] **WHEN** the (IRM A) OUT backlight illuminates, **THEN**

RELEASE the DRIVE OUT pushbutton. ☐

6.0 SYSTEM OPERATIONS

6.1 Bypassing an IRM Channel

CAUTION

[QA/QC] NPG-SPP 10.4 requires approval of the Plant Manager or his designee prior to any planned operation with IRMs bypassed unless bypassing is specifically allowed within approved procedures. [ISE-NPS-92-RO1]

NOTES

1) It is not necessary for a bypassed IRM channel to have its detector inserted into the core.
2) Only one IRM in each trip system can be bypassed at a time.
3) All operations are performed on Panel 2-9-5 unless specifically stated otherwise.

[1] **REVIEW** all precautions and limitations in Section 3.0. ☐

[2] **PLACE** the appropriate IRM Bypass selector switch to the BYPASS position:

- IRM BYPASS, 2-HS-92-7A/S4A ☐
- IRM BYPASS, 2-HS-92-7A/S4B ☐

[3] **CHECK** that the Bypassed light is illuminated. ☐

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. ☐

BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 17 of 20

NOTES

1) All IRM rod block trips are automatically bypassed when the reactor mode switch is in the RUN position.
2) All IRM scram trips are automatically bypassed when the reactor mode switch is in RUN.
3) More than one IRM detector may be inserted at a time if needed.
4) All IRM's should be inserted simultaneously following a Reactor Scram with all Control Rods Fully Inserted per 2-AOI-100-1 or as directed by the EOI's..
5) All operations are performed on Panel 2-9-5 unless specifically stated otherwise.
6) 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.

6.4 Inserting IRMs

[1] **SELECT** the desired IRM(s) drives by:

PERFORMING the following: (Refer to notes above.)

[1.1] **DEPRESS** SRM/IRM DETECTOR POSITION, 2-HS-92-7C/S1 pushbutton.

AND

OBSERVE the background light ILLUMINATES. ☐

[1.2] **DEPRESS** the appropriate IRM SELECT pushbutton for the IRMs being inserted. (Refer to notes above.)

AND

OBSERVE the background light ILLUMINATES. ☐

[1.3] **DEPRESS** the DRIVE IN, 2-HS-92-7C/S2 pushbutton.

AND

OBSERVE the background light ILLUMINATES. ☐

Excerpts from OPL171.020 Lesson Plan:



Panel 9-12 **Panel 9-5**

Image 3 - Panel Indications

13. Trip Circuits
    a) The trip circuits used with the IRMs are identical to the trip circuits used by the SRMs with the exception of setpoints.
    b) There are two dual trip units in each IRM drawer
    c) Each trip unit contains two independent trip circuits to provide alarms, and in some cases, protective interlocks (rod block or scram)

OPL171.020 , Intermediate Range Monitor System, Rev# 13U1

D. Instrumentation
1. Control Room Instrumentation
   a) Dual-Input Recorders on Panel 9-5 (4)
   b) 8 Meters on Panel 9-12
      (1) Use 0-125 scale only
2. Annunciators and alarm indication
   a) Annunciators
      (1) 9-5A Window 7 - Control Rod Withdrawal Block
         (a) IRM is one of many inputs into this alarm
         (b) Action D.4 directs that when the REACTOR MODE SWITCH is in START-UP position, then CHECK IRM detectors NOT FULL IN.
            (i) Only IRM control rod block that does not have another alarm associated with it.
      (2) 9-5A Window 19 - IRM DOWNSCALE
         (a) ≤7.5 on 125 scale
         (b) Results in rod withdrawal block if IRM range switch is NOT in position 1 or REACTOR MODE SWITCH in RUN.
         (c) Directs first ranging the IRM, and if ranging the IRM does not reset the alarm, then bypassing the associated IRM.
      (3) 9-5A Window 26 - IRM HIGH
         (a) ≥ 90 on 125 scale (Unit 1, 2, an 3)
         (b) Results in control rod block except with REACTOR MODE SWITCH in RUN.
         (c) Directs either ranging or bypassing the associated IRM.
      (4) 9-5A Window 33(34) - IRM CH A, C, E, G (B, D, F, H) HI-HI/INOP
         (a) ≥ 116.4 on 125 scale OR
         (b) INOP signal
            (i) High voltage low
            (ii) Module Unplugged
            (iii) Function switch not in OPERATE
            (iv) Loss of ± 24 VDC to monitor
         (c) Results in control rod block and 1/2 scram on 'A'(B) RPS channel (except REACTOR MODE SWITCH in RUN).
         (d) Directs either ranging or bypassing the associated IRM channel.
         (e) Directs resetting the half-scram signal.
   b) Indicating Lights
      (1) Each of the IRM channels has a set of 4 lights above the IRM range switches.

Examination Outline Cross-reference:

295018 (APE 18) Partial or Complete Loss of CCW / 8

AK3.04 (10CFR 55.41.5)

Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW):

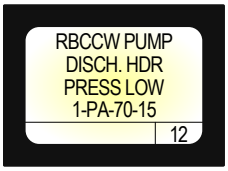
- Starting the standby pump

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295018AK3.04	
Importance Rating	3.5	------

Proposed Question: **# 7**

Unit 1 is operating at 100% RTP when the following conditions occur:

- RBCCW PUMP DISCHARGE HEADER PRESSURE LOW (1-9-4C, Window 12) alarms
- RBCCW Header Pressure is 50 psig
- **NO** Operator actions have been taken



Given the conditions above and in accordance with the given ARP, the spare RBCCW Pump can be started from _____.

A. Unit 1 **ONLY**

B. Unit 2 **ONLY**

C. Unit 1 and 2 **ONLY**

D. Unit 3 **ONLY**

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached)* In accordance with the given ARP, whether Unit 1, 2 or 3, it directs the Operator to start the spare RBCCW Pump using each Unit's respective AOI-70-1, Loss of RBCCW, and OI-70, RBCCW. Each Unit's RBCCW System consists of two RBCCW Pumps, however there is only one spare RBCCW Pump shared among all three Units. All applicable ARPs, OIs, AOIs will direct Operators back to Unit 1 since only Unit 1 can start the spare RBCCW Pump from Panel 1-9-4.

B INCORRECT: Incorrect but plausible in that the spare RBCCW Pump is normally aligned for immediate use on Unit 2, but still requires the Unit 1 Operator assistance from Panel 1-9-4.

C INCORRECT: Incorrect but plausible in that the CRD System has a spare CRD Pump 1B that is shared/controlled between Unit 1 and 2 only.

D INCORRECT: Incorrect but plausible in that the Unit 3 CRD System has a spare CRD Pump 3B that is for Unit 3 operation only.

RO Level Justification: Tests the candidate's knowledge of the procedural requirements for the operation of the spare RBCCW Pump as it relates to Unit 1, 2, and 3 operations and controls. 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): OPL171.047, Rev. 14U1 (Attach if not previously provided)
1-ARP-9-4C, Rev. 31
1-OI-70, Rev. 64
2-OI-70, Rev. 74
1-AOI-70-1, Rev. 17
OPL171.005, Rev. 24U3

Proposed references to be provided to applicants during examination: RBCCW PUMP DISCHARGE HEADER PRESSURE LOW (1-9-4C, Window 12)

Learning Objective: OPL171.047 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 **X**
Comprehension or Analysis

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

Comments:

Excerpts 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
3. Annunciation	
a Unit 3 annunciation 9-4 C "DRYWELL OUTAGE CHILLER SYSTEM ABNORMAL" alarm annunciates on any of the following conditions:	
(1) Chilled water low flow	
(2) Chilled water pump fault	
(a) electrical overload	
(b) breaker trip	
(3) Chiller fault alarms (established by the vendor)	
(a) Water temperature low	
(b) Ambient air temperature high/low	
(c) VAC undervoltage	
(d) Compressor current high/low	
(e) Compressor suction pressure low	
(f) Compressor oil pressure low	
(g) Compressor discharge pressure high	
(h) Pump down incomplete	
G. Unit differences	
1. The RBCCW systems are the same for all three units.	
2. The difference is in the control room panels	Obj. ILT 7
3. Each unit RBCCW system has two pumps and two heat exchangers. A third pump and heat exchanger serves as a spare for all three units.	Obj. LOR 3 Obj. NLO/NLOR 8
4. There are a total of six MOV's associated with the spare pump (two valves per unit). The valves are located on the suction and discharge of the spare pump. All of these MOV's are controlled from Unit 1 panel 9-4.	

OPL171.047, Reactor Building Closed Cooling Water System, Rev. 14

Lesson Plan Content

Outline of Instruction			Instructor Notes and Methods
	5.	Unit 2 and Unit 3 have installed four isolation valves on the RBCCW penetrations to and from the drywell. The supply and return lines each have two isolation valves. The reason for this installation is to provide for leak rate testing on the FCV-70-47 and the inlet check valve. The Unit 2 valves have motors but are not connected to a power source. The motors may later be used for a PCIS isolation added to the RBCCW system.	
	6.	The spare pump is aligned by OI-70 to be immediately available for use on Unit 2. All actions to place the spare pump in service on Unit 2 are accomplished from Unit 1 panel 9-4.	
	7.	On Unit 2, Rotork operator installed but not electrically connected. Operator installed as contingency if automatic isolation required by the NRC. Issue resolved with automatic isolation not required. On unit 3, Manual hand crank with gear reducer operator installed.	
	8.	Unit 3 480V Load Shed does not send a close signal to FCV-70-48, however Unit 1 and 2 480V Load Shed does.	
	9.	On Unit 3, the RWCU outlet temperature controls the RBCCW flow through the Non-Regenerative Heat exchanger	
	10.	"Drywell Outage Chiller System Abnormal" exist only on Unit 3. See 3-OI-70, ILL 2 and 3-ARP-9-4C-W7.	
H.		System interrelationships	
	1.	See II.B.2 for system loads. A loss of RBCCW to any of these loads would result in a rise of components temperatures. Manual/auto shutdown of the component may be required to prevent damage due to heating.	Obj. ILT 7 Obj. LOR 3 Obj. NLO/NLOR 9
	2.	Raw cooling water system is the normal cooling medium to the RBCCW heat exchangers. A loss of RCW results in EECW supplying the cooling medium as previously discussed.	

Excerpt from 1-ARP-9-4C:

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

Excerpt from 1-AOI-70-1:

BFN Unit 1	Loss of Reactor Building Closed Cooling Water	1-AOI-70-1 Rev. 0017 Page 6 of 14

4.2 Subsequent Actions (continued)

[1] **IF** Reactor is at power **AND** Drywell Cooling cannot be immediately restored, **THEN**

PERFORM the following (otherwise N/A):

[1.1] **IF** core flow is above 60%, **THEN**

REDUCE core flow to between 50-60%.

[1.2] **MANUALLY SCRAM** the Reactor and **PLACE** Mode Switch in SHUTDOWN. (**REFER TO** 1-AOI-100-1)

[1.3] **SHUT DOWN** both Recirc Pumps as follows:

- **DEPRESS** RECIRC DRIVE 1A SHUTDOWN, 1-HS-96-19.
- **DEPRESS** RECIRC DRIVE 1B SHUTDOWN, 1-HS-96-20.

[1.4] **INITIATE** a 90°F/HR cooldown rate and **REFER TO** 1-AOI-100-1.

[2] **IF** any EOI entry condition is met, **THEN**

ENTER appropriate EOI(s) (otherwise N/A).

[3] **IF** RBCCW Pump(s) has tripped and it is desired to restart the tripped RBCCW pump, **THEN**

PERFORM the following (otherwise N/A):

[3.1] **INSPECT** the tripped RBCCW pump and its associated breaker for any damage or abnormal conditions.

[3.2] **IF** no damage or abnormal conditions are found, **THEN**

ATTEMPT to restart tripped RBCCW pump(s).

[4] **IF** unable to restart a tripped pump, **THEN**

PLACE Spare RBCCW Pump in service in accordance with 1-OI-70. (otherwise N/A).

Excerpt from 1-OI-70:

BFN Unit 1	Reactor Building Closed Cooling Water System	1-OI-70 Rev. 0064 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.

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]

BFN Unit 1	Reactor Building Closed Cooling Water System	1-OI-70 Rev. 0064 Page 30 of 86

8.2 Placing Spare Pump in Service

CAUTION

1) Damage to the RBCCW pumps or RBCCW heat exchangers may occur if the number of RBCCW pumps in operation is not equal to the number of RBCCW heat exchangers in operation.
2) Valving in the Spare RBCCW Pump may affect RBCCW Surge Tank level.

[1] **ENSURE** RBCCW System in operation.

[2] **PLACE** the Spare RBCCW Pump in service to support Unit 1 RBCCW System as follows:

[2.1] **NOTIFY** Units 2 and 3 that the Spare RBCCW Pump will be placed in service to support Unit 1.

[2.2] **ENSURE CLOSED** the following valves:

- 0-FCV-70-67, using SPARE PUMP SUCT VLV U-2, 0-HS-70-67A
- 0-FCV-70-68, using SPARE PUMP DISCH VLV U-2, 0-HS-70-68A
- 0-FCV-70-69, using SPARE PUMP SUCT VLV U-3, 0-HS-70-69A
- 0-FCV-70-70, using SPARE PUMP DISCH VLV U-3, 0-HS-70-70A

[2.3] **OPEN** RBCCW SP PMP SUCT VLV U1, 0-FCV-70-13.

[2.4] **IF** time allows, **THEN**

VENT the Spare RBCCW Pump using SPARE PUMP CASE VT, 0-70-615 until a solid steam of water is discharged.

[2.5] **START** RBCCW SPARE PUMP, using 0-HS-70-11A.

[2.6] **OPEN** RBCCW SP PMP DISCH VLV U1, 0-FCV-70-14 using 0-HS-70-14A on Panel 1-9-4.

8.2 Placing Spare Pump in Service (continued)

[3] **PLACE** the Spare RBCCW Pump in service to support Unit 2 RBCCW System as follows:

[3.1] **NOTIFY** Unit 3 that the Spare RBCCW Pump will be placed in service to support Unit 2.

[3.2] **ENSURE CLOSED** the following valves:

- RBCCW SP PMP SUCT VLV U1, 0-FCV-70-13 using 0-HS-70-13A on Panel 1-9-4
- RBCCW SP PMP DISCH VLV U1, 0-FCV-70-14 using 0-HS-70-14A on Panel 1-9-4
- 0-FCV-70-69, using SPARE PUMP SUCT VLV U-3, 0-HS-70-69A
- 0-FCV-70-70, using SPARE PUMP DISCH VLV U-3, 0-HS-70-70A

CAUTION

Valving in the Spare RBCCW Pump may affect Unit 2 RBCCW Surge Tank level.

[3.3] **OPEN** SPARE PUMP SUCT VLV U-2, 0-FCV-70-67 using 0-HS-70-67A on Panel 1-9-4.

[3.4] **IF** time allows, **THEN**

VENT the Spare RBCCW Pump using SPARE PUMP CASE VT, 0-70-615 until a solid steam of water is discharged.

[3.5] **START** RBCCW SPARE PUMP, using 0-HS-70-11A.

[3.6] **OPEN** SPARE PUMP DISCH VLV U-2, 0-FCV-70-68using 0-HS-70-68A on Panel 1-9-4.

[3.7] **CHECK** Spare RBCCW Pump suction pressure greater than 18 psig, as indicated on RBCCW SPARE PUMP SUCT, 0-PI-070-0010 (locally).

8.2 Placing Spare Pump in Service (continued)

[3.8] **REQUEST** Unit 2 perform the following:

- **STOP** Unit 2 RBCCW PUMP A(B), using 2-HS-70-5A(8A)(Panel 2-9-4).
- **CLOSE** PUMP A(B) DISCH, 2-70-501A(B), locally.

NOTE

Spare RBCCW Pump differential pressure is obtained by subtracting Spare RBCCW Pump suction pressure, indicated on RBCCW SPARE PUMP SUCT, 0-PI-070-0010 from Spare RBCCW Pump discharge pressure, indicated on RBCCW SPARE PUMP DISCH, 0-PI-070-0012.

[3.9] **CHECK** Spare RBCCW Pump differential pressure normal, 47 to 52 psig

- **IF** differential pressure is **NOT** normal, **THEN**

  VENT the Spare RBCCW Pump using SPARE PUMP CASE VT, 0-70-615 until a solid steam of water is discharged.

- **IF** Spare RBCCW Pump differential pressure does **NOT** return to normal after venting, **THEN**

  ADJUST RBCCW differential pressure in accordance with Section 6.5.

[3.10] **NOTIFY** Unit 2 control room that the Spare RBCCW Pump is in service supplying Unit 2.

[4] **PLACE** the Spare RBCCW pump in service to support Unit 3 RBCCW system as follows:

[4.1] **NOTIFY** Unit 2 that the Spare RBCCW Pump will be placed in service to support Unit 3.

Excerpt from 2-OI-70:

BFN Unit 2	Reactor Building Closed Cooling Water System	2-OI-70 Rev. 0074 Page 27 of 74

8.2 Placing Spare Pump in Service

CAUTIONS

1) Damage to the RBCCW Pumps and/or RBCCW Heat Exchangers may occur if the number of RBCCW Pumps in operation is not equal to the number of RBCCW Heat Exchangers in operation.
2) Valving in the Spare RBCCW Pump could affect RBCCW Surge Tank level.

[1] **ENSURE** RBCCW System in operation.

[2] **REQUEST** Unit 1 to place the Spare RBCCW Pump in service to support Unit 2 operation. REFER TO 1-OI-70.

[3] **IF** the Spare RBCCW HX is inservice, **THEN**

PERFORM the following:

[3.1] While maintaining RBCCW Pump Differential between 47 and 52 psid, until SPARE HX TO UNIT 2, 0-ISV-70-633 is fully closed.

- **THROTTLE OPEN**, SPARE HX TO UNIT 1, 0-SHV-70-644 (R3-Q).
- **THROTTLE CLOSE** SPARE HX TO UNIT 2, 0-ISV-70-633 (R13-R).

[3.2] **ENSURE CLOSED** SPARE HX TO UNIT 2, 0-ISV-70-633 (R13-R).

[3.3] **CHECK** RBCCW Pump Differential between 47 and 52 psid.

Excerpt from OPL171.005: Supports Distractors C, D

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 24U3

3) CRD System Pumps
   a) Two pumps per subsystem
   b) 100 % capacity
   c) Multistage Centrifugal
   d) Electric Motor-driven
   e) Pump 1A is specific to Unit 2
   f) Pump 1B can be used for Unit 1 or Unit 2
   g) Pump 2A is assigned to Unit 2
   h) Pumps 3A and 3B assigned to Unit 3.
   i) Normal rated operation - 76 gpm at 1500 psig, plus 20 gpm to the minimum flow line
   j) Reactor scram rating - pump will be required to deliver a total flow of approximately 200 gpm at 1350 psig (maximum rating) in order to recharge the accumulators
   k) A speed increaser is provided between the motor and the pump
   l) The pump has NPSH protection in the form of a low suction pressure trip at 18" HgA and electrical protection trips.
   m) Suction strainers are provided to protect the pump seals, etc. (ALARA consideration, can be prohibited High Rad area)
   n) Each pump is provided with a manual suction and discharge isolation valve
   o) The discharge has a stop check valve to prevent reverse flow in the idle pump
   p) The spare pump has motor-operated valves in the common discharge line which directs flow either to Unit 1 or Unit 2 CRD Hydraulic System
   q) A minimum flow line is provided upstream of the discharge valve to prevent overheating of the pump if the discharge valve is advertently shut
   r) The minimum flow line is locked open and recirculates 20 gpm to the CST through an orifice when the pump is running
   s) The minimum flow line also has a stop check valve to prevent backflow through the idle pump

Each stage allows higher final discharge pressure

System pressure will be approximately equal to reactor pressure due to leakage into the RPV

Unit 3 has 2 CRD Pumps

2-EOI App. 3B, Alt. SLC injection uses 1B CRD pump from U-1 SLC tank & modified suction strainer with 1 1/2" rubber hose connection.

Q: What is a stop check valve?

A: A stop check is a check valve with a hand wheel to allow manually "locking" the valve closed.

1 or 2-FCV-85-8; no 85-8 on Unit 3

Min. flow from CRD pumps to CST goes to steel header.

Unit difference: 2A CRD pump min. flow path to RCIC suction from CST. (2-47E813-

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

Examination Outline Cross-reference:

295031 (EPE 8) Reactor Low Water Level / 2

EA2.01 (10CFR 55.41.10)

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:

- Reactor Water Level

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295031EA2.01	
Importance Rating	4.7	------

Proposed Question: **# 8**

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



- REACTOR LEVEL ABNORMAL (2-9-5A, Window 8) alarms

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

The given annunciator alarms when Reactor Water Level lowers to less than or equal to (≤) **(1)** and rises to greater than or equal to (≥) **(2)**.

A. (1) 27 inches
(2) 36 inches

B. (1) 27 inches
(2) 39 inches

C. (1) 30 inches
(2) 36 inches

D. (1) 30 inches
(2) 39 inches

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible if the candidate confuses the setpoint to be (+) 3 inches from the normal Reactor Water Level of (+) 33 inches at which the rising Reactor Water Level results in the alarm.

B **CORRECT**: *(See attached)* Normal Reactor Water Level is (+) 33 inches. The given alarm annunciates at (+/-) 6 inches rising/lowering Reactor Water Level. In accordance with the given ARP, when Reactor Water Level lowers ≤ 27 inches, the alarm annunciates. This informs the crew of malfunctions that could result in a SCRAM at (+) 2 inches if the lowering trend continues. Second part, when Reactor Water Level rises to ≥ 39 inches, the alarm annunciates. This informs the crew of malfunctions that could result in HPCI/RCIC tripping at (+) 51 inches, then the Main Turbine tripping at (+) 55 inches requiring a SCRAM if the rising trend continues.

C INCORRECT: First part is incorrect but plausible if the candidate confuses the setpoint to be (-) 3 inches from the normal Reactor Water Level of (+) 33 inches at which the lowering Reactor Water Level results in the alarm. 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 determine and interpret Reactor Water Level as it applies to lowering or rising Reactor Water Level. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to containment isolations.

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. 65 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: REACTOR LEVEL ABNORMAL (2-9-5A, Window 8)

Learning Objective: OPL171.003 Obj. 19 (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-ARP-9-5A:

REACTOR
WATER LEVEL
ABNORMAL
2-LA-3-53

8

(Page 1 of 1)

Sensor/Trip Point:
RFW Control System
2-RLY-003-0053LL ≤ 27 inches
2-RLY-003-0053LH ≥39 inches
Average of valid Narrow Range Level signals from RFW Control System. Signals originate from 2-LT-3-53, 3-60, 3-206 and 3-253.

Sensor Location:
2-RLY-003-0053LL in Panel 2-9-97 (Unit 2 Control Room)
2-RLY-003-0053LH in Panel 2-9-97 (Unit 2 Control Room)
2-LT-3-53 on Panel 25-5B (Rx Bldg El 593', R10-S)
2-LT-3-60 on Panel 25-6B(Rx Bldg El 593', R12-P)
2-LT-3-206 on Panel 2-925-426(Rx Bldg El 593', R10-T)
2-LT-3-253 on Panel 25-6D(Rx Bldg El 593', R12-R)

Probable Cause:
A. Reactor water level high or low.
B. Malfunction of sensor.

Automatic Action:
None (Reactor scram on low level at +2 in)
(Main Turbine/RFPT trip and subsequent scram if greater than or equal to 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 2-XR-3-53 recorder, 2-LI-3-53, 2-LI-3-60, 2-3-206 and 2-LI-3-253 on Panel 2-9-5.
B. **IF** alarm is valid, **THEN**
REFER TO 2-AOI-3-1 or 2-OI-3.
C. **IF** 2-LI-3-53, 2-LI-3-60, 2-LI-3-206 and 2-LI-3-253 has failed or is invalid, **THEN**
With SRO permission, **BYPASS** the affected level instrument.
REFER TO 2-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:

2-45E620-6	2-729E895-10	2-47E610-46-1
2-729E895-1	2-AOI-3-1	2-OI-3

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

Examination Outline Cross-reference:	Level	RO	SRO
295006 (APE 6) SCRAM / 1 **AK2.01** (10CFR 55.41.7) Knowledge of the relationship between SCRAM and the following systems or components: • RPS	Tier #	1	------
	Group #	1	------
	K/A #	295006AK2.01	
	Importance Rating	4.5	------

Proposed Question: **# 9**

Unit 2 startup is in progress, Reactor Power is approximately 20% with the following conditions:

- 2-HS-99-5A/S1, REACTOR MODE SWITCH, in RUN
- Main Turbine has just been paralleled with the grid
- Main Turbine First Stage Shell Pressure is 101 psig
- No IRMs are bypassed
- No APRMs are bypassed

Given the conditions above, which **ONE** of the following will result in a **FULL** Reactor SCRAM in accordance with 2-OI-99, Reactor Protection System (RPS)?

A. IRMs 2A and 2F failing UPSCALE

B. Critical Self-Test Fault on APRM #2

C. A Main Turbine Trip resulting from a Main Generator Breaker fault

D. EHC System failure resulting in Reactor Pressure rising to 1080 psig

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that in accordance with 2-OI-92A, Intermediate Range Monitors, and 2-OI-99, Reactor Protection System, IRMs are separated into RPS Trip System A/B, respectively, for channels required for an automatic SCRAM. RPS-A is IRM A/C/E/G and RPS-B is IRM B/D/F/H. A half SCRAM occurs in either RPS Trip System A or B with any 2 of 4 taken-once logic. For a full SCRAM to occur, at least one-out-two is required so IRM 2A (RPS-A) and IRM 2F (RPS-B) meet the criteria. However, with the MODE SWITCH in RUN, these automatic SCRAM signals are bypassed.

B INCORRECT: Incorrect but plausible in that in accordance with 2-OI-92B, Average Power Range Monitoring, an APRM INOP Trip signal for a Critical Self-Test Fault is when one APRM channel is detected, but no alarm or RPS output signal occurs. If **two** APRMs are detected, then RPS output signal to all four Voters results in a Full SCRAM.

C INCORRECT: Incorrect but plausible in that in accordance with 2-OI-47, Turbine-Generator System, and 2-OI-99, a Main Turbine Trip will result in an automatic SCRAM if Reactor Power was 26% and Main Turbine First Stage Shell Pressure is 118 psig/116 psig. However, this is not the case with the parameters given in the stem.

D **CORRECT**: *(See attached)* In accordance with 2-OI-99, in the event an EHC System failure occurs resulting in the loss of Reactor Pressure Control, an automatic SCRAM occurs when Pressure rises to 1073 psig.

RO Level Justification: Tests the candidate's knowledge of the relationship between RPS System SCRAM logic as it relates to multiple associated systems. 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-OI-47, Rev. 209 (Attach if not previously provided)
2-OI-92A, Rev. 29
2-OI-92B, Rev. 44
2-OI-99, Rev. 101

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.028, Obj. 4b (As available)

Question Source:

Bank #	OPL171.028-09 004 #1092
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:

1092. OPL171.028-09 004

Given the following:

- Unit 2 is performing a reactor startup.
- Unit 2 Reactor Mode Switch is in RUN.
- Reactor power is approximately 20% reactor power.
- The Main Turbine has just been paralleled with the grid.
- First stage shell pressure on the Unit 2 Main Turbine is 115 psig.

Which ONE of the following would result in a full reactor scram?

A. IRMs '2A' and '2F' failing upscale with all IRMs unbypassed.

B. A Main Turbine Trip resulting from a Main Generator Breaker fault.

C. Critical Self-Test Fault on APRM #2 with all APRMs unbypassed.

D. EHC System Failure resulting in Reactor Pressure rising to 1080 psig.

Excerpts from 2-OI-99:

Attachment 2
(Page 1 of 2)

Unit 2 Reactor Scram Initiation Signals

The following conditions initiate a Reactor Scram:

NOTE:

If any of the events listed below occur, a Reactor Scram should be automatically initiated. If an automatic scram initiating event occurs and the scram is **NOT** initiated, the operator is required to verify the condition and then manually scram the Reactor.

	Scram	Setpoint	Bypass
1.	Manual	NA	NA
2.	Mode Switch in SHUTDOWN	NA	2-seconds after MODE SWITCH to SHUTDOWN
3.	SRM Hi Hi	2 x 10^5 cps	Shorting links installed
4.	IRM INOP	NOT in OPERATE High Voltage Low Module unplugged Loss of ± 24 VDC	Mode Switch in RUN
5.	IRM Hi Hi	116.4/125	Mode Switch in RUN
6.	APRM Hi Hi	12%	Mode Switch in RUN
7	APRM Hi Hi	DLO: ≥ (0.61W + 67.4%) [W = Recirc loop flow in % rated] SLO: ≥ (0.55(W-ΔW) + 64.5%) [W = Recirc Loop Flow in % rated].	N/A N/A
8.	APRM Hi Hi	119%	NA
9.	APRM INOP	Self-test Critical fault. Loss of Input Power. Chassis Mode Sw. NOT in OPERATE. Firmware Watchdog timer timed out.	NA

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0101 Page 101 of 112

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

Excerpts from 2-OI-92A:

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. ☐

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

Excerpt 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

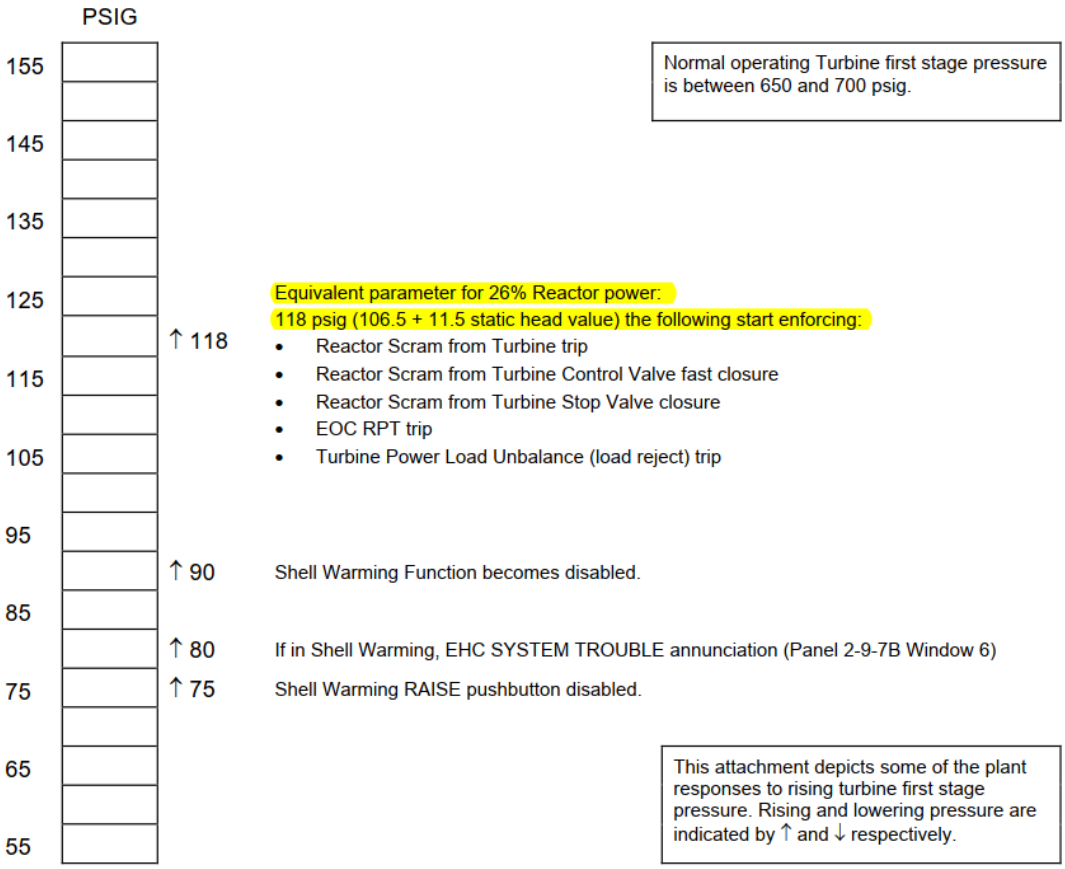
TRIP SIGNAL	SETPOINT	ACTION
APRM Downscale	≤5%	1. Rod Block if REACTOR MODE SWITCH in RUN.
APRM Inop	1. APRM Chassis Mode not in OPERATE (keylock to INOP). 2. Loss of Input Power to APRM. 3. Self Test detected Critical Fault in the APRM instrument. 4. Firmware Watchdog timer has timed out	1. One Channel 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.

Excerpt from 2-OI-47:

Attachment 3
(Page 2 of 3)

Turbine Operating Spectrums

TURBINE FIRST STAGE PRESSURE



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

Outline Cross-reference:	Level	RO	SRO
239002 (SF3 SRV) Safety Relief Valves **K1.02 (**10CFR 55.41.7/8) Knowledge of the physical connections and/or cause and effect relationships between the SAFETY RELIEF VALVES and the following systems • Plant process computer/parameter display systems	Tier #	2	------
	Group #	1	------
	K/A #	239002K1.02	
	Importance Rating	3.0	------

Proposed Question: **# 10**

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



- MAIN STEAM RELIEF VALVE OPEN (2-9-3C, Window 25) alarms

Which **ONE** of the following completes the statements below pertaining to SRVs?

Given the condition above, the Operator can monitor SRV parameters on **(1)** .

The SRVs discharge to the Suppression Pool via the **(2)** to enhance thermal mixing.

A. (1) Panels 2-9-3, 2-9-47 **AND** ICS
(2) ring header

B. (1) Panels 2-9-3, 2-9-47 **AND** ICS
(2) T-quenchers

C. (1) Panels 2-9-3, 2-9-47 **ONLY**
(2) ring header

D. (1) Panels 2-9-3, 2-9-47 **ONLY**
(2) T-quenchers

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that the level of the Suppression Chamber ring header is equivalent to the bottom of the vacuum breaker which is threshold Suppression Pool Level between the SAFE/ACTION REQUIRED region of Curve 4, SRV Tail Pipe Level Limit.

B **CORRECT**: (*See attached*) In accordance with the given ARP, the Operator is required check 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE, on Panel 2-9-47 and SRV Tailpipe Flow Monitor on Panel 2-9-3 for raised temperature and flow indications. Additionally, 2-AOI-1-1, Relief Valve Stuck Open, states once a MSRV is operated, a time delay of 15 – 30 seconds can be expected before a response can be detected on 2-TR-1-1, ICS can be used to monitor the discharge tailpipe temperature, but the appropriate indications on 2-TR-1-1 must be confirmed.
Second part, the SRVs discharge to the Suppression Pool via the T-quenchers to enhance thermal mixing. This is illustrated on 0-TI-394, Att. 7, showing SRV T-quencher level at 5.5 feet in the Suppression Pool.

C INCORRECT: First part is incorrect but plausible in that 0-OI-48, ICS, states, the SPDS component of ICS is not qualified instrumentation and therefore cannot be used as a sole guide in operating the plant. However, the question is only asking from where can SRV parameters be monitored. Second part incorrect but plausible (*See A*).

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

RO Level Justification: Tests the candidate's knowledge of the physical connections and/or cause and effect relationships between the Safety Relief Valves and plant process computer/parameter display systems. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to SRV monitoring.

Technical Reference(s): 0-TI-394, Att. 7, Rev. 15 (Attach if not previously provided)
2-AOI-1-1, Rev. 30
2-ARP-9-3C, Rev. 29
2-EOI Appendix-11A, Rev.5
2-EOI-2, Curve 4, Rev. 19
0-OI-48, Rev. 56

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 # BFN NRC 1205 #42 (Note changes or attach parent)
New

Question History: Last NRC Exam 2012

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:

QUESTION 42

Which ONE of the following completes the statements below?

The SRVs discharge to the suppression pool via the __(1)__, this enhances thermal mixing in the suppression pool.

Direct pressurization of the Suppression Chamber air space from an SRV discharge would occur when suppression pool level lowers to less than __(2)__.

A. (1) T-Quenchers
(2) 11.5 feet

B. (1) T-Quenchers
(2) 5.5 feet

C. (1) ring header
(2) 11.5 feet

D. (1) ring header
(2) 5.5 feet

Correct: B

Excerpt from 2-ARP-9-3C:

MAIN STEAM RELIEF VALVE OPEN

2-FA-1-1

25

(Page 1 of 1)

Sensor/Trip Point:

2-FMT-1-4
SRV Tailpipe Flow Monitor

Sensor Location: SRV TAILPIPE FLOW MONITOR, 2-FMT-1-4, on Panel 2-9-3.

Probable Cause: Relief valve is open or leaking.

Automatic Action: None

Operator Action:

A. **CHECK** MSRV DISCHARGE TAILPIPE TEMPERATURE, 2-TR-1-1, on Panel 2-9-47 and SRV Tailpipe Flow Monitor on Panel 2-9-3 for raised temperature and flow indications.
B. **REFER TO** 2-AOI-1-1.
C. **IF** alarm is due to sensor malfunction, **THEN** **REFER TO** 0-OI-55 and OPDP-4.

References: 2-47E610-1-1 2-45N620-2 GE 730E929-2

Excerpt from 2-AOI-1-1:

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;

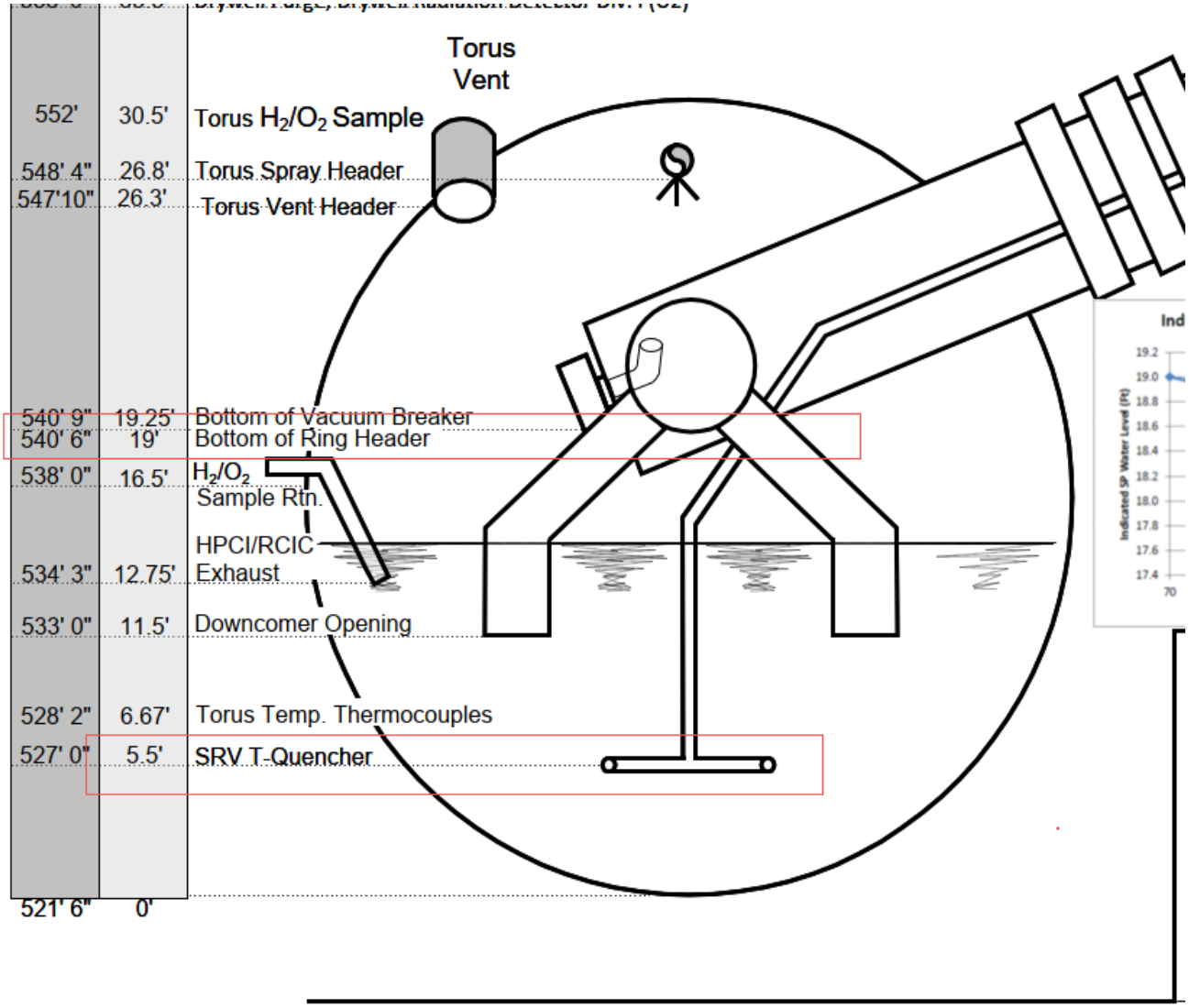
CYCLE the affected relief valve control switch as required up to three times:

- CLOSE to OPEN to CLOSE positions ☐

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

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

Excerpt from 0-TI-394, Att. 7: Also supports Distractors A2, C2



Severe Accident Management Guidelines	
Technical Support Guidelines	
BROWNS FERRY NUCLEAR PLANT	
0-TI-394 Attachment 7	Rev 15

Excerpt from 2-EOI Appendix-11A:

BFN Unit 2	Alternate RPV Pressure Control Systems MSRVs	2-EOI Appendix-11A Rev. 0005 Page 3 of 5

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Control Room
ATTACHMENTS	None

[1] **IF** Drywell Control Air is NOT available, **THEN**

EXECUTE EOI Appendix 8G, CROSSTIE CAD TO DRYWELL CONTROL AIR, **CONCURRENTLY** with this procedure. ☐

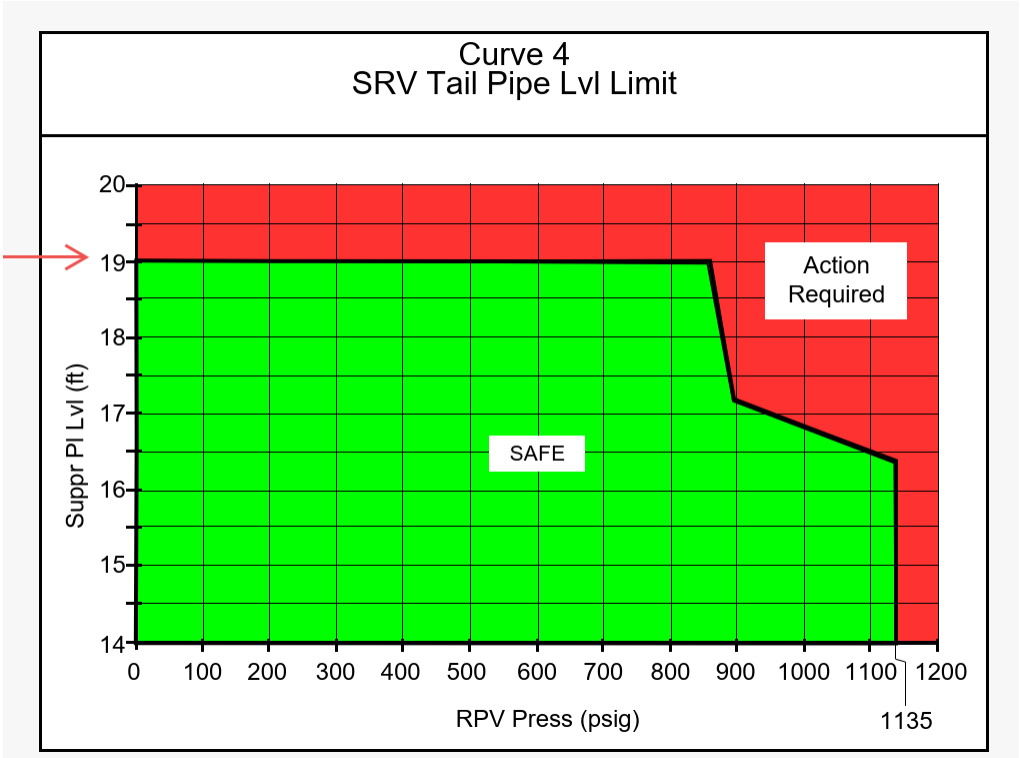
[2] **IF** Suppression Pool level is at or below 5.5 ft, **THEN**

CLOSE MSRVs and **CONTROL** RPV pressure using other options. ☐

[3] **OPEN** MSRVs using the following sequence to control RPV pressure as directed by SRO:

1	2-PCV-1-179	MN STM LINE A RELIEF VALVE	☐
2	2-PCV-1-180	MN STM LINE D RELIEF VALVE	☐
3	2-PCV-1-4	MN STM LINE A RELIEF VALVE	☐
4	2-PCV-1-31	MN STM LINE C RELIEF VALVE	☐
5	2-PCV-1-23	MN STM LINE B RELIEF VALVE	☐
6	2-PCV-1-42	MN STM LINE D RELIEF VALVE	☐
7	2-PCV-1-30	MN STM LINE C RELIEF VALVE	☐
8	2-PCV-1-19	MN STM LINE B RELIEF VALVE	☐
9	2-PCV-1-5	MN STM LINE A RELIEF VALVE	☐
10	2-PCV-1-41	MN STM LINE D RELIEF VALVE	☐
11	2-PCV-1-22	MN STM LINE B RELIEF VALVE	☐
12	2-PCV-1-18	MN STM LINE B RELIEF VALVE	☐
13	2-PCV-1-34	MN STM LINE C RELIEF VALVE	☐

Excerpt from 2-EOI-2, Curve 4: Supports Distractors A2, C2



2-EOI-2 Page 1 of 1

PRIMARY CONTAINMENT CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT

Rev. 19

Excerpt from 0-OI-48, Rev. 56: Supports Distractors C1, D1

3.0 PRECAUTIONS AND LIMITATIONS

A. The Safety Parameter Display System (SPDS) component of the Integrated Computer System (ICS) is **NOT** qualified instrumentation, and therefore cannot be used as the sole guide in operating the plant. It is a highly reliable operator's aid. Installed, qualified plant instruments are to be checked to back up any information shown by the SPDS before any actual plant manipulations are performed.

B. Loss of ICS or ICS Multiplexer, 1(2)(3)-MUX-261-0017, for any reason (INOP, NIS, OOS) will cause the following to be inoperable:

1. 1(2)(3)-XA-55-1, Turbine Tripped Alarms on the first out annunciator
2. 1(2)(3)-XA-55-6C, window #29, RFPT TRIPPED
3. 1(2)(3)-XA-55-7B, window 5, EHC POWER ABNORMAL
4. 1(3)-XA-55-7B, window 6, EHC/TSI SYSTEM TROUBLE
5. 2-XA-55-7B, window 6, EHC SYSTEM TROUBLE
6. 1(3)-XA-55-7A, window #18, MAIN TURB OIL TANK LEVEL LOW-LOW
7. 1-XA-55-6A, window # 26, EXHAUST HOOD A, B, OR C TEMP HIGH HIGH 1-TA-2-113A

C. The time the ICS Computer Power Supply is on the maintenance feed should be minimized due to the automatic battery backup being inoperative for the power supply.

D. The battery backup to the ICS Computer power supply only picks up when the normal power supply (480V Common Board supply) is lost.

E. The Operations Duty Specialist (ODS) should be notified anytime the ICS Computer is going to be removed from service or down powered. The ODS will lose the computer inter-link from BFN during this process [BFN PER 01-003158-000].

F. Operation of the ICS Inverter without the DC source is **NOT** recommended by the Vendor. Operation without the DC source can result in failure of the static switch control components and the total loss of power to equipment normally supplied from the ICS Inverter. (BFPER 01-012718-000)

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

Examination Outline Cross-reference:	Level	RO	SRO
215004 (SF7 SRMS) Source-Range Monitor System	Tier #	2	------
A3.04 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operation of the SOURCE RANGE MONITOR SYSTEM including:	K/A #	215004A3.04	
• Control Rod Block system	Importance Rating	3.7	------

Proposed Question: **# 11**

A Unit 3 startup is in progress with the following conditions:

- 3-HS-99-5A/S1, REACTOR MODE SWITCH, in STARTUP/HOT STANDBY
- **ALL** IRMs are on Range 1
- SRM indications are as follows:
  - SRM A: 2 cps, full in
  - SRM B: 130 cps, full in
  - SRM C: 160 cps, partially withdrawn
  - SRM D: 8.6×10^3, partially withdrawn
- **NO** Operator actions have been taken

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

The SRM Rod Block is _____.

A. NOT received

B. received due to an SRM High signal

C. received due to an SRM Downscale signal

D. received due to an SRM Retract Permit signal

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that there are multiple easily and often confused SRM associated Rod Block outputs as specified in 2-OI-92, Attachment 1, SRM Trip Outputs.

B INCORRECT: Incorrect but plausible in that the SRM Rod Block High signal occurs at ≥ 6.8×10^4 cps unless IRMs on Range 8 or higher or MODE SWITCH in RUN.

C **CORRECT**: (*See attached*) In accordance with 2-OI-92, SRMs, the trip signal for SRM Downscale setpoint is 5 cps in which a Rod Block occurs unless IRMs are on Range 3 or higher (given on Range 1) or REACTOR MODE SWITCH in RUN (given in STARTUP/HOT STANDBY). For the condition given, SRM A is at 2 cps, thus less than the minimum required 5 cps, resulting in an SRM Rod Block.

D INCORRECT: Incorrect but plausible in that SRM Retract Permit Rod Block occurs at less than 145 cps unless detector is full-in, IRMs on Range 3 (or higher) or MODE SWITCH in RUN.

RO Level Justification: Tests the candidate's ability to monitor automatic operation of the SRMs as it relates to Control Rod Block 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.

Technical Reference(s): 3-OI-92, Rev. 18 (Attach if not previously provided)
3-GOI-100-1A, Rev. 148

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.019 Obj. 4a (As available)

Question Source:
Bank #
Modified Bank # FP NRC #37 (Note changes or attach parent)
New

Question History: Last NRC Exam 2020

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:

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

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A3.04
	Importance Rating	3.6

Source Range Monitor

Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: Control rod block status

Proposed Question: #37

A plant startup is in progress with the following:

- The Reactor Mode Switch is in ***START & HOT STBY***.
- All IRMs are on Range 1.
- All SRMs are partially withdrawn.
- A malfunction of the SRM drive control circuitry results in the following indications:
  - SRM A: 60 cps
  - SRM B: 130 cps
  - SRM C: 7×10^3 cps
  - SRM D: 3×10^4 cps

Which one of the following describes the status of the SRM rod block?

The SRM rod block is…

A. NOT received.

B. received due to an SRM Hi Flux signal.

C. received due to an SRM Downscale signal.

D. received due to an SRM Detector Not Full In signal.

Proposed Answer: D

Explanation: The SRM Detector Not Full In rod block occurs when an SRM is not fully inserted with < 100 cps, associated IRMs on range 1 or 2, and the Mode Switch is not in RUN. This prevents control rod withdrawal with RMCS under the given conditions due to SRM A.

A. Incorrect – A control rod withdrawal block is active due to SRM A being <100 cps and partially withdrawn. Plausible because this would be correct if SRM A were fully inserted or >100 cps.
B. Incorrect – SRM Hi Flux rod block occurs with SRM counts above 10^5 cps. All SRMs are less than 10^5 cps. Plausible because SRMs B and D are indicating fairly high.
C. Incorrect – SRM Downscale rod block occurs with SRM counts at 3 cps or less. All SRMs are above 3 cps. Plausible because SRM A is causing a rod block because counts are too low for its position.

Technical Reference(s): OP-16, SDLP-07B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.c.1

Question Source: Bank - 14-1 NRC #5

Question History: 14-1 NRC #5

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Excerpts from 3-OI-92:

3.0 PRECAUTIONS AND LIMITATIONS

A. To prevent a rod withdrawal block when withdrawing SRMs, SRM count rate is required to be above Retract Permit (145 counts per second) and indicating above Downscale Trip Point (5 cps) OR all un-bypassed IRM channels are set to Range 3 or above.

B. Only one SRM channel can be bypassed at a time.

C. In order to prevent inadvertent rod withdrawal block or Reactor scram (with shorting links removed) while operating the SRM BYPASS selector switch,;

Verify that the previously bypassed channel returns to normal status by observing the applicable HIGH HIGH and HIGH or INOP status lights are extinguished prior to selecting any other channel to be bypassed.

After bypassing a channel, the applicable BYPASSED status light should be illuminated prior to testing, operating, or working on that channel.

D. To prevent SRM detector drive damage, the CRD service platform should be locked in the stored position with key removed to allow free movement of SRMs.

E. In order to minimize their exposure, SRM detectors should be fully withdrawn from the core when the IRMs are on range 3 or above and indicating above their downscale trip point.

F. Attachment 1 lists trip signals and associated actions for the Source Range Monitoring System.

G. The Reactor Protection System in conjunction with the Neutron Monitoring System (SRM, and IRM) has non-coincident trip logic if all eight shorting links are removed. If the yellow, green, and red shorting links (six total) are removed, the SRM High-High trips will be placed in a one-out-of-two taken twice logic.

H. The time required to drive a detector from full-out to full-in is approximately 3 minutes.

I. The INOP TRIP BY-PASS switches, located on Panel 3-9-12 SRM drawers, bypass the SRM switch position out-of-operate trip. These switches are to be used only during testing of SRM channels.

J. [NRC/C] Upon return to service of 24 VDC Neutron Monitoring Battery A or B, Instrument Maintenance is required to perform functional tests on SRMs and IRMs that are powered from the affected battery board. [NRC IE Inspector Followup Item 86-40-03]

BFN Unit 3	Source Range Monitors	3-OI-92 Rev. 0018 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

Excerpt from 3-GOI-100-1A:

BFN Unit 3	Unit Startup	3-GOI-100-1A Rev. 0148 Page 103 of 218

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

[5] **RECORD** SRM count rate readings in step 5.4[4] on PIP 95-119 on Panel 3-9-5.

(R) _____ _____ _____
Initials Date Time
1st

(R) _____ _____ _____
Initials Date Time
Reactor Engineer

[6] **IF** bypassed, **THEN**

RECORD SRM BYPASS, 3-HS-92-7A/S3, joystick position. Otherwise, **MARK** N/A.

Channel _____ bypassed

(R) _____ _____ _____
Initials Date Time

[7] **ENSURE** the following Panel 3-9-5 SRM display lights extinguished:

- HIGH HIGH.
- HIGH OR INOP.
- DNSCL.
- BYPASSED (Will be illuminated if channel bypassed.)
- RETRACT PERMIT (NA if above setpoint.)
- PERIOD.

(R) _____ _____ _____
Initials Date Time

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

Examination Outline Cross-reference:	Level	RO	SRO
300000 (SF8 IA) Instrument Air	Tier #	2	------
K3.10 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following systems or system parameters:	K/A #	300000K3.10	
• Control rod drive hydraulic system	Importance Rating	3.6	------

Proposed Question: **# 12**

Unit 1 is operating at 100% RTP when events occur resulting in the following conditions:

- A manual SCRAM is inserted
- After stabilizing the Unit, the SCRAM signal is **RESET**
- **ALL** eight (8) SCRAM solenoid group lights are **ON**

Subsequently,

- RFP 1A (1B) (1C) MINIMUM FLOW VALVES are **OPEN**
- 1-FCV-85-11A(B), CRD SYSTEM FLOW CONTROL VALVES, are **CLOSED**
- OUTBOARD MSIVs **CLOSED**, INBOARD MSIVs remain **OPEN**
- SCRAM INLET and OUTLET VALVES are **OPEN**

Which **ONE** of the following completes the statement below?

Given the conditions above, a Loss of _____ has occurred.

A. Control Air

B. Both RPS Buses

C. Drywell Control Air

D. 9-9 Cabinet 5, Unit Non-Preferred

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 1-AOI-32-2, RFP 1A(1B)(1C) MIN FLOW VALVES, 1-FCV-003-0020(0013)(006), fail OPEN, CRD SYS FLOW CONTROL VLV 1A(1B), 1-FCV-085-0011A(B), fail CLOSED, outboard MSIV accumulator air will be routed to close the outboard MSIVs, the INBOARD MSIVs remain OPEN, and CRD SCRAM INLET and OUTLET VALVES, 1-FCV-085-0039A(B), fail OPEN.

B INCORRECT: Incorrect but plausible in that with a loss of both RPS buses a SCRAM occurs. Subsequently, RFP 1A (1B) (1C) MINIMUM FLOW VALVES open when flow to the vessel lowers due to lack of demand and loss of motive steam supply, the in service 1-FCV-85-11A(B), CRD SYSTEM FLOW CONTROL VALVE, closes when all flow directed to the CRD charging header, the OUTBOARD and INBOARD MSIVs will close on loss of power, and the SCRAM INLET and OUTLET VALVES will open from the loss of RPS

C INCORRECT: Incorrect but plausible in that loss Drywell Control Air will cause the INBOARD MSIVs to drift close. Subsequently, RFP 1A (1B) (1C) MINIMUM FLOW VALVES open when flow to the vessel lowers due to lack of demand and loss of motive steam supply. If Drywell temperature rises above 2.45 psig due to lack of cooling a SCRAM signal will open the SCRAM INLET and OUTLET VALVES.

D INCORRECT: Incorrect but plausible in that a loss of 9-9 Cabinet 5, Unit Non-Preferred, is easily confused with a loss of 9-9, Cabinet 6, Unit Preferred. 1-FCV-85-11A(B), CRD SYSTEM FLOW CONTROL VALVES, will close on a loss of 9-9 Cabinet 6, Unit Preferred, and open on a loss of 9-9 Cabinet 5, Unit Non-Preferred.

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 Control rod drive hydraulic 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.

Technical Reference(s): 1-AOI-32-2, Rev. 9 (Attach if not previously provided)
1-AOI-32A-1, Rev. 14
1-AOI-57-4, Rev. 40
1-AOI-99-1, Rev 24

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.054 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

Excerpts from 1-AOI-32-2:

Attachment 1
(Page 1 of 7)

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 outboard MSIV accumulator air will be routed to close the outboard 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 outboard MSIV closure.

C. STEAM SEAL REGULATOR, 1-PCV-001-0147, will fail CLOSED on loss of air. Steam seal pressure can be controlled using STEAM SEAL REG BYPASS VALVE, 1-FCV-001-0145.

D. STEAM TO OFF GAS PREHEATER 1A(1B), 1-PCV-001-0175A(B), fail OPEN on loss of air. STEAM TO OFF GAS PREHEATER 1A(1B), 1-FCV-001-0176A(B), must be closed after loss of air.

E. STEAM TO SJAE 1A(1B) STAGES I & II, 1-PCV-001-0151(0153), fail CLOSED on loss of air.

F. STEAM TO SJAE 1A(1B) STAGE III, 1-PCV-1-0166(0167), fail CLOSED on loss of air.

G. Main Steam Line Drain valves 1-FCV-1-58, 185, 168, 169, 170, and 171 fail open.

2.0 CONDENSATE

A. COND BSTR PMPS DISCH BYPASS TO COND 1C(1B), 1-FCV-002-0029A(B), fail CLOSED on loss of air.

B. HOTWELL HIGH LEVEL DUMPBACK CONTROL, 1-LCV-002-0003, fails CLOSED on loss of air. CONDENSATE DUMPBACK BYPASS VLV, 1-LCV-002-0004 can be used for hotwell reject.

C. HOTWELL LOW LEVEL MAKEUP CONTROL, 1-LCV-002-0006, fails CLOSED on loss of air. CONDENSATE MAKEUP BYPASS VLV, 1-HS-2-7A can be used for hotwell makeup.

D. Condensate demin influent, effluent, and drain valves fail as is.

E. CNDS DEMIN BYPASS VALVE, 1-FCV-002-0130, fails OPEN.

Attachment 1
(Page 2 of 7)

Expected System Responses

3.0 FEEDWATER

A. RFP 1A(1B)(1C) MIN FLOW VALVE, 1-FCV-003-0020(0013)(006), fail OPEN and can be isolated locally at RFP rooms.

B. RFW STARTUP LCV, 1-LCV-003-0053, fails as is and can be manually isolated or bypassed locally.

C. Loss of RFPT Turning Gear remote engagement capability.

4.0 HEATER DRAINS AND VENTS

A. FW HTR 1A(B)(C)1 DRAIN TO HTR 1A(B)(C)2, 1-LCV-006-0001(0019)(0037), fail OPEN.

B. FW HTR HEATER 1A(B)(C)2 DRAIN TO HTR 1A(B)(C)3, 1-LCV006-0004A(0022A)(0040A), fail CLOSED.

C. FW HTR 1A(B)(C)2 COND DUMP, 1-LCV-006-0004B(0022B) (0040B), fail OPEN.

D. FW HTR 1A(B)(C)3 DRAIN TO HTR 1A(B)(C)4, 1-LCV-006-0007(0025)(0043), fail OPEN.

E. FW HTR 1A(B)(C)4 DRAIN TO HTR 1A(B)(C)5, 1-LCV-006-0011A(0029A)(0047A), fail CLOSED.

F. FW HTR 1A(B)(C)4 COND DUMP, 1-LCV-006-0011B (0029B)(0047B), fail OPEN.

G. FW HTR 1A(B)(C)5 DR TO CNDR, 1-LCV-006-0014A(0032A)(0050A), fail CLOSED.

H. FEEDWATER HEATER 1A(B)(C)5 DUMP TO CNDR, 1-LCV-006-0014B(0032B)(0050B), fail OPEN. These LCVs can be isolated by closing LP HEATER 1A(B)(C)5 CNDR DUMP SHUTOFF using 1-HS-6-97A(98A)(99A) from Panel 1-9-7.

I. Moisture Separator level control reservoir high level dump level control valves to condenser fail OPEN.

Attachment 1
(Page 6 of 7)

Expected System Responses

16.0 HPCI

A. HPCI STM LINE INBD(OUTBD) DRAIN VLV, 1-FCV-073-0006A(B), fail CLOSED. (Normal position while RCIC is operating)

B. HPCI HOTWELL PUMP INBD(OUTBD) ISOL VLV, 1-FCV-073-0017A(B), fail CLOSED. (Normal position while HPCI is operating)

17.0 CORE SPRAY

A. PSC PUMP SUCTION INBD(OUTBD) ISOL, 1-FCV-075-0057(0058), fail CLOSED on loss of air. ECCS discharge piping pressure must be maintained greater than TRM 3.5.4 limits by the condensate storage and supply system.

18.0 FUEL POOL COOLING

A. MAIN CNDR FCV FPC HDR DR, 1-FCV-078-0007, fails CLOSED.

B. FPCL F/D 1A INFL(EFFL) VLV, 1-FCV-078-0019(0026), fail CLOSED.

C. FPC F/D 1A HOLDING PMP DISCH VLV, 1-FCV-078-0033, fails OPEN.

19.0 CRD

A. CRD SCRAM INLET and OUTLET VALVES, 1-FCV-085-0039A(B), fail OPEN.

B. EAST CRD SCRAM DISCH VOL VENT CONT VLV A(B), 1-FCV-085-0083(0083A), and WEST CRD SCRAM DISCH VOL VENT CONT VLV A(B), 1-FCV-085-0082(0082A) fail CLOSED on loss of air.

C. EAST CRD SCRAM DISCH VOL DR CONT VLV, 1-FCV-085-0037E(F) and WEST CRD SCRAM DISCH VOL DR CONT VLV, 1-FCV-085-0037C(D), fail CLOSED on loss of air.

D. CRD SYS FLOW CONTROL VLV 1A(1B), 1-FCV-085-0011A(B), fail CLOSED on loss of air. Valves can be manually opened if required.

Excerpts from 1-AOI-99-1:

4.2 Subsequent Actions (continued)

[11.12] **OBSERVE** the following ILLUMINATED:

- RPS MG SET 1B GEN AVAILABLE, 1-IL-099-0001BB
- RPS MG SET 1B GEN CONNECTED, 1-IL-099-0001BA

[11.13] **NOTIFY** Unit 1 Operator that RPS MG Set 1B is in service to the 1B RPS Bus.

[11.14] **PROCEED** to Step 4.2[12].

[12] **RESET** the RPS trip logic half scram at Panel 1-9-5 as follows:

NOTE

The eight CONTROL ROD TEST SCRAM SOLENOID GROUP A AND B lights should illuminate.

[12.1] **MOMENTARILY PLACE** the SCRAM RESET switch, 1-HS-99-5A/S5, at Panel 1-9-5, to the following positions:

[12.2] RESET FIRST

[12.3] RESET SECOND

[12.4] NORMAL

[13] **CHECK** the following conditions:

[13.1] All eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights ILLUMINATED.

[13.2] SYSTEM A BACKUP SCRAM VALVE, 1-IL-99-5A/AB, ILLUMINATED.

[13.3] SYSTEM B BACKUP SCRAM VALVE, 1-IL-99-5A/CD, ILLUMINATED.

[14] **RESET** the PCIS trip logic at Panel 1-9-4 as follows:

[14.1] **MOMENTARILY PLACE** PCIS DIV I RESET switch, 1-HS-64-16A-S32, to the left and right RESET positions.

Excerpt from 1-AOI-32A-1:

BFN Unit 1	Loss of Drywell Control Air	1-AOI-32A-1 Rev. 0014 Page 3 of 9

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, and operator actions for the loss of Drywell Control Air System.

2.0 SYMPTOMS

A. DRYWELL CONTROL AIR PRESS LOW (1-XA-55-3E, Window 35) at ≤ 92 psig

B. MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW (1-XA-55-3D, Window 18) at ≤ 82 psig

C. Inboard MSIV's close or start to close

D. Drywell cooler dampers close

3.0 AUTOMATIC ACTIONS

None

Excerpts from 1-AOI-57-4:

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.

3.0 AUTOMATIC ACTIONS

A. Unit 1 Panel 1-9-9 Cabinet 6 Unit Preferred automatically transfers to the alternate power source.

B. Reactor Feed Water Control System Panel Display Stations are disabled and the system continues controlling on the last known signal.

C. CRD SYS FLOW CONTROL VLV 1A / B, 1-FCV-85-11 A / B CLOSES on loss of Cabinet 6 and OPENS if only Cabinet 5 is de-energized.

D. UPSTREAM MSL DRAIN TO CONDENSER, 1-FCV-1-58 and BYPASS VLV UPSTREAM MSL DRAIN TO CONDENSER, 1-FCV-1-185 auto open.

E. The following systems swap to the alternate auctioneered ICS power source:

1. EHC Control System (This includes the PLU)
2. Reactor Feed Water Control System
3. Recirc Flow Control System
4. RFPT 1C Woodward Governor.

F. TSI Panel 1-9-46 is powered through auctioneered power supply from both Unit Preferred Panel 1-9-9 Cabinet 6 and Unit Non-preferred Panel 1-9-9 Cabinet 5. On loss of either supply the other will automatically provide power to TSI instruments.

G. Power may be interrupted to various power outlets in the MCR which could result in computer and/or internet access being affected.

Attachment 2
(Page 6 of 15)

Unit Preferred Load Listing

2.0 UNIT 1 PANEL 1-9-9 CABINET 5 UNIT NON-PREFERRED (continued)

L. BKR-514 PANEL 1-9-19 CRD HYDRAULIC SYS

Loss of power to the following CRD instrumentation

1-LPNL-925-0018 Designation	Panel 1-9-5 Designation	Instrument Description
1-FI-085-0025B	1-FI-85-25A	CRD CLG WTR HDR FLOW
1-FI-085-0015B	1-FI-85-15A	CRD DRIVE WTR HDR FLOW
1-FI-085-0030B	1-FI-85-30A	CRD RETURN WATER FLOW
1-PI-085-0013B	1-PI-85-13A	CRD ACCUM CHG WTR HDR PRESS
	1-FIC-85-11	CRD SYS FLOW CONTROL
1-FI-085-0011B		CRD SYSTEM FLOW

M. BKR-515 AIR PART MON 1-RM-90-52, 54, & 56

Air particulate monitors

1. 1-RM-090-0056
2. 1-RM-090-0054

N. BKR-516 AIR PART MON 1-RM-90-51

Loss of Radiation Monitor 1-RM-090-0051

O. BKR-517 PANEL 1-9-2 PURGE AND CHECK SOURCE

Loss of power to Purge and Check Source Network PNL 1-9-2, Stack Radiation Monitoring

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

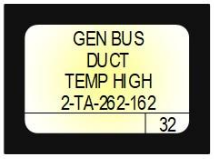
Examination Outline Cross-reference:	Level	RO	SRO
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary	Tier #	2	------
K4.06 (10CFR 55.41.7)	Group #	2	------
Knowledge of MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design features and/or interlocks that provide for the following:	K/A #	245000K4.06	
• Generator protection	Importance Rating	3.5	------

Proposed Question: **# 13**

Unit 2 is operating at 100% RTP when the following conditions occur:

- '**A**' GENERATOR BUS DUCT COOLING FAN trips
- '**B**' GENERATOR BUS DUCT COOLING FAN **FAILED** to manually start
- GEN BUS DUCT FAN FAILURE (2-9-7A, WINDOW 31) alarms
- GEN BUS DUCT TEMPERATURE HIGH (2-9-7A, WINDOW 32) alarms
- Main Generator output amps currently indicate 32,400 amps





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

In accordance with the given ARPs, the crew is required to **IMMEDIATELY** _____.

A. insert a Recirculation Core Flow Runback **ONLY**

B. limit Main Generator output to 7726 amps

C. insert a SCRAM **AND** trip the Main Generator

D. limit Main Generator output to 16,500 amps

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that in accordance with 2-ARP-9-7A, Window 32, if alarm is valid with available bus duct fans in service, then the correct action would be to lower Reactor Power until the Main Generator output amps are lower than 16,500.

B INCORRECT: Incorrect but plausible in that the Turbine Trip Timer is initiated if a low stator cooling water flow condition exists concurrent with Main Generator stator amp ≥ 7726 amps. Conditions where an operational limit or trip is required for the Main Turbine or Generator are easily and often confused.

C **CORRECT**: (*See attached*) In accordance with 2-ARP-9-7A, Window 31, if no Bus Duct Fans are operating and the Generator is tied to the grid with 16,500 amps, then immediately SCRAM and trip the Main Generator.

D INCORRECT: Incorrect (*see A*).

RO Level Justification: Tests the candidate's knowledge of Main Generator and Auxiliary System design features and/or interlocks that provide for generator protection. 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): 2-ARP-9-7A, Rev. 37 (Attach if not previously provided)
2-ARP-9-8A, Rev. 51

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.134 Obj. 10 (As available)

Question Source:

Bank #	OPL171.136-11 003 #3022
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:

QUESTIONS REPORT
for ILT EXAM BANK 08 22 2018

3022. OPL171.136-11 003

Unit 2 is operating at 100% power when the following events occur:

-The running bus duct cooling fan has tripped
-The standby bus duct cooling fan cannot be manually started
-GEN BUS DUCT FAN FAILURE (9-7A Window 31) in alarm
-GEN BUS DUCT TEMPERATURE HIGH (9-7A Window 32) in alarm
-Main Generator output amps currently indicate 32,400 amps

Determine which ONE of the following **manual** actions is required as a result of this condition?

A. Perform an emergency power reduction below 8,000 Main Generator amps.

B✔ Scram the reactor and remove the main generator from service.

C. Insert a Recirculation Core Flow Runback **ONLY** .

D. Perform an emergency power reduction below 16,500 Main Generator amps.

Excerpts from 2-ARP-9-7A:

BFN Unit 2	Panel 9-7 2-XA-55-7A	2-ARP-9-7A Rev. 0037 Page 39 of 46

GEN BUS DUCT FAN FAILURE
2-EA-262-161

31

(Page 1 of 2)

Sensor/Trip Point:	10 in. WC
2-FS-057-0161A(Fan A) 2-FS-057-0161B(Fan B)	Low Air Flow with Gen Bus Cooling Fan 2-HS-262-1A (Fan A) or 2-HS-262-2A (Fan B) in start or normal-after-start.

Sensor Location: Generator bus duct
El 586'
Turbine Building

Probable Cause:
A. Fan normal feeder breaker tripped.
B. Blown control power fuses.
C. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** Main Bus Cooling Fans, 2-HS-262-1A or 2-HS-262-2A, indicates running on Panel 2-9-8 .

CAUTION

Starting Generator Bus Duct Heat Exchanger Fan while rotating greater than 100 rpm in the reverse direction creates potential to damage the fan. U-2 GEN BUS DUCT HTX FAN 2A(2B) BACKDRAFT DMPR, 2-DMP-262-0051(0052), should preclude reverse rotation of the associated fan at speeds greater than 100 rpm. It is acceptable to try restarting the fan one time immediately if the breaker trips on the first attempt.

B. **START** GEN BUS DUCT HX FAN A(B) using 2-HS-262-1A(2A), on panel 2-9-8 to start the standby fan.

C. **IF** no Fans are operating and the Generator is tied to the grid and loaded to greater than the self cooled bus rating of 16,500 amps **THEN**,
PERFORM the following:
1. **IMMEDIATELY INSERT** a manual reactor scram.
2. **TRIP** the Main Generator.

Continued on Next Page

BFN Unit 2	Panel 9-7 2-XA-55-7A	2-ARP-9-7A Rev. 0037 Page 41 of 46

GEN BUS DUCT TEMP HIGH
2-TA-262-162
32

(Page 1 of 3)

Sensor/Trip Point:

2-TS-262-0003A3, B3, C3	72°C (161.6°F)
2-TS-262-0003A1, A2, B1, B2, C1, C2	75°C (167°F)
2-TS-262-0003AA, AC, BA, BB, CB, CC	80°C (176°F)

Sensor Location:
2-TS-262-0003A3, B3, C3 - Main bus
2-TS-262-0003A1, A2, B1, B2, C1, C2 on generator bus Turbine and Collector ends
2-TS-262-0003AA, AC, BA, BB, CB, CC on Transformer

Probable Cause:
A. Fan malfunction.
B. Cooler malfunction.
C. Sensor malfunction.

Automatic Action:
None

Operator Action:

A. **IF** two fans are not operating, **THEN**

START GEN BUS DUCT HX FAN A(B) using 2-HS-262-1A(2A), on panel 2-9-8 to start the standby fan, **AND**
ENSURE system operation in accordance with 2-OI-47.

B. **IF** no Fans are operating and the Generator is tied to the grid and loaded to greater than the self cooled bus rating of 16,500 amps **THEN**,

IMMEDIATELY
INSERT a manual reactor scram, **AND** ☐
TRIP the Main Generator.

C. **IF** while executing this procedure, the Bus Duct Temperature is at or above the Temperature Excursion limit of 120°C **THEN**

IMMEDIATELY
INSERT a manual reactor scram, **AND** ☐
TRIP the Main Generator.

D. **CHECK** the status of alarm GEN BUS DUCT FAN FAILURE, 2-EA-262-161 (2-XA-55-7A, Window 31).

Continued on Next Page

GEN BUS DUCT TEMP HIGH
Window 32
(Page 2 of 3)

Operator Action: (Continued)

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 monitoring device locally at the 15 in-service thermostats. **REFER TO** Figure 1.

NOTE
With two fans operating at the design airflow of 45,000 scfm, the maximum allowable Bus Duct Cooler discharge temperature is 114°F.

G. **IF** alarm is still valid with available fans in service, **THEN**

REDUCE generator power to the self cooled bus rating of 16,500 amps (approximately 50% power) **OR** until the alarm clears.

H. **IF** the fan(s) and/or the cooling coils are **NOT** operable, **THEN**

LIMIT generator output to the self cooling bus rating of 16,500 amps (approximately 50% power).

I. **ENSURE** the system is operating in accordance with 2-OI-47.

Continued on Next Page

Excerpt from 2-ARP-9-8A:

BFN Unit 2	Panel 9-8 2-XA-55-8A	2-ARP-9-8A Rev. 0051 Page 4 of 47

TURBINE TRIP TIMER INITIATED
2-XA-57-139

1

(Page 1 of 1)

Sensor/Trip Point:

62C5 (TDPU 10 sec) ENERGIZED

2-TS-035-0071A(B)(C) ≥ 85°C (185°F)(2 out of 3 logic)

OR

2-FS-035-0065 A(B)(C) ≤ 508 gpm with stator amps ≥ 7726 amps (2 out of 3 logic)

Sensor Location: Stator Temperature Recorder, Panel 2-9-8, Control Room

Probable Cause:
A. Stator Coolant System malfunction.
B. H_2 Cooling malfunction.
C. High reactive load.
D. Sensor malfunction.

Automatic Action: Turbine trip 50 seconds after receiving alarm (total of 60 seconds).

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%.) [PER 03-006187-000]

C. **IF** all of the following conditions exist:
- Alarm fails to reset,
- Low Stator Cooling Water flow OR High Generator or Stator Cooling temperatures are observed on ICS,
- Reactor Power is above turbine bypass valve capability,

THEN,
SCRAM the reactor.

D. **DISPATCH** personnel to Stator Coolant Unit to investigate.

References: 2-45E606-1 2-45E602-16

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	------
EK2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of the relationship between HIGH DRYWELL TEMPERATURE and the following systems or components:	K/A #	295028EK2.01	
• RHR/LPCI: containment spray system mode	Importance Rating	3.9	------

Proposed Question: **# 14**

In accordance with EOI-2, Primary Containment Control, which **ONE** of the following completes the statements below?

Drywell Sprays are required **BEFORE** Drywell Temperature reaches **(1)**.

Operators are required to shutdown **(2)** prior to initiating Drywell Sprays.

A. (1) 280 ºF
(2) Recirc Pumps **ONLY**

B. (1) 280 ºF
(2) Recirc Pumps **AND** Drywell Blowers

C. (1) 350 ºF
(2) Recirc Pumps **ONLY**

D. (1) 350 ºF
(2) Recirc Pumps **AND** Drywell Blowers

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B).* Second part incorrect but plausible in that in accordance with DW/T leg of EOI-2, Recirc Pumps are required to be shutdown prior to initiating Drywell Sprays, however Drywell Blowers are also required.

B **CORRECT**: *(See attached)* In accordance with EOI-2, Primary Containment Control, Drywell Temperature leg (DW/T) states that BEFORE Drywell Temperature rises to 280 ºF, Drywell Sprays are required to initiated. Second part, in accordance with DW/T leg of EOI-2, both Recirc Pumps and Drywell Blowers are required to be shutdown prior to initiating Drywell Sprays.

C INCORRECT: First part is incorrect but plausible in that EOI-2, DW/T leg states that when Drywell Temperature cannot be restored and maintained below 350 ºF, Emergency Depressurization is required. 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 High Drywell Temperature and RHR Containment (Drywell) Spray System mode requirements. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to Drywell Sprays and Drywell Temperature.

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.

Technical Reference(s): 2-EOI-2, Rev. 19 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 Obj. 5b (As available)

Question Source:

Bank #	
Modified Bank #	

(Note changes or attach parent)

New	X

Question History:

Last NRC Exam	

Question Cognitive Level:

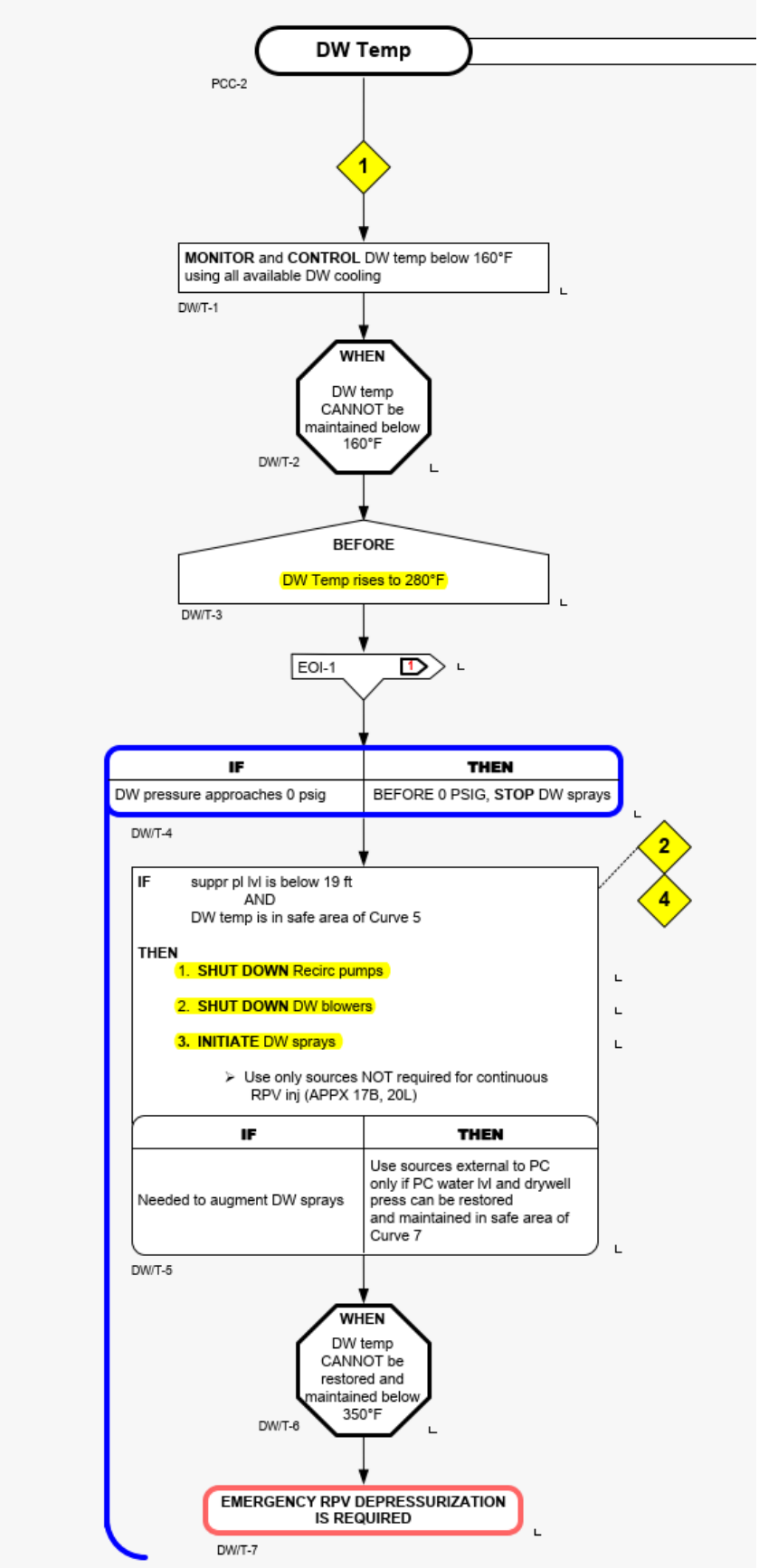
Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

Excerpt from 2-EOI-2:



Examination Outline Cross-reference:

295019 (APE 19) Partial or Complete Loss of Instrument Air / 8

AA2.01 (10CFR 55.41.10)

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

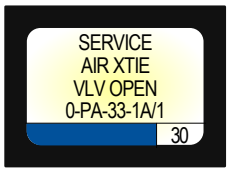
- Instrument air pressure

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295019AA2.01	
Importance Rating	4.1	------

Proposed Question: **# 15**

Unit 1 is operating at 100% RTP when a Control Air leak develops, resulting in the following conditions:

- SERVICE AIR CROSSTIE VALVE OPEN (1-9-20B, Window 30) alarms
- Control Air Pressure is 75 psig, slowly lowering



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

In accordance with the given ARP, 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE, opened/will open at less than or equal to **(1)** Control Air Pressure.

In accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors, if Control Air Pressure **LOWERS** to 65 psig then 1-PCV-32-3901, Unit 1 to Unit 2 Control Air Crosstie, will automatically **(2)**.

A. (1) 70 psig
(2) close

B. (1) 70 psig
(2) open

C. (1) 85 psig
(2) close

D. (1) 85 psig
(2) open

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 0-OI-32, Control Air System, Attachment 1 – Control Air System Pressure Spectrum, lists CONTROL AIR PRESSURE LOW alarms at 70 psig Control Air Pressure. 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 fails to recognize that 1-PCV-32-3901, Unit 1 to Unit 2 Control Air Crosstie, automatically closes when Control Air System Pressure reaches the setpoint of 65 psig versus opening to supplement one Unit to the next. This action prevents a Control Air failure on any one Unit from resulting in a multi-Unit SCRAM. The valve will reopen when Control Air Pressure returns to 85 psig.

C **CORRECT**: (*See attached*) In accordance with the given SERVICE AIR CROSSTIE VALVE OPEN (1-9-20B, Window 30), 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE, opens at 85 psig Control Air Pressure. Second part, in accordance with 0-AOI-32-1, if Control Air Pressure continues to lower to 65 psig then 1-PCV-32-3901, Unit 1 to Unit 2 Control Air Crosstie, will automatically close.

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 interpret instrument air pressure as it relates to partial or complete loss of Control (Instrument) Air. 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 to determine system responses as it relates to multiple Unit and systems impacted that potentially could result in multi-Unit SCRAMs. 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-OI-32, Rev. 143 (Attach if not previously provided)
1-ARP-9-20B, Rev. 40
0-AOI-32-1, Rev. 59

Proposed references to be provided to applicants during examination: SERVICE AIR CROSSTIE VALVE OPEN (1-9-20B, Window 30)

Learning Objective: OPL171.054 Obj. 7 (As available)

Question Source: Bank # | Modified Bank # BFN NRC 2204 #21 | New (Note changes or attach parent)

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

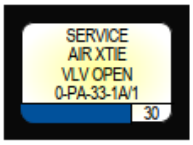
Comments:

Copy of Bank Question:

Proposed Question: **# 21**

Unit 1 is operating at 100% RTP when a Control Air leak develops, resulting in the following conditions:

- SERVICE AIR CROSSTIE VALVE OPEN (1-9-20B, Window 30) alarms
- Control Air Pressure is 69 psig and slowly lowering



Given the conditions above and in accordance with the appropriate Alarm Response Procedure, 0-FCV-33-1, SERVICE AIR CROSSTIE VALVE opened at **(1)** Control Air Pressure.

A manual Reactor SCRAM is required if Control Air Pressure continues to lower to less than **(2)** as indicated by 1-PI-32-20 on Panel 1-9-20 in accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors.

A. (1) 70 psig
(2) 55 psig

B. (1) 70 psig
(2) 66 psig

C. (1) 85 psig
(2) 55 psig

D. (1) 85 psig
(2) 66 psig

Proposed Answer: **C**

Excerpt from 1-ARP-9-20B:

BFN Unit 1	Panel 9-20 1-XA-55-20B	1-ARP-9-20B Rev. 0040 Page 34 of 39

SERVICE AIR XTIE VLV OPEN 0-PA-33-1A/1	30

(Page 1 of 1)

Sensor/Trip Point:

Relay 20C 0-PS-33-1 85 psig

Sensor Location: Relay 20C, Panel 9-20 — 0-PS-33-1, Elevation 565, Col. T-2 N-LINE

Probable Cause:
A. Air compressor malfunction.
B. Excessive air usage.
C. System leakage.
D. Sensor malfunction.

Automatic Action: FCV-33-1 opens to supply Control Air from Service Air.

Operator Action:
A. **CHECK OPEN** FCV-33-1, using 0-HS-33-1A/1, on Panel 1-9-20, or **OPEN** CONTROL AIR BYPASS, 0-33-501 and **MONITOR** CONTROL AIR PRESSURE, 1-PI-32-20. (**COORDINATE** with Unit 3.)
B. **DISPATCH** personnel to Control Air Compressor G Control Panel 0-LPNL-925-0692 and Control Panel 0-LPNL-925-0118 to investigate local alarms and indications.
C. **NOTIFY** Unit SRO.
D. **IF** unable to maintain air pressure, **THEN PERFORM** the following:
- **CHECK** running or **START** all available Control and Service Air Compressors locally. (**COORDINATE** with Unit 3.)
- **ENSURE** Control Air Compressors are loading as required. (G Compressor can be monitored on ICS).

E. **IF** Control Air Compressors are **NOT** loading as required, **THEN HAVE** personnel perform Alternate Method for Manually Loading Control Air Compressors. **REFER TO** 0-OI-32.
F. **INITIATE** a search to find and isolate air leaks. (**COORDINATE** with other units).
G. **REFER TO** 1-AOI-32-2. (**COORDINATE** with other units).

References: 0-45E769-5 1-45E620-12-2 0-45E781-3

Excerpt from 0-AOI-32-1:

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0059 Page 5 of 38

3.0 AUTOMATIC ACTIONS

- Service Air crosstie to Control Air valve, 0-FCV-33-1, opens at control air header pressure less than or equal to 85 psig.
- Control Air Compressors A,B,C,D will start as their on-line pressure setpoints are reached.
- The Emergency Control Bay Air Compressor will start at Control Air Header pressure less than or equal to 73 psig.
- Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches equal to or less than to 65 psig at the valve.
- Unit 1 to Unit 2 Control Air Crosstie, 1-PCV-032-3901, will close when Control Air Header pressure reaches equal to or less than to 65 psig at the valve.

Excerpt from 0-OI-32:

BFN Unit 0	**Control Air System**	**0-OI-32 Rev. 0143 Page 82 of 117**

Attachment 3
(Page 2 of 2)

CONTROL AIR UNIT SEPARATION

The automatic unit separation feature for the Control Air System was installed between Units to isolate the main control air header between the units during a Loss of Control Air. This modification reduces the potential for a scram in one unit when a loss of Control Air occurs in the adjacent unit. This brings the Control Air System into agreement with the FSAR, Section F.4.g of Appendix F, which states "Incidents or accidents in one unit shall not directly influence the operation of other units".

Figure 4 depicts the unit separation feature between Units 2 and 3 in its normal lineup. These valves are located in the Turbine Building Elevation 565' on the main Control Air header between Unit 2 and Unit 3. Valve 2-PCV-032-3901 is held open by Control Air routed through SPV-032-3905, 3906, and 3910 to the diaphragm of PCV-032-3901.

Figure 5 shows how the unit separation feature reacts to a low control air pressure condition in Unit 2. When header pressure drops below 65 psig on the Unit 2 side of PCV-032-3901, SPV-032-3904 realigns, isolating SPV-032-3910 from the main header. SPV 3904 vents the air off the diaphragm of SPV 3910. This, in turn, vents the air off the diaphragm of PCV 3901, causing the valve to close. The main header is now isolated in between units. Once header pressure returns to 85 psig, these valves will realign to allow PCV 3901 to reopen and unisolate the main header. Main header pressure must be greater than 85 psig on both sides of PCV 3901 for the valve reopen. The Low Select Relay keeps the lowest unit pressure aligned to the diaphragm of SPV 3910, thus allowing the lowest unit control air pressure to dictate the operation of PCV 3901.

The unit separation feature between Units 1 and 2 operates in a similar manner using 1-PCV-032-3901.

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

Examination Outline Cross-reference:

295030 (EPE 7) Low Suppression Pool Water Level / 5

EA1.04 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:

- Suppression pool makeup system(s)

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295030EA1.04	
Importance Rating	3.5	------

Proposed Question: **# 16**

Unit 3 was operating at 100% RTP when an event resulted in lowering Suppression Pool Water Level.

Which **ONE** of the following completes the statements below?

In accordance with 3-EOI Appendix-18, Suppression Pool Water Inventory Removal and Makeup, the systems that are available for make-up are **(1)**.

With lowering Suppression Pool Water Level, the **(2)** will be reached **FIRST**.

A. (1) RCIC, Standby Coolant **ONLY**
(2) Downcomer Opening

B. (1) HPCI, RCIC **AND** Standby Coolant
(2) Downcomer Opening

C. (1) RCIC, Standby Coolant **ONLY**
(2) HPCI Turbine exhaust

D. (1) HPCI, RCIC **AND** Standby Coolant
(2) HPCI Turbine exhaust

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the candidate fails to recall all of the available systems including HPCI, that can be used in EOI Appendix-18, Suppression Pool Makeup. Second part is incorrect but plausible in that as Suppression Pool Water lowers, the Downcomer Opening will be reached at 11.5 ft.

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

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

D **CORRECT**: *(See attached)* In accordance with EOI Appendix-18, Suppression Pool makeup systems that are available are HPCI, RCIC, and Standby Coolant. Second part, in accordance with EOIPM Section 0-V(G) EOI 2, Primary Containment Control Bases, HPCI is required to be locked out by 12.75 feet Suppression Pool Water Level unless it's needed for Adequate Core Cooling. HPCI Turbine exhaust normally discharges under the Suppression Pool Water Level, where the suppression function condenses the steam and prevents pressurizing the Suppression Chamber itself.

RO Level Justification: Tests the candidate's ability to operate and monitor Suppression Pool makeup systems as it relates to Low Suppression Pool Water Level. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to containment 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.

Technical Reference(s): 0-TI-394, Rev. 15 (Attach if not previously provided)
EOIPM 0-V(G), Rev. 2
3-EOI Appendix-18, Rev. 4

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 Obj. 15a, b (As available)

Question Source:
Bank #
Modified Bank # BFN NRC 2204 #24 (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 **X**
55.43

Comments:

Copy of Bank Question:

Proposed Question: **# 24**

Unit 1 was operating at 100% RTP when an event occurred resulting in the following conditions:

- Suppression Pool Water Temperature is 150 °F and rising
- Suppression Chamber Pressure is 3 psig and rising
- SRVs are being cycled for Reactor Pressure Control
- Suppression Pool Water Level is 13 feet and lowering

Given the conditions above, the **(1)** will be uncovered **FIRST**.

(2) will be more effective at mitigating the **CURRENT** Heat Capacity limitations of Primary Containment.

A. **(1)** HPCI Turbine exhaust
(2) Suppression Chamber Sprays

B. **(1)** HPCI Turbine exhaust
(2) Suppression Pool Cooling

C. **(1)** Downcomer opening
(2) Suppression Chamber Sprays

D. **(1)** Downcomer opening
(2) Suppression Pool Cooling

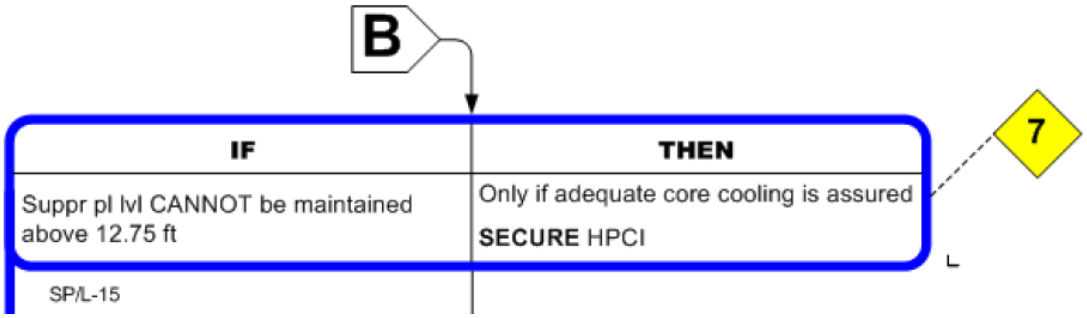
Proposed Answer: **B**

Excerpt from EOIPM 0-V(G):

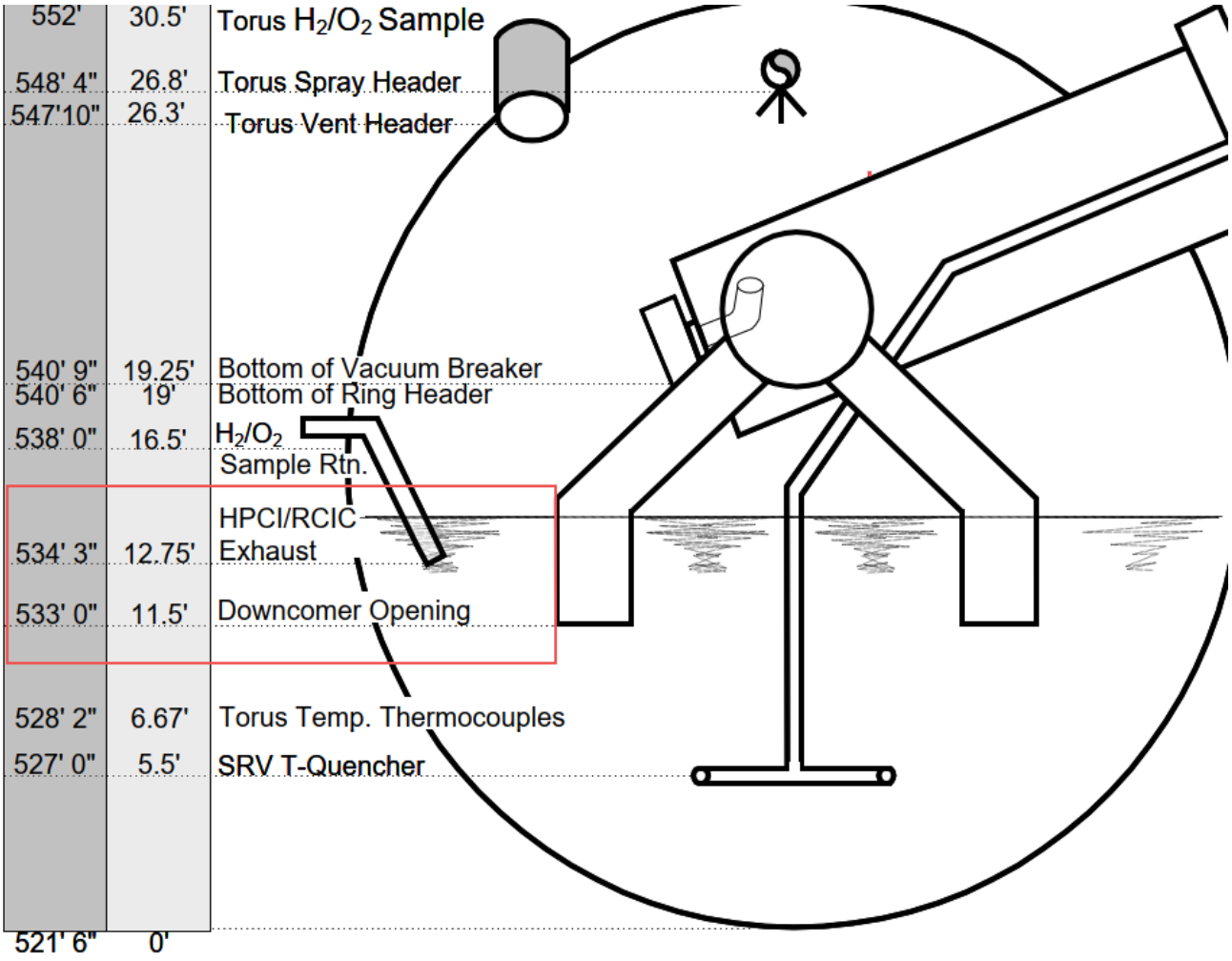
BFN **Unit 0**	**EOI-2, Primary Containment Control Bases**	**EOIPM Section 0-V(G)** **Rev. 0002** **Page 122 of 129**

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

SP/L-15



Excerpt from 0-TI-394, Att. 7:



Excerpts from 3-EOI Appendix-18:

BFN Unit 3	Suppression Pool Water Inventory Removal and Makeup	3-EOI Appendix-18 Rev. 0004 Page 3 of 8

1.0 INSTRUCTIONS

Location:	Unit 3 Control Room
Attachments:	None

NOTE

All panel operations performed at Control Room Panel 3-9-3 unless otherwise stated.

CAUTION

[NRC/C] Suppression Pool water will be highly radioactive after a LOCA. Chemical Engineering recommendations are used to determine location to pump contaminated water.
[NRC Inspection Report 89-16]

[1] **IF** Suppression Pool Water makeup is required, **THEN**

CONTINUE in this procedure at Step 1.0[5].

[2] **IF** Gross fuel failure is suspected, **THEN**

OBTAIN SED/SRO permission to pump down Suppression Pool BEFORE continuing in this procedure.

[3] **IF** Directed by SRO, **THEN**

REMOVE water from 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 DR PUMP A (B) DISCH SHUTOFF VALVE.

[3.1.2] **OPEN** the following valves:

- 3-SHV-074-0564A(B), RHR DR PUMP A(B) SEAL WTR SPLY
- 3-SHV-074-0529A (B), RHR DR PUMP A (B) SHUTOFF VLV.

[3.1.3] **UNLOCK** and **OPEN** 3-SHV-074-0765A (B), RHR DR PMP A(B) DISCH.

1.0 INSTRUCTIONS (continued)

[4.2] **CLOSE** 3-FCV-74-108, RHR DR PUMP 3A/B DISCH HDR VALVE.

[4.3] **ENSURE CLOSED** 3-FCV-74-62, RHR MAIN CNDR FLUSH VALVE.

[4.4] **ENSURE CLOSED** 3-FCV-74-63, RHR RADWASTE SYS FLUSH VALVE.

[4.5] **WHEN** Suppression Pool level can be maintained between -1 in. and -5.5 in., **THEN**

EXIT this procedure.

[5] **IF** Directed by SRO to Emergency Makeup to the Suppression Pool from Standby Coolant, **THEN**

CONTINUE in this procedure at Step 1.0[9].

[6] **IF** Directed by SRO to add water to suppression pool, **THEN**

MAKEUP water to Suppression Pool as follows:

[6.1] **ENSURE OPEN** 3-FCV-73-40, HPCI CST SUCTION VALVE.

[6.2] **OPEN** 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

[6.3] **IF** HPCI is NOT available for Suppression Pool makeup, **THEN**

MAKEUP water to Suppression Pool using RCIC as follows:

[6.3.1] **ENSURE OPEN** 3-FCV-71-19, RCIC CST SUCTION VALVE.

[6.3.2] **OPEN** 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE.

1.0 INSTRUCTIONS (continued)

NOTE

Step 1.0[9] is performed ONLY if directed by Step 1.0[5] to Emergency Makeup to the Suppression Pool.

[9] **IF** Directed by SRO to Emergency Makeup to the Suppression Pool from Standby Coolant Supply, **THEN**

MAKEUP water to the Suppression Pool as follows:

[9.1] **ENSURE CLOSED** the following valves:

- 3-FCV-74-61, RHR SYS I DW SPRAY INBD VLV
- 3-FCV-74-60, RHR SYS I DW SPRAY OUTBD VLV
- 3-FCV-74-58, RHR SYS I SUPPR CHBR SPRAY VALVE
- 3-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV
- 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VLV.

[9.2] **ENSURE** RHR Pumps 3A and 3C are NOT running.

[9.3] **PLACE** 3-BKR-074-0100, RHR HTX A-C DISCH XTIE (TO U-2) VLV FCV-74-100 (M010-171) to ON (480V RMOV Board 3B, Compartment 19A).

[9.4] **START** RHRSW Pumps B1 and B2.

[9.5] **NOTIFY** Unit 1 Operator to **ENSURE CLOSED** 1-FCV-23-46, RHR HEAT EXCHANGER B COOL WATER OUTLET VLV (Panel 1-9-3).

[9.6] **NOTIFY** Unit 2 Operator to perform the following (Panel 2-9-3):

[9.6.1] **ENSURE CLOSED** 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV

[9.6.2] **OPEN** 2-FCV-23-57, STANDBY COOLANT VLV FROM RHRSW.

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	------
EA1.02 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:	K/A #	295037EA1.02	
• Redundant reactivity control system	Importance Rating	4.0	------

Proposed Question: **# 17**

Unit 2 was operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual Reactor SCRAM was inserted
- 2-AOI-100-1, Reactor SCRAM, ATWS Actions are reported as complete
- **ALL** 8 red lights for RPS Group A (4) and B (4), remain illuminated on Panel 2-9-5
- Reactor Water Level is (-) 55 inches, steady
- Reactor Power at 30%, lowering
- The crew has **NOT** completed any Subsequent or EOI actions

Given the conditions above and in accordance with 2-AOI-100-1, which **ONE** of the following completes the statement below?

The crew **(1)** initiated Alternate Rod Insertion System (ARI), and SLC **(2)** injected.

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: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* Given that both RPS Group A and B red lights remain illuminated on Panel 9-5, this indicates that an Electrical ATWS condition has occurred. This is due to the RPS failure to automatically de-energize the SCRAM solenoid relays to vent the SCRAM air header which allows the Control Rods to insert. As given, ATWS Actions are complete in accordance with 2-AOI-100-1, which implies that Reactor Water Level has been lowered to less than (-) 50 inches (given in stem at (-) 55 inches). Prior to performing ATWS Actions, the crew is required to perform IMMEDIATE ACTIONS which, if Control Rods can NOT be verified fully inserted after the SCRAM, then ARI must be initiated.
Second part, in accordance with 2-AOI-100-1, if Reactor Power is above 5% or unknown, then as part of continuing with ATWS Actions, the crew is required to initiate SLC and ensure injection. Given steady Reactor Water Level in conjunction with reactor power continuing to lower with all rods out, is indicative that SLC injection has/is occurring.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible if the candidate fails to equate the completion of the ATWS Actions after the initially IMMEDIATE ACTIONS of 2-AOI-100-1 and incorrectly conclude that injecting SLC is part of the Subsequent or EOI actions. Additionally, for the given conditions, including reactor power continuing to lower, SLC injection is indicated.

C INCORRECT: First part is incorrect but plausible if the candidate fails to equate the completion of the ATWS Actions after the initially IMMEDIATE ACTIONS of 2-AOI-100-1 and incorrectly concludes that initiating ARI is part of the Subsequent or EOI actions. 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 operate and monitor redundant reactivity control systems (ARI and SLC) 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.

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-1A, Rev. 7 (Attach if not previously provided)
2-AOI-100-1, Rev. 123

Proposed references to be provided to applicants during examination: NONE

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

Learning Objective: OPL171.202 Obj.21 (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:

Excerpts from 2-AOI-100-1:

4.0 OPERATOR ACTIONS

4.1 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.

[3] **IF** all control rods can NOT be verified fully inserted, **THEN**

INITIATE ARI.(otherwise **MARK** N/A).

[4] **IF** Reactor Power is 5% or BELOW, **THEN:** (otherwise **MARK** N/A)

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

BFN Unit 2	Reactor Scram	2-AOI-100-1 Rev. 0123 Page 6 of 72

4.1 Immediate Actions (continued)

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 step4.1[5.8] if Reactor Water Level is slowly lowering.

[5] **IF** Reactor Power is ABOVE 5% or unknown, **THEN**

PERFORM the following:
(otherwise **MARK** steps N/A).

[5.1] **REPORT** the following to the Unit SRO:

- REACTOR SCRAM, Mode Switch in SHUTDOWN
- Control Rods out
- Reactor power
- Continuing with ATWS Actions

[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 CS per APP 4.

[5.5] **INITIATE** SLC and **ENSURE** injection. ☐

[5.6] **INHIBIT** ADS LOGIC.

[5.7] **BYPASS** Group 1 RPV Low-Low-Low Level isolation interlocks per APP 8A.

4.1 Immediate Actions (continued)

[5.8] **WHEN** 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 IS/IS NOT Injecting.

[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 IS/IS NOT Injecting.

[5.10] **IF** Any Main steam line is OPEN, **THEN**

CALL for APP 8E, Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation interlocks

4.2 Subsequent Actions

NOTES

1) Steps in this section are written in general order of importance for most anticipated events; however, they are NOT required to be performed in order, but as required to maintain stable conditions. When a step is entered, all associated sub steps are required to be completed in order except those in Step 4.2[32] (Return to Service). Steps which are NOT applicable for this scram should be marked N/A.
2) For Scram Response logic to initiate, all of the following conditions must be met:
   - Scram Response Logic is not inhibited (amber light at SCRAM RESPONSE INHIBIT/RESET switch, 2-HS-46-5 on Panel 2-9-5, is extinguished).
   - REACTOR WATER LEVEL CONTROL PDS, 2-LIC-46-5 on Panel 2-9-5, is in AUTO and at least one individual RFPT Speed Control PDS in AUTO.
   - Either RPS A or B Backup Scram channel activates.
   - Reactor Level (narrow range) falls below 0 inches within 60 seconds of first Backup Scram channel activating.
3) If Programmed Scram Response is initiated, the logic is reset by ANY of the following conditions:
   1. Placing REACTOR WATER LEVEL CONTROL PDS, 2-LIC-46-5 on Panel 2-9-5 in MANUAL.
   2. Reactor level (narrow range) exceeding level setpoint.
   3. Five minutes expire from the time the Scram Response logic was activated.
   4. Depressing SCRAM RESPONSE INHIBIT/RESET Switch, 2-HS-46-5, on Panel 2-9-5.

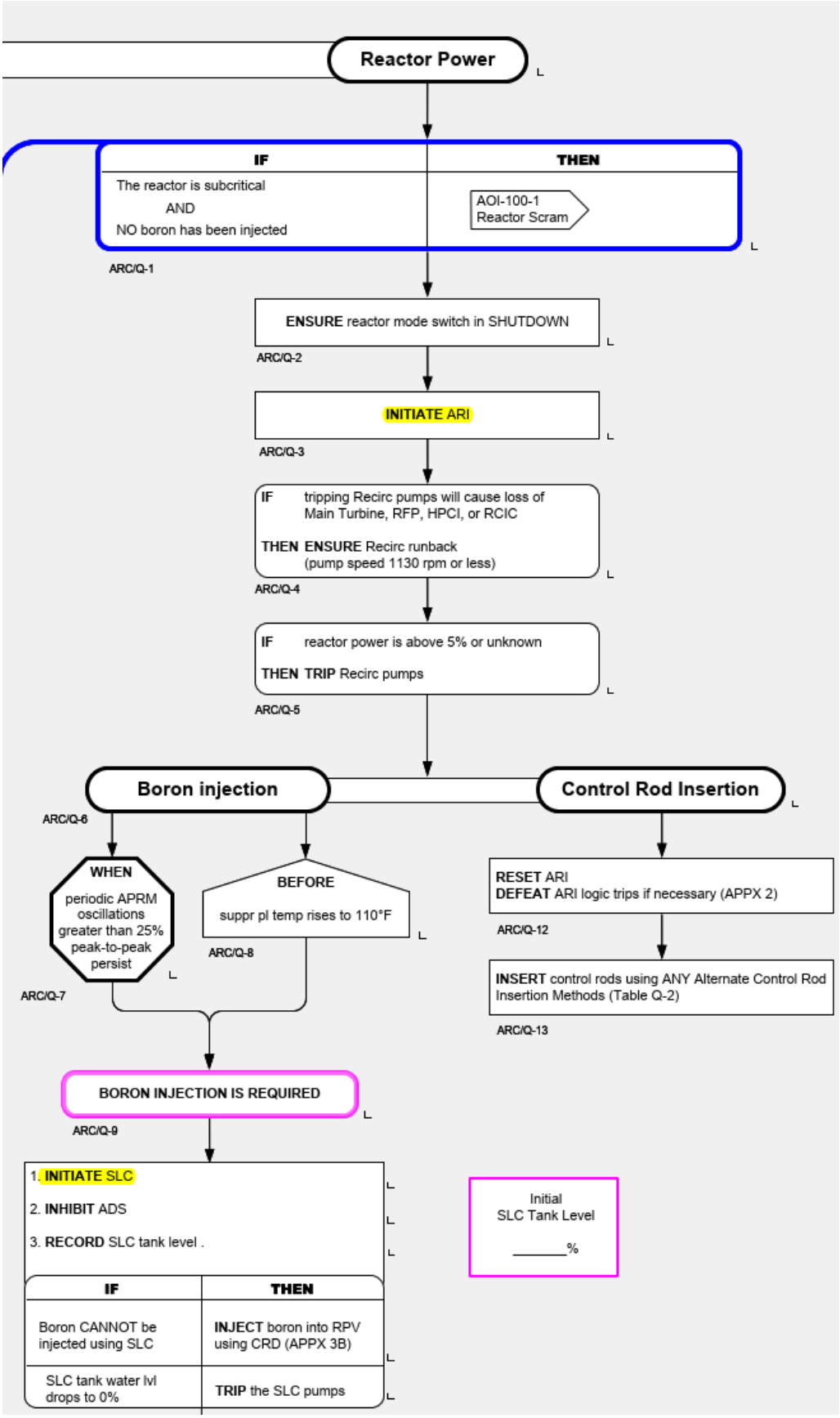
[1] **ANNOUNCE** Reactor SCRAM over PA system.

[2] **DRIVE** in all IRMs and SRMs from Panel 2-9-5 as time and conditions permit.

[2.1] **DOWNRANGE** IRMs as necessary to follow power as it lowers.

[3] **ENSURE** SCRAM DISCH VOL VENT & DR VLVS CLOSED by green indicating lights at SDV Display on Panel 2-9-5.

Excerpt from 2-EOI-1A:



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

Examination Outline Cross-reference:

295038 (EPE 15) High Offsite Radioactivity Release Rate / 9

EK2.15 (10CFR 55.41.7)

Knowledge of the relationship between the HIGH OFFSITE RADIOACTIVITY RELEASE RATE and the following systems or components:

- Primary containment system and auxiliaries

Proposed Question: **# 18**

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295038EK2.15	
Importance Rating	3.7	------

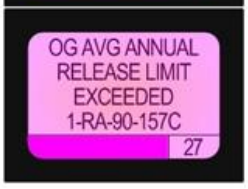
Unit 1 has been operating for one week at 85% RTP with the following conditions:



- MAIN STEAM LINE RADIATION HIGH (1-9-3A, Window 7) in alarm
- Rising amounts of fuel bundle leaks
- Suppression efforts have been unsuccessful
- A trigger value has been set for shutting down the Reactor on excessive Stack Release Rates

Subsequently, the following alarms are received:





- MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27)
- OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED (1-9-4C, Window 27)

Given the conditions above and in accordance with the given ARPs, which **ONE** of the following completes the statements below?

The crew is required to perform a **(1)**.

The MAIN STEAM LINE RADIATION HIGH-HIGH alarm specific sensor/trip setpoint is based upon radiation levels of **(2)** (times) normal full power background including N-16 contribution and Hydrogen Water Chemistry (HWC) System injection.

A. (1) SCRAM **ONLY**
(2) 1.5 x

B. (1) SCRAM **ONLY**
(2) 3.0 x

C. (1) Core Flow runback to 50-60%, **THEN** insert a SCRAM
(2) 1.5 x

D. (1) Core Flow runback to 50-60%, **THEN** insert a SCRAM
(2) 3.0 x

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the given Reactor Power is below 100% and the candidate could conclude that a manual Recirc runback has already been performed thereby only requiring a manual SCRAM. Second part is incorrect but plausible in that in accordance with the ARP for MAIN STEAM LINE RADIATION HIGH (1-9-3A, Window 7), 2-RM-136/137 setpoints are 1.5 x normal full power background including N-16 contribution and HWC System injection.

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 for MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27), if alarm is valid and SCRAM has not occurred, then if Core Flow is above 60%, lower Core Flow to between 50-60% then manually SCRAM the Reactor. Second part, in accordance with the ARP for Window 27, 2-RM-136/137 setpoints are 3.0 x normal full power background including N-16 contribution and HWC System injection.

RO Level Justification: Tests the candidate's knowledge of the relationship between High Offsite Radiation Release Rate and the Primary Containment System as it relates to radiation monitoring and mitigation actions. This question is rated as C/A due to the requirement to assemble, sort, and integrate at multiple parts of the question to predict an outcome for 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. 56
1-ARP-9-4C, Rev. 31
1-OI-68, Rev. 55
(Attach if not previously provided)

Proposed references to be provided to applicants during examination:
MAIN STEAM LINE RADIATION HIGH (1-9-3A, Window 7)
MAIN STEAM LINE RADIATION HIGH-HIGH (1-9-3A, Window 27)
OFFGAS AVERAGE ANNUAL RELEASE RATE EXCEEDED (1-9-4C, Window 27)

Learning Objective: OPL171.030, Obj 10 (As available)

Form 4.2-1	**Written Examination Question Worksheet**		
Question Source:	Bank #		
	Modified Bank #	BFN NRC 2204 #34	(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

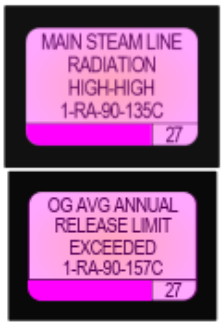
Comments:

Copy of Bank Question 2204 #34:

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-9-4C:

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0031 Page 34 of 43

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

Excerpts from 1-ARP-9-3A:

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. 0056 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	

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0056 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. 0056 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

Excerpt from 1-OI-68:

BFN Unit 1	Reactor Recirculation System	1-OI-68 Rev. 0055 Page 198 of 221

Attachment 2
(Page 3 of 5)

Recirc Flow Control System Manual Runback Controls

DESCRIPTION

The manual runback controls consist of three blue backlit push-buttons which can be utilized to reduce Reactor Power by reducing Recirc Pump speed in response to abnormal operating conditions such as Condensate Pump trips, Reactor Feedpump trips, Feedwater Heater isolations, etc. During normal rated plant conditions, these push-buttons are illuminated, telling the operator that total steam flow and core flow are above the manual runback setpoints and Recirc Pump speeds are above the manual runback lower speed limits.

The manual runback controls are disabled when Recirc Drive Speed Demand is 25 rpm or more below Calc Speed.

When a manual runback push-button is depressed, a demand signal is sent to the Recirc Drive to reduce recirc pump speed until either the runback setpoint or the lower Recirc Pump speed limit is reached. The manual setpoints and lower speed limits are as follows:

MANUAL RUNBACKS		
Manual Runback	**Setpoint**	**Pump Speed Limit**
High Power Runback	90% Total Steam Flow (14.8 Mlbm/hr)	700-750 RPM
Mid Power Runback	66% Total Steam Flow (approximately 10.9 Mlbm/hr)	700-750 RPM
Core Flow Runback	58% Total Core Flow (60 Mlbm/hr)	700-750 RPM

Depressing a manual runback push-button will initiate a runback of Recirc Pump speed. A manual runback can be stopped by simply depressing the appropriate push-button a second time. Depressing the push-button a third time will start the manual runback and depressing the push-button a fourth time will stop the manual runback. This pattern can be repeated until either the setpoint or the Recirc Pump speed limit is reached.

Depressing any speed push-button (raise or lower) will stop the manual runback.

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	------
AK3.02 (10CFR 55.41.10)	Group #	1	------
Knowledge of the reasons for the following responses or actions as they apply to PLANT FIRE ON SITE:	K/A #	600000AK3.02	
• Steps called out in the site fire protection plan, fire protection system manual, and fire zone manual	Importance Rating	3.3	------

Proposed Question: **# 19**

Fire Operations has responded to a fire in the Unit 3 Auxiliary Instrument Room.

The Incident Commander has requested securing ventilation.

Which **ONE** of the following completes the statements below in accordance with 0-AOI-26-1, Fire Response?

If ventilation **CANNOT** be secured from the control switch on Attachment 1, Ventilation Controls, an Assistant Unit Operator (AUO) **(1)** be directed to trip the breaker for the Air Handling Unit.

Performing the above action will support the **(2)**.

A. (1) will
(2) closure of fire dampers due to loss of air flow

B. (1) will
(2) automatic initiation of the CO_2 System

C. (1) will NOT
(2) closure of fire dampers due to loss of air flow

D. (1) will NOT
(2) automatic initiation of the CO_2 System

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 0-AOI-26-1, Fire Response, when directed by Incident Commander or Shift Manager, trip fans, air handling units, or cooling unit for affected area (Attachment 1). If components fail to trip at listed handswitch locations, then trip components at listed breakers. Second part, 0-AOI-26-1, states that tripping fans or air handling units will support the closure of fire dampers due to loss of air flow.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that 0-AOI-26-1 discusses the discharge of CO_2, however in accordance with 0-OI-39, CO_2 System, CO_2 is required to be manually

initiated for the Auxiliary Instrument Rooms. Additionally, 0-OI-39 contains a section for automatic CO_2 initiation.

C INCORRECT: First part is incorrect but plausible in that by manually closing the dampers to the affected, the air flow path could be isolated, however 0-AOI-26-1, does not direct manually isolating dampers. 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 reasons for procedure steps taken by the crew related to 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-AOI-26-1, Rev. 28 (Attach if not previously provided)
0-OI-39, Rev. 39

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.067 Obj. 2i (As available)

Question Source:

Bank #	BFN NRC 1703 #24	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam 2017

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:

QUESTION 24 Rev 1

Fire OPS has responded to a fire in the U3 Aux Instrument Room.

The Incident Commander has requested securing ventilation.

Which one of the following completes both statements below in accordance with 0-AOI-26-1, Fire Response?

If the ventilation cannot be secured from the control switch listed in 0-AOI-26-1 Attachment 1, Ventilation Controls, an Auxiliary Unit Operator (AUO) will be dispatched to __ (1) __.

Performing this action will support the __ (2) __.

A. (1) trip the breaker for the Air Handling Unit
(2) closure of fire dampers

B. (1) trip the breaker for the Air Handling Unit
(2) automatic initiation of the CO_2 system

C. (1) isolate ventilation by manually closing Aux Instr Rm 3 supply damper 3-31-727
(2) closure of fire dampers

D. (1) isolate ventilation by manually closing Aux Instr Rm 3 supply damper 3-31-727
(2) automatic initiation of the CO_2 system

Answer: A

Excerpts from 0-AOI-26-1:

4.2 Subsequent Actions (continued)

NOTES

1) Tripping fans or air handling units will support the closure of fire dampers due to loss of air flow.
2) The ventilation systems can be restarted for smoke removal provided a PCIS Group 6 isolation is **NOT** required and radioactive release limits will **NOT** be exceeded.
3) It may be necessary to wear self-contained breathing apparatus to manually align ventilation dampers.
4) Damper locations are given in 1(2)(3)-OI-30B and 1(2)(3)-OI-30A valve lineup checklists.
5) If SBGT train is run to remove smoke, the train will be technically inoperative thereafter. System Engineering should be contacted to determine tests and SI/SRs necessary to re-establish operability. Since the SGT System is common to all three units, the Unit Operator on each unit should stay fully aware of the system status and condition at all times. The Unit Operators should communicate to each other any change in system status.
6) The Fire Brigade Leader will coordinate changes to ventilation system alignment through the Incident Commander to the Control Room.
7) The Emergency Responders which are not being used for EOI Appendices or FSS Attachments can be utilized to align ventilation system dampers/fans as necessary.

[7] **WHEN** directed by Incident Commander or Shift Manager, **THEN**

- **TRIP** fans, air handling units, or cooling unit for affected area. **REFER TO** Attachment 1.
  - **IF** components fail to trip at listed handswitch locations, **THEN**

    TRIP components at listed breakers.
- **MANUALLY ALIGN** ventilation system dampers as necessary.
- **PLACE** ventilation systems in service to remove smoke from affected area.

[8] **WHEN** directed by Incident Commander or Shift Manager, **THEN**

USE additional smoke removal equipment such as portable fans and smoke ejectors for smoke removal.

4.2 Subsequent Actions (continued)

NOTES

1) When 0-HS-31-124 is placed in EMERGENCY, 0-FCO-31-124A closes to isolate Battery Room No. 1 exhaust to Battery and Board Room Exhaust Fan. 0-FCO-31-124C closes to isolate Auxiliary Instrument Room No. 1 air return to Unit 1 and 2 Elevation 593 AHU. 0-FCO-31-124B opens to route Auxiliary Instrument Room No. 1 return to Battery and Board Room Exhaust Fan.
2) When 0-HS-31-125 is placed in EMERGENCY, 0-FCO-31-125C closes to isolate Battery Room No. 2 and Communications Battery Room exhaust to Battery and Board Room Exhaust Fan. 0-FCO-31-125B closes to isolate Auxiliary Instrument Room No. 2 air return to Unit 1 and 2 Elevation 593 AHU. 0-FCO-31-125A opens to route Auxiliary Instrument Room No. 2 return to Battery and Board Room Exhaust Fan.
3) When 0-HS-31-126 is placed in EMERGENCY, 0-FCO-31-126C closes to isolate Battery Room No. 3 exhaust to Battery and Board Room Exhaust Fan. 0-FCO-31-126A closes to isolate Auxiliary Instrument Room No. 3 air return to Unit 3 Elevation 593 AHU. 0-FCO-31-126B opens to route Auxiliary Instrument Room No. 3 return to the Battery Board Room Exhaust Fan.

CAUTION

Control Switch, 0-HS-31-124(125, 126), should be returned to NORMAL as soon as possible to reopen battery room exhaust dampers to prevent buildup of explosive hydrogen concentration.

[19] **IF** smoke or CO_2 is present in Auxiliary Instrument Room No. 1, **THEN**

PERFORM the following:

[19.1] **PLACE** AUX INST RM NO 1 EMERGENCY EXHAUST, 0-HS-31-124, on 0-LPNL-925-0165 Panel A in EMERGENCY to ventilate affected room.

[19.2] **WHEN** Auxiliary Instrument Room No. 1 smoke/CO2 removal has been completed, **THEN**

PLACE AUX INST RM NO 1 EMERGENCY EXHAUST, 0-HS-31-124, on 0-LPNL-925-0165 Panel A in NORMAL.

4.2 Subsequent Actions (continued)

[20] **IF** smoke or CO2 is present in Auxiliary Instrument Room No. 2, **THEN**

PERFORM the following:

[20.1] **PLACE** AUX INST RM NO 2 EMERGENCY EXHAUST, 0-HS-31-125, on 0-LPNL-925-0165 Panel B in EMERGENCY to ventilate affected room.

[20.2] **WHEN** Auxiliary Instrument Room No. 2 smoke/CO2 removal has been completed, **THEN**

PLACE AUX INST RM NO 2 EMERGENCY EXHAUST, 0-HS-31-125, on 0-LPNL-925-0165 Panel B in NORMAL.

[21] **IF** smoke or CO2 is present in Auxiliary Instrument Room No. 3, **THEN**

PERFORM the following:

[21.1] **PLACE** AUX INST RM NO 3 EMERGENCY EXHAUST, 0-HS-31-126, on 0-LPNL-925-0165 Panel C in EMERGENCY to ventilate affected room.

[21.2] **WHEN** Auxiliary Instrument Room No. 3 smoke/CO2 removal has been completed, **THEN**

PLACE AUX INST RM NO 3 EMERGENCY EXHAUST, 0-HS-31-126, on 0-LPNL-925-0165 Panel C in NORMAL.

[22] **IF** smoke is present in 1A or 1B Electrical Board Room, **THEN**

PERFORM the following:

[22.1] **START** 1A Shutdown Board Room Exhaust Fan.
REFER TO 0-OI-31.

[22.2] **WHEN** Electrical Board Room smoke removal has been completed, **THEN**

STOP 1A Shutdown Board Room Exhaust Fan.
REFER TO 0-OI-31.

Attachment 1
(Page 9 of 10)

Unit 2 Control Bay, El 617 Fire Pre-Plan Cb 2-617	Handswitch	Handswitch Location	Elect Panel	Compt No.
RELAY RM AIR HANDLING UNIT A	0-HS-031-133	U-1 Mech Eq Rm El 617	CB Vent Bd A	2E
Unit 3 Control Bay, El 593 Fire Pre-Plan Cb 2-617	**Handswitch**	**Handswitch Location**	**Elect Panel**	**Compt No.**
BATTERY & BOARD RM EXHAUST FAN 3A/3B	0-HS-31-97A	U-1 Mech Eq Rm El 617 0-LPNL-925-0165 Pnl A	CB Vent Bd B	3C
BATTERY & BOARD RM EXHAUST FAN 3B	0-HS-31-98A	U-1 Mech Eq Rm El 617 0-LPNL-925-0165 Pnl C	CB Vent Bd A	6F
BATTERY & BD RM EMER EXHAUST FAN 3C	0-HS-31-119	U-1 Mech Eq Rm El 617	CB Vent Bd B	6A
EL 593 AIR HANDLING UNIT 3A	0-HS-31-107	U-3 Mech Eq Rm El 593	CB Vent Bd A	6D
El 593 AIR HANDLING UNIT 3B	0-HS-31-108	U-3 Mech Eq Rm El 593	CB Vent Bd B	R2D
Unit 3 Control Bay, EL 606 Fire Pre-Plan CB 3-606	**Handswitch**	**Handswitch Location**	**Elect Panel**	**Compt No.**
BATTERY & BOARD RM EXHAUST FAN 3A/3B	0-HS-31-97A	U-1 Mech Eq Rm El 617 0-LPNL-925-0165 Pnl A	CB Vent Bd B	3C
BATTERY & BOARD RM EXHAUST FAN 3B	0-HS-31-98A	U-1 Mech Eq Rm El 617 0-LPNL-925-0165 Pnl C	CB Vent Bd A	6F
BATTERY & BD RM EMER EXHAUST FAN 3C	0-HS-31-119	U-1 Mech Eq Rm El 617	CB Vent Bd B	6A
EL 593 AIR HANDLING UNIT 3A	0-HS-31-107	U-3 Mech Eq Rm El 593	CB Vent Bd A	6D
El 593 AIR HANDLING UNIT 3B	0-HS-31-108	U-3 Mech Eq Rm El 593	CB Vent Bd B	R2D
UNIT 3 TOILET RM EXHAUST FAN	0-HS-31-137	Vent Tower 3	Ltg Cab LC-307	21
Unit 3 Control Bay, EL 617 Fire Pre-Plan CB 3-617	**Handswitch**	**Handswitch Location**	**Elect Panel**	**Compt No.**
UNIT 3 CONTROL RM AIR HANDLING UNIT 3A	0-HS-31-104	U-3 Mech Eq Rm El 617	CB Vent Bd A	3F
UNIT 3 CONTROL RM AIR HANDLING UNIT 3B	0-HS-31-105	U-3 Mech Eq Rm El 617	CB Vent Bd B	R2C
UNIT 3 CONTROL BAY SUPPLY FAN 3A	0-HS-031-0138	Vent Tower 3	CB Vent Bd B 5B	
UNIT 3 CONTROL BAY SUPPLY FAN 3B	0-HS-031-0139	Vent Tower 3	480 RMOV Bd 3B 17A	
UNIT 3 TOILET RM EXHAUST FAN	0-HS-31-137	Vent Tower 3	Ltg Cab LC-307	21

Excerpt from 0-OI-39:

BFN Unit 0	CO_2 System	0-OI-39 Rev. 0039 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 within one hour 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.

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	------
K5.02 (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 REACTOR CORE ISOLATION COOLING SYSTEM:	K/A #	21700K5.02	
• Flow control	Importance Rating	3.8	------

Proposed Question: **# 20**

Which **ONE** of the following completes the statements below in accordance with 3-OI-71, Reactor Core Isolation Cooling System?

The **(1)** illuminated status light on 3-FIC-71-36A, RCIC FLOW CONTROLLER, indicates a critical failure.

When 3-FIC-71-36A, is in MANUAL, RCIC **(2)** OPERABLE.

A. (1) amber
(2) is NOT

B. (1) amber
(2) is

C. (1) red
(2) is NOT

D. (1) red
(2) is

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the ALM (amber light) on RCIC Flow Controller can indicate a number of things on 3-FIC-36A, RCIC Flow Controller. If the 3-FIC-36A amber light is continuously lit for short periods of time during transients including RCIC startups and shutdowns, then this is expected and consistent with system design. A continuously lit amber light during steady state operation or for extended periods of time on 3-FIC-36A, indicates a high or low limit alarm has actuated or the input/output signal line is open. 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 RCIC Flow Controller OPERABILITY can be confusing as it relates to being functional and/or available. For example, when the RCIC Flow Controller amber light is continuously lit for short periods of time during transients including RCIC startups and shutdowns, it is expected and consistent with system design and RCIC is still considered OPERABLE.

C **CORRECT**: (*See attached*) In accordance with 3-OI-71, RCIC, the FAIL (red light) indicates a critical failure has occurred in 3-FIC-36A, RCIC Flow Controller in Main Control Room on Panel 3-9-3. Second part, in accordance with 3-OI-71, any time the RCIC Main Control Room 3-FIC-36A is placed in MANUAL, RCIC is to be considered NOT OPERABLE, however RCIC is still available.

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 or cause and effect relationships of flow control as it applies to Reactor Core Isolation Cooling System. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to RCIC Flow Controller operation.

Technical Reference(s): 3-OI-71, Rev. 75 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.040 Obj. 9 (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

Excerpts from 3-OI-71: Also supports Distractors A1, B1

3.1 General Precautions (continued)

DD. When Condition Reports (CR) are initiated for RCIC FLOW CONTROLLER, 3-FIC-71-36A, alarms or features, they should contain a request for the Instrument Mechanics to use the ALARM DISPLAY to obtain fault and alarm codes. This information is useful during apparent or root cause analysis performed by the technical staff to document the cause the cause of the failure.

EE. Any time the RCIC Main Control Room Controller is placed in "MANUAL" RCIC is to be considered "**NOT** OPERABLE" however RCIC is still available.

RCIC FLOW CONTROLLER, 3-FIC-71-36A(B) also contains a hard manual backup control circuit & thumbwheel. This is a completely independent analog control circuit that is not reliant on controller software. Hard manual operation is described in Attachment 2 1.0 and should only be considered in conditions when controller has failed.

FF. Due to design leakage, 3-ISV-75-1 (CS System I Suppr Pool Suct Isolation Valve) is **NOT** suitable for a clearance boundary without the use of a ring header plug.

GG. During summer operation of RCIC, CST suction temperatures may be high due to seasonal outside high temps. Under these conditions, RCIC OIL COOLER OUTLET OIL TEMP HIGH, Window 23 on XA-55-3C may alarm. This elevated drain temperature is not an issue as long as the RCIC bearing temperatures are not in alarm. (Refer to Functional Eval. #42260)

HH. Reducing RCIC System flow below 450 GPM (75% of rated flow) while injecting to the reactor vessel may result in control system instability. If control system instability occurs, RCIC SYSTEM FLOW/CONTROL, 3-FIC-71-36A, should be operated in MANUAL. This action will provide stable, constant turbine speed control, but will require operator action to maintain the desired vessel injection rate.

II. The BFN Generic Letter (GL) 89-10 Program has excluded 3-FCV-071-0037 from its program maintenance requirements. Consequently, administrative actions must be taken to ensure RCIC is declared inoperable and a limiting condition for operation (LCO) entered if this valve is repositioned from its normal (open) position when the unit is in Mode 1, 2, or 3.

JJ. RCIC TURB TRIP/THROT VLV, 3-FCV-071-0009 should not be tripped using 3-HS-71-9A from the full open position without steam pressure (main or boiler). Impact without a steam cushion between the valve disc and valve seat could damage the valve seat.

3.1 General Precautions (continued)

Z. (II/C) During routine plant evolutions, notify RADCON prior to making changes in the RCIC System which could cause a rise in area radiation levels. Confirmation that RADCON has implemented appropriate radiological controls/barriers for the expected RCIC System alignment is required prior to performing the alignment. (BFPER961778)

AA. RCIC Turbine oil level, as seen in 3-LG-071-0509, is expected to drop out of the marked fill band while the turbine is operating, but should stay within the sight glass at all times. [BFPER 98-007964]

BB. RCIC FLOW CONTROLLER, 3-FIC-71-36A, Status Lights

1. The FAIL (red light) indicates a critical failure has occurred in the controller. The RCIC System is inoperable during this condition.

2. The ALM (amber light) can be continuously lit for short periods of time during transients including RCIC startups and shutdowns. This is expected and consistent with system design. The RCIC System is operable during these conditions.

3. The ALM (amber light) being continuously lit during steady state operation or for extended periods of time indicates the controllers high or low limit alarm is actuated or the input/output signal line is open. The RCIC System is inoperable during these conditions.

CC. RCIC FLOW CONTROLLER, 3-FIC-71-36B, Status Lights

1. The FAIL (red light) indicates a critical failure has occurred in the controller. The RCIC System is inoperable during this condition.

2. The ALM (amber light) can be continuously lit for short periods of time during transients including RCIC startups and shutdowns. This is expected and consistent with system design. The RCIC System is operable during these conditions.

3. The ALM (amber light) being continuously lit during steady state operation or for extended periods of time indicates the controllers high or low limit alarm is actuated or the input/output signal line is open. The RCIC System is inoperable during these conditions.

4. The ALM (amber light) is normally lit for this controller.
When 3-XS-71-36B (on Pnl 25-31) is in NORM, the controller is not in the circuit and an alarm condition is displayed.
When 3-XS-71-36B (on Pnl 25-31) is in EMERG, the controller is in the circuit and the alarm condition should clear. (1 - 3 above would apply)

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

Examination Outline Cross-reference:	Level	RO	SRO
295008 (APE 8) High Reactor Water Level / 2	Tier #	1	------
AK1.02 (10CFR 55.41.10)	Group #	2	------
Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to HIGH REACTOR WATER LEVEL:	K/A #	295008AK1.02	
• Component erosion/damage	Importance Rating	3.1	------

Proposed Question: **# 21**

Unit 2 is operating at 100% RTP when an event occurs resulting in the following conditions:

- REACTOR LEVEL ABNORMAL (2-9-5A, Window 8) alarms
- Reactor Water Level is (+) 50 inches, rising



Given the conditions above, which **ONE** of the following completes the statements below in accordance with 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High / Low?

A manual SCRAM **(1)** required.

The SRVs will potentially fail **(2)** following discharge of water or two-phase flow.

A. (1) is
(2) OPEN

B. (1) is
(2) CLOSED

C. (1) is NOT
(2) OPEN

D. (1) is NOT
(2) CLOSED

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with the given ARP, the alarm annunciates when Reactor Water Level lowers to less than or equal to 27 inches or when it rises to greater than or equal to 39 inches. The stem states Reactor Water Level is (+) 50 inches, rising and the ARP states that Main Turbine/RFPT automatically trip and subsequent SCRAM occur if greater than or equal to 26% Reactor Power when Reactor Water Level reaches (+) 55 inches. The given ARP references 2-AOI-3-1, which states that a manual SCRAM is required when Reactor Water Level approaches (+) 55 inches. Second part, in accordance with 2-AOI-3-1, the SRVs will potentially fail OPEN following discharge of water or two-phase flow which occurs when Reactor Water Level rises and left unattended.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible if the candidate concludes that the High Reactor Water Level condition could bind the SRV CLOSED.

C INCORRECT: First part is incorrect but plausible if the candidate concludes that a manual SCRAM is not required until Reactor Water Level actually reaches (+) 55 inches or since an automatic SCRAM occurs when Reactor Water Level lowers to 2 inches. 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 operational implications and/or cause and effect relationships of component erosion/damage specifically Main Turbine/RFPT as it relates to High Reactor Water Level. This question is rated as C/A due to the requirement to correctly assemble given abnormal plant conditions as it relates to the Reactor Water Level. 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): 2-ARP-9-5A, Rev. 65 (Attach if not previously provided)
2-AOI-3-1, Rev. 28
2-OI-99, Rev. 101

Proposed references to be provided to applicants during examination: REACTOR LEVEL ABNORMAL (2-9-5A, Window 8)

Learning Objective: OPL171.026, Obj. 9 (As available)

Question Source:

Bank #	
Modified Bank #	BFN NRC 1306 #21
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2013

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 21

In accordance with 1-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low, which ONE of the following identifies a consequence of allowing water level to rise to the point that water enters the main steam lines?

A. The safety relief valves could fail to OPEN

B. The safety relief valves could fail to CLOSE

C. RCIC damage could occur

D. HPCI damage could occur

Answer: **B**

Excerpt from 2-ARP-9-5A:

REACTOR
WATER LEVEL
ABNORMAL
2-LA-3-53

8

(Page 1 of 1)

Sensor/Trip Point:
RFW Control System
2-RLY-003-0053LL ≤ 27 inches
2-RLY-003-0053LH ≥39 inches
Average of valid Narrow Range Level signals from RFW Control System. Signals originate from 2-LT-3-53, 3-60, 3-206 and 3-253.

Sensor Location:
2-RLY-003-0053LL in Panel 2-9-97 (Unit 2 Control Room)
2-RLY-003-0053LH in Panel 2-9-97 (Unit 2 Control Room)
2-LT-3-53 on Panel 25-5B (Rx Bldg El 593', R10-S)
2-LT-3-60 on Panel 25-6B(Rx Bldg El 593', R12-P)
2-LT-3-206 on Panel 2-925-426(Rx Bldg El 593', R10-T)
2-LT-3-253 on Panel 25-6D(Rx Bldg El 593', R12-R)

Probable Cause:
A. Reactor water level high or low.
B. Malfunction of sensor.

Automatic Action:
None (Reactor scram on low level at +2 in)
(Main Turbine/RFPT trip and subsequent scram if greater than or equal to 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 2-XR-3-53 recorder, 2-LI-3-53, 2-LI-3-60, 2-3-206 and 2-LI-3-253 on Panel 2-9-5.
B. **IF** alarm is valid, **THEN**
REFER TO 2-AOI-3-1 or 2-OI-3.
C. **IF** 2-LI-3-53, 2-LI-3-60, 2-LI-3-206 and 2-LI-3-253 has failed or is invalid, **THEN**
With SRO permission, **BYPASS** the affected level instrument.
REFER TO 2-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:

2-45E620-6	2-729E895-10	2-47E610-46-1
2-729E895-1	2-AOI-3-1	2-OI-3

Excerpts from 2-AOI-3-1:

BFN Unit 2	Loss of Reactor Feedwater or Reactor Water Level High/Low	2-AOI-3-1 Rev. 0028 Page 4 of 17

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 2-3-208A, 208B, 208C and 208D):

1. MAIN TURBINE SHUTDOWN (2-XA-55-8A, Window 11)
2. RFPT TRIPPED (2-XA-55-6C, Window 29)
3. RFPT A ABNORMAL (2-XA-55-6C, Window 1)
4. RFPT B ABNORMAL (2-XA-55-6C, Window 8)
5. RFPT C ABNORMAL (2-XA-55-6C, Window 15)

B. Annunciator HPCI TURBINE TRIPPED (2-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 2-3-208B and 208D).

C. Annunciator REACTOR WATER LEVEL ABNORMAL (2-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 (2-XA-55-4A, Window 35)
2. RECIRC LOOP B FLOW LIMITER ENFORCING (2-XA-55-4B, Window 35)
3. RFP DISCH FLOW LOW (2-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(2-XA-55-5B, Window 24)
2. RFWCS FAILED TO MANUAL (2-XA-55-6C, Window 21)
3. RFWCS GROSS FAILURE (2-XA-55-6C, Window 7)
4. RFWCS INPUT FAILURE (2-XA-55-6C, 14)
5. RFWCS TROUBLE (2-XA-55-6C, Window 28)

2.0 SYMPTOMS (continued)

I. Possible mismatch between Feedwater Line Flows (FI -3-78A, FI-3-78B) during HPCI or RCIC inadvertent injections.

J. Possible Reactor power rise due to improved moderation from 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 2-3-208A, 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 2-3-208A, 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 individual RFP < 16%.
2. Reactor Scram.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** Condensate Booster Pump(s) OR Reactor Feedwater Pump(s) low suction pressure annunciators are in alarm due to loss of Condensate Pump(s), **THEN**

PERFORM the following:

[1.1] **DEPRESS** RECIRC PUMPS MID POWER RUNBACK push-button, 2-HS-68-43.

[1.2] **CHECK** reactor power lowers to 66% or as directed by Unit SRO.

[2] **IF** RPV water level approaches +2 inches, **THEN**

INITIATE a manual reactor SCRAM.

[3] **IF** RPV water level approaches +55 inches, **THEN**

INITIATE a manual reactor SCRAM.

4.2 Subsequent Actions

CAUTIONS

1) [NRC/C] Operations outside of 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 for required actions and monitoring to be performed during power reduction. [NCO 940245001]
2) Monitor and maintain reactor feed pump turbine (RFPT) speed to less than or equal to 5800 rpm.
3) Nuclear Heat Balance is only accurate when reactor water level is stable. During periods when reactor water level is changing, APRMs should be utilized to monitor reactor power.

[1] **ENSURE** applicable automatic actions.

6.0 HIGH REACTOR WATER LEVEL

CAUTIONS

The following could lead to steamline break or other malfunction if water level is left unattended:

1) The added dead weight and potential seismic loads placed the main steam line and its supports should main steamline be flooded.
2) The loads placed on main steam lines as a result of potential for rapid collapse of steam voids resulting in water hammer.
3) The potential for safety relief valves sticking open following discharge of water or two-phase flow.
4) The potential inoperability of the main steamline isolation valves (MSIVs), reactor feed pump turbine (RFPT) valves, main turbine stop or bypass valves from the effects of water or two-phase flow.

[1] **ENSURE** HPCI or RCIC are not inadvertently injecting into Reactor (if Reactor pressure < 450 psi, RHR, Core Spray and/or Condensate & Feedwater could be injecting).

- **STOP** injection by tripping pumps or closing discharge valves, as appropriate.

Excerpt from 2-OI-99:

Attachment 2
(Page 2 of 2)

Unit 2 Reactor Scram Initiation Signals

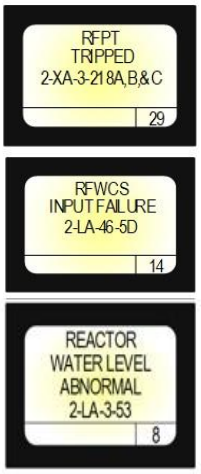
	Scram	Setpoint	Bypass
10.	OPRM TRIP	Any one of four algorithms, period, growth, amplitude or CDA exceeds its trip value setpoint for an operable OPRM cell.	Reactor is NOT operating in the AUTO ENABLE Region of the Power/Flow Map or ABSP Enabled
11.	Low RPV Water Level (Level 3)	+2.0"	NA
12.	Hi RPV Pressure	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

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

Examination Outline Cross-reference:	Level	RO	SRO
295009 (APE 9) Low Reactor Water Level / 2 **AA2.02** (10CFR 55.41.10) Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: • Steam flow/feed flow mismatch	Tier #	1	------
	Group #	2	------
	K/A #	295009AA2.02	
	Importance Rating	3.9	------

Proposed Question: **# 22**

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



- RFPT TRIPPED (2-9-6C, Window 29) alarms
- RFWCS INPUT FAILURE (2-9-6C, Window 14) alarms
- REACTOR WATER LEVEL ABNORMAL (2-9-5A, Window 8) alarms
- Operator reports 2C RFPT tripped
- On Panel 2-9-5, 2-FI-46-3, Main Steam Line 'C' Flow, indicates UPSCALE

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

Both Recirculation Pumps **(1)** automatically runback.

Reactor Feedwater Control System (RFWCS) will **(2)**.

A. (1) will
(2) automatically swap to Single Element

B. (1) will
(2) remain in Three Element

C. (1) will NOT
(2) automatically swap to Single Element

D. (1) will NOT
(2) remain in Three Element

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that the Three Element criteria for RFWCS is complex and easily and often confused by candidates. If associated permissives are misunderstood, a candidate could conclude that the UPSCALE condition results in an automatic swap to Single Element. RFWCS control will automatically transfer Single Element when more than two steam line flows declared invalid or 1or 2 steam flows are invalid AND Main Turbine First Stage Pressure is invalid.

B **CORRECT**: (*See attached*) Given that the stem states that 2C RFPT tripped and the REACTOR WATER LEVEL ABNORMAL is in alarm, it is indicated that Reactor Water Level has lowered to less than or equal to (+) 27 inches. In accordance with 2-OI-68, Reactor Recirc System, Att. 2, the automatic runback criteria are an individual RFP discharge flow less than 16% AND Reactor Water Level less than 27 inches. This results in the 75% Limiter automatically running back both Recirc Pumps to 1130 RPM.
Second part, in accordance with the given ARP for RFWCS INPUT FAILURE, 2-FI-46-3, Main Steam Line 'C' (given as UPSCALE), is listed as an input to the alarm. This single input failure is automatically bypassed and indicated on Panel 2-9-5 by an amber light. 2-OI-3, RFW System, lists the permissives for RFWCS to be in Three Element as a Valid Total Steam Flow (Four good steam line flows, or at least two good steam line flow signals and a valid Main Turbine First Stage Pressure signal). The criteria are met with at least two good steam line flows, therefore RFWCS will remain in Three Element.

C INCORRECT: First part is incorrect but plausible in that the Recirc Flow Control 28% Limiter Automatic Runback occurs when **total** FW Flow is less than 16% for 15 seconds. This runback does not have any associated steam flow criteria permissives. 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 determine and/or interpret steam flow/feed flow mismatch as it applies to Low 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. 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.
(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-3, Rev. 173 (Attach if not previously provided)
2-OI-68, Rev. 173
2-ARP-9-5A, Rev. 65
2-ARP-9-6C, Rev. 29

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

Proposed references to be provided to applicants during examination: RFPT TRIPPED (2-9-6C, Window 29)
RFWCS INPUT FAILURE (2-9-6C, Window 14)
REACTOR WATER LEVEL ABNORMAL (2-9-5A, Window 8)

Learning Objective: OPL171.012 Obj. 3, 5b (As available)

Question Source:
Bank # _____
Modified Bank # BFN NRC 2204 #44 (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

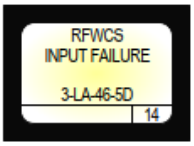
Comments:

Copy of Bank Question:

Proposed Question: **# 44**

Unit 3 is operating at 100% RTP with the following conditions:

- RFWCS INPUT FAILURE (3-9-6C, Window 14) alarms
- The OATC reports 3-FI-3-78A, FW LINE A FLOW on Panel 3-9-5 indicates UPSCALE



Given the conditions above, 3-FI-3-78A, FW LINE A FLOW ___**(1)**___ automatically bypass.

Once FW LINE A FLOW is bypassed, Feedwater Level Control will be in ___**(2)**___ element control.

A. (1) will
(2) three

B. (1) will
(2) single

C. (1) will NOT
(2) three

D. (1) will NOT
(2) single

Proposed Answer: **A**

Excerpt from 2-ARP-9-5A:

BFN Unit 2	Panel 9-5 2-XA-55-5A	2-ARP-9-5A Rev. 0065 Page 13 of 48

REACTOR
WATER LEVEL
ABNORMAL
2-LA-3-53

8

(Page 1 of 1)

Sensor/Trip Point:
RFW Control System
2-RLY-003-0053LL ≤ 27 inches
2-RLY-003-0053LH ≥39 inches
Average of valid Narrow Range Level signals from RFW Control System. Signals originate from 2-LT-3-53, 3-60, 3-206 and 3-253.

Sensor Location:
2-RLY-003-0053LL in Panel 2-9-97 (Unit 2 Control Room)
2-RLY-003-0053LH in Panel 2-9-97 (Unit 2 Control Room)
2-LT-3-53 on Panel 25-5B (Rx Bldg El 593', R10-S)
2-LT-3-60 on Panel 25-6B(Rx Bldg El 593', R12-P)
2-LT-3-206 on Panel 2-925-426(Rx Bldg El 593', R10-T)
2-LT-3-253 on Panel 25-6D(Rx Bldg El 593', R12-R)

Probable Cause:
A. Reactor water level high or low.
B. Malfunction of sensor.

Automatic Action:
None (Reactor scram on low level at +2 in)
(Main Turbine/RFPT trip and subsequent scram if greater than or equal to 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 2-XR-3-53 recorder, 2-LI-3-53, 2-LI-3-60, 2-3-206 and 2-LI-3-253 on Panel 2-9-5.
B. **IF** alarm is valid, **THEN**
REFER TO 2-AOI-3-1 or 2-OI-3.
C. **IF** 2-LI-3-53, 2-LI-3-60, 2-LI-3-206 and 2-LI-3-253 has failed or is invalid, **THEN**
With SRO permission, **BYPASS** the affected level instrument.
REFER TO 2-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:

2-45E620-6	2-729E895-10	2-47E610-46-1
2-729E895-1	2-AOI-3-1	2-OI-3

Excerpts from 2-ARP-9-6C:

BFN Unit 2	Panel 9-6 2-XA-55-6C	2-ARP-9-6C Rev. 0029 Page 39 of 46

RFPT TRIPPED

2-XA-3-218A,B,&C

29

(Page 1 of 1)

Sensor/Trip Point:

Alarm is activated through a diode blocking circuit internal to the sequential events recorder (SER) by SER Point #074, 075, or 076 via 2-PS-3-218A (B)(C).

Sensor Location: Panel 9-37, Elevation 593', Auxiliary Instrument Room

Probable Cause:

A. Condenser vacuum at 7" Hg. (2 - out - of 3 logic).
B. Suction valve NOT full open.
C. RFPT electrical overspeed at 105%(6150 rpm).
D. RFPT mechanical overspeed at 110%(6500 rpm).
E. RFP A low suction pressure at 150 psig lowering, for 80 seconds.
F. RFP B low suction pressure at 150 psig lowering for 100 seconds.
G. RFP C low suction pressure at 150 psig lowering for 120 seconds.
H. RFP bearing oil pressure at 6 psig.
I. RFPT bearing oil pressure at 4 psig.
J. Excessive thrust bearing wear at 40 psig active or inactive face.
K. High Reactor water level at +55".
L. Manual trip.
M. Loss of speed feedback signal from both MPUs.

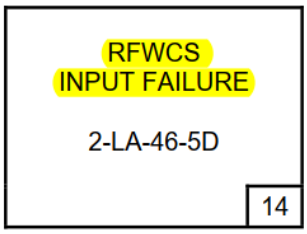
Automatic Action: None

Operator Action:

A. **ENSURE** reactor power is within the capacity of operating RFPs.
B. **CHECK** core limits.
C. **WHEN** RFPT coasts down to zero speed, unless RFPT is rolling on minimum flow, **THEN**
ENSURE turning gear motor starts and engages.
D. **REFER TO** 2-AOI-3-1 or 2-OI-3, Section 8.1.

References: 2-45E620-7 2-45E612-3 2-47E610-3-1

BFN Unit 2	Panel 9-6 2-XA-55-6C	2-ARP-9-6C Rev. 0029 Page 21 of 46



(Page 1 of 2)

Sensor/Trip Point:

2-RLY-46-5D - A process input to the RFW Control System declared Bad or Invalid. A signal is declared bad when it has failed or is off scale. A signal is declared invalid when it fails the validation process described in 2-OI-3 Illustration 8.

Sensor Location: Panel 2-9-97 (Behind Panel 2-9-5).

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, 2-FI-3-20,13,6 (Panel 2-9-6).
   - RFW Line Flows, 2-FI-3-78A, 78B (Panel 2-9-5).
   - Main Steam Line Flows, 2-FI-46-1, 2, 3, 4(Panel 2-9-5).
   - WR Reactor Pressure 2-PI-3-54, 61,207 (Panel 2-9-5).
   - Reactor Water Level 2-LI-3-53, 60,206,253 (Panel 2-9-5).

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 2-OI-3).

Continued on Next Page

Excerpt from 2-OI-68:

Attachment 2
(Page 4 of 4)

Recirc Flow Control System Manual Runback Controls

An automatic runback always takes precedence over a manual runback. If an automatic runback occurs while a manual runback is in progress, the automatic runback signal will be sent to the recirc drive. The manual runback will be locked out and the manual runback pushbutton backlight will stop blinking. After a 30 second time delay, the manual runback may be reinitiated by depressing the applicable pushbutton a second time. Automatic runback setpoints to the 28% and 75% Limiters are as follows:

AUTOMATIC RUNBACKS		
Limiter	**Runback Setpoint**	**PUMP Speed After Runback**
28%	Total Feedwater Flowless than 16% for 15 seconds OR recirc pump discharge valve less than 90% open	480 RPM
75%	Individual RFP discharge flow less than16% and Reactor Water Level less than 27 inches OR Reactor scram	1130 RPM

FAILURE MECHANISMS:

During a loss of Unit Preferred concurrent with a loss of ICS power, all of the manual runbacks, as well as the Recirc Flow Control System, will be rendered inoperative. A loss of any one of these two power supplies will result in the RECIRC FLOW SYSTEM TROUBLE ALARM, 2-XA-55-4A, Window 23, to alarm. The lights for those manual runback pushbuttons will become extinguished.

A loss of the total steam flow signal to the Recirc Flow Control System will render the high power manual runback and the mid power runbacks inoperative. The lights for those manual runback pushbuttons will become extinguished. A loss of this signal will result in RECIRC FLOW SYSTEM TROUBLE ALARM (2-XA-55-4A, Window 23) and the RFWCS INPUT FAILURE annunciation (2-XA-55-6C, window 14) to alarm.

Excerpts from 2-OI-3:

3.0 PRECAUTIONS AND LIMITATIONS

A. During normal power operations with three RFPs in automatic control on master controller (2-LIC-46-5), the following is suggested lineup for Panel Display Stations on Panel 2-9-5:

1. RFW START-UP LEVEL CONTROL, 2-LIC-3-53, in Manual (amber light illuminated) with Column 3 (CO) selected.
2. RFPT 2A(2B)(2C) SPEED CONTROL, 2-SIC-46-8(9)(10), in Auto (blue light illuminated) with Column (SP) selected.
3. REACTOR WATER LEVEL CONTROL, 2-LIC-46-5, in Auto (blue light illuminated) with Column 2 (SP) selected.

B. The manufacturer of Panel Display Stations furnished one type of display with different indications for AUTO/MANUAL Indicator. They are blue light illuminated for Automatic and amber light illuminated for Manual. Attachments 1, 2 and 3 can be referred to for general information on Panel Display Stations located on Panel 2-9-5.

C. All of the following permissives are required to be met for RFW Control System to allow Three Element control:

1. Reactor Water Level Control, 2-LIC-46-5, in Auto (blue light illuminated) and at least one RFPT Speed Control PDS in Auto.
2. A valid Total Steam Flow signal exists.
3. A valid Total Feedwater Line Flow signal exists.
4. Total Steam Flow greater than 16%.
5. At least one valid Narrow Range Level signal exists.

D. Attachment 4 has general information on Single and Three Element control.

E. Programmed scram response is normally enabled during power operations. Scram response can be inhibited with Unit SRO approval. Attachment 5 has general information on this feature.

F. Attachment 10 has information concerning RFP/RFPT Vibration Monitoring.

G. Periodically monitor level instruments 2-LI-3-208A, 208B, 208C, and 208D during normal operations and level transients. These instruments provide high reactor water level trips for Main Turbine and RFPTs at 55 inches, as sensed by 2-LI-3-208A and 2-LI-3-208C **OR** 2-LI-3-208B and 2-LI-3-208D (two out of two taken once logic).

Attachment 8
(Page 2 of 7)

RFWCS Instrumentation

1.4 Failure Mechanisms

When the RFWCS declares a level signal bad or invalid, the level instrument is automatically bypassed (amber light illuminates on the instrument). This will cause annunciation RFWCS INPUT FAILURE (2-XA-55-9C, Window 14) to alarm.

When all four level instruments become bad or invalid, the RFW Control System will automatically transfer to SINGLE ELEMENT control and the Reactor Water Level Control PDS, 2-LIC-46-5, trips to MANUAL mode. The annunciation RFWCS FAILED TO MANUAL (2-XA-55-6C, Window 21) will alarm when this condition occurs.

2.0 MAIN STEAM LINE FLOW AND TURBINE FIRST STAGE PRESSURE

2.1 Components

2-FI-46-1

2-FI-46-2

2-FI-46-3

2-FI-46-4

2.2 Description

The instruments are located on Panel 2-9-5 but are manually bypassed at the Engineering Work Station in Unit 1/2 Computer Room. Steam line flows are compensated for changes in Reactor pressure. Flow signals are sent to the RFWC to produce a Total Steam Flow signal. Total Steam Flow is used for the following:

- Correcting level demand signal for RFWCS when system is in THREE ELEMENT control.
- Initiating Rod Worth Minimizer Low Power Alarm Point (< 27% rated flow) and Enable setpoint (< 22% rated flow).

Attachment 8
(Page 3 of 7)

RFWCS Instrumentation

2.3 System Operation

The RFW Control System will use a Main Steam Line Flow signal provided the system determines signal to be good and valid. A **GOOD** steam flow signal is one that has **NOT** failed and is on scale. A **VALID** steam flow signal is determined by a validation process described in the next paragraph.

Total Steam Flow is validated by comparison to Turbine First Stage Pressure signal steam flow equivalent. If those two signals differ by more than 0.8 x 10^6 lbm/hr for more than 3 seconds AND Total Steam Flow is greater than 16%, then each of the Steam line flow signals will be compared to average steam line flow. If any individual signal deviates from the average by more than 1.2 x 10^6 lbm/hr, then the signal is declared invalid and is automatically bypassed.

2.4 Failure Mechanisms

When the RFWCS declares a Steam Line Flow signal bad or invalid, the flow instrument is automatically bypassed. The amber light on the affected instrument will illuminate and the RFWCS INPUT FAILURE annunciation (2-XA-55-6C, Window 14) will alarm.

When the RFWCS declares the Turbine First Stage Pressure signal to be bad, the RFWCS INPUT FAILURE annunciation (2-XA-55-6C, Window 14) will alarm.

The following events will automatically transfer RFWCS control to SINGLE ELEMENT:

- Total Steam Flow < 16%.
- More than two Steam line flows declared invalid.
- If 1 or 2 steam line flows are invalid AND Turbine First Stage Pressure is invalid.

When Total Steam Flow to Turbine First Stage Pressure deviates by more than 0.8 x 10^6 lbm/hr for more than 30 seconds, MAIN STEAM LINE VS STEAM FLOW MISMATCH annunciation (2-XA-55-5B, Window 24) will alarm.

8.31 Transferring RFWCS Between Single Element and Three Element

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** THREE ELEMENT pushbutton, 2-HS-46-6/3, and **CHECK** green backlight for pushbutton illuminates.

[1.4] **CHECK** green backlight for SINGLE ELEMENT pushbutton, 2-HS-46-6/1, is extinguished.

[1.5] **CHECK** Reactor water level stable.

[2] **IF** desired to transfer level control from Three Element to Single Element, **THEN** (Otherwise N/A):

[2.1] **DEPRESS** SINGLE ELEMENT pushbutton, 2-HS-46-6/1, and **CHECK** green backlight for pushbutton illuminates.

[2.2] **CHECK** green backlight for THREE ELEMENT pushbutton, 2-HS-46-6/3, is extinguished.

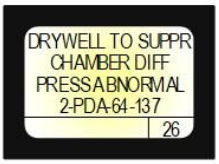
[2.3] **CHECK** Reactor water level stable.

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

Examination Outline Cross-reference:	Level	RO	SRO
295010 (APE 10) High Drywell Pressure / 5 **AA1.11** (10CFR 55.41.7) Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: • Drywell/suppression chamber differential pressure (Mark I, II)	Tier #	1	------
	Group #	2	------
	K/A #	295010AA1.11	
	Importance Rating	3.6	------

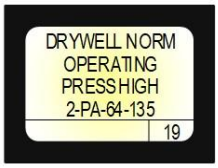
Proposed Question: **# 23**

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



- DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL (2-9-3B, Window 26) alarms
- Drywell to Suppression Chamber Differential Pressure (DP) is 1.40 psid

Subsequently,



- DRYWELL NORMAL OPERATING PRESSURE HIGH (2-9-3B, Window 19) alarms
- The crew is monitoring for a possible Drywell Control Air System leak

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

The Drywell DP Compressor **___(1)___** receive an automatic **STOP** signal.

The above 2-9-3B, Window 19, alarm setpoint is **___(2)___**.

A. (1) did
(2) 1.6 psig

B. (1) did
(2) 1.96 psig

C. (1) did NOT
(2) 1.6 psig

D. (1) did NOT
(2) 1.96 psig

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with the given ARP for DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL, which alarms at either 1.13 psid low OR 1.34 psid high, the Drywell DP Compressor automatically starts on lowering DP (1.15 psid) and stops on rising DP high (1.28 psid). Second part, in accordance with the given ARP for DRYWELL NORMAL OPERATING PRESSURE HIGH (2-9-3B, Window 19), the alarm setpoint is 1.6 psig.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that DRYWELL PRESSURE ABNORMAL (2-ARP-9-5A, Window 31) alarms at 1.96 psig.

C INCORRECT: First part is incorrect but plausible if the candidate fails to correlate that the Drywell DP Compressor does have an automatic start/stop signal that is within the parameter band for 2-9-3B, Window 26, alarm setpoints. Additionally, 2-SR-2, Instrument Checks and Observations, for Drywell – Suppression Chamber DP limit is ≥ 1.1 psid and ≤ 1.33 psid, therefore candidate could conclude that the compressor did not stop. 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 operate and/or monitor Drywell/Suppression Chamber DP 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):

2-ARP-9-3B, Rev. 40	(Attach if not previously provided)
2-ARP-9-5B Rev. 33	
2-SR-2, Rev. 95	
2-SIMI-64B, Rev. 52	

Proposed references to be provided to applicants during examination:

DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE ABNORMAL (2-9-3B, Window 26)
DRYWELL NORMAL OPERATING PRESSURE HIGH (2-9-3B, Window 19)

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

Learning Objective: OPL171.016 Obj. 10c (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:

Excerpts from 2-ARP-9-3B:

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0040 Page 30 of 39

DRYWELL TO SUPPR CHAMBER. DIFF PRESS ABNORMAL
2-PDA-64-137
26

(Page 1 of 1)

Sensor/Trip Point:

2-PDS-64-137A or 64-138A	1.13 psid low
2-PDS-64-137C or 64-138C	1.34 psid high

Sensor Location:
Panel 9-19
Elevation 593'
Auxiliary Instrument Room
Column R-8 P-LINE

Probable Cause:
A. Drywell DP compressor malfunction.
B. Excessive N2 makeup to the drywell/torus.
C. Excessive venting from drywell/torus.
D. Sensor malfunction.

Automatic Action:
A. 2-PDS-64-137B and/or 138B.
1. Starts compressor on lowering DP.
2. Stops compressor on rising DP.

Operator Action:
A. **CHECK** alarm by checking Drywell to Suppression Chamber DP.
B. **REFER TO** 2-AOI-64-1.
C. **REFER TO** Tech Spec Section 3.6.2.6.

References:
2-45E620-8 2-45E777-21 2-47E610-64-2
Technical Specifications
3.6.2.6

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0040 Page 23 of 39

DRYWELL NORM OPERATING PRESS HIGH
2-PA-64-135
19

Sensor/Trip Point:
2-PS-64-135
or
2-PS-64-136
1.6 psig

(Page 1 of 1)

Sensor Location:
Panel 9-19
Elevation 593'
Auxiliary Instrument Room

Probable Cause:
A. Excessive N_2 makeup.
B. Low pressure front moving through area.
C. Drywell DP compressor malfunction.
D. Drywell blowers failure.
E. Steam or water leak inside drywell.
F. Possible Drywell Control Air System In leakage.
G. Sensor malfunction.
H. Reactor Start-up.

Automatic Action:
None

Operator Action:
A. **CHECK** drywell pressure and temperature for rise.

B. **CHECK** weather report for atmospheric pressure.

C. **IF** Drywell DP Compressor is running, **THEN** **STOP** compressor.

D. **CHECK** N_2 makeup valves to Suppression Chamber and Drywell closed.

E. **CHECK** Drywell Control Air System Flow Elements 2-FIQ-032-00092 (Rx Bldg 565' R10-S) and 2-FIQ-032-0075 (Rx Bldg 565' R20-T0) < 0.5 SCFM.

F. **IF** pressure rise is due to normal startup, **THEN** **REFER TO** 2-OI-64 for normal venting instructions.

G. **IF** Drywell pressure is high, **THEN** **REFER TO** 2-AOI-64-1.

References:
2-45E620-8 2-45E777-21

Excerpt from 2-SIMI-64B: Supports ARP setpoints

BFN Unit 2	Primary Containment Systems Scaling and Setpoint Documents	2-SIMI-64B Rev. 0052 Page 219 of 322

ATTACHMENT 5

INST NO: 2-PDS-64-137B	BROWNS FERRY NUCLEAR PLANT	SCALING INFORMATION SHEET
VENDOR NO: NA	SCALING AND SETPOINT DOCUMENT	SHEET: 7 OF 9

FUNCTION: DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

LOCATION: Unit 2 Auxiliary Instrument Room, Panel 2-9-19

SETTINGS	REFERENCE	ACCURACY	REFERENCE
1.15 PSI DEC 1.28 PSI INC	MEL	±0.5% SPAN	BFN-VTM-R290-0060

CALIBRATION RANGE: 0-2 PSID	ACTION: DIRECT
INPUT 1: 4-20 mA	OUTPUT 1: CONTACT CLOSE DEC dP,
INPUT 2: NA	OPEN INC dP
	OUTPUT 2: CONTACT CLOSE DEC dP,
	OPEN INC dP

CALCULATIONS/SPECIAL NOTES:

NOTE 1: FAIL SAFE ALARM UNIT. RELAYS DEENERGIZE ON DECREASING dP AND CONTACTS CLOSE TO START DRYWELL COMPRESSOR AND OPEN DRYWELL COMPRESSOR DISCHARGE VALVE 2-FCV-64-140.

NOTE 2 RELAYS ENERGIZE ON INCREASING dP AND CONTACTS OPEN TO STOP DRYWELL COMPRESSOR AND CLOSE DRYWELL COMPRESSOR DISCHARGE VALVE 2-FCV-64-140.

NOTE 3 2-PDS-64-137B AND 2-PDS-64-138B OUTPUT CONTACTS ARE WIRED IN PARALLEL.

NOTE 4 When setting up a new switch verify that switch is configured for Low Level Fail Safe Mode. The switch should have a link connected between points "A" and "B" and between points "C" and "D". Reference VTM-R290-0060 as necessary. [BFNPER970237].

SETPOINT CALCULATIONS

$$\left(\frac{1.15\,\text{PSI}}{2.0\,\text{PSI}}\right)(16\,\text{mA})+4\,\text{mA}=13.04\,\text{mA} \qquad \left(\frac{1.28\,\text{PSI}}{2.0\,\text{PSI}}\right)(16\,\text{mA})+4\,\text{mA}=14.24\,\text{mA}$$

REFERENCES:	DRAWINGS:
MANUFACTURER: ROBERTSHAW	2-47E610-64-2
MODEL: 553-B2-B2-B1	2-45E777-21
CONTRACT#: 821125	2-791E290
VENDOR MANUAL: BFN-VTM-R290-0060	

PREPARED BY: Benny Freeman

Excerpt from 2-ARP-9-5B:

BFN Unit 2	Panel 9-5 2-XA-55-5B	2-ARP-9-5B Rev. 0033 Page 36 of 43

DRYWELL PRESSURE ABNORMAL
2-PA-64-56
31

(Page 1 of 1)

Sensor/Trip Point:

2-PS-64-56E	1.96 psig rising
2-PS-64-56F	0.1 psig lowering

Sensor Location: Panel 25-5B
Elevation 593
Col R-3 S-LINE

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** 2-AOI-70-1.
C. **REFER TO** 2-AOI-64-1.

References:

2-45E620-6	2-47E610-64-1	2-730E915-17
2-AOI-70-1	2-AOI-64-1	

Excerpt from 2-SR-2:

BFN Unit 2	Instrument Checks and Observations	2-SR-2 Rev. 0095 Page 42 of 161

Attachment 2
(Page 21 of 90)
Surveillance Procedure Data Package - Modes 1, 2, & 3

TABLE 1.17 DRYWELL - SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE **DAY SHIFT** WEEK: _____ to _____

APPLICABILITY: Mode 1 with THERMAL POWER > 15%RTP
Readings are required at all times.

Surveillance Requirements: 3.6.2.6.1 Technical Requirements Manual 3.3.5.1

LOCATION: Panel 2-9-3

	TIME	2-PDI-64-137 (psid)	2-PDI-64-138 (psid)	LIMITS (AC)	MAX DEV (AC)	Review Initials: UO	Unit SRO
Friday	0800			≥ 1.1 psid & ≤ 1.33 psid (Note 1)	0.10 psid		
Saturday	0800						
Sunday	0800						
Monday	0800						
Tuesday	0800						
Wednesday	0800						
Thursday	0800						

(1) The Drywell-Suppression Chamber Differential Pressure should not exceed 1.33 psid.

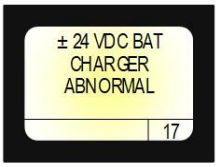
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295004 (APE 4) Partial or Complete Loss of D.C. Power / 6	Tier #	1	------
G2.1.20 (10CFR 55.41.10)	Group #	1	------
Ability to interpret and execute procedure steps	K/A #	295004G2.1.20	
	Importance Rating	4.6	------

Proposed Question: **# 24**

Unit 2 is in **MODE 2** with a startup in progress when the following occurs:

At 0700 a loss of I&C Bus 'A' occurs with the following:



- +/- 24 VDC BATTERY CHARGER ABNORMAL (2-9-5B, Window 17) alarms
- Control Bay AUO reports:
  - 2-EI-283-0000A1-2, 24V Battery Charger A1-2 Voltmeter, indicates 22 VDC
  - 2-EI-283-0000A2-2, 24V Battery Charger A2-2 Voltmeter, indicates 22 VDC
- **NO** other Operator actions have been taken

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

At 1100, in accordance with the given ARP, a loss of associated IRMs **___(1)___** occurred and a **HALF** SCRAM **___(2)___** automatically occurred.

A. (1) has
   (2) has NOT

B. (1) has
   (2) has

C. (1) has NOT
   (2) has NOT

D. (1) has NOT
   (2) has

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible if the candidate fails to correlate that the long-term loss (greater than three hours) of I&C Bus 'A' power supply to the Neutron Monitoring battery chargers, and that the corresponding degrading voltage results in a loss of half of IRM detectors and a subsequent half SCRAM. Candidate could misapply the 0-OI-57D, 7.11 Caution that states the battery capable of supplying respective loads up to 3 hours.

B **CORRECT:** *(See attached)* In accordance with 2-AOI-57-5A, Loss of I&C Bus A, a loss of I&C Bus 'A', results in a de-energization of +/- 24 VDC Neutron Monitoring Battery Chargers A1-2 and A2-2. Long-term (greater than three hours), this charger loss would result in a loss of battery capability for associated Division I IRMs, and subsequent protective actions. Second part, the given report from the AUO of 22 VDC indicates that the normal +/- 24 VDC supplied to Channel 'A' for IRMs is degrading (ARP states undervoltage is +/- 22 VDC). Over the provided elapsed time (four hours), the batteries would be beyond their three-hour capacity, resulting in half SCRAM from IRMs A, C, E, and G. Additionally, it is given that the Unit is in MODE 2, thus the REACTOR MODE SWITCH is not in RUN (Tech Spec 1.0). Therefore, IRM trip logic is not bypassed.

C INCORRECT: First part is incorrect but plausible in that the AC and DC Electrical Distribution System is quite complex and easily and often confused. The candidate could easily conclude that IRMs are powered from 48 VDC. 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 interpret and execute procedure steps as it relates to a partial or complete loss of DC power. 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 IRM and RPS trip logic.

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-57D, Rev. 203 (Attach if not previously provided)
2-AOI-57-5A, Rev. 63
2-OI-92A, Rev. 29
2-ARP-9-5B, Rev. 33
Unit 2 Tech Spec 1.0, Amend. 253
0-45E702-3, Rev. 67

Proposed references to be provided to applicants during examination: +/- 24 VDC BATTERY CHARGER ABNORMAL (2-9-5B, Window 17)

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

Learning Objective: OPL171.037 Obj. 4c (As available)

Question Source:

Bank #	
Modified Bank #	BFN NRC 2304 #14
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2023

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: **# 14**

Intermediate Range Monitor (IRM) drawers on Panel 9-12 are **POWERED** by **(1)**.

The IRM Detectors **OPERATE** at **(2)**.

A. (1) 24VDC
(2) 100VDC

B. (1) 24VDC
(2) 350VDC

C. (1) 120VAC
(2) 100VDC

D. (1) 120VAC
(2) 350VDC

Proposed Answer: **A**

Excerpts from 2-ARP-9-5B:

BFN Unit 2	Panel 9-5 2-XA-55-5B	2-ARP-9-5B Rev. 0033 Page 20 of 43

± 24 VDC BAT CHARGER ABNORMAL

17

(Page 1 of 2)

Sensor/Trip Point:

Chargers A1,A2,B1

Relays — K1 -120V AC Failure (I&C Bus A or B).
K2 -Undervoltage (±22V DC).
K3 -Overvoltage (± 29V DC).

CB-2 - Shunt trip.

Charger B2

AC Failure
Charger Failure
Low DC - (±22V DC)
High DC - (± 28.75V DC)
Shutdown - (± 29.3V DC)

Sensor Location: Elevation 593 in Battery Board Room 2.
Battery Charger A1-2, A2-2, B1-2 and B2-2.

Probable Cause:

Chargers A1,A2,B1
AC fail alarm - (I&C Bus A or B)
Undervoltage - (battery undervoltage, charger problem)
Overvoltage - (battery overvoltage, charger problem)
Shunt trip

Charger B2
AC fail alarm - (I&C Bus B or AC breaker tripped)
Charger Fail - (charger low current output)
Low DC - (battery undervoltage, charger problem)
High DC - (battery overvoltage, charger problem)
Shutdown - (battery overvoltage, charger problem)

Automatic Action: Loss of ± 24V DC Ch A (B) results in loss of IRM & SRM Div I (II), Reactor power indication on Panel 2-9-5 and 2-9-12, and half-scram if Rx Mode Sw. NOT in RUN.

Operator Action:

A. **IF** I&C Bus A(B) is deenergized, **THEN**
REFER TO 2-AOI-57-5A (5B).
B. **IF** I&C Bus A and B are energized, **THEN**
REQUEST IMs to check ± 24 VDC charger for problem cause.
C. **REFER TO** 0-OI-57D for proper alignment of + 24V Battery and Charger.
D. **REFER TO** 0-OI-57D for list of affected loads.

Continued on Next Page

± 24 VDC BAT CHARGER ABNORMAL, Window 17
(Page 2 of 2)

Operator Action: (Continued)

E. **NOTIFY** Unit SRO.

F. [NRC/C] **WHEN** ± 24V DC is restored, **THEN**
NOTIFY IMs functional tests are required to be performed on affected radiation monitors, SRMs and IRMs. [RPT 86-40]

G. **IF** half-scram is received, **THEN**
With SRO permission, **RESET** half-scram. **REFER TO** 2-OI-99.

H. **IF** alarm is invalid, **THEN**
REFER TO 0-OI-55.

NOTE
Other parameters will be masked from alarming while this alarm is sealed in.

I. **IF** this alarm is valid, frequently, **THEN**
MONITOR other parameters that input to it.

References:

2-45E620-6	0-45E703-2	2-AOI-57-5A (5B)
0-OI-57D	0-OI-55	

Excerpts from 0-OI-57D:

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0203 Page 147 of 399

5.10 Placing Unit 2 ± 24V DC Neutron Monitoring Battery A(B) in Service to Battery Board 2 (continued)

[4.3] **OBSERVE** the following Channel B indications for normal operation on Panel 12:

- Battery Channel B DC Voltmeters indicate greater than 24 Volts. (Both voltmeters)
- Battery Channel B DC Ammeters indicate greater than zero amps. (Both voltmeters)

NOTE

If Unit 2 ± 24V Neutron Monitoring Battery Charger Channel A(B) is in service in accordance with Section 5.13, Steps 5.10[5] through 5.10[9] need NOT be performed.

[5] **IF** the Reactor Mode switch is **NOT** in RUN, **THEN**

HAVE the Unit Operator reset Reactor Auto Scram Channel A (B).

[6] **ENSURE** the following equipment is in normal operation in accordance with 2-OI-90 Radiation Monitoring System, 2-OI-92 Source Range Monitoring, and 2-OI-92A Intermediate Range Monitoring:

[6.1] ± 24V Neutron Monitoring Channel A.

A. Panel 9-5

- IRM recorders A, C, E and G
- SRM Channel A and C indicators, recorders and period meters

B. Panel 9-10

- 2-RM-90-266A, OG POST-TREATMENT CH A RAD MON RTMR

5.10 Placing Unit 2 ± 24V DC Neutron Monitoring Battery A(B) in Service to Battery Board 2 (continued)

[6.2] ± 24V Neutron Monitoring Channel B.

A. Panel 9-5

- IRM recorders B, D, F and H
- SRM Channel B and D indicators, recorders and period meters

B. Panel 9-10

- 2-RM-90-265A, OG POST-TREATMENT CH B RAD MON RTMR

[7] **TURN** IRM and SRM drawers RESET switch and **CHECK** alarm lights extinguished on drawers.

[8] **PLACE** the ± 24V Neutron Monitoring Battery Chargers Channel A (B) in service in accordance with Section 5.13.

[9] [NRC/C] **NOTIFY** Instrument Maintenance that functional tests are required to be performed on SRMs and IRMs (N/A functional testing for the IRMs and the SRMs if the Unit is in RUN Mode and the IRMs and SRMs are inoperable). [NRC IE Inspect Follow-up Item 86-40]

[10] [NRC/C] **OBSERVE** readings on the affected radiation monitors return to approximately the same reading they were at prior to de-energizing (~ 25% as compared to the gas log or the chart recorders). [NRC IE Inspect Follow-up Item 86-40]

[11] **IF** battery requires it, **THEN**

HAVE Electrical Maintenance place ± 24V Neutron Monitoring Battery Chargers Channel A (B) on Equalizer charge upon completion of Section 5.13 (Otherwise N/A).

7.11 Removing Unit 2 ± 24V Neutron Monitoring Battery Charger Channel A(B) from Service (continued)

CAUTION

Battery voltage should be monitored frequently when the ± 24/v Neutron Monitoring Battery Charger is out of service. Low voltage on the effected channel results in a loss of SRM/IRM detectors powered from that channel and a 1/2 scram. The battery is capable of supplying connected loads for 3 hours.

TABLE 12

UNIT	CHANNEL A CHARGER	CHANNEL B CHARGER
2	A1-2	B1-2
2	A2-2	B2-2

[4] **IF** removing Battery Chargers A2-1,A2-2, or B2-1, **THEN**

PERFORM the following (Otherwise N/A):

[4.1] **OPEN** the AC CIRCUIT BREAKERS on the ± 24 Volt Neutron Monitoring Battery Chargers (applicable charger, see Table 12), by placing the breakers in OFF.

[4.2] **OPEN** the ± 24V DC CHARGER (Applicable units charger, see Table 11) TIE TO BAT BD. 2, 0-BKR-280-002/1201(1221), by placing breakers in OFF.

[4.3] **CHECK** amperage indicates zero on ± 24V Chargers (applicable charger, see Table 12).

Excerpt from Unit 2 Tech Spec 1.0:

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel(a) or Startup/Hot Standby	NA
3	Hot Shutdown(a)	Shutdown	> 212
4	Cold Shutdown(a)	Shutdown	≤ 212
5	Refueling(b)	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Excerpt from 2-OI-92A:

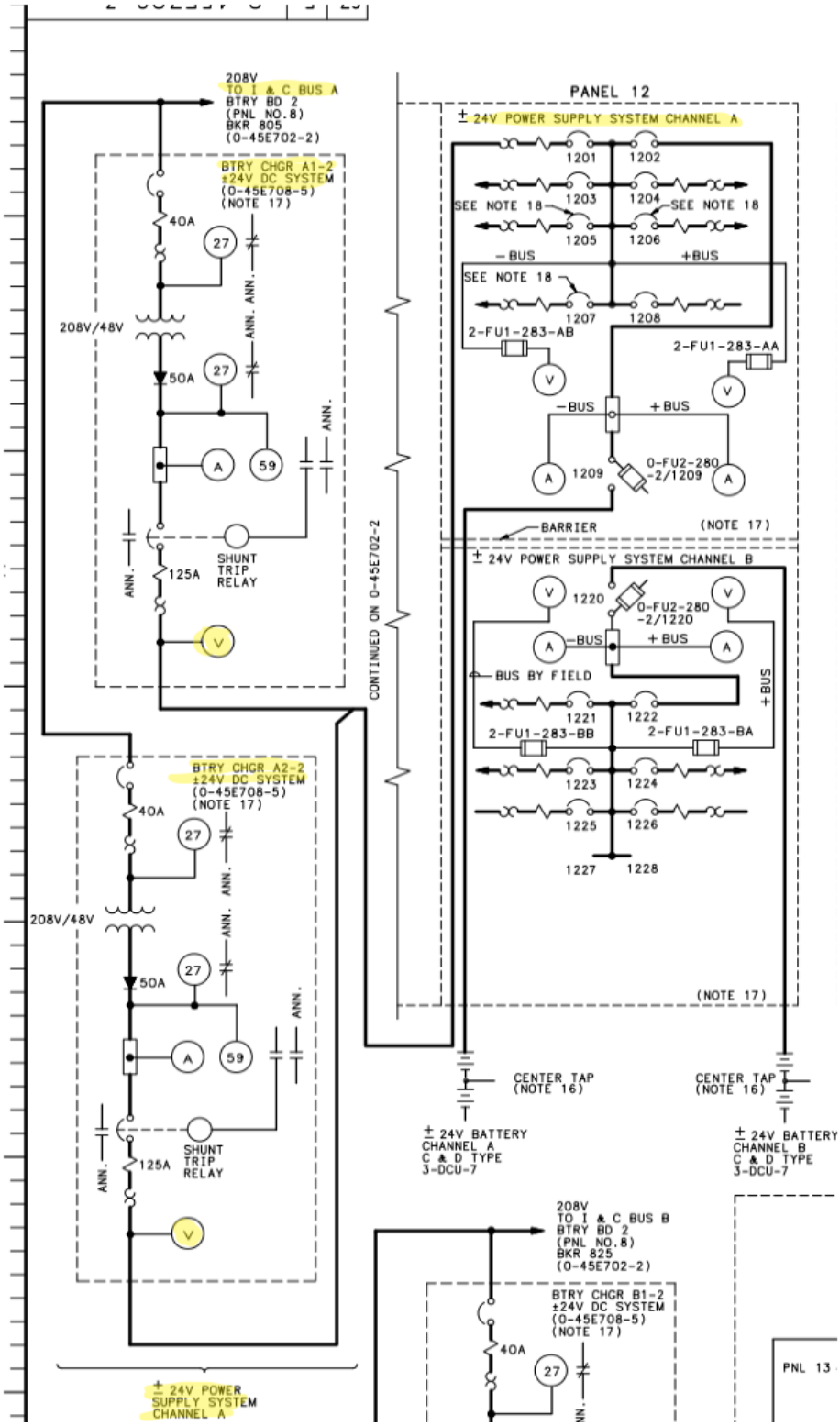
BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0029 Page 20 of 20

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

Excerpt from 0-45E702-3:



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

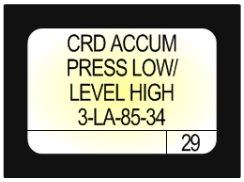
Examination Outline Cross-reference:	Level	RO	SRO
295022 (APE 22) Loss of Control Rod Drive Pumps / 1	Tier #	1	------
AK3.01 (10CFR 55.41.5)	Group #	2	------
Knowledge of the reasons for the following responses or actions as they apply to LOSS OF CONTROL ROD DRIVE PUMPS:	K/A #	295022AK3.01	
• Reactor SCRAM	Importance Rating	4.0	------

Proposed Question: **# 25**

Unit 3 is operating at 100% RTP at the End of Core Life with **ALL** Control Rods fully withdrawn.

CRD Pump 3A has been tagged out for repairs when the following plant conditions occur:

- At 05:15 – 4KV Shutdown Board 3EA de-energizes and locks out due to a fault
- At 05:16 - Charging Water Header Pressure is 930 psig and slowly lowering
- At 05:19 - CRD ACCUMULATOR PRESS LOW/LEVEL HIGH (3-9-5A, Window 29) alarms
- At 05:23 - AUO reports Control Rod 30-31 Accumulator Pressure is 900 psig
- At 05:24 - AUO reports Control Rod 22-19 Accumulator Pressure is 920 psig



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

At 0524, in accordance with 3-AOI-85-3, CRD System Failure, a manual SCRAM is required within ___**(1)**___ due to the risk of ___**(2)**___.

A. (1) one (1) hour
(2) drifting Control Rods

B. (1) one (1) hour
(2) loss of SCRAM capability

C. (1) 20 minutes
(2) drifting Control Rods

D. (1) 20 minutes
(2) loss of SCRAM capability

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that there are various completion times associated with conditions for Control ods and accumulators specified in Tech Spec 3.1.3, Control Rod OPERABILITY, and 3.1.5, Control Rod Scram Accumulators. Times include 20 minutes, 1 hour, 2 hours, 3 hours, 4 hours, 8 hours, and 12 hours. Additionally, the candidate must recognize conditions presented with the loss of the 3EA 4KV Shutdown Board and resulting 3B CRD Pump loss, as well as additional symptoms provided, then apply specific requirements in Subsequent Actions of 3-AOI-85-3. Second part is incorrect but plausible if the candidate confuses specific SCRAM Valve leakage differential pressure symptom sufficient enough to permit the respective Control Rod to drift in.

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-85-3, CRD System Failure, if Reactor Pressure is ≥ 900 psig **AND** Charging Water Header Pressure can NOT be restored and maintained > 940 psig within 20 minutes from discovery of two or more SCRAM accumulators INOP and associated Control Rod NOT fully inserted, **THEN** REDUCE Core Flow to between 50-60% and MANUALLY SCRAM the Reactor. Second part, normal SCRAMs occur using both Accumulator Pressure and Reactor Pressure. Given that 3A CRD Pump is tagged out and 4KV Shutdown Board 3EA de-energizes, the candidate must be able to recognize that 3B CRD Pump has lost its power supply which justifies Charging Water Header Pressure at 930 psig, lowering. This results in reduced SCRAM capability using Reactor Pressure only. If a loss of Reactor Pressure occurs, a SCRAM becomes much more difficult to accomplish. Also, the given alarm has been validated by the AUO's report of the Control Rod Accumulator Pressures to be less than 940 psig.

RO Level Justification: Tests the candidate's knowledge of the reasons for Reactor SCRAM as it applies to a loss of Control Rod Drive Pumps. 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.(3) The progression of an event.

Technical Reference(s): (Attach if not previously provided)

3-AOI-85-3, Rev. 14
Unit 3 Tech Specs 3.1.3, Amend. 212, 260
Unit 3 Tech Specs 3.1.5, Amend. 212
3-ARP-9-5A, Rev. 57
OPL171.005, Rev. 24U3
OPL171.006, Rev. 11U5

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

Proposed references to be provided to applicants during examination: CRD ACCUMULATOR PRESS LOW/LEVEL HIGH (3-9-5A, Window 29)

Learning Objective: OPL171.005 Obj. 28j. (As available)

Question Source:

Bank #		
Modified Bank #	BFN NRC 1909 #78	(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	

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

Copy of Bank Question:

Examination Outline Cross-reference:

201001 (SF1 CRDH) CRD Hydraulic

A2.03 (10CFR 55.43.2 – SRO Only)

Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequence of those abnormal conditions or operations:

- Power supply failures

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	201001A2.03	
Importance Rating	------	3.1

Proposed Question: **# 78**

Unit 3 is operating at 100% RTP at the End of Core Life with **ALL** Control Rods fully withdrawn.

CRD Pump 3A has been tagged out for repairs when the following plant conditions occur:

- At 05:15 - 4kV Shutdown Board 3EA de-energizes and locks out due to a fault
- At 05:16 - Charging Water Header Pressure is 930 psig and slowly lowering
- At 05:19 - CRD ACCUMULATOR PRESS LOW/LEVEL HIGH (3-9-5A, Window 29) alarms
- At 05:23 - AUO reports Control Rod 30-31 Accumulator Pressure is 900 psig
- At 05:24 - AUO reports Control Rod 22-19 Accumulator Pressure is 920 psig



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

In accordance with Tech Specs, the **ACCUMULATORS** for Control Rods 30-31 and 22-19 are ___**(1)**___.

In accordance with 3-AOI-85-3, CRD System Failure, the Unit Supervisor is required to direct a manual SCRAM **NO LATER THAN** ___**(2)**___.

A. **(1)** OPERABLE
(2) 0544

B. **(1)** OPERABLE
(2) 0536

C. **(1)** INOPERABLE
(2) 0544

D. **(1)** INOPERABLE
(2) 0536

Proposed Answer: C

Excerpts from 3-ARP-9-5A:

BFN Unit 3	Panel 9-5 3-XA-55-5A	3-ARP-9-5A Rev. 0057 Page 38 of 48

CRD ACCUM
PRESS LOW/
LEVEL HIGH
3-LA-85-34

29

(Page 1 of 3)

Sensor/Trip Point:

3-PS-85-34 (1 thru 185)	**REFER TO** 3-SIMI -85B for pressure setpoint
3-LS-85-34 (1 thru 185)	37 cc water

Sensor Location: Both sensors located at HCU's in Rx Bldg, El 565'.

Probable Cause:
A. Leaking nitrogen fitting.
B. Leaking piston seals.
C. Single rod scram or Reactor scram.
D. Sensor malfunction.
E. Cleared fuse.

Automatic Action: None

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 N_2 pressure.

Continued on Next Page

BFN **Unit 3**	**Panel 9-5** **3-XA-55-5A**	**3-ARP-9-5A** **Rev. 0057** **Page 39 of 48**

CRD ACCUM PRESS LOW/ LEVEL HIGH 3-LA-85-34, Window 29
(Page 2 of 3)

Operator Action: (Continued)

F. **IF** alarm is valid, **THEN**
REFER TO 3-OI-85 and 3-AOI-85-3.

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. **IF** the Reactor is in MODE 3,4 or 5, **THEN**
REFER TO TECH SPECs 3.9.5, 3.10.3, 3.10.4, and 3.10.5

Continued on Next Page

Excerpts from 3-AOI-85-3:

BFN Unit 3	CRD System Failure	3-AOI-85-3 Rev. 0014 Page 4 of 12

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a CRD System Failure.

2.0 SYMPTOMS

- CRD Pump A breaker tripped.
- CRD PUMP A SUCT PRESS LOW annunciator, (3-XA-55-5A, Window 2) in alarm.
- CRD DRIVE WTR FILTER DIFF PRESS HIGH annunciator, (3-XA-55-5A, Window 3) in alarm.
- CRD PUMP B OVERCURRENT annunciator, (3-XA-55-5A, Window 9) in alarm.
- CRD ACCUM CHG WTR HDR PRESS HIGH annunciator (3-XA-55-5A, Window 10) in alarm.
- CRD PUMP B BREAKER TRIPPED annunciator, (3-XA-55-5A, Window 16) in alarm.
- CONTROL ROD DRIVE UNIT TEMP HIGH annunciator, (3-XA-55-5A, Window 17) in alarm.
- CRD PUMP B SUCT PRESS LOW annunciator, (3-XA-55-5A, Window 23) in alarm.
- CRD ACCUM PRESS LOW/LEVEL HIGH annunciator, (3-XA-55-5A, Window 29) in alarm.
- Low pressure indication on any or all of the following on Panel 3-9-5:
- CRD ACCUM CHG WTR HDR PRESS, 3-PI-85-13A, less than 1475 psig.
- CRD DRIVE WTR HDR DP, 3-PDI-85-17A, less than 250 psid.
- CRD CLG WTR HDR DP, 3-PDI-85-18A, less than 15 psid.
- High or Low indication on CRD DRIVE WTR HDR FLOW, 3-FI-85-15A, on Panel 3-9-5, while control rod is selected and driving.
- Failure of CRD to move on normal insert or withdraw signal.
- ACCUM status light on Full Core Display in alarm at accumulator pressure less than 970 psig.

BFN Unit 3	CRD System Failure	3-AOI-85-3 Rev. 0014 Page 9 of 12

4.2 Subsequent Actions (continued)

[2] **IF** Reactor Pressure is greater than or equal to 900 PSIG

AND

- Charging Water Pressure can NOT be restored and maintained above 940 PSIG within 20 minutes,

AND

- Two or more Scram accumulators inop and associated control rod not fully inserted, **THEN**

PERFORM the following: (Otherwise N/A):

[2.1] **IF** core flow is above 60%, **THEN**

REDUCE core flow to between 50-60%.

[2.2] **MANUALLY SCRAM** Reactor, place the reactor mode switch in the shutdown position immediately.

[2.3] **REFER TO** 2-AOI-100-1.

Excerpts from TS 3.1.3: Supports Distractors A1, B1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	----------NOTE---------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. --- A.1 Verify stuck control rod separation criteria are met. AND A.2 Disarm the associated control rod drive (CRD). AND	Immediately 2 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	AND		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	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.	3 hours
	AND		
	C.2	Disarm the associated CRD.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ----------NOTE---------- Not applicable when THERMAL POWER > 10% RTP. ---------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. OR D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

Excerpts from Tech Specs 3.1.5: Supports Distractors A1, B1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig.	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."	8 hours
	OR		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig.	B.1 Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND	
	B.2.1 ----------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."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted. AND C.2 Declare the associated control rod inoperable.	Immediately upon discovery of charging water header pressure < 940 psig 1 hour
D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met.	D.1 ----------NOTE---------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. ---------- Place the reactor mode switch in the shutdown position.	 Immediately

Excerpts from OPL171.006 Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
2) Belleville washer (For BWR-4 drives) and graphitar seal damage a) Sufficiently high scram velocities will invert the lower Belleville washers, rendering them useless as a shock absorber when the drive piston hits the stop piston. b) Impact loading of the stop piston can cause breakage of the graphitar seals, especially the buffer seals on the drive piston and lower stop piston seals. c) This leads to improper notching or inability to move the drive without increased drive water pressure. Note: High velocity scrams are prevented by adjusting the accumulator gas pressure to accommodate the pressure drops in a particular plant's hydraulic piping.	
2. Scramming With The Over-Piston Flow Path Closed a. Scramming a drive with either the scram discharge valve closed or the manual valves in the withdraw port closed will cause extensive internal damage and failure of the drive. 1) Pressures in the region of 6000 to 7000 psig are generated in the P-over volumes of the drive due to the area ratio between the under- and over-piston areas. 2) The primary effect is reduced ID of the inner tube just below the bottom of the collet piston. a) In serious overpressure situations, this squeezes the inner tube against the circumference of the index tube. b) The index tube is then held in the insert overtravel position and often cannot be withdrawn. 3) Bulging of the index tube as described above also occurs. b. Extensive procedural controls are specified to prevent improper valving of the hydraulic module.	OBJ. ILT/NLO-13.b, LOR-1.b IEN 86-68 – See Significant Industry Events section.
3. Scram Capability a. Piston areas 1) Under-piston area equals 4.0 in^2 2) Over-piston area equals 2.8 in^2 b. Normal scram forces 1) During a normal scram condition, the over-piston area is opened to the scram discharge volume which is initially at atmospheric pressure. 2) Accumulator and/or reactor pressure is simultaneously applied to the under-piston area. The net initial force	

Figure 2: Marathon Control Rod Blade

OPL171.006, Control Rod Blade and Drive Mechanism (CRDM), Rev. 11U5

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
F_{net} = (Up Forces) - (Down Forces) F_{net} = (Rx Pressure x Under-Piston Area) – (Rx Pressure x Area of Index Tube + Weight of Blade + Friction) F_{net} = (400 psig x 4.0 in^2) – (400 psig x 2.8 in^2 + 255 lbs. + ~225 lbs.) F_{net} = (1600 lbs.) – (1120 lbs. + 255 lbs. + 225 lbs.) F_{net} = 0 lbs.	
c) Frictional force is proportional to the differential pressure across the seals. d) Therefore, frictional loss is much less at lower reactor pressures than at rated pressure.	
4) Accumulator-pressure-only scram a) From FIG.-9 it can be seen that, assuming normal accumulator pressure, the drive scram time increases with reactor pressure due to there being less and less net upward force.	FIG.-9
5) Normal scram (accumulator plus reactor pressure) a) Normal scrams are accomplished using a combination of accumulator and reactor pressures (accumulator pressure at the beginning of the stroke and accumulator plus reactor pressure once the accumulator pressure decays to reactor pressure).	
e. Average scram times (normal drive) 1) Technical Specifications state that scram times are to be obtained without reliance on the CRD pumps. 2) Consequently, the charging water must be valved out on the drive to be tested. 3) Maximum scram time for a typical drive occurs at 800 psig reactor pressure. 4) This is why Technical Specifications specify that scram times are to be taken at 800 psig or greater reactor pressure.	FIG.-9
f. Abnormal scram conditions 1) Scram outlet valve failure to open	

Figure 2: Marathon Control Rod Blade

OPL171.006, Control Rod Blade and Drive Mechanism (CRDM), Rev. 11U5

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
a) Drive will slowly scram on seal leakage as long as accumulator charging water pressure stays greater than reactor pressure. b) If the accumulator is not available, the drive will not scram (this is a double failure).	
E. Interrelationships with the CRDM	OBJ. ILT-14.a
1. Control Rod Drive Hydraulic System a. Provides the normal drive water pressure for incremental movement of the control rods. b. Provides accumulator pressure to the CRDM for fast rod insertion. c. Provides cooling for the seals in each CRDM. d. If the CRDH system is lost, the control rod can still be scrammed using reactor pressure, but the time to scram will dramatically increase.	**Operator Fundamental 5** Having a solid understanding of plant design and system interrelationships. OBJ. ILT-21.a (OF5)
2. Reactor Water a. Completes a SCRAM after initiation by the HCU accumulators. b. It would be difficult to complete a SCRAM with a loss of reactor pressure.	OBJ. ILT-14.b OBJ. ILT-21.b (OF5)
3. Rod Position Information System a. Provides indication of each control rod within the reactor core. b. Gives the operator an indication that a control rod has drifted from its intended position. c. Gives indication that a control rod has over-traveled.	OBJ. ILT-14.c
4. Reactor Vessel a. Provides a physical barrier interface, giving the CRDM a place on which to be bolted. b. Provides a housing for the CRDM's.	OBJ. ILT-14.d
5. CRDM Temperature Monitor a. provides an indication of the temperature of each CRDM	OBJ. ILT-14.e High temp can accelerate wear of the graphitar seals
F. Procedure Review 1. Rod Control Drive System, 1/2/3-OI-85 2. Control Rod Drive System Testing and Troubleshooting, 0-TI-20	OBJ. ILT-23 a/b, LOR-3 a/b

Figure 2: Marathon Control Rod Blade

Excerpts from OPL171.005 Lesson Plan: Supports Distractors A2, C2

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 24U3

d)	When a scram occurs, both valves open by internal spring pressure on loss of control air pressure, after the scram pilot air valves open to vent.	
e)	Sequencing is required to prevent buildup of high pressures in the control rod drive, so the outlet valves open slightly ahead of the inlet valves. The outlet valves have stronger opening spring tension.	Obj. ILT 7
f)	Position indication is provided by means of spring-mounted position switches. When both valves open, position switches causing a blue rod scram light to be illuminated on the Rod Status Display, Panel 9-5.	
g)	Air pressure in the valves is maintained at 73 psig by regulating valves (85-66 and 67) which can be set with setpoints of 73 psig and 62 psig (either can be lead controller) such that the backup controller will open to supply air if pressure drops to 62 psig. Pressures >73 psig would cause deformation of the Teflon seats, with resultant valve leakage and slightly increased scram times, particularly for the initial phase of rod movement.	Locking wire with lead seals added to instrument checklist for these regulators. DCN T30146 (U2) DCN T35937 (U3) ARP alarm setpoints: Scram pilot air pressure low: 66 psig
h)	Scram inlet valve leakage can cause over-pressurization of the drive under piston area and possibly rod insertion. May cause rod double notching.	Need >60 psig prior to reset scram per AOI 32-2 & > 50 psig to control 85-11A/11B.
i)	Scram outlet valve leakage Causes depressurization of the over-piston area. Cooling water pressure on the under piston area can cause the drive to drift in.	Obj. ILT 8 Obj. NLO/NLOR 4 Control rod drive high temperatures may result

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 24U3

2. Venting	
a. The CRD Hydraulic System should be well vented prior to starting CRD pumps to prevent pressure surges (water hammer) in the CRD System lines.	
b. The pressure over piston and pressure under piston lines should be well vented to ensure proper control rod drive operation.	
c. Problems of air in CRD pressure over-piston lines:	Obj. ILT 26.a Obj. LOR 15.a
1) Erratic rod response to control occurs.	
2) Rod movement is sluggish.	
3) The rod may not settle fully.	
4) The CRD may fail to notch out because of a loss of unlatching differential pressure.	
d. Problems with air in CRD pressure under piston line:	Obj. ILT 26.b Obj. LOR 15.b
1) Erratic rod response to control occurs.	
2) Rod movement is sluggish.	
3) The rod may not settle fully.	
4) The CRD may fail to notch in, because sufficient pressure will not be produced to move the rod to the next latched position.	
3. Scram Valve Leakage	Obj. ILT 26.c Obj. LOR 15.c
a. If the scram valves leak, a differential pressure may be caused across the piston resulting in rod insertion. This will be seen as "rod drift."	
b. Outlet valve leaking may draw in sufficient reactor water to cause high temperature alarm. A severe leak would permit sufficient DP with cooling water to permit the rod to drift in.	

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

Examination Outline Cross-reference:	Level	RO	SRO
295029 (EPE 6) High Suppression Pool Water Level / 5	Tier #	1	------
G2.1.7 (10CFR 55.41.5)	Group #	2	------
Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation	K/A #	295029G2.1.7	
	Importance Rating	4.4	------

Proposed Question: **# 26**

Events occurred on Unit 2 requiring a manual Reactor SCRAM and EHC Pumps locked out resulting in the following conditions:

- Reactor Pressure is 958 psig, rising
- Reactor Water Level lowered to (-) 25 inches, currently (-) 5 inches
- Main Condenser Vacuum is 20 inches Hg, stable
- SUPPRESSION POOL LEVEL HIGH (2-9-3F, Window 12) in alarm
- 2-LI-64-54A, Suppression Pool Water Level, is (+) 6.0 inches
- **NO** other Operator actions have been taken



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

In accordance with 2-EOI-1, RPV Control MODES 1-3, _____ are currently available to augment Reactor Pressure Control?

A. RCIC **AND** Steam Seals

B. RWCU **AND** Steam Seals

C. HPCI **AND** Main Steam drains

D. RFPTs on minimum flow **AND** Main Steam drains

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that RCIC is not available due to HPCI automatically swapping suction from CST to Suppression Pool (*See C*) thereby closing 2-FCV-73-36, HPCI/RCIC CST TEST VALVE. Steam Seals are available.

B INCORRECT: Incorrect but plausible in that RWCU is not available due to the PCIS Group 3 Isolation signal from Low Reactor Water Level (+) 2 inches. Steam Seals are available.

C INCORRECT: Incorrect but plausible in that HPCI is not available due to HPCI automatically swapping suction from CST to Suppression Pool at (+) 5.25 inches thereby automatically closing 2-FCV-73-36, HPCI/RCIC CST TEST VALVE. If Suppression Pool Level cannot be maintained below (+) 5.25 inches, then 2-EOI Appendix-11C does direct 2-EOI Appendix-16E to bypass HPCI High Suppression Pool Water Level Suction Transfer Interlock. However, no other Operator actions have been taken. Second part is correct (*See C*).

D **CORRECT**: (*See attached*) In accordance with 2-OI-3, Reactor Feedwater System, RFPTs automatically trip when Condenser Low Vacuum 7 inches Hg is reached. Given as 20 inches Hg and stable leaving RFPTs available for Reactor Pressure Control. Reactor Pressure 958 psig and rising indicates MSIVs have not isolated on a PCIS Group 1 signal leaving Main Steam drains available for Reactor Pressure Control.

RO Level Justification: Tests the candidate's ability to evaluate plant performance and make operational judgements based on operating characteristics, Reactor behavior, and instrument interpretation as it relates to High Suppression Pool 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. 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.
(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-EOI-1, Rev. 22 (Attach if not previously provided)
2-OI-3, Rev. 173
2-OI-47, Rev. 209
2-OI-73, Rev. 107
2-ARP-9-3F, Rev. 43
2-EOI Appendix-11C, Rev. 8
2-EOI Appendix-16E, Rev. 4

Proposed references to be provided to applicants during examination: SUPPRESSION POOL LEVEL HIGH (2-9-3F, Window 12)

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

Learning Objective: OPL171.042 Obj. 4 (As available)

Question Source:

	Bank #	BFN 1501 NRC #25
	Modified Bank #	
	New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2015

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 25 Rev 2

Unit 3 has inserted a manual Reactor scram and locked out the EHC pumps due to lowering EHC tank level.

The following conditions currently exist:

- Reactor Water Level (-) 5 inches, lowest indicated level was (-) 25 inches.
- Reactor Pressure is 958 psig and rising
- Drywell Pressure remains stable at 1.5 psig
- Main Condenser vacuum has lowered to (-) 15 inches Hg
- 3-LI-64-54A Suppression Pool Water Level is reading (+) 6.0 inches
- 3-LI-64-66 Suppression Pool Water Level is reading (+) 5.5 inches

Which of the following systems listed in RC/P-11 are currently available to augment RPV pressure control?

A. HPCI and Main Steam system drains

B. RFPTs on min flow and Main Steam system drains

C. Steam Seals and RWCU

D. RCIC and RFPTs on min flow

Answer B

Excerpt from 2-ARP-9-3F:

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0043 Page 16 of 41

SUPPR POOL LEVEL HIGH

2-LA-73-57

12

(Page 1 of 1)

Sensor/Trip Point:
LS-73-57A OR 57B — 5.25 inches above instrument zero after 7-second time delay

Sensor Location: Rx BLdg, El 519', NW wall next to Suppr Chbr on a platform (Bay 7)

Probable Cause:
A. Rising water level in Suppression Pool.
B. Sensor malfunction.

Automatic Action: Automatic transfer of HPCI suction from CST to Suppression Pool.

Operator Action:
A. **CHECK** CST 2 and Suppression Pool level using multiple indications.
B. **ENSURE** HPCI Suction automatically transfers to the Suppression Pool.
C. **IF** automatic transfer fails, **THEN** **REFER TO** 2-OI-73, Section 6.1.
D. **REFER TO** Tech Spec 3.5.1 and 3.5.2.

References: 2-45E620-1 2-47E610-64-1 2-730E928-3
Technical Specifications 3.5.1 and 3.5.2

Excerpt from 2-OI-73:

3.7 Interlocks

A. When any of the following signals are received, HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27, and HPCI SUPPR POOL INBD SUCT VALVE, 2-FCV-73-26 automatically open, unless a HPCI isolation signal is present.

1. Suppression Pool Level High at +5.25 in.

2. HPCI Pump Suction Condensate Header Level Low at approximately 7000 gallons (El. 552'6" on 2-LS-73-56A and -56B).

B. When HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27 and HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26 are fully open, HPCI CST SUCTION VALVE, 2-FCV-73-40, automatically closes.

C. When either HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27, or HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26, is FULL OPEN, the HPCI/RCIC CST TEST VLV, 2-FCV-73-36, and HPCI PUMP CST TEST VLV, 2-FCV-73-35, will close.

D. When starting HPCI manually the HPCI Aux Oil pump will not start until the HPCI TURBINE STEAM SUPPLY VLV, 2-FCV-73-16, starts to open.

E. When the HPCI TURBINE STEAM SUPPLY VALVE, 2-FCV-73-16, is opened, the following valves will close:

1. HPCI HOTWELL PUMP INBD ISOL VLV, 2-FCV-73-17A.

2. HPCI HOTWELL PUMP OUTBD ISOL VLV, 2-FCV-73-17B.

3. HPCI STEAM LINE INBD DRAIN VLV, 2-FCV-73-6A.

4. HPCI STEAM LINE OUTBD DRAIN VLV, 2-FCV-73-6B.

Excerpt 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

RPV CONTROL-MODES 1-3
UNIT 2
BROWNS FERRY
NUCLEAR PLANT

Rev. 22

Excerpt from 2-OI-3:

3.0 PRECAUTIONS AND LIMITATIONS (continued)

M. Reactor Feedwater Pump Turbines will trip on any of the following:

1. Manual Trip (Local or Remote).
2. Reactor Vessel High Water Level (55" as sensed by LT-3-208A and 208C or 208B and 208D).
3. RFPT Electrical Overspeed 105% (6150 rpm)
4. RFPT Mechanical Overspeed 110% (6435 rpm)
5. RFPT Thrust Bearing Active Face Excessive Wear (40 psig).
6. RFPT Thrust Bearing Inactive Face Excessive Wear (40 psig).
7. RFPT Bearing Oil Low Pressure (4 psig).
8. Reactor Feedwater Pump (RFP) Bearing Oil Low Pressure (6 psig).
9. Condenser Low Vacuum (7" Hg).
10. RFP A Low Suction Pressure (150 psig for 80 seconds).
11. RFP B Low Suction Pressure (150 psig for 100 seconds).
12. RFP C Low Suction Pressure (150 psig for 120 seconds).
13. RFP Suction Valve not full open.
14. Loss of both 24V DC power supplies to Woodward Governor

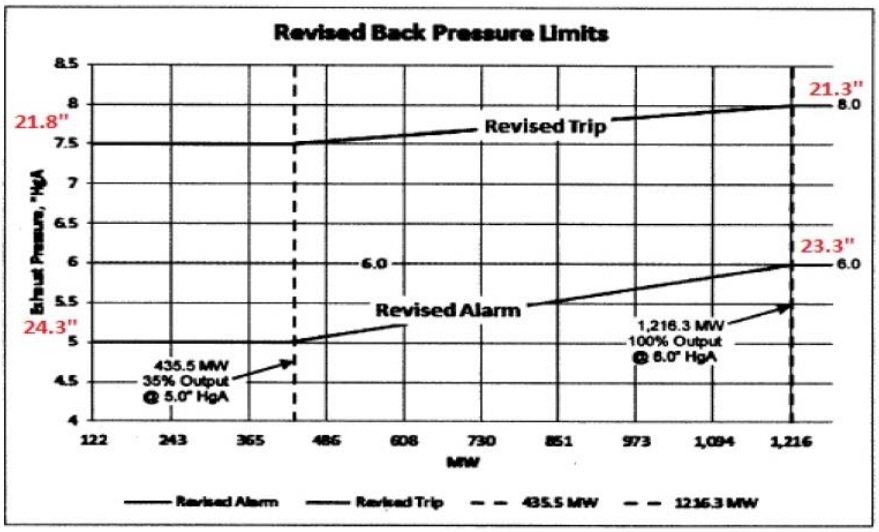
Excerpt from 2-OI-47:

Attachment 4
(Page 2 of 5)

Turbine Trip Logic

The following is a list of main turbine trips:

P.	Loss of Condenser Vacuum	21.8 inches Hg at 35% power up to 21.3 inches Hg at 100% power (REFER TO Revised Back Pressure Limits) .
Q.	Main Turbine manual trips	Trip pushbutton (Panel 2-9-7) EHC WORK STATION computer screen (Panels 2-9-7 and Panel 2-9-31) Trip pushbutton (front standard)
R.	All modules failed	Occurs when two out of three micronet processors (kernals) fail.
S.	Power supply failure	Loss of both power supplies located inside Panel 2-9-31.



Excerpts 2-EOI Appendix-11C:

BFN Unit 2	Alternate RPV Pressure Control Systems HPCI Test Mode	2-EOI Appendix-11C Rev. 0008 Page 3 of 6

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Control Room
ATTACHMENTS	None

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 140°F may result in equipment damage.

[1] **IF** Suppression Pool level drops below 12.75 ft, **THEN**

TRIP HPCI and **CONTROL** RPV pressure using other options.

[2] **IF** Emergency RPV Depressurization is required,

OR

Steam Cooling is required, **THEN**

EXECUTE EOI Appendix 16C and 16D as necessary to bypass HPCI Low RPV pressure and Test Mode Isolation Interlocks.

[3] **IF** Suppression Pool level CANNOT be maintained below 5.25 in., **THEN**

EXECUTE EOI Appendix 16E concurrently with this procedure to bypass HPCI High Suppression Pool Level Suction Transfer Interlock.

1.0 INSTRUCTIONS (continued)

[5] **START** HPCI as follows:

A. **ENSURE** at least one SGTS Train in operation.

B. **ENSURE** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller in AUTO and set for 5300 gpm.

NOTE

HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open.

C. **PLACE** 2-HS-73-47A, HPCI AUXILIARY OIL PUMP handswitch, in START.

D. **PLACE** 2-HS-73-10A, HPCI STEAM PACKING EXHAUSTER, in START.

E. **OPEN** the following valves:

- 2-FCV-73-36, HPCI/RCIC CST TEST VLV
- 2-FCV-73-35, HPCI PUMP CST TEST VLV
- 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

F. **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, to start HPCI Turbine.

G. **ENSURE** HPCI Auxiliary Oil Pump starts and turbine accelerates above 2400 rpm.

[6] **CHECK** proper HPCI minimum flow valve operation as follows:

A. **IF** HPCI flow is above 1200 gpm, **THEN**

CHECK CLOSED 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

B. **IF** HPCI flow is below 600 gpm, **THEN**

CHECK OPEN 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

[7] **THROTTLE** 2-FCV-73-35, HPCI PUMP CST TEST VLV, to control HPCI pump discharge pressure at or below 1100 psig.

Excerpt 2-EOI Appendix-16E:

BFN Unit 2	Bypassing HPCI High Suppression Pool Water Level Suction Transfer Interlock	2-EOI Appendix-16E Rev. 0004 Page 3 of 4

1.0 INSTRUCTIONS

Location:	Unit 2 Auxiliary Instrument Room
Attachments:	1. Tools and Equipment

[1] **REFER** to Attachment 1 and **OBTAIN** necessary tools and equipment. ☐

[2] **LOCATE** terminal strip CC inside Panel 2-9-39, Rear. ☐

[3] **LOCATE** white wire and gray wire attached to terminal CC-13. ☐

[4] While holding white wire with needle-nose pliers, **REMOVE** terminal screw at terminal CC-13. ☐

[5] **REMOVE** and **TAPE** lugged end of white wire lifted from terminal CC-13. ☐

[6] **REPLACE** terminal screw to secure gray wire on CC-13. ☐

[7] **NOTIFY** Unit Operator that HPCI High Suppression Pool Water Level Suction Transfer Interlock is bypassed. ☐

END OF TEXT

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

Examination Outline Cross-reference:	Level	RO	SRO
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode **A3.01** (10CFR 55.41.7)	Tier #	2	------
Ability to monitor automatic operation of RHR/LPCI: INJECTION MODE including: • Valve operation	Group #	1	------
	K/A #	203000A3.01	
	Importance Rating	4.0	------

Proposed Question: **# 27**

Unit 1 was operating at 100% RTP when a LOCA occurred resulting in the following conditions:

- Reactor Pressure is 400 psig
- Reactor Water Level is (-) 140 inches and slowly lowering

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

1-FCV-74-52, RHR SYSTEM I LPCI OUTBOARD INJECTION VALVE, is **(1)**, and

1-FCV-74-7, RHR SYSTEM I MINIMUM FLOW VALVE, is **(2)**.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See Attached)* Reactor Water Level is below the Common Accident Signal of (-) 122 inches, which causes RHR Pumps to start for injection. Reactor Pressure is below the 450 psig RHR Injection Valve interlock; therefore RHR Pumps are running and the RHR System Injection Valves are open. Second part, Reactor Pressure is above the shutoff head value of 320 psig for RHR Pumps. Therefore, RHR System Flow is below 5800 gpm, so the Minimum Flow Valve will also be open.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that since the RHR Pumps are running and the Injection Valves are open, the candidate may assume that RHR is injecting to the Reactor, which would cause the Minimum Flow Valve to close at proper flow rate.

C INCORRECT**:** First part is incorrect but plausible in that in accordance with 1-EOI-1, RPV Control, the shutoff head for RHR Pumps is 320 psig. However, the candidate may confuse the shutoff head with the Injection Valve Interlock, and since Reactor Pressure is above the shutoff head pressure believe the Injection Valve has not opened yet. 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 how the RHR System Valves operate with an accident signal present and Reactor Pressure above the shutoff head for 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. The candidate must analyze the conditions and integrate several pieces of mental data to determine a solution.

Technical Reference(s): 1-OI-74, Rev. 127 (Attach if not previously provided)
1-EOI-1, Rev. 10
1-ARP-9-3C, Rev. 31

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 Obj. 4d, 4e (As available)

Question Source:

Bank #	BFN NRC 2104 #30	
Modified Bank #		(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:

Proposed Question: **# 30**

Unit 1 was operating at 100% RTP when a LOCA occurred resulting in the following conditions:

- Reactor Pressure is 400 psig
- Reactor Water Level is (-) 140 inches and slowly lowering

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

1-FCV-74-52, RHR SYSTEM I LPCI OUTBOARD INJECTION VALVE is **(1)**, and

1-FCV-74-7, RHR SYSTEM I MINIMUM FLOW VALVE is **(2)**.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Proposed Answer: **A**

Excerpt from 1-ARP-9-3C:

BFN Unit 1	Panel 1-9-3 1-XA-55-3C	1-ARP-9-3C Rev. 0031 Page 34 of 41

RX WTR LVL
LOW LOW LOW
ECCS/ESF INIT
1-LA-3-58A

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(Page 1 of 1)

Sensor/Trip Point:

1-LIS-003-0058A	≤ -122 inches
1-LIS-003-0058B	RPV Low-Low-Low level (Level 1)
1-LIS-003-0058C	
1-LIS-003-0058D	

Sensor Location:
1-LIS-003-0058A&B 1-PNLA-009-0081
1-LIS-003-0058C&D 1-PNLA-009-0082

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 (LPCI mode) System
- Diesel Generators
- RHRSW (EECW) Pump

B. ADS Blowdown Logic Input

Operator Action:
A. **CHECK** RPV Level using multiple indications.
B. **IF** alarm is valid, **THEN**
REFER TO EOIs.
C. **IF** alarm is **NOT** valid, **THEN**
RETURN auto start systems to Standby Readiness per respective OI's.
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:

1-45E620-2-1	1-47E610-3-1	1-47W600-58
GE 730E930-3	GE 0-730E930 -4, and -9	Tech Spec 3.3.5.1

Excerpt from 1-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

Excerpts from 1-OI-74:

3.2 RHR Pumps

A. To minimize system vibration, RHR Pump operation should be minimized below 7,000 gpm or above 10,000 gpm, or for more than 3 minutes at minimum flow.

B. [NRC/C] The RHR Pumps are considered to be operable without the seal cooler under the following conditions:

1. Always operable in the LPCI and Containment Cooling Mode.

2. During Shutdown Cooling, operable up to a suction temperature of 215°F.

3. Operable for an emergency with suction temperature between 215°F and 400°F. Operation in this condition for more than two days will require an inspection of the seal surfaces. [NRC LER 296/83047 R1]

C. To prevent Recirculation Pump operation at shutoff head, RHR pumps must not be operated in parallel with Recirculation Pumps.

D. Unit 1 RHR Pumps may be considered available for make-up and Shutdown Cooling Mode of operation with no EECW cooling water available for the associated room and pump seal water coolers with the reactor in Mode 5.

3.3 LPCI

A. LPCI will initiate on any of the following signals:

1. Reactor Vessel low-low-low water level (-122 inches)(Level 1).

2. High Drywell Pressure (2.45 psig) with low Reactor Vessel Pressure (450 psig).

B. Manually stopping an RHR Pump after LPCI Initiation disables automatic restart of that pump until the initiation signal is reset. The affected RHR Pump can still be started manually.

C. Upon an automatic LPCI Initiation with normal power available, RHR Pump 1A will start immediately, THEN 1B, 1C, 1D sequentially start at 7-second intervals. Otherwise, all RHR Pumps start immediately once Diesel power is available (and normal power unavailable).

3.6 Interlocks

A. The RHR System is equipped with pump and valve interlocks to assure the following:

1. All RHR Pump flow is directed to the LPCI Injection path during ECCS initiation.

2. Protection of low pressure piping from high reactor pressures.

3. A pump suction path is fully open prior to pump start.

4. Suction Path Interlocks:

   a. An RHR Pump will not start or will trip, if running, unless its corresponding torus suction valve is open or the SDC suction valve and the SDC suction supply valves, 1-FCV-74-47 and 48, are open.

   b. The Torus Suction valves cannot be opened unless the corresponding pumps SDC suction valve is fully closed.

   c. The SDC suction valves cannot be opened unless the corresponding pumps Torus Suction valve is fully closed.

5. RHR Minimum Flow Valve Interlocks:

   a. The RHR minimum flow valves auto close if both pumps in the corresponding loop are off and either pump's SDC suction valve is open.

   b. The minimum flow valves open and close on a low flow of 5800 gpm. The tolerance of the flow switch may allow the setpoint of the min flow valve to be from 4500 gpm to 7000 gpm. Operation outside of this expanded range should be investigated. The analytical limit as listed in design criteria BFN-50-7074 is 11000 gpm for min flow valve closure. [PER-238791-01, 06]

   c. Placing the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), in INHIBIT, will simulate a high flow and the minimum flow valve will remain closed regardless of flow.

3.6 Interlocks (continued)

d. Opening RHR SYSTEM I(II)MIN FLOW VALVE, 1-HS-74-7A(30A), from 1-PNL-9-3, with the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), in INHIBIT, will cause the minimum flow valves to travel full open and full close unless the RHR SYSTEM I(II) MIN FLOW VALVE, 1-HS-74-7A(30A) is placed in closed position to break the OPEN seal in contacts.

e. [II/C] Local operation of the RHR minimum flow valves will bypass the intended function of the Minimum Flow Inhibit switch and can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFPER941099]

f. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), with the respective RHR Loop in standby readiness, can cause inadvertent damage to that loop RHR Pump(s) should RHR Pump(s) auto start. [BFA-890790003P]

g. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), with the respective RHR loop in Shutdown Cooling, can cause inadvertent drainage of the Reactor Vessel to the Suppression Pool. [BFA-890790003P]

6. RHR Outboard LPCI Injection valves, 1-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(II) OUTBD INJ VLV BYPASS SEL keylock switch, 1-HS-74-155A(155B) is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is greater than 450 psig if its in-line companion valve 1-FCV-74-53(67) is not fully closed.

7. If Unit 1 Reactor Pressure exceeds 100 psig or a Group II isolation occurs on Unit 1 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, 1-FCV-74-47 and 1-FCV-74-48, close, thus tripping operating Unit 1 RHR Pumps.

(Group II) RHR SYS I and II LPCI INBD INJECT VALVEs, 1-FCV-74-53 and 1-FCV-74-67, close and RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 1-FCV-74-47 and 1-FCV-74-48, close, thus tripping operating Unit 1 RHR Pumps.

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	------
K6.08 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE):	K/A #	205000K6.08	
• Service water	Importance Rating	3.3	------

Proposed Question: **# 28**

Unit 1 is in MODE 4, with the following conditions:

- 1A RHR Pump is aligned to Shutdown Cooling (SDC) Mode
- EECW/RHRSW is in a normal lineup
- A2 RHRSW Pump tripped and **CANNOT** be restarted
- **NO** Operator actions have been taken

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

1-FCV-23-34, 1A RHR HEAT EXCHANGER OUTLET VALVE, **(1)** automatically close.

In accordance with 1-AOI-74-1, Loss of Shutdown Cooling, Operators are required to start **(2)** RHRSW Pump.

A. (1) will
(2) A1

B. (1) will
(2) C2

C. (1) will NOT
(2) A1

D. (1) will NOT
(2) C2

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 0-OI-23, RHRSW System, if the RHRSW Pump in the associated header is not operating, then the RHR Heat Exchanger Cooling Water Outlet Valves are interlocked to auto close. In the conditions given, A2 (or A1) RHRSW Pump tripped resulting in 1-FCV-23-34 automatically closing. Second part, in the event of a loss/trip of RHRSW aligned to support shutdown cooling, 1-AOI-74-1 directs starting the associated loop's standby RHRSW Pump (A1).

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that for this common use system for U1, U2, and U3, there are 12 pumps in the RHRSW/EECW System with #3 pumps specific for EECW use only, #2 pumps specific for RHRSW use only, and #1 pumps can serve as either EECW or RHRSW Pumps as needed. Along with the valve alignment and logic and heat exchanger alignment, the systems are easily and often confused by candidates. Since the 1A RHR Pump is running, then either A2 or A1 RHRSW Pumps are needed to serve the 'A' heat exchanger header. C2 RHRSW Pump would be correct if 1C RHR Pump was running and the RHRSW Pump outlet valve FCV-23-40 is in use.

C INCORRECT: First part is incorrect but plausible in that if A1, B1, C1, or D1 RHRSW Pump is aligned and running for EECW use, then the associated RHRSW Discharge Valve will not automatically close when the #2 RHRSW Pump trips and will cause a Low RHRSW Pressure condition to exist. 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 effect of service water system malfunctions, or component malfunctions on the RHR Shutdown Cooling Mode of 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.

Technical Reference(s): 0-OI-23, Rev.109 (Attach if not previously provided)
1-AOI-74-1, Rev. 11
OPL171.046, Rev. 020
OPL171.044, Rev. 023

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 Obj. 4i (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

Excerpt from 1-AOI-74-1:

4.2 Subsequent Actions (continued)

[4] **ENTER** (Go to) the applicable step for the cause of Loss of Shutdown Cooling

- Inadequate RHRSW Flow - Step 4.2[4.1]
- Loss of running RHR pump - Step 4.2[4.2]
- Loss of RHR Loop with opposite Loop available - Step 4.2[4.3]
- All RHR unavailable - Step 4.2[4.4]
- Valid Group 2 Isolation - 4.2[4.5]
- Invalid Group 2 Isolation - Step 4.2[4.6]
- LPCI Initiation - Step 4.2[4.7]

[4.1] INADEQUATE RHRSW FLOW

[4.1.1] **IF** the running RHRSW Pump trips on the header in Shutdown Cooling and the other pump on the header is available, **THEN**

START the standby RHRSW pump for the appropriate header. **REFER TO** 0-OI-23. (Otherwise N/A)

[4.1.2] **IF** both RHRSW Pumps become unavailable on the header in Shutdown Cooling, **THEN**

ENSURE the RHR pump associated with the available RHRSW Header is in service with appropriate RHR and RHRSW flows to control RCS temperature and RHRSW outlet temperature. **REFER TO** 1-OI-74. (Otherwise N/A).

[4.1.3] **IF** the affected loop of RHRSW cannot be placed back in service, **THEN**

PLACE the alternate loop of RHR in Shutdown Cooling. **REFER TO** 1-OI-74. (Otherwise N/A)

[4.1.4] **GO TO** step 4.2[4.8]

Excerpts from 0-OI-23:

3.2 Operability and LCO's (continued)

E. Technical Specification 3.7.2, EECW System and UHS, bases Ultimate Heat Sink (UHS) Operability on the UHS having a maximum water temperature of 95°F. 2-SR-2, Instrument Checks and Observations, satisfies SR 3.7.2.1 for all three units by verifying that the average water temperature of the UHS is less than or equal to 95°F every 24 hours.

F. There shall be at least 2 RHRSW pumps, associated with the selected RHR pumps, aligned for RHR heat exchanger service for each reactor vessel containing irradiated fuel per Technical Specification 3.7.1.

G. When any three components of the RHRSW System (on any or all fueled units) are inoperable, the capability to meet the safety function must be evaluated by all fueled units. When an RHRSW pump is credited by one fueled unit for maintaining the RHRSW safety function, then the other two fueled units cannot also credit this same RHRSW pump with maintaining their RHRSW safety function since the capacity of a single RHRSW pump is not sufficient to support the required heat removal function of more than one RHR heat exchanger. Therefore, in this condition, the RHRSW pump credited with maintaining RHRSW safety function on a fueled unit must be considered inoperable for the other fueled units for the purposes of determining if RHRSW safety function is maintained. The other fueled units must then include the additional inoperable RHRSW pump(s) with the total number of inoperable components when determining if RHRSW safety function is maintained.

H. Consideration should be given to room inoperability any time a flood barrier (example: water tight door) becomes degraded.

3.3 Interlocks

A. If the RHRSW Pump in the associated header is not operating, then the RHR Heat Exchanger Cool Water Outlet Valves are interlocked to auto close.

RHRSW Pump	Valve
A1, A2	FCV-023-0034
B1, B2	FCV-023-0046
C1, C2	FCV-023-0040
D1, D2	FCV-023-0052

B. If A1, B1, C1 or D1 RHRSW Pump is aligned and running for EECW, then the associated RHRSW Discharge Valve will not auto close when the #2 RHRSW Pump trips and will cause a Low RHRSW Pressure condition to exist.

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.

8.6 Restoring RHRSW Flow After a Pump Trip (continued)

[1] **IF** desired to reestablish RHRSW flow through RHR Heat Exchanger 1(2)(3)A, **THEN**:

PERFORM the following: (Otherwise N/A)

[1.1] **ENSURE CLOSED** RHR HX 1(2)(3)A RHRSW OUTLET VLV's 1(2)(3)-FCV-23-34.

[1.2] **START** the RHRSW PUMP A2(A1) using one of the following:

- 0-HS-23-5A/1(1A/1) for Unit 1
- 0-HS-23-5A/2(1A/2) for Unit 2
- 0-HS-23-5A/3(1A/3) for Unit 3

[1.3] **THROTTLE** the RHR HX 1(2)(3)A RHRSW OUTLET VLV, 1(2)(3)-FCV-23-34, to maintain a cumulative RHRSW flow through the A RHR in-service heat exchangers ≥ 4000 gpm (sum of the values on 1, 2, and 3-FI-23-36), and ≤ 4500 gpm through any one heat exchanger.

[2] **IF** desired to reestablish RHRSW flow through RHR Heat Exchanger 1(2)(3)B, **THEN**:

PERFORM the following: (Otherwise N/A)

[2.1] **ENSURE CLOSED** RHR HX 1(2)(3)B RHRSW OUTLET VLV's 1(2)(3)-FCV-23-46.

[2.2] **START** the RHRSW PUMP B2(B1) using one of the following:

- 0-HS-23-19A/1(15A/1) for Unit 1
- 0-HS-23-19A/2(15A/2) for Unit 2
- 0-HS-23-19A/3(15A/3) for Unit 3

[2.3] **THROTTLE** the RHR HX 1(2)(3)B RHRSW OUTLET VLV(s), 1(2)(3)-FCV-23-46, to maintain a cumulative RHRSW flow through the B RHR in-service heat exchangers ≥ 4000 gpm (sum of the values on 1, 2, and 3-FI-23-48), and ≤ 4500 gpm through any one heat exchanger.

Excerpt from OPL171.044 Lesson Plan:

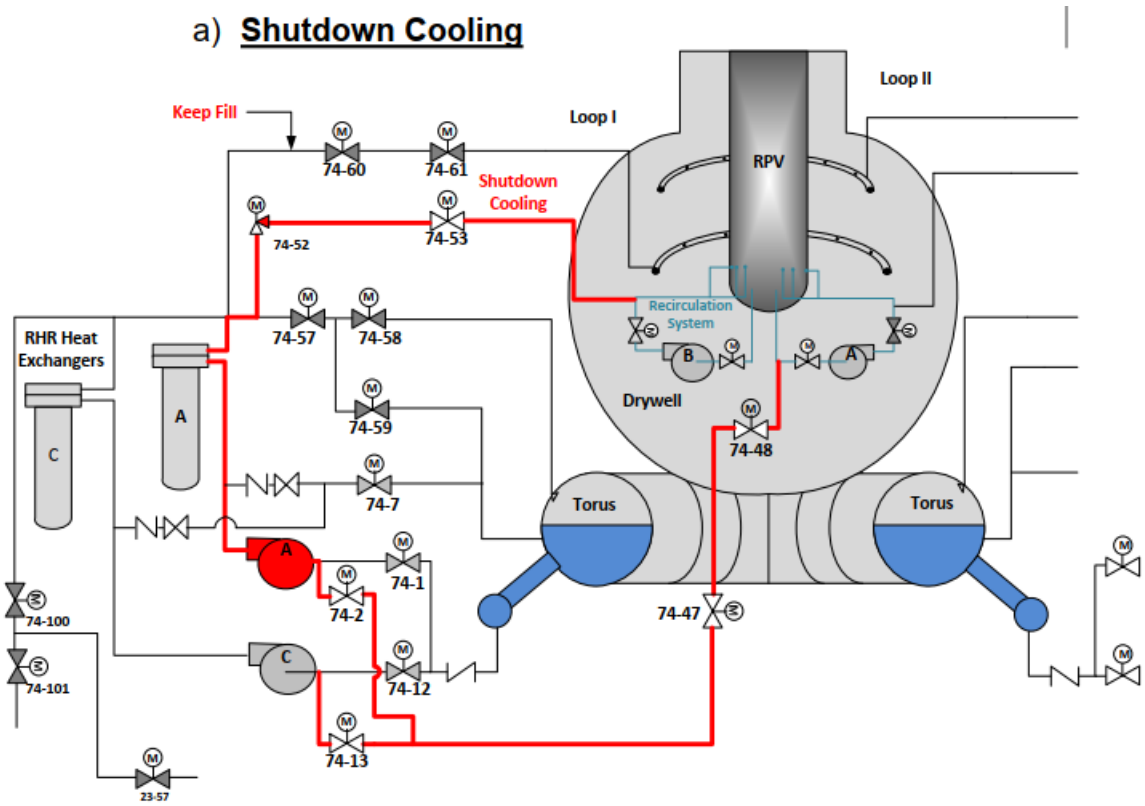
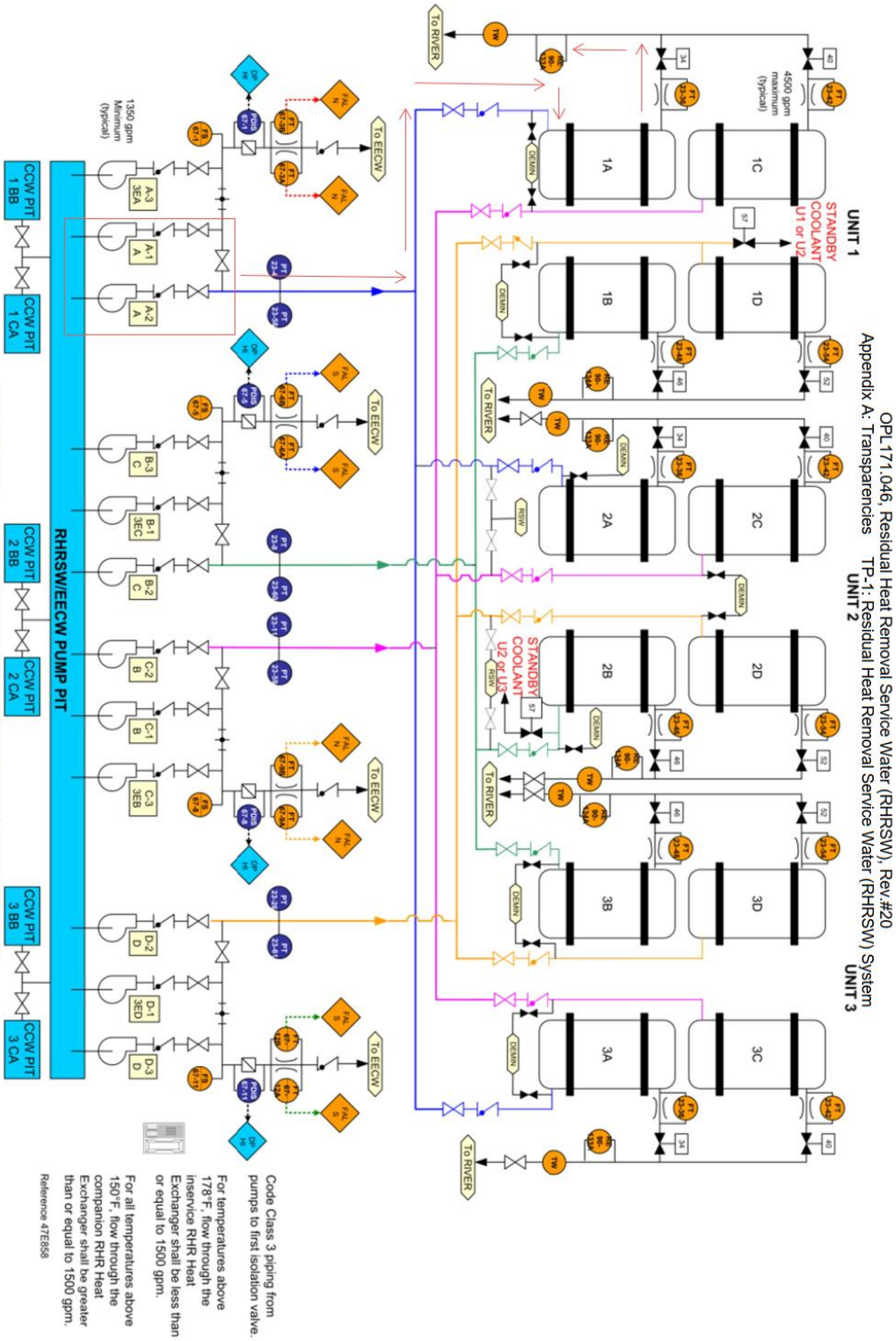


Figure 2 – Shutdown Cooling Flowpath

Excerpts from OPL171.046 Lesson Plan:

	to the EECW header. When the mode switches are placed to EECW mode, Alarm on 1(3)-XA-55-3E.
The manual mode switch is located at the corresponding RHRSW/EECW pump breaker (4KV Shutdown Board). The mode switch will select RHRSW mode when closed and EECW mode when cross-tie valve is open, aligning the RHRSW swing pump to EECW.	Two concerns: (1) No flow path & pump damage from deadheading
Mode Switch disables the auto start capability when aligned for RHRSW operation, but only designed for EECW operation.	(2) EDG overloading from spurious start.
5. The A header supplies 1A (Unit 1), 2A (Unit 2), and 3A (Unit 3) RHR HX's; the B header supplies 1B, 2B, and 3B RHR Hx, the C header supplies 1C, 2C, and 3C RHR Hx, the D header supplies 1D, 2D, and 3D RHR Hx	Additionally, this condition could produce a potential flow diversion path for EECW into RHRSW or back and requires local OMAs to manually close the hand wheel.
6. RHRSW headers enter the plant through 6 tunnels on south side of plant (2 per unit). Each unit has 4 inlet supply header penetrations (A-D headers.)	
7. RHRSW headers have continuous charging from Raw Service Water. These connections are located at Unit 2 Elev. 565 south wall, providing inlet header and heat exchanger tube-side charging. Since the headers are all common between Units 1, 2, & 3, only one charging point per header is needed.	Obj. 7.e ILT Obj. 6.e LOR
8. RHR heat exchangers (4 per Unit) allow service water to make two passes before leaving heat exchangers. Heat exchanger outlet flow control valves are motor-operated from Main Control Room (MCR) for temperature control (FCV-23-34, 46, 40 & 52).	
9. The Heat Exchanger outlet valve is interlocked as follows:	Obj. 3 ILT Obj. 2 LOR Obj. 7 NLO
a) Cannot be opened unless a corresponding # 1 or # 2 RHRSW pump associated with that header is running.	
b) The valve will auto close, if open, should the running pump(s) trip or be shut off. This is to prevent draining the RHRSW header.	Valve logic only looks at the motor contactors (52a). TP-1 and TP-2.
c) The valve(s) can be opened and will remain open even if the # 1 pump on that loop has been re-aligned to supply and is running to supply EECW vice RHRSW.	TP-1


OPL171.046, Residual Heat Removal Service Water (RHRSW), Rev.#20
Appendix A: Transparencies TP-1: Residual Heat Removal Service Water (RHRSW) System
UNIT 1
UNIT 2
UNIT 3
STANDBY COOLANT U1 or U2
STANDBY COOLANT U2 or U3
4500 gpm maximum (typical)
1350 gpm Minimum (typical)
To RIVER
To EECW
DEMIN
RSW
RHRSW/EECW PUMP PIT
CCW PIT 1 BB
CCW PIT 1 CA
CCW PIT 2 BB
CCW PIT 2 CA
CCW PIT 3 BB
CCW PIT 3 CA
A-3 3EA
A-1 A
A-2 A
B-3 C
B-1 3EC
B-2 C
C-2 B
C-1 B
C-3 3EB
D-2 D
D-1 3ED
D-3 D
1A
1B
1C
1D
2A
2B
2C
2D
3A
3B
3C
3D
Code Class 3 piping from pumps to first isolation valve.
For temperatures above 178°F, flow through the inservice RHR Heat Exchanger shall be less than or equal to 1500 gpm.
For all temperatures above 150°F, flow through the companion RHR Heat Exchanger shall be greater than or equal to 1500 gpm.
Reference 47E856

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

Examination Outline Cross-reference:	Level	RO	SRO
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection **K5.10**(10CFR 55.41.5)	Tier #	2	------
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the HIGH PRESSURE COOLANT INJECTION SYSTEM: • Reactor Pressure Control	Group #	1	------
	K/A #	206000K5.10	
	Importance Rating	3.9	------

Proposed Question: **# 29**

Which **ONE** of the following completes the statements below in accordance with 2-EOI Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode?

When operating HPCI for Reactor Pressure Control, **(1)** is(are) required to be OPEN.

HPCI Turbine is required to be operated no less than **(2)** rpm.

Note: 2-FCV-73-35, HPCI PUMP CST TEST VALVE
2-FCV-73-36, HPCI/RCIC CST TEST VALVE

A. (1) **ONLY** 2-FCV-73-35
(2) 1200

B. (1) **ONLY** 2-FCV-73-35
(2) 2400

C. (1) 2-FCV-73-35 **AND** 2-FCV-73-36
(2) 1200

D. (1) 2-FCV-73-35 **AND** 2-FCV-73-36
(2) 2400

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that when in standby readiness, HPCI and RCIC have numerous valves that are normally open (NO). Given that 2-FCV-73-36, HPCI/RCIC CST TEST VALVE, is operated from both RCIC and HPCI procedures, the candidate could incorrectly conclude that it is a normally open (NO) valve. Second part is incorrect but plausible if the candidate confuses 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, operation setpoints (1200 gpm, close) with turbine speed requirements.

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 Appendix-11C, when starting HPCI for Alternative Reactor Pressure Control Mode, both 2-FCV-73-35 and 36 are required to OPEN. Second part, upon startup for Reactor Pressure Control Mode, HPCI Auxiliary Oil Pump starts and the turbine accelerates automatically above 2400 rpm.

RO Level Justification: Tests the candidate's knowledge of the operational implications or cause and effect relationships of the Reactor Pressure Control as it applies to the High Pressure Coolant Injection System. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to HPCI Reactor Pressure Control Mode.

Technical Reference(s): 2-EOI Appendix-11C, Rev. 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.042 Obj. 4 (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-EOI Appendix-11C:

BFN Unit 2	Alternate RPV Pressure Control Systems HPCI Test Mode	2-EOI Appendix-11C Rev. 0008 Page 5 of 6

1.0 INSTRUCTIONS (continued)

[5] **START** HPCI as follows:

A. **ENSURE** at least one SGTS Train in operation.

B. **ENSURE** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller in AUTO and set for 5300 gpm.

NOTE

HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open.

C. **PLACE** 2-HS-73-47A, HPCI AUXILIARY OIL PUMP handswitch, in START.

D. **PLACE** 2-HS-73-10A, HPCI STEAM PACKING EXHAUSTER, in START.

E. **OPEN** the following valves:

- 2-FCV-73-36, HPCI/RCIC CST TEST VLV
- 2-FCV-73-35, HPCI PUMP CST TEST VLV
- 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

F. **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, to start HPCI Turbine.

G. **ENSURE** HPCI Auxiliary Oil Pump starts and turbine accelerates above 2400 rpm.

[6] **CHECK** proper HPCI minimum flow valve operation as follows:

A. **IF** HPCI flow is above 1200 gpm, **THEN**

CHECK CLOSED 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

B. **IF** HPCI flow is below 600 gpm, **THEN**

CHECK OPEN 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE.

[7] **THROTTLE** 2-FCV-73-35, HPCI PUMP CST TEST VLV, to control HPCI pump discharge pressure at or below 1100 psig.

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

Examination Outline Cross-reference:	Level	RO	SRO
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray	Tier #	2	------
K1.02 (10CFR 55.41.8)	Group #	1	------
Knowledge of the physical connections and/or cause and effect relationships between the LOW PRESSURE CORE SPRAY SYSTEM and the following systems:	K/A #	209001K1.02	
• Primary containment	Importance Rating	3.7	------

Proposed Question: **# 30**

Which **ONE** of the following completes the statements below in accordance with 2-OI-75, Core Spray?

The Core Spray System normal suction supply is from **(1)**.

When Primary Containment is required, the preferred method for maintaining the Core Spray System charged is the **(2)**.

A. (1) CST
(2) PSC Head Tank

B. (1) CST
(2) CST System

C. (1) Suppression Pool
(2) PSC Head Tank

D. (1) Suppression Pool
(2) CST System

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the CST is the normal suction source for both HPCI and RCIC and the backup for Core Spray and RHR. 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 the PSC Head Tank is not available, then the CST System may be aligned via 2-SHV-75-700, CORE SPRAY SYSTEM LOOP I & II FILL FROM CONDENSATE SHUTOFF VALVE.

C **CORRECT**: (*See attached*) In accordance with 2-OI-75, Core Spray System, for prestartup/standby readiness requirements, Suppression Pool Level is between (-) 5.5 inches to (-) 2.0 inches. Additionally, 2-OI-75, contains steps to transfer from normal suction Suppression Pool to CST, and alarms exist when Core Spray System I or II suction valves are not open in which the Operator is required to ensure the applicable Loop's Suppression Pool suction valves are OPEN. Second part, in accordance with 2-OI-75, when Primary Containment is required, the preferred method for maintaining the Core Spray System charged is the PSC Head Tank.

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 physical connections and/or cause and effect relationships between Primary Containment and Low Pressure Core Spray. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to Core Spray and Primary Containment.

Technical Reference(s): 2-ARP-9-3C, Rev. 29 (Attach if not previously provided)
2-OI-75, Rev. 124

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.045 Obj. 4a (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-75:

4.0 PRESTARTUP/STANDBY READINESS REQUIREMENTS (continued)

- 480V Reactor Building Vent Board 2B
- 250V DC Reactor MOV Board 2A
- 250V DC Reactor MOV Board 2B
- 240V Lighting Board 2A
- Panel 2-9-9, Cabinet 2
- Panel 2-9-9, Cabinet 3
- Panel 2-9-9, Cabinet 5, Unit Non-Preferred 120V Distribution

E. The following system lineup checklists are current:

- Attachment 1, Core Spray System Valve Lineup Checklist, Unit 2
- Attachment 2, Core Spray System Panel Lineup Checklist, Unit 2
- Attachment 3, Core Spray System Electrical Lineup Checklist, Unit 2
- Attachment 4, Core Spray System Instrument Inspection Checklist, Unit 2

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

- Suppression Pool Level is -5.5 inches to -2.0 inches.
- Control Air is in service to supply PSC PUMP SUCTION INBD and OUTBD ISOL VALVEs, 2-FCV-75-57 and 2-FCV-75-58. REFER TO 0-OI-32.
- Emergency Equipment Cooling Water is in Standby Readiness to provide cooling water to Core Spray Pump Room Coolers. REFER TO 0-OI-67.

[3] **NOTIFY** Unit SRO Core Spray System is now in Standby Readiness Condition.

NOTES

1) When Primary Containment is required, the preferred method for maintaining the Core Spray system charged is the PSC Head Tank. If the PSC head tank is NOT available, then the CST System may be aligned via CS SYSTEM I & II FILL FROM CONDENSATE SHUTOFF VALVE 2-SHV-075-0700.
2) If personnel were stationed at 2-SHV-075-0582A, then they may be released when 2-SHV-075-0582A is LOCKED CLOSED

8.1.3 Charging Core Spray System I When Primary Containment Is Required

[1] **CHECK** Section 8.1.1 has been completed.

[2] **IF** PSC Head Tank is available **THEN**:

PERFORM the following: (Otherwise N/A)

[2.1] **LOCK CLOSED** CS SYSTEM I CNDS FLUSH & FILL SHUTOFF VALVE, 2-SHV-075-0582A.

[2.2] **CHECK LOCKED CLOSED** CS SYSTEM I & II FILL FROM CONDENSATE SHUTOFF VALVE, 2-SHV-075-0700.

[2.3] **CHECK LOCKED OPEN** PSC WATER HEAD TANK FILL & DRAIN SHUTOFF VALVE, 2-SHV-075-0616.

[2.4] **LOCK OPEN** PSC WATER TO CS SYSTEM I FILL SHUTOFF VALVE, 2-SHV-075-0608.

[3] **IF** PSC Head Tank is **NOT** available , **THEN**:

PERFORM the following: **(**Otherwise N/A)

[3.1] **LOCKOUT** PSC HEAD TANK PUMPS 2A and 2B using 2-HS-75-75A and 2-HS-75-76A on Panel 2-9-3.

[3.2] **UNLOCK** and **CLOSE** PSC WATER HEAD TANK FILL & DRAIN SHUTOFF VALVE, 2-SHV-075-0616.

[3.3] **CHECK LOCKED CLOSED** CS SYSTEM I CNDS FLUSH & FILL SHUTOFF VALVE, 2-SHV-075-0582A.

[3.4] **UNLOCK** and **OPEN** CS SYSTEM I & II FILL FROM CONDENSATE SHUTOFF VALVE, 2-SHV-075-0700.

8.21 Returning Core Spray System I Suction to the Suppression Pool

[1] **CHECK** Core Spray System I is **NOT** in operation.

[2] **DISPATCH** personnel to **CLOSE** the Condensate suction valve for the Core Spray Pump with the Suppression Pool suction valve CLOSED. (N/A Pump **NOT** aligned)

- 2-SHV-075-0003, CS PUMP 2A CNDS SUCT SHUTOFF VALVE.
- 2-SHV-075-0012, CS PUMP 2C CNDS SUCT SHUTOFF VALVE.

[3] **OPEN** the Core Spray Minimum Flow Shutoff Valve for pumps being aligned to the Suppression Pool.
(N/A Pump **NOT** aligned)

- CS PUMP 2A MINIMUM FLOW SHUTOFF VALVE, 2-SHV-075-0008.
- CS PUMP 2C MINIMUM FLOW SHUTOFF VALVE, 2-SHV-075-0017.

[4] **CHECK OPEN** the following valves:

- 2-FCV-75-2, CORE SPRAY PUMP 2A SUPPR POOL SUCT VLV.
- 2-FCV-75-11, CORE SPRAY PUMP 2C SUPPR POOL SUCT VLV.

[5] At Panel 2-9-3

CHECK RESET CORE SPRAY SYS II SUCTION VLVS NOT OPEN, (2-XA-55-3C, Window 26).

8.19 Swapping Core Spray System I Suction to the CST

8.19.1 Swapping Core Spray 2A Suction to the CST

NOTES

1) The following step will cause CORE SPRAY SYS I SUCTION VLVS NOT OPEN annunciator (2-XA-55-3C, Window 26) to ALARM.
2) A Core Spray Pump is inoperable when its associated Suppression Pool Suction valve is outside its normal safety position (Open).

[1] **CHECK** Core Spray System I is NOT in operation.

[2] **CLOSE** 2-FCV-75-2 using CORE SPRAY PUMP 2A SUPPR POOL SUCT VLV, 2-HS-75-2A.

[3] At Panel 2-9-3

CHECK CORE SPRAY SYS I SUCTION VLVS NOT OPEN, (2-XA-55-3C, Window 26) is in ALARM.

[4] **CHECK CLOSED** 2-FCV-75-22, CORE SPRAY SYS I TEST VALVE.

[5] At U2 RB NW, Platform above El 519'

[5.1] **CLOSE** CS PUMP 2A MINIMUM FLOW SHUTOFF VALVE, 2-SHV-075-0008

[5.2] **OPEN** 2-SHV-075-0003, CS PUMP 2A CNDS SUCT SHUTOFF VALVE

[6] **MONITOR** CST level and **CROSSTIE** with other Unit CST as directed by the Unit SRO.

8.21 Returning Core Spray System I Suction to the Suppression Pool

[1] **CHECK** Core Spray System I is **NOT** in operation.

[2] **DISPATCH** personnel to **CLOSE** the Condensate suction valve for the Core Spray Pump with the Suppression Pool suction valve CLOSED. (N/A Pump **NOT** aligned)

- 2-SHV-075-0003, CS PUMP 2A CNDS SUCT SHUTOFF VALVE.
- 2-SHV-075-0012, CS PUMP 2C CNDS SUCT SHUTOFF VALVE.

[3] **OPEN** the Core Spray Minimum Flow Shutoff Valve for pumps being aligned to the Suppression Pool.
(N/A Pump **NOT** aligned)

- CS PUMP 2A MINIMUM FLOW SHUTOFF VALVE, 2-SHV-075-0008.
- CS PUMP 2C MINIMUM FLOW SHUTOFF VALVE, 2-SHV-075-0017.

[4] **CHECK OPEN** the following valves:

- 2-FCV-75-2, CORE SPRAY PUMP 2A SUPPR POOL SUCT VLV.
- 2-FCV-75-11, CORE SPRAY PUMP 2C SUPPR POOL SUCT VLV.

[5] At Panel 2-9-3

CHECK RESET CORE SPRAY SYS II SUCTION VLVS NOT OPEN, (2-XA-55-3C, Window 26).

Excerpt from 2-ARP-9-3C:

BFN Unit 2	Panel 9-3 2-XA-55-3C	2-ARP-9-3C Rev. 0029 Page 33 of 42

CORE SPRAY
SYS I SUCTION
VLVS NOT OPEN

2-ZA-75-2

26

(Page 1 of 1)

Sensor/Trip Point:

Limit switches on 2-FCV-75-2 and 2-FCV-75-11.

Sensor Location: Rx Bldg, El 519', R-9 0-LINE

Probable Cause:
A. CORE SPRAY Pump 2A or 2C suction valve is **NOT** fully open.
B. SI/SR is in progress.
C. Sensor Malfunction.

Automatic Action: None

Operator Action:
A. **ENSURE OPEN** the following valves:
- CORE SPRAY PUMP 2A SUPPR POOL SUCT VLV, 2-FCV-75-2.
- CORE SPRAY PUMP 2C SUPPR POOL SUCT VLV, 2-FCV-75-11.

B. **DISPATCH** personnel to locally check the following valves full open:
- CORE SPRAY PUMP 2A SUPPR POOL SUCT VLV, 2-FCV-75-2.
- CORE SPRAY PUMP 2C SUPPR POOL SUCT VLV, 2-FCV-75-11.

C. **REFER TO** Tech Spec Section 3.5.1.

References: 2-45E620-2 45N779-13
Technical Specifications 3.5.1

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	------
AA1.01 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING:	K/A #	295021AA1.01	
• Reactor Water Cleanup System	Importance Rating	3.4	------

Proposed Question: **# 31**

Unit 2 is operating in **MODE 4** when a spurious High Reactor Pressure Isolation signal causes a Loss of RHR Shutdown Cooling.

Which **ONE** of the following completes the statement below?

Given the conditions above, _____ will provide **BOTH** alternate decay heat removal and Reactor Coolant temperature indication in place of RHR Shutdown Cooling.

A. ADHR

B. RWCU

C. FPC

D. Core Spray

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the given stem states MODE 4 which is Cold Shutdown with all Reactor Pressure Vessel head closure bolts fully tensioned. ADHR can be used as an alternative to RHR Shutdown Cooling, but since ADHR takes suction from and discharges to the Spent Fuel Pool (SFP), ADHR use for RHR SDC alternative is limited to when RPV Head and SFP gates are removed.

B **CORRECT**: *(See attached)* In accordance with 2-OI-74, RHR System, the High Reactor Pressure Isolation signal results in a Loss of Shutdown Cooling. The RWCU System has the capability to remove decay heat using the Non-Regenerative Heat Exchanger rejecting Reactor Coolant heat to the heat sink from the RBCCW System. This is why OI-74 requires that RWCU is removed from service prior to the initiation/operation of RHR Shutdown Cooling. Additionally, OI-74 states that if Shutdown Cooling is secured, RWCU is required to be in service to provide accurate temperature indication for 2-TE-56-8, REACTOR BOTTOM HEAD DRAIN. 2-AOI-74-1, Loss of Shutdown Cooling, requires that RWCU flow rate be raised to maximum and maximize RWCU blowdown as required to maintain Reactor Coolant Temperatures less than 200 °F.

C INCORRECT: Incorrect but plausible in that FPC can be used to remove decay heat from the Vessel, if the Vessel head has been removed (MODE 5), RPV flooded up, and the Fuel Pool gates removed.

D INCORRECT: Incorrect but plausible in that Core Spray is used to maintain Adequate Core Cooling by providing proper the flowrate in order to maintain at least a predetermined Reactor Water Level, not alternate decay heat removal.

RO Level Justification: Tests the candidate's ability to monitor the RWCU System as it relates to a Loss of Shutdown Cooling during MODE 4. This question is rated as C/A due to the requirement to correctly assemble 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 the MODE and available systems as it relates to the complexity of being able to still remove decay heat and provide Reactor Coolant temperature indication in place of the RHR Shutdown Cooling 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): 2-OI-74, Rev. 202 (Attach if not previously provided)
Unit 2 Tech Spec 1.0, Amend. 253
2-AOI-74-1, Rev. 45
0-OI-72, Rev. 72

Proposed references to be provided to applicants during examination: NONE

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

Question Source:

Bank #	Riverbend NRC #47
Modified Bank #	
New	

(Note changes or attach parent)

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

Copy of Bank Question:

2019 RBS NRC Examination

Examination Outline Cross Reference	**Level**	RO
295021 (APE 21) Loss of Shutdown Cooling Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.7 / 45.6) AA1.01 Reactor water cleanup system	**Tier**	1
	Group #	1
	K/A	295021 AA1.01
	Rating	3.4
	Revision	0
Revision Statement:		

Question: 47

Operating in MODE 4, a spurious Level 3 isolation has caused a loss of RHR shutdown cooling.

With the isolation signal sealed in, what system could provide BOTH alternate decay heat removal AND coolant temperature indication in place of RHR shutdown cooling?

A. SPC-ADHR

B. RWCU

C. RHR-FPC

D. LPCS

Answer: B

Excerpts from 2-OI-74:

8.8.1 Initiation / Operation of RHR Loop I in Shutdown Cooling (continued)

[9] [II/C] **ENSURE** a Caution Order is in place on the SYSTEM I MIN FLOW VALVE, 2-FCV-74-7, stating "Operation of this valve can cause inadvertent drainage of the Reactor Vessel to the Suppression Pool. **DO NOT OPERATE** without Unit SRO permission." (Tags should be placed on all points of control.) [BFPER941099]

[10] **ENSURE** RWCU system removed from service per 2-OI-69.

[11] **IF** desired to use RHR Heat Exchanger 2A as the primary Shutdown Cooling Heat Exchanger, **THEN**

PERFORM the following: (Otherwise N/A all of step 8.8.1[11].)

[11.1] **START** a C RHRSW Pump to establish flow through RHR Heat Exchanger 2C.

CAUTIONS

1) **DO NOT EXCEED** 4500 gpm RHRSW flow through any RHR Heat Exchanger.
2) When operating RHRSW through the heat exchangers, damage can occur to the RHRSW discharge valves for the RHR Heat Exchanger if operating at low flows and high differential pressures for long periods. In order to lower the differential pressure the valves experience, flow through the in service heat exchanger(s) should be established such that the total header flow is greater than or equal to 4000 gpm. When operating RHRSW in split mode with other units, this is calculated by adding the individual flows from each of the in-service RHR Heat Exchangers. BFPER 00-003901-000

[11.2] **THROTTLE** 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV, to obtain a flow of 3000 to 4000 gpm.

[11.3] **ESTABLISH** RHRSW cooling flow by performing ONE of the following:

- **REQUEST** another unit START an A RHRSW Pump to be used for Shutdown Cooling and establish minimum flow. REFER TO 0-OI-23.

OR

- **START** an A RHRSW Pump and have another unit establish minimum flow. REFER TO 0-OI-23.

8.8.1 Initiation / Operation of RHR Loop I in Shutdown Cooling (continued)

NOTE

The following is a list of Alternate Methods of Decay Heat Removal that may be used as desired when available: REFER TO the applicable Tech Specs Bases, B 3.4.7, B 3.4.8, B 3.9.7, B 3.9.8.

- ADHR system. REFER TO 0-OI-72.
- Fuel Pool Cooling system. REFER TO 2-OI-78.
- RWCU system. REFER TO 2-OI-69.
- Ambient losses with Natural or Forced circulation.

[41.2] **IF** operating under Tech Specs 3.4.7.E, 3.4.8.C, 3.9.7.C, OR 3.9.8.C, **THEN**: (Otherwise N/A)

- **CHECK** reactor coolant temperature hourly, as required by Tech Spec 3.4.7, 3.4.8, 3.9.7 or 3.9.8.
- **CHECK** reactor coolant circulation by an alternate method one (1) hour from discovery of no reactor coolant circulation AND ONCE per 12 hours thereafter. **TRACK** in the Tech Spec LCO Tracking log.

[41.3] **ENSURE** a Reactor Recirculation Pump is running, **OR** Reactor water level is greater than 70 inches as indicated on 2-LI-3-55.

8.8.1 Initiation / Operation of RHR Loop I in Shutdown Cooling (continued)

[41.4] **IF** reactor vessel cavity is flooded, **AND** the SFSP gates are removed, **THEN**: (Otherwise N/A)

[41.4.1] **DETERMINE** the time for reactor water to heat up. The following maximum allowed values should be used:

- **USE** 195°F if reactor vessel head is installed.
- **USE** 140°F if reactor vessel head is removed.

[41.4.2] **IF** the difference between time for reactor vessel water to heat up and the expected out of service time for shutdown cooling is less than 2 hours, **THEN**: (Otherwise N/A)

- **OBTAIN** Plant Manager, Operations Manager, and Operations Superintendent permission
- **PROVIDE** contingency plans prior to removing shutdown cooling from service.

[41.5] **IF** Reactor vessel cavity is **NOT** flooded, **OR** the SFSP gates are installed, **THEN**: (Otherwise N/A)

- **WHILE** shutdown Cooling is secured, **ENSURE** RWCU System is in service to provide accurate temperature indication for the bottom head drain, 2-TE-56-8, **AND**
- **MONITOR** ICS shutdown cooling screen (RHRSHUT) if available, **OR** Attachment 16 for possible stratification.

8.8.2 Initiation / Operation of RHR Loop II in Shutdown Cooling (continued)

NOTES

1) **IF** Reactor pressure exceeds 100 psig **OR** Group II isolation occurs while Shutdown Cooling is in operation, **THEN** the following actions will occur for the given condition:
   - **(100 psig)** RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47(48), close, thus tripping operating Unit 2 RHR Pumps.
   - **(Group II)** RHR SYS I and II LPCI INBD INJECT VALVEs, 2-FCV-74-53(67) close and RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47(48), close, thus tripping operating Unit 2 RHR Pumps.
2) To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET, 2-XS-74-126(132) pushbutton must be depressed after either of the following occur:
   - Isolation signal has been reset **OR**:
   - 2-FCV-74-47 or 2-FCV-74-48 are fully closed.
3) All operations are performed at Panel 2-9-3 unless otherwise noted.

CAUTION

Shutdown Cooling operation at saturated conditions (212°F) with two RHR pumps operating at or near combined maximum flow (20,000 gpm) could cause Jet Pump cavitation. Indications of Jet Pump cavitation are as follows:

1) Rise in RHR System flow without a corresponding rise in Jet Pump flow
2) Fluctuation of Jet Pump flow
3) Louder "rumbling" noise heard when vessel head is off

Corrective action for any of these symptoms would be to reduce RHR flow until the symptom is corrected.

[36] **OPERATE** Shutdown Cooling as follows

[36.1] **MAINTAIN** Reactor pressure less than 100 psig. REFER TO applicable GOI.

Excerpt from 2-AOI-74-1:

BFN Unit 2	Loss of Shutdown Cooling	2-AOI-74-1 Rev. 0045 Page 17 of 32

4.2 Subsequent Actions (continued)

[4.8.3] **REQUEST** the SRO to **ESTIMATE** the following times at least once per shift until a method of decay heat removal is restored:

A. **DETERMINE** the time since shutdown.

B. **DETERMINE** the current RPV heat-up rate from 2-SR-3.4.9.1(1), or, if reactor coolant stratification is suspected, **USE** Attachment 1.

- **IF** additional information is required to determine the heat-up rates, **THEN**

  CONTACT Reactor Engineer.

C. **DETERMINE** the reactor coolant temperature or use the last valid reactor coolant temperature available.

D. **IF** the Reactor Vessel head is removed and the cavity is flooded with the fuel pool gates installed, **THEN** (Otherwise N/A)

   ESTIMATE the time for reactor coolant temperature to reach 125°F and 150°F using a plot of the actual heatup rate or Attachment 1.

E. **ESTIMATE** the time for reactor coolant temperature to reach 212°F, using data obtained in Steps 4.2[4.8.3]A through 4.2[4.8.3]C.

[4.8.4] **RAISE** RWCU flow rate to maximum AND **MAXIMIZE** RWCU blowdown as required to maintain reactor coolant temperatures less than 200°F on all indications. **REFER TO** 2-OI-69. (N/A if RWCU is unavailable)

[4.8.5] **RAISE** CRD flow. **REFER TO** CRD Pump Operation at Elevated Flow section. **REFER TO** 2-OI-85 (N/A if CRD is unavailable)

Excerpt from U2 Tech Spec 1.0:

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel(a) or Startup/Hot Standby	NA
3	Hot Shutdown(a)	Shutdown	> 212
4	Cold Shutdown(a)	Shutdown	≤ 212
5	Refueling(b)	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Excerpt from 0-OI-72:

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General Precautions

A. ADHR System may be aligned to the Unit 1, 2, or 3 Fuel Pool. This system may also be used in conjunction with the Fuel Pool Cooling System and/or Shutdown Cooling Mode of RHR and/or the Supplemental Mode of Fuel Pool Cooling of RHR. The ADHR System is to be used as required, at SRO direction.

B. Secondary Containment Integrity requirements must be adhered to any time Secondary Containment is required.

C. 0-AOI-72-1, Auxiliary Decay Heat Removal System Failures, is referred to when Loss of Power occurs, Primary or Secondary pumps trip, or Fuel Pool temperature approaches 125°F for a Unit utilizing the ADHR System.

D. All maintenance activities to be performed for this instruction will be done by an approved work order.

E. Root valves may need to be throttled to reduce erratic readings on instruments.

F. Loop operations

1. Single-Loop operation consists of one Primary pump (A or B) and one Secondary pump (A or B) in service.

2. Dual-Loop operation consists of two Primary pumps and two Secondary pumps in service. See P&L 3.7A for Dual-Loop operation.

G. Secondary Loop could contain tritium, which has potential for groundwater contamination.

H. Backflow preventers must be used when connecting external water supplies to ADHR piping.

3.2 Operability and LCO's

A. The following is a list of Alternate Methods of Decay Heat Removal that may be used as desired when available: (**REFER TO** the applicable Tech Spec Bases B 3.4.7, B 3.4.8, B 3.9.7, B 3.9.8.)

1. ADHR system. **REFER TO** 0-OI-72.

2. Fuel Pool Cooling system. **REFER TO** 1(2)(3)-OI-78.

3. RWCU system. **REFER TO** 1(2)(3)-OI-69.

4. Ambient losses with Natural or Forced circulation.

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

Examination Outline Cross-reference:

212000 (SF7 RPS) Reactor Protection
A4.18 (10CFR 55.41.7)
Ability to manually operate and/or monitor in the control room:

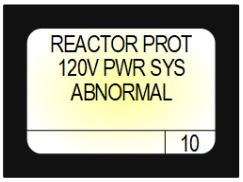
- Transferring RPS power supplies

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	212000A4.18	
Importance Rating	3.6	------

Proposed Question: **# 32**

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

- 2B Reactor Protection System (RPS) Motor Generator (MG) trips
- REACTOR PROTECTION 120V POWER SYSTEM ABNORMAL (2-9-5B, Window 10) alarms
- 2B RPS is then transferred to its alternate power supply
- **NO** other Operator actions have been taken



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

(1) closed.

In accordance with the given ARP, a half-SCRAM **(2)** occur.

Note: 2-FCV-64-139, DRYWELL TO SUPPRESSION CHAMBER DP COMPRESSOR SUCTION VALVE

2-FCV-64-140, DRYWELL TO SUPPRESSION CHAMBER DP COMPRESSOR DISCHARGE VALVE

A. (1) **ONLY** 2-FCV-64-139
(2) did

B. (1) **ONLY** 2-FCV-64-139
(2) did NOT

C. (1) 2-FCV-64-139 **AND** 2-FCV-64-140
(2) did

D. (1) 2-FCV-64-139 **AND** 2-FCV-64-140
(2) did NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that only specific system valves will close based on RPS A or RPS B de-energizing as illustrated in 2-OI-99, Attachment 1. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that RPS has numerous complexities associated with alignments of RPS A, RPS B, RPS circuit protectors, normal and alternate power supplies, etc. Additionally, numerous BFN electrical supply system alignments and fault responses are easily and often confused.

C **CORRECT**: (*See attached*) In accordance with the given ARP, de-energizing one RPS channel actuates every SCRAM-related alarm for the affected channel and the Operator is required to reference 2-AOI-99-1, Loss of Power to One RPS Bus. This is where an overview of automatic actions for a loss of RPS Bus A(B) are listed, however 2-OI-99, Illustration 1, is the detailed list referenced for more specific actions. This shows that for a loss or power transfer of RPS Bus A or B, 2-FCV-64-139 and 140 both close. Second part, in accordance with the given ARP, 2-AOI-99-1 is referenced which also states an RPS trip logic A(B) half-SCRAM occurs.

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 manually operate and/or monitor from the Control Room the Reactor Protection System equipment as it relates to transferring or a loss of RPS power supplies. 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): 2-OI-99, Rev. 101 (Attach if not previously provided)
2-ARP-9-5B, Rev. 33
2-AOI-99-1, Rev. 33

Proposed references to be provided to applicants during examination: REACTOR PROT 120V PWR SYS ABNORMAL (2-9-5B, Window 10)

Learning Objective: OPL171.028 Obj. 10a (As available)

Question Source: Bank # _____
Modified Bank # BFN NRC 2104 #34 (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

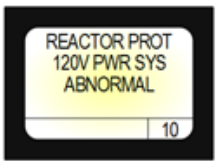
Comments:

Copy of Bank Question:

Proposed Question: **# 34**

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

- 2A Reactor Protection System (RPS) Motor Generator (MG) trips
- REACTOR PROTECTION 120V POWER SYSTEM ABNORMAL (2-9-5B, Window 10) alarms



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

(1) RPS logic channels are impacted. As a result, a **(2)** occurs.

A. (1) A1 and A2 **ONLY**
(2) Full SCRAM

B. (1) A1 and A2 **ONLY**
(2) Half-SCRAM **ONLY**

C. (1) A1, A2 **AND** A3
(2) Full SCRAM

D. (1) A1, A2 **AND** A3
(2) Half-SCRAM **ONLY**

Proposed Answer: **D**

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. 0033 Page 4 of 20

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.

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.

3.0 AUTOMATIC ACTIONS

NOTE

An overview of automatic actions for RPS Bus A(B) is provided here. A detailed list of actions is provided in 2-OI-99, Attachment 1, which lists actions that occur when RPS buses are de-energized on a transfer of power supply.

A. RPS trip logic A(B) half-scram occurs.

B. PCIS Group 1 half-trip logic de-energizes.

C. PCIS Group 2 isolation, RHR Shutdown Cooling Mode:

1. Bus A inboard.
2. Bus B outboard.

D. PCIS Group 3 isolation, RWCU:

1. Bus A inboard and outboard.
2. Bus B outboard.

E. PCIS Group 6 isolation, Primary Containment Vent and Purge and Reactor Building Ventilation:

1. Bus A or B inboard and outboard.

F. Group 8 isolation, TIP.

G. Control Room Emergency Ventilation System start.

H. Standby Gas Treatment System starts.

Excerpts from 2-OI-99:

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0101 Page 96 of 112

Attachment 1
(Page 1 of 4)

RPS Bus A or B Power Transfer

A. Loss of power supply to either RPS Bus A or B will result in the following events:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-64-17	Drywell/Suppression Chamber purge inlet	CLOSES
FCV-64-18	Drywell purge inlet inboard	CLOSES
FCV-64-19	Suppression Chamber purge inlet inboard	CLOSES
FCV-64-29	Drywell purge exhaust inboard	CLOSES
FCV-64-30	Drywell purge exhaust outboard	CLOSES
FCV-64-31	Drywell purge exhaust bypass to SGTS	CLOSES
FCV-64-32	Suppression Chamber purge exhaust inboard	CLOSES
FCV-64-33	Suppression Chamber purge exhaust outboard	CLOSES
FCV-64-34	Suppression Chamber purge exhaust bypass to SGTS	CLOSES
FCV-64-36	Drywell/Suppr Chbr purge exhaust to SGTS	CLOSES
FCV-64-139	Drywell to Suppr Chbr DP compressor suction	CLOSES
FCV-64-140	Drywell to Suppr Chbr DP compressor discharge	CLOSES
FCV-76-17	Drywell/Suppression Chamber nitrogen purge inlet	CLOSES
FCV-76-24	Drywell/Suppression Chamber nitrogen purge inlet	CLOSES
FCV-76-18	Drywell nitrogen purge inlet	CLOSES
FCV-76-19	Suppression Chamber nitrogen purge inlet	CLOSES
FCV-76-49	Containment Inerting System A sample	CLOSES
FCV-76-50	Containment Inerting System A sample	CLOSES
FCV-76-55	Containment Inerting System A sample	CLOSES
FCV-76-56	Containment Inerting System A sample	CLOSES
FCV-76-57	Containment Inerting System A sample	CLOSES
FCV-76-58	Containment Inerting System A sample	CLOSES
FCV-76-59	Containment Inerting System B sample	CLOSES
FCV-76-60	Containment Inerting System B sample	CLOSES
FCV-76-65	Containment Inerting System B sample	CLOSES
FCV-76-66	Containment Inerting System B sample	CLOSES
FCV-76-67	Containment Inerting System B sample	CLOSES
FCV-76-68	Containment Inerting System B sample	CLOSES
FCV-84-20	Drywell or Suppr Chbr exhaust to SGTS	CLOSES
FCV-90-254A	Drywell radiation monitoring sample	CLOSES

7.3 De-energization of RPS Bus 2B

[1] **REVIEW** all Precautions and Limitations in Section 3.0.

[2] **PERFORM** Scram Solenoid Temperature Checks per Section 8.14.

[3] **REVIEW** Attachment 1 to determine affected systems and components.

NOTE

The following Step 7.3[4] will de-energize RPS Bus 2B.

CAUTIONS

1) De-energization of RPS Bus 2B causes a half-scram and associated isolations. Operation of RHR Shutdown Cooling and Reactor Water Cleanup System is **NOT** possible while RPS Bus 2B is de-energized.
2) If any MSIV has a failed solenoid circuitry, deenergizing a RPS BUS may cause a fast closure of the MSIV. Fast closure of a MSIV at 100% Power results in a Reactor Scram.

[4] **IF** RPS MG Set 2B is providing power to RPS Bus 2B, **THEN**

[4.1] **ENSURE** an inadvertent fast closure of an MSIV is **NOT** expected due to the RPS bus de-energization.

[4.2] **IF** a fast closure is expected with the unit at power, **THEN**

REFER TO 2-OI-1

[4.3] **IF** De-energizing the "2B" RPS Bus, **THEN**

At Battery Board 2, **OPEN** RPS MG SET B TIE TO BAT BD 2, 0-BKR-280-0002/952.

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0101 Page 101 of 112

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

Supports Distractors A1, B1: illustrates variation of PCIS responses to RPS losses

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0101 Page 98 of 112

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
FCV-74-48	RHR shutdown cooling inboard suction	CLOSES
FCV-74-53	RHR System I inboard injection	CLOSES
FCV-74-102	RHR System HP flush/vent	CLOSES
FCV-74-103	RHR System LP flush/vent	CLOSES
FCV-75-57	PSC Pump Suction Inboard Isolation Valve	CLOSES
FCV-77-15A	Drywell equipment drain discharge	CLOSES
FCV-77-2A	Drywell floor drain discharge	CLOSES
FCV-69-1	RWCU inlet	CLOSES
FCV-69-2	RWCU inlet	CLOSES
FCV-69-12	RWCU outlet	CLOSES
FCV-1-14	MSIV AC control power	DE-ENERGIZES
FCV-1-26	MSIV AC control power	DE-ENERGIZES
FCV-1-37	MSIV AC control power	DE-ENERGIZES
FCV-1-51	MSIV AC control power	DE-ENERGIZES
FCV-1-55	Main Steam Line drain inboard	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
FCV-74-47	RHR shutdown cooling outboard suction	CLOSES
FCV-74-67	RHR System II inboard injection	CLOSES
FCV-74-119	RHR System HP flush/vent	CLOSES
FCV-74-120	RHR System LP flush/vent	CLOSES
FCV-75-58	PCS Pump Suction Outboard Isolation Valve	CLOSES
FCV-77-15B	Drywell equipment drain discharge	CLOSES
FCV-77-2B	Drywell floor drain discharge	CLOSES
FCV-69-2	RWCU inlet	CLOSES
FCV-69-12	RWCU outlet	CLOSES
FCV-1-15	MSIV AC control power	DE-ENERGIZES
FCV-1-27	MSIV AC control power	DE-ENERGIZES
FCV-1-38	MSIV AC control power	DE-ENERGIZES
FCV-1-52	MSIV AC control power	DE-ENERGIZES
FCV-1-56	Main Steam Line drain outboard	CLOSES

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

Examination Outline Cross-reference:	Level	RO	SRO
295016 (APE 16) Control Room Abandonment / 7	Tier #	1	------
AA2.03 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT:	K/A #	295016AA2.03	
• Reactor Pressure	Importance Rating	4.5	------

Proposed Question: **# 33**

In accordance with 2-AOI-100-2, Control Room Abandonment, which **ONE** of the following completes the statements below regarding Reactor Pressure Control?

At Backup Control Panel 2-25-32, Reactor Pressure is **controlled** **(1)**.

RCIC **(2)** be used for Reactor **PRESSURE** Control from Panel 2-25-32.

A. (1) manually **ONLY**
(2) can

B. (1) manually **ONLY**
(2) can NOT

C. (1) manually **OR** automatically
(2) can

D. (1) manually **OR** automatically
(2) can NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part incorrect but plausible in that RCIC can be used for both Reactor Pressure Control and Reactor Water Level Control normally at Panel 2-9-3 in the MCR.

B **CORRECT**: *(See attached)* First part is correct in that AOI-100-2 specifically establishes manual pressure control using SRVs to establish and execute a cooldown to cold shutdown conditions. Second part, in accordance with AOI-100-2, RCIC is only used for Reactor Water Level Control from Panel 2-25-32.

C INCORRECT: First part is incorrect but plausible in that while automatic SRV relief setpoint operation remains active from the Backup Control Panel, it is neither controlling pressure to establish and maintain a cooldown, nor is use in this manner directed in AOI-100-2. Second part 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 the status of Reactor Pressure Control as it applies to Control Room Abandonment. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to 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. 62 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.040, 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	

10 CFR Part 55 Content:

55.41	X
55.43	

Excerpts from 2-AOI-100-2:

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	☐

4.2 Unit 2 Subsequent Actions (continued)

[3] **PLACE** the following MSRV disconnect switches in DISCT at Panel 2-25-32:

Switch No.	Description	
2-XS-1-18	MAIN STM LINE B RELIEF VALVE DISCT	☐
2-XS-1-4	MAIN STM LINE A RELIEF VALVE DISCT	☐
2-XS-1-42	MAIN STM LINE D RELIEF VALVE DISCT	☐
2-XS-1-23	MAIN STM LINE B RELIEF VALVE DISCT	☐
2-XS-1-41	MAIN STM LINE D RELIEF VALVE DISCT	☐
2-XS-1-180	MAIN STM LINE D RELIEF VALVE DISCT	☐

[4] **PLACE** the following MSRV transfer switches in EMERG at Panel 2-25-32:

Switch No.	Description	
2-XS-1-22	MAIN STM LINE B RELIEF VALVE XFR	☐
2-XS-1-5	MAIN STM LINE A RELIEF VALVE XFR	☐
2-XS-1-30	MAIN STM LINE C RELIEF VALVE XFR	☐
2-XS-1-34	MAIN STM LINE C RELIEF VALVE XFR	☐

NOTE

Use of the following sequence when opening MSRVs should distribute heat evenly in the Suppression Pool.

[5] **MAINTAIN** Reactor Pressure between 800 and 1000 psig using the following sequence at Panel 2-25-32:

A. 2-HS-1-22C, MAIN STM LINE B RELIEF VALVE

B. 2-HS-1-5C, MAIN STM LINE A RELIEF VALVE

C. 2-HS-1-30C, MAIN STM LINE C RELIEF VALVE

D. 2-HS-1-34C, MAIN STM LINE C RELIEF VALVE

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) RCIC Turbine should start and flow should stabilize at 620 gpm.
2) Reference Attachment 12 for RCIC controller operation including HARD MANUAL mode that may be used in the case of a controller failure.

[9.4] **CHECK** turbine speed 2100 rpm or above using RCIC TURBINE SPEED, 2-SI-71-42B at Panel 2-25-32.

[9.5] **PLACE** RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, 2-HS-071-0034C, in CLOSE at 250V DC RMOV Bd 2B, compt. 5D. (Unit 2 Turbine Building AUO)

[9.6] **ADJUST** flowrate as necessary using RCIC SYSTEM FLOW/CONTROL, 2-FIC-71-36B at Panel 2-25-32.

[9.7] **MAINTAIN** Reactor Water Level between +2 and +50 inches using RX WATER LEVEL A & B, 2-LI-3-46A & B at Panel 2-25-32.

NOTE

The following step prevents HPCI operation and automatic opening of HPCI MAIN PUMP MINIMUM FLOW VALVE, 2-FCV-73-30.

[10] At 250V Reactor MOV Bd 2A, **PERFORM** the following:

[10.1] **ENSURE CLOSED** HPCI STEAM SUPPLY VALVE TO TURB FCV-73-16 at Compt. 3D. (MO 23-14).

[10.2] **PLACE** HPCI TURBINE STEAM SUP VLV TRANS, 2-XS-73-16, in EMERG at Compt. 3D.

[10.3] **IF** desired to verify HPCI MIN FLOW BYPASS TO SUPPRESSION CHAMBER VALVE, 2-FCV-73-30, closed prior to opening breaker, **THEN**
(Otherwise N/A)

DIRECT operator to verify locally.

[10.4] **PLACE** HPCI MAIN PUMP MIN FLOW VLV FCV-73-30, breaker in OFF Compt. 8D.

4.2 Unit 2 Subsequent Actions (continued)

[11] **ESTABLISH** plant cooldown as follows:

NOTE

Use of the following sequence when opening MSRVs should distribute heat evenly in the Suppression Pool.

[11.1] At Panel 2-25-32, **DEPRESSURIZE** and **COOL DOWN** the reactor at 90°F/hr or less by periodically opening MSRVs using the following sequence:

A. 2-HS-1-22C, MAIN STM LINE B RELIEF VALVE

B. 2-HS-1-5C, MAIN STM LINE A RELIEF VALVE

C. 2-HS-1-30C, MAIN STM LINE C RELIEF VALVE

D. 2-HS-1-34C, MAIN STM LINE C RELIEF VALVE

[11.2] **RECORD** the following:

- Time cooldown commenced _____
- Initial Reactor Pressure _____

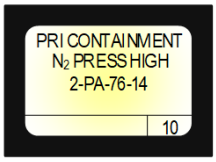
[11.3] **REFER TO** Attachment 8, Plant Cooldown Chart, to determine cooldown rate.

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

Examination Outline Cross-reference:	Level	RO	SRO
295024 (EPE 1) High Drywell Pressure / 5	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 #	295024G2.1.31	
	Importance Rating	4.6	------

Proposed Question: **# 34**

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



- PRIMARY CONTAINMENT NITROGEN PRESSURE HIGH (2-9-3B, Window 10) alarms
- Drywell Pressure is 1.5 psig, rising
- NUSO directs venting the Drywell in accordance with 2-OI-64, Primary Containment Systems

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

Standby Gas Train **(1)** will be started from Panel 9-25 on **(2)** .

A. (1) 'A'
(2) Unit 2

B. (1) 'B'
(2) Unit 2

C. (1) 'C'
(2) Unit 2

D. (1) 'C'
(2) Unit 1

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that SGT A can only be started from Unit 1 or 3's Panel 9-25.

B INCORRECT: Incorrect but plausible in that SGT B can only be started from Unit 1 or 3's Panel 9-25.

C **CORRECT**: *(See attached)* In accordance with the given ARP, in order to vent the Drywell, 2-OI-64 is referenced. 2-OI-64, Section 5.2, SGT Manual Initiation, states that SGT C is started from Unit 2's Panel 9-25.

D INCORRECT: Incorrect but plausible in that SGT C can only be started from Unit 2 or 3's Panel 9-25.

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

RO Level Justification: Tests the candidate's ability to locate Control Room switches, controls, and indications to determine they correctly reflect the desired plant lineup as it relates to the Standby Gas Treatment System required for 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. 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-OI-64, Rev. 132 (Attach if not previously provided)
3-OI-64, Rev. 71
0-OI-65, Rev. 57
2-ARP-9-3B, Rev. 40

Proposed references to be provided to applicants during examination: PRIMARY CONTAINMENT NITROGEN PRESSURE HIGH (2-9-3B, Window 10)

Learning Objective: OPL171.018 Obj. 11 (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-ARP-9-3B:

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0040 Page 14 of 39

PRI CONTAINMENT N_2 PRESS HIGH
2-PA-76-14

10

(Page 1 of 1)

Sensor/Trip Point:
PT-76-14 1.5 psig (alarm from recorder).

Sensor Location: Panel 25-6B
Rx Bldg, El 593, R-12 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** 2-OI-64, Venting the drywell with standby gas treatment fan.

References: 45E620-3 47W600-57A 2-47E610-76-1
GE 730E943-1

Excerpt from 2-OI-64:

6.1.1 Venting Lineup (continued)

NOTE

1) A Standby Gas Treatment Fan must be running for 2-FCV-84-20 to operate.
2) Depending upon plant conditions, it is possible that flow may indicate less than 100 SCFM even with the thumbwheel of 2-FIC-84-20 on Panel 2-9-55 adjusted to the fully open position.
3) Drywell pressure is usually controlled to maintain the Drywell and Suppression Chamber differential pressure between 1.15 and 1.30 psid to provide a margin to the Tech Spec limit.

CAUTION

Path A and B Vent valves, 2-FCV-84-19 and 2-FCV-84-20, isolate on Standby Gas duct high pressure of 1 psig.

[11] **PLACE** a Standby Gas Treatment Train in service as follows: (N/A this section if SBGT is already in service.)

[11.1] **CONTACT** Unit 1 to determine which SBGT Train to be started.

[11.2] **START** the desired Standby Gas Treatment Train using the appropriate hand-switch listed below:

A. On Panel 1-9-25

- SGTS TRAIN A FAN, 0-HS-65-18A/1
- SGTS TRAIN B FAN, 0-HS-65-40A/1

B. On Panel 2-9-25

- SGTS TRAIN C FAN, 0-HS-65-69A/2 .

[11.3] **RECORD** in the Narrative Logs the SBGT Train started and the Appropriate Filter DP.

Excerpts from 0-OI-65:

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0057 Page 18 of 44

5.2 Standby Gas Treatment System Manual Initiation (continued)

[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.

[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.

- REACTOR ZONE EXH TO SGTS dampers, 1-HS-64-40 and 1-HS-64-41 on Panel 1-9-25
- REACTOR ZONE EXH TO SGTS dampers, 2-HS-64-40 and 2-HS-64-41 on Panel 2-9-25
- REACTOR ZONE EXH TO SGTS dampers, 3-HS-64-40 and 3-HS-64-41 on Panel 3-9-25

[3.3] **IF** alignment to Refuel Zone Ventilation suction path is desired, **THEN**

ENSURE OPEN REFUEL ZONE EXH TO SGTS, 1-HS-64-44 and 1-HS-64-45 on Panel 1-9-25.

[4] **START** SGT FAN A(B)(C) as follows:

[4.1] **IF** starting SGT FAN A(B) from Panel 1-9-25, **THEN**

PLACE SGTS TRAIN A(B) FAN, 0-HS-65-18A/1(40A/1) in START.

[4.2] **IF** starting SGT FAN C from Panel 2-9-25, **THEN**

PLACE SGTS FAN C, 0-HS-65-69A/2 in START.

[4.3] **IF** starting SGT FAN A(B)(C) from Panel 3-9-25, **THEN**

DEPRESS SGTS TRAIN A(B)(C) FAN, 0-HS-65-18A/3(40A/3)69A/3) push-button.

7.1 Standby Gas Treatment System Shutdown (continued)

[3] **ENSURE** reactor/refuel zone normal ventilation systems in service. **REFER TO** 1(2)(3)-OI-30A and 1(2)(3)-OI-30B.

NOTE

Filter bank differential pressure(s) obtained in Step 7.1[4] may be recorded in the Narrative Log after the SBGT Train has been secured.

[4] **RECORD** filter bank differential pressure for SGT Train A(B)(C) as follows:

[4.1] **IF** SGT FAN A(B) is to be secured, **THEN**

RECORD FILTER BANK DIFFERENTIAL PRESSURE, 0-PDI-65-5(27) on Panel 1-9-25, in the Narrative Log.

[4.2] **IF** SGT FAN C is to be secured, **THEN**

RECORD FILTER BANK DIFFERENTIAL PRESSURE, 0-PDI-65-53 on Panel 2-9-25, in the Narrative Log.

NOTE

Due to SBGT configuration, the SBGT Train Filter Bank Heater Temperature High alarm may occur up to 30 minutes after the A and/or B trains have been shutdown, but should clear shortly thereafter.

[5] **MOMENTARILY PLACE** SGTS TRAIN A(B) FAN and SGTS TRAIN C FAN, 0-HS-65-18A/1(40A/1) and 69A/2 on Panel 1-9-25(1-9-25) and 2-9-25 in STOP.

Excerpt from 3-OI-64:

8.4 Venting the Suppression Chamber when Drywell DP Compressor and DRYWELL CPRSR BYPASS 3-FCV-64-141 are Out of Service (continued)

[8] **NOTIFY** Unit 1 and Unit 2 Control Rooms that Unit 3 Suppression Chamber venting with Standby Gas Treatment Train A (B) (C) is about to be started.

[9] **START** one of the following SGT trains by depressing the appropriate push-button (REFER TO 0-OI-65):

A. For SGT FAN A, 0-HS-65-18A/3 on Panel 3-9-25

B. For SGT FAN B, 0-HS-65-40A/3 on Panel 3-9-25

C. For SGTS TRAIN C FAN, 0-HS-65-69A/3 on Panel 3-9-25

[10] **OPEN** DW/SUPPR CHBR VENT TO SGT, 3-FCO-64-36, using 3-HS-64-36 (Panel 3-9-3).

[11] **MONITOR** Drywell/Torus DP on DW/SUPPR CHBR DIFF PRESS, 3-PDI-64-137 and 138 (Panel 3-9-3).

[12] **WHEN** DW/SUPPR CHBR DIFF PRESS, 3-PDI-64-137 or 138 is between 1.15 psid and 1.30 psid (Panel 3-9-3), **THEN**

STOP Suppression Chamber venting by performing the following:

[12.1] **CLOSE** DW/SUPPR CHBR VENT TO SGT, 3-FCO-64-36, using 3-HS-64-36 (Panel 3-9-3).

[12.2] **NOTIFY** Unit 1 and Unit 2 Control Rooms that Unit 3 Suppression Chamber venting has been completed and **REQUEST** the SGT Fan started in Step 8.4[9] to be secured.

[12.3] **CLOSE** SUPPR CHBR VENT INBD ISOLATION VLV, 3-FCV-64-32, using 3-HS-64-32 (Panel 3-9-3).

[12.4] **CLOSE** SUPPR CHBR VENT OUTBD ISOLATION VLV, 3-FCV-64-33, using 3-HS-64-33 (Panel 3-9-3).

[13] **PLACE** the following switches in the NORMAL position (Panel 3-9-3).

- PC PURGE DIV I RUN MODE BYPASS, 3-HS-64-24
- PC PURGE DIV II RUN MODE BYPASS, 3-HS-64-25

[14] **CHECK** Drywell/Suppression Chamber DP between 1.15 and 1.30 psid on DW/SUPPR CHBR DIFF PRESS, 3-PDI-64-137 and 138 (Panel 3-9-3).

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

Examination Outline Cross-reference:	Level	RO	SRO
215004 (SF7 SRMS) Source-Range Monitor System **K4.03** (10CFR 55.41.7)	Tier #	2	------
Knowledge of SOURCE RANGE MONITOR SYSTEM design features and/or interlocks that provide for the following: • Gamma compensation	Group #	1	------
	K/A #	215004K4.03	
	Importance Rating	2.7	------

Proposed Question: **# 35**

A Unit 2 startup is in progress and the crew is monitoring the SRMs for Reactor Power changes.

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

The output of the SRMs are compensated for gamma interactions with the detectors

(1) Reactor Power levels since gamma interactions produce **(2)** detector pulses than neutron interactions.

A. (1) at **ALL**
(2) smaller

B. (1) at **ALL**
(2) larger

C. (1) **ONLY** when shutdown or at **LOW**
(2) smaller

D. (1) **ONLY** when shutdown or at **LOW**
(2) larger

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that IRMs and other detectors are still monitoring Reactor Power at higher levels while SRMs are withdrawn once IRM overlap criteria has been met. 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 gamma/neutron interactions associated with SRM detectors which require larger operating voltage (350VDC) than IRM detectors (100VDC).

C **CORRECT**: (*See attached*) The output of the SRMs are compensated for gamma interactions with the detectors only when shutdown or low Reactor Power levels. The SRM pulse preamplifier raises the gamma pulse from the SRM prior to routing to the SRM drawer in the Control Room, thereby improving the signal-to-noise ratio. The amplified pulse then enters the pulse height discriminator to eliminate the gamma pulse. During a startup, once IRMs are on Range 3 or above and meet all requirements for SRM/IRM overlap, the SRMs are withdrawn. Second part, SRMs are compensated for gamma interactions since gammas produce smaller detector pulses than neutron interactions.

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 Source Range Monitor System design features and/or interlocks that provide gamma compensation. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to SRMs.

Technical Reference(s): GFE Ch.7 Sensors and Detectors, Rev. 4 (Attach if not previously provided)
OPL171.019, Rev. 15
2-GOI-100-1A, Rev. 198

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.019, Obj. 4c (As available)

Question Source:

Bank #	GFE CH. 7 B3112
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:

Excerpt from GFE Ch.7 Sensors and Detectors:

BPC07Gr4_Sensors 2011May

7 / 0 QID: B3112 Operation of fission chambers, ion chambers	
BWR TOPIC: 291002	PWR TOPIC: 191002
KNOWLEDGE: BK1.19 [3.0/3.1]	KNOWLEDGE:
Similar questions	
Recent BWR exams	Recent PWR exams
Dec2009, Dec2007, Sep2006, Sep2004, Jun2003, Oct2002, Jun2001	Not used in 2001 through 2009Dec

Fission chamber detectors are used to monitor reactor power/neutron level in a shutdown reactor as well as a reactor operating at full power (and all power levels in between). At what power levels and why is it necessary to compensate the output of the detectors for gamma interactions with the fission chambers?

A. At all power levels, because gamma interactions produce larger detector pulses than neutron interactions.

B. At all power levels, because gamma interactions produce smaller detector pulses than neutron interactions.

C. Only when shutdown or at low power levels, because gamma flux is not proportional to reactor power at low power levels.

D. Only when operating at high power levels, because gamma flux is not proportional to reactor power at high power levels.

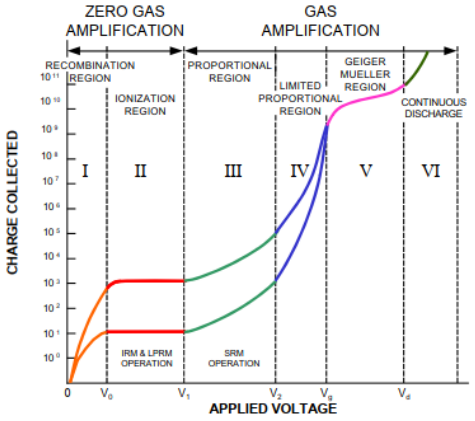
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BPC07Gr4_Sensors 2011May

7 / 0 QID: B3112 Operation of fission chambers, ion chambers

Explanation

Fission chamber detectors are used to monitor reactor power/neutron level in a shutdown reactor as well as a reactor operating at full power (and all power levels in between). At what power levels and why is it necessary to compensate the output of the detectors for gamma interactions with the fission chambers?



A. At all power levels (FALSE), because gamma interactions produce larger detector pulses than neutron interactions (FALSE).
Incorrect - gamma compensation is only required at shut down or low power conditions. At high power operations gamma compensation is not needed because the neutron to gamma current ratio is about 1,000 to 1. Therefore, it is not necessary to compensate for gammas in the power range detector. Gamma interactions produce SMALLER detector pulses than neutron interactions

B. At all power levels (FALSE), because gamma interactions produce smaller detector pulses than neutron interactions (TRUE but not a concern).
Incorrect - gamma compensation is only required at shut down or low power conditions. At high power operations gamma compensation is not needed because the neutron to gamma current ratio is about 1,000 to 1. Therefore, it is not necessary to compensate for gammas in the power range detector.

C. Only when shutdown or at low power levels (TRUE), because gamma flux is not proportional to reactor power at low power levels (TRUE).
CORRECT - When the reactor is operating at power, the neutron to gamma current ratio is about 1,000 to 1. Therefore, it is not necessary to compensate for gammas in the power range detector.

D. Only when operating at high power levels (FALSE), because gamma flux is not proportional to reactor power at high power levels (TRUE but not a concern).
Incorrect - gamma compensate is NOT require when the reactor is operating at high power, because the neutron to gamma current ratio is about 1000 to 1.

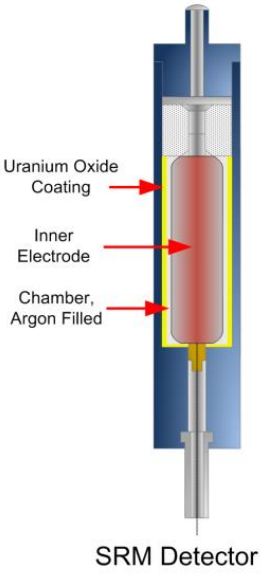
The correct answer is ANSWER C. Only when shutdown or at low power levels, because gamma flux is not proportional to reactor power at low power levels.

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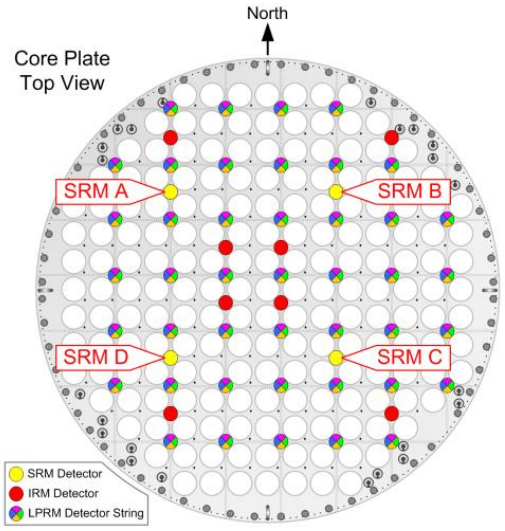
Excerpts from OPL171.019 Lesson Plan:

In-core SRM Detector

The purpose of the in-core detector (detection chamber) is to generate an electrical signal proportional to the neutron flux level in the core. The SRM detection (fission) chambers sit in gaps between the fuel bundles on the opposite corner from the control rod at strategic locations within the core. The SRM detector is ≈1.6 inch by ≈0.3 inches with an active fission chamber length of ≈1 inch. It is argon gas filled to 1 atmosphere and has an inner and outer electrode (two electrodes). The inner surface of the outer electrode is coated with a very thin coat (6.25 mg) of U_3O_8; which is 90 percent enriched in U235. When a thermal neutron penetrates the coating, there is a chance that it will fission a U235 atom causing fission products to recoil into the chamber. The fission products, being heavy, highly charged, energetic particles, will cause ionization by stripping electrons from the argon gas in the chamber.



SRM Detector



The detector inner and outer electrode is supplied with 350 VDC by a high voltage power supply; which causes the inner electrode to be positive with respect to the outer electrode. The bias voltage on the electrodes will cause the negative ions (primarily electrons) to be collected by the inner electrode and the positive ions (primarily argon atoms with electrons stripped away) to be collected by the outer electrode. (For knowledge purposes only, the heavy fission fragments tend to collect on the outer electrode.) As the electrons strike the inner electrode, they cause a current pulse to be generated in the external circuit as do the argon ions when they strike the outer electrode. These pulses are then passed to the Pulse Preamplifier.

Gamma radiation can cause a detector response also. Gammas come from two sources: fission gammas and fission product decay gammas. A gamma entering the chamber can directly ionize the argon in the space between the electrodes. The argon ions and the electrons are collected and current pulses generated just as in the neutron event. The specific ionization

(ions produced per unit track length) of the gamma is much less than for the fission products; therefore, many less ions are produced by the gamma than by the neutron event. The gamma current pulses applied to the external circuit are therefore much smaller than the neutron pulses. This difference in pulse size allows the pulse height discriminator section of the SRM to eliminate the gamma pulses while passing neutron pulses.

A visible neutron count rate in the core is necessary to ensure the SRMs are operating properly. Technical Specifications require that the SRMs indicate at least 3 cps to be considered operable.

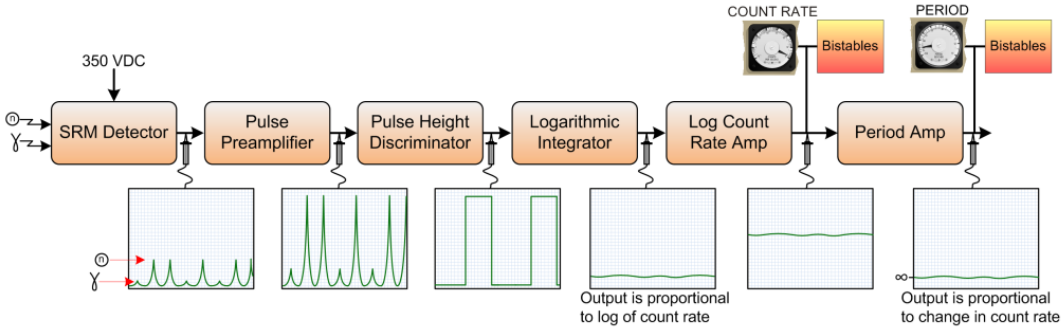
During initial fuel loading, initial startup and periods of long shutdown conditions installed neutron sources may be required to maintain ≥ 3 cps. For Unit 2, Cycle 6, Californium-252 installed sources were utilized. The spontaneous fission of Californium-252 in these installed sources provides an adequate neutron count level for SRM operability.

Once a reactor has achieved a significant operating history, installed sources are usually no longer required and intrinsic neutron sources produce sufficient neutrons to maintain ≥ 3 cps. This is primarily due to the production of Curium-242 and 244 isotopes during the fission process; which, spontaneously fission or decay to generate neutrons through nuclear reactions.

With a large portion of a core unloaded, or after long periods of shutdown conditions, SRM detectors may not indicate the necessary 3 cps. Under these conditions, special detectors (Fuel Loading Chambers (FLC) also known as dunking chambers) may be placed in the core near fuel as it is loaded. The FLC will be connected to the normal SRM electronics for indication, alarm and trip purposes.

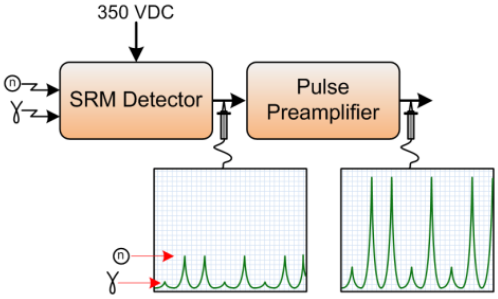
The normal range of detector operation is from source range to a count level which equals approximately 10^{-3} percent reactor power. The operating range of the SRMs can be extended into higher reactor flux ranges by withdrawing the detector partially or fully from the core. The correlation between core flux and SRM count rate is lost when the detector is not fully inserted. However, relative changes in flux will still cause proportional changes in count rate. The period indication is valid as long as the detector is not moving when the reading is being taken. With the detector fully withdrawn, the SRM range is extended to approximately 6% power.

Refer to the drawing below during the following discussions:



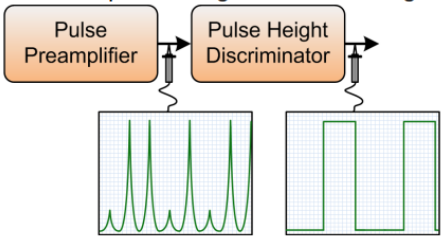
Pulse Preamplifier

The purpose of the pulse preamplifier is to amplify the pulses from the detector prior to routing them to the SRM drawer in the control room, thereby improving the signal-to-noise ratio. The pulse preamplifier is located immediately outside the drywell. This location is as close to the detector as possible, while still allowing it to be accessible during operation. It is necessary to have the pulse preamplifier located near the associated detector since the output pulses from the detector are very small and would not be detectable above cable noise if they had to travel a significant distance before being amplified. The signal output is composed of pulses generated by neutron events as well as pluses generated by gamma events.



Pulse Height Discriminator

The purpose of the pulse height discriminator is to eliminate pulses from gamma radiation or signal noise while allowing pulses caused by neutron events to pass through to the remaining SRM circuitry. As discussed under detector theory, the pulse generated by a neutron event is much larger than that from a gamma radiation or signal noise. The Pulse Height Discriminator is set to pass only those pulses with a magnitude greater than the discrimination level. The magnitude of the gamma pulses or signal noise is smaller than that required by the pulse height discriminator and



Excerpts from 2-GOI-100-1A:

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

[19] **ENSURE** Reactor Engineer records applicable criticality data in 2-SR-3.1.1.1.

_____ _____ _____
Initials Date Time

[20] **ENSURE** reactor period greater than 60 seconds.

(R) _____ _____ _____
Initials Date Time

NOTES

1) Completing paper closure of 2-SR-3.3.1.1.5 is NOT required prior to performing Step 5.4[21]. However, all AC steps are required to be VERIFIED COMPLETED SATISFACTORY prior to withdrawing SRMs.
2) Tech Spec Bases states that overlap between SRMs and IRM's exists when IRM downscale indications have cleared and IRM readings are on-scale and trending higher prior to SRMs reaching 10^5 cps.

[21] **ENSURE** SRM/IRM overlap by obtaining data and completing 2-SR-3.3.1.1.5, SRM and IRM Overlap Verification.

(R) _____ _____ _____
Initials Date Time
Reactor Engineer

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

NOTES

1) SRMs are fully withdrawn when IRM's are on Range 3 or above and indicating above their downscale trip point.
2) Operating experience has shown SRM withdrawal has caused momentary spikes/noise of IRMs resulting in RPS actuation (half and full scrams). It is preferred to withdraw SRMs individually, as plant conditions.
3) If a shutdown margin test has been performed using a different rod sequence, 2-SR-3.1.3.5(A) will provide required actions to insert all control rods, establish normal sequence and perform the subsequent start up with re-entry at Step 5.3[13].

[22] **WITHDRAW** SRMs as necessary, to maintain them on scale between 10^2 cps and 10^5 cps.

_____ _____ _____
Initials Date Time

[23] **MAINTAIN** IRM's on scale between approximately 25 and 75 using IRM range switches.

_____ _____ _____
Initials Date Time

[24] **WHEN** all operable IRM's are on Range 3 or above and all acceptance criteria is met for 2-SR-3.3.1.1.5, SRM and IRM Overlap Verification, **THEN**

WITHDRAW all operable SRMs.

(R) _____ _____ _____
Initials Date Time

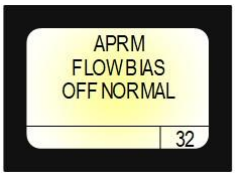
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor	Tier #	2	------
A3.05 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operation of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR including • Flow converter/comparator signals	K/A #	215005A3.05	
	Importance Rating	3.2	------

Proposed Question: **# 36**

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

- APRM FLOW BIAS OFF NORMAL, (2-9-5A, Window 32) alarms
- APRM Flow signal indicates 110%



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

In accordance with the given ARP, a Rod Block **(1)** automatically occur.

The crew can monitor the APRM Flow signal indication on the bar graph from Panel(s) **(2)**.

A. (1) will
(2) 2-9-14 **ONLY**

B. (1) will
(2) 2-9-5 **AND** 2-9-14

C. (1) will NOT
(2) 2-9-14 **ONLY**

D. (1) will NOT
(2) 2-9-5 **AND** 2-9-14

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that Panel 2-9-14 has 4 separate APRM bays with each bay also containing the two-out-of-four logic voter modules, where Panel 2-9-5 only has the 2 APRM displays total with 2 APRMs shared on each display.

B **CORRECT**: (*See attached*) In accordance with the given ARP, a Rod Block will automatically occur when either APRM Channel Recirc Flow ≥ 107% or has failed Upscale. 2-OI-92B, Attachment 1, APRM Trip Outputs, lists the Recirc Flow Upscale trip signal setpoint at 107% Flow monitor resulting in a Rod Block. Second part, in accordance with the given ARP, the Flow signal APRM bar graph indicates either greater than 107% or failed Upscale seen by three upward facing arrows on the APRM Display screen located on Panel 2-9-5 and 2-9-14.

C INCORRECT: First part is incorrect but plausible in that the APRM High High SCRAM setpoint is ≥ 119% APRM Flux. Second part is correct (*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 automatic operation of the Average Power Range Monitor/Local Power Range Monitor including Flow converter/comparator signals in the Main Control Room. 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 related to Power Range Neutron Monitoring logic.

Technical Reference(s): 2-OI-92B, Rev. 44 (Attach if not previously provided)
2-ARP-9-5A, Rev. 65
OPL171.148, Rev. 18U4

Proposed references to be provided to applicants during examination: APRM FLOW BIAS OFF NORMAL, (2-9-5A, Window 32)

Learning Objective: OPL171.148 Obj. 11 (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-ARP-9-5A:

BFN Unit 2	Panel 9-5 2-XA-55-5A	2-ARP-9-5A Rev. 0065 Page 43 of 48

APRM
FLOW BIAS
OFF NORMAL

32

(Page 1 of 1)

Sensor/Trip Point:

APRM Channel 1,2,3,4
FLOW ≥ 107% or failed.

(1) Either APRM Channel Recirc
Flow ≥ 107% or has failed upscale.

Sensor Location: 2/4 Logic Modules (Voters) located in Panel 2-9-14, Main Control Room.

Probable Cause:
A. Malfunction of flow circuit instrumentation.
B. Testing in progress.
C. Malfunction of Sensor.

Automatic Action: Rod block.
FLOW signal on APRM bargraph indicates either greater than 107% or Failed upscale as seen by "↑↑↑" on the Display screen.

Operator Action:
A. **CHECK** Rod Block.
B. **REQUEST** IMs to check for high output or mismatch.
C. **CHECK** flow by either the greater than 107% FLOW or "↑↑↑" indicated on the APRM Display.
D. **IF** flow signal is failed upscale, **THEN** **BYPASS** APRM channel. **REFER TO** 2-OI-92B.
E. **REFER TO** Tech Spec Table 3.3.1.1-1 TRM Table 3.3.4-1, Sect 5.3.1.

References: 2-45E620-6. GEK 103936
Technical Specifications Technical Requirements Manual-TRM

Excerpts from 2-OI-92B:

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).		

BFN Unit 2	Average Power Range Monitoring	2-OI-92B Rev. 0044 Page 26 of 30

Attachment 1
(Page 6 of 6)

APRM Trip Outputs

5.0 PANEL 2-9-14 OPERATOR DISPLAY.

A. The display is divided into 3 sections (upper, middle, and lower).

B. The upper-display section contains general instrument information for each channel of the APRM/OPRM instrument connected to them. This includes self-test status, channel identification, trip status and instrument mode. The center portion of the upper-display section is reserved for trip indication.

C. The mid-display section is dependent upon user actions. Normally, this section displays total flow, APRM flux, and STP(Simulated Thermal Power) for each channel of the APRM instrument connected to them in bar graph and numerical form. All numerical data is given in the form of percent of rated. The flow is presented only in numerical form.

D. The lower-display section consists of one line at the bottom of the display and contains the softkey menu.

Excerpts from OPL171.148 Lesson Plan:

c) Each RBM receives the four Total Recirculation Flow values from the APRM channels to determine the status of the flow compare alarm.

Displays on Input Status display

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)

Inputs to flow upscale alarm APRM flow alarm is when flow is >107 % or upscale, generates rod block.

(2) Flow Compare Alarm (generated by the RBM, display alarm only)

e) The table below shows the relationship between the flow monitors and each APRM channel.

Discuss effects if Flow transmitters fail

	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

Show drawing 2-47E610-68-1

See information on TMOD bfn-2020-068-003 and TMOD 1-2022-068-004 for Unit 1 differences.

f) APRM Channel 4 also provides input into the flow recorder (XR-68-2/5).

g) APRM Channel 3 also provides indications on Panel 9-4 (FI-68-5, FI-68-81).

(1) A loss of power, or if the APRM chassis is unplugged would interrupt the flow signals for panel 9-4 indications.

LOR Objective 3

h) Each APRM calculates a Total Recirculation Flow value by averaging its assigned loop A and loop B recirculation flow values.

(1) Each APRM uses its Total Recirculation Flow value for its flow biased trips and alarms.

(2) In addition, Total Recirc Flow is also used for enabling OPRM trips and alarms.

i) The Total Recirculation Flow signal for each APRM channel is continually processed even when the APRM channel is bypassed.

(1) If APRM channels 3 or 4 are bypassed, the outputs to the meters /recorders are still available.

Provides chassis alarm if greater than 5% difference

OPL171.148, POWER RANGE NUCLEAR MONITORING SYSTEM, REV.# 18U4

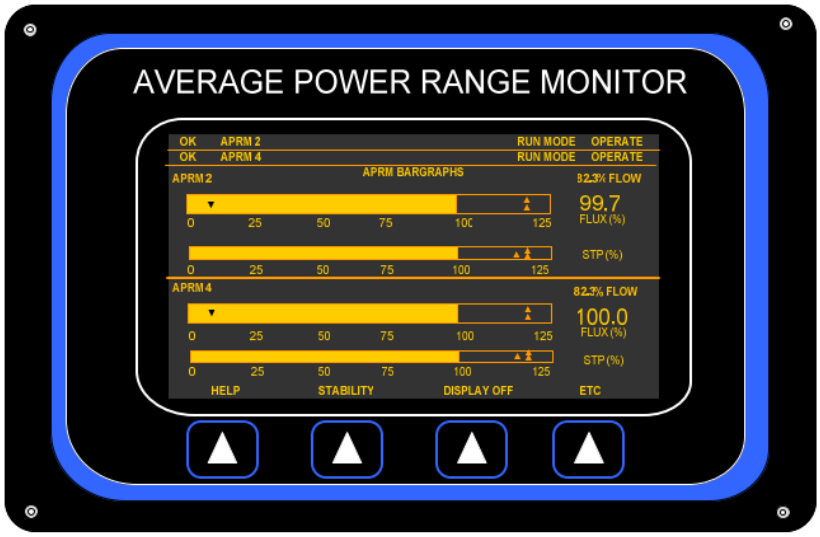


Figure 10 - Operator Display Assembly

4. Operator Display Assembly (ODA) indication

   a. The Operator Display Assembly (ODA) on Main Control Room Panel 9-5 provides APRM, LPRM, OPRM, RBM, and Recirculation flow indications for the operators.

   b. There a total of 4 ODAs, 2 for the APRMs and 1 each per RBM channel.

   c. Each APRM ODA is shared by 2 APRMs.

      1) APRMs 1 and 3 input to one ODA, APRMs 2 and 4 input to the other ODA.

   d. The APRM ODA provides indication only.

   e. The following information is transmitted to the ODA from associated APRM instruments:

      1) Average neutron flux
      2) Simulated thermal power
      3) Total flow signal
      4) LPRM detector signals
      5) Upscale and downscale trips and set points
      6) Instrument status (i.e., Instrument ID and Mode, Self-Test, Bypass)
      7) Oscillation Power Range Monitor displays

Right Unit/Train/Component

OPL171.148, POWER RANGE NEUTRON MONITORING SYSTEM, REV.# 18U4

The NEXT PAGE indications are the miscellaneous indications, including flow adjustments, test options, and test outputs.



The TRIP STATUS screen shows the status of any trips, including trips that were received and cleared. This screen also allows clearing of the memory of the trips. The next two screens show trip status with trips sealed in, then the results when the memory is reset.



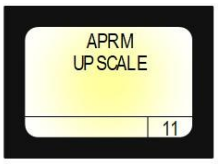
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
215005 (SF7 PRMS) Average Power Range Monitor / Local Power Range Monitor	Tier #	2	------
K3.10 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters:	K/A #	215005K3.10	
• OPRM	Importance Rating	3.7	------

Proposed Question: **# 37**

Unit 1 is operating at 100% RTP when the following occurs:

- APRM UPSCALE (1-9-5A, Window 11) alarms
- Operator manually bypasses APRM 1



Given the conditions above, which **ONE** of the following completes the statements below in accordance with 1-OI-92B, Average Power Range Monitoring?

When APRM 1 is bypassed using 1-HS-92-7B/ S3A, APRM BYPASS, **(1)** is (are) bypassed.

Subsequently, if one more APRM fails UPSCALE, then an automatic SCRAM **(2)** occur.

A. (1) **ONLY** APRM 1
(2) will NOT

B. (1) **ONLY** APRM 1
(2) will

C. (1) both APRM 1 **AND** OPRM 1
(2) will NOT

D. (1) both APRM 1 **AND** OPRM 1
(2) will

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that when bypassing SRMs or IRMs on Panel 9-5, only that specific channel is bypassed. Second part correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that bypassing an APRM does not preclude testing a Voter, such that with an APRM in bypass, the Voters can still be tested and produce half SCRAMs since Voters are not bypassed with the APRM joystick.

C **CORRECT:** (*See attached*) In accordance with 2-OI-92B, APRMs, bypassing an APRM bypasses both the ARPM and OPRM for the channel selected.

Second part, a bypassed APRM channel is unseen by the logic such that the VOTER function remains unchanged. That is, when operating in a 2 out of 4 Voting configuration, the first un-bypassed input will be seen as a single input with trip output. When the second un-bypassed signal of the same type (either APRM function or OPRM function) is received, it will also be seen by all 4 Voters resulting in a trip output from all 4 Voters, consequently producing a full Reactor SCRAM. Given designed operation logic and the given condition, if one more un-bypassed APRM fails UPSCALE, it will be seen as the first APRM trip output to all 4 Voters, therefore an automatic SCRAM will not occur.

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 that a loss or malfunction of the Average Power Range Monitor/Local Power Range Monitor will have on the Oscillating Power Range Monitors. 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-92B, Rev. 18 (Attach if not previously provided)
1-ARP-9-5A, Rev. 35

Proposed references to be provided to applicants during examination: APRM UPSCALE (1-9-5A, Window 11)

Learning Objective: OPL171.148 Obj. 13b, c (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 1-ARP-9-5A:

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0035 Page 16 of 48

APRM
UPSCALE

11

(Page 1 of 1)

Sensor/Trip Point:

Relay K14	(0.61W + 53.0%) (Two loop operation with Rx Mode Sw. in Run) TMOD BFN-1-2020-068-003 TMOD BFN-1-2022-068-004
A6K1 in 2/4 Logic Module (Voters)	8% with Rx Mode Sw. NOT in Run
	113% Neutron Flux Clamped
	(0.55 (W-ΔW) + 47%) (Single loop operation with Mode Switch in RUN) TMOD BFN-1-2020-068-003 TMOD BFN-1-2022-068-004

Sensor Location: Control Room Panel 1-9-14.

Probable Cause:
A. Flux level at or above setpoint.
B. Testing in progress.
C. Failed sensor.

Automatic Action: Rod withdrawal block of all control rods.

Operator Action:
A. **CHECK** alarm by multiple indications.
B. **CHECK** for rod block.
C. **STOP** any power rise or rod withdrawal in progress.
D. **NOTIFY** Reactor Engineer.
E. **REFER TO** TRM Table 3.3.4-1.

References: 1-45E620-6-1 1-197R114-16 1-107E5784-3, 20

Excerpts from 1-OI-92B:

2.2.4 Miscellaneous Documents

SIL-111 Revision 1, Neutron Monitor and Flow Bypass Switch Modification

3.0 PRECAUTIONS AND LIMITATIONS

A. Each operable APRM channel requires a minimum of 20 LPRM inputs and at least 3 LPRM inputs per level (REFER to Tech Spec Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation, and associated basis).

B. Only one APRM/OPRM can be bypassed at a time. The APRM BYPASS selector switch, 1-HS-92-7B/S3 on Panel 1-9-5, bypasses both the APRM AND OPRM for the channel selected.

C. In order to prevent an inadvertent rod withdrawal block or Reactor scram while operating the APRM BYPASS selector switch;1-HS-92-7B/S3:

Ensure that the previously bypassed channel returns to normal status by observing the blue BYPASSED light extinguished on all four of the Voters at Panel 1-9-14 for the applicable APRM channel.

After bypassing a channel, the applicable blue BYPASSED lights on all four of the Voters are required to be illuminated prior to testing, operating, or working on that channel.

D. Attachment 1, APRM/OPRM Trip Outputs and PRNMS Overview lists trip signals and associated actions for the Average Power Range Monitoring System/Oscillation Power Range Monitoring.

E. If the blue shorting links (two total links) are removed, the Voters and IRMs are placed in non-coincident trip logic where any one channel, if tripped, will produce a full Reactor scram.

F. If any one APRM/OPRM channel rod withdrawal block signal is tripped, then withdrawal of all control rods is blocked.

If a Rod Block Monitor is tripped, it applies only to the control rod selected but does not imply that another control rod would be blocked.

G. Any time an LPRM is bypassed, or removed from bypass, the Reactor Engineer should be notified. Bypassing of LPRMs is required to be accomplished by the Unit Operator or by Instrument Maintenance with Unit Operator's cognizance.

H. [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-RO1]

3.0 PRECAUTIONS AND LIMITATIONS (continued)

I. Each of the four APRM/OPRM channels input to the four Voters, such that when a signal is generated from an APRM/OPRM channel, all four Voters see and reflect that signal. Each Voter is directly associated with one RPS sub-channel.

When operating in a 2 out of 4 voting configuration, the first un-bypassed input will be seen as a single input with no trip outputs. When the second un-bypassed signal of the same type [The SAME TYPE inferring that one type is an APRM function and a different type is an OPRM function] is received it will also be seen by all four Voters resulting in a trip output from all four Voters consequently producing a full reactor scram.

J. Bypassing an APRM does not preclude testing a Voter, such that with an APRM in bypass, the Voters can still be tested and produce half scrams. Voters are not bypassed with the APRM joystick.

K. The Recirc Flow Indication and the Voters are never bypassed unless they are removed for testing. There is no bypass capability for the Recirc flow signal input or Voters.

L. A reactor scram will be produced when at least two of the SAME TYPE of trip inputs are received by the Voters:

Either: APRM HIGH/INOP {i.e., APRM High Flux/STP Flow Biased Scram/INOP}

OR:

Any one of the three algorithms, period, growth, or amplitude for an operable OPRM cell has exceeded its trip value.

OR:

Confirmation Density Algorithm trip value exceeded.

Attachment 1
(Page 5 of 6)

APRM/OPRM Trip Outputs and PRNMS Overview

3.0 2/4 LOGIC PROCESSING

A. Each 2/4 Logic Module (Voter) receives trip status from each APRM/OPRM instrument and APRM/OPRM channel bypass information for each channel. Based on this information, each 2/4 Logic Module determines whether to send trip signals to the RPS or not. Trip signals are sent if any two of the same type of non-bypassed APRM/OPRMs are providing trip signals to the 2/4 Logic Module. A separate, non-safety section of the 2/4 Logic Module serves as an interface between the associated APRM channel and other plant equipment outside Panel 1-9-14, such as the annunciator system, indicators on Panel 1-9-5, and the Reactor Manual Control System.

B. The 2/4 logic does not latch the input trip conditions. This means that no reset is required and no output trip signal occurs if one APRM/OPRM instrument generates a trip input to the logic and then clears before another APRM/OPRM instrument generates a trip. A trip output occurs only if two or more of the same type of inputs indicate a trip condition.

C. The 2/4 logic produces a trip state at two redundant trip outputs(X and Y) if any two of the same type of non-bypassed APRM/OPRM channels indicate a trip state. If one of the APRM/OPRM channels is bypassed, the logic automatically reverts to 2/3 logic. The 2/4 logic monitors for an active input from the channels. A non-bypassed input channel is processed as if in a trip state, if the input is not active.

D. The 2/4 logic modules do not latch the trip signals, trip memory indicators on both the input and output trips provide indication of a past or present trip and clear by manual reset.

4.0 RECIRCULATION FLOW PROCESSING

Each APRM powers and processes the Recirculation flow signals for one transmitter monitoring loop A and for one transmitter monitoring loop B. The PRNMS monitors the Recirculation System Flow in order to calculate a value for Total Recirculation Flow Rate. This value is then used by the APRM System to determine the flow-biased upscale trips and alarms for STP. Flow data is also provided to remote recorders and meters and to the Process Computer.

Most of the Recirculation System Flow signal processing is performed by the APRM instruments; however, the RBM instruments compare the Total Flow values generated by and received from each of the APRM instruments. The RBM generates a Flow Compare alarm if any two Total Recirculation Flow values differ by an amount greater than or equal to the setpoint.

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

Examination Outline Cross-reference:

700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6

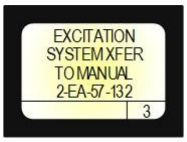
G2.4.46 (10CFR 55.41.10)

Ability to verify that the alarms are consistent with the plant conditions

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	700000G2.4.46	
Importance Rating	4.2	------

Proposed Question: **# 38**

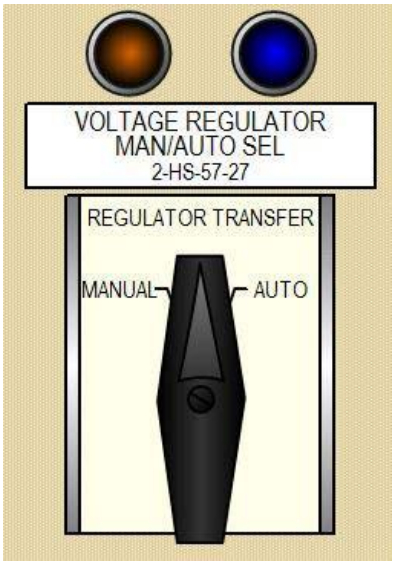
Unit 2 is operating at 100% RTP when the following occurs:



- EXCITATION SYSTEM TRANSFER TO MANUAL (2-9-8A, Window 3) alarms
- 2-HS-57-27, VOLTAGE REGULATOR MANUAL/AUTO SELECT, is positioned as indicated

Given the conditions above, which **ONE** of the following completes the statements below in accordance with the given ARP?

CURRENT expected light indications for 2-HS-57-27 are amber light **(1)** and blue light **(2)**.



A. (1) illuminated
(2) extinguished

B. (1) illuminated
(2) illuminated

C. (1) extinguished
(2) extinguished

D. (1) extinguished
(2) illuminated

Proposed Answer: **A**

Explanation (Optional):

A

CORRECT: *(See attached)* In accordance with the given ARP, the generator voltage regulator auto-trips to manual in response to certain events (e.g., failure of automatic mode in both Voltage Regulator channels, Vols/Hertz level 1). The indicating light scheme for 2-HS-57-27 provides information as to the status of voltage regulator operation. Specifically, operation in Manual is indicated by Amber light illuminated, Blue light extinguished. Operation in Automatic is indicated by Amber light extinguished, Blue light illuminated. For the condition given, candidate must recognize the trip to Manual as well as recall expected indications. It is vitally important for the crew to know the status of the Generator Voltage

Regulator, normally in automatic, but especially the indications when it's in manual for the given conditions.

B INCORRECT: First part is correct *(See A).* Second part is incorrect but plausible in that normally the Generator Voltage Regulator is in automatic with the blue light illuminated.

C INCORRECT: First part is incorrect but plausible in that normally the Generator Voltage Regulator is in automatic with the amber light extinguished. 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 verify the alarms are consistent with plant conditions as it relates to the Generator Voltage Regulator position indication during 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): 2-ARP-9-8A, Rev. 51 (Attach if not previously provided)
2-OI-47, Rev. 209
0-AOI-57-1E, Rev. 17

Proposed references to be provided to applicants during examination: EXCITATION SYSTEM TRANSFER TO MANUAL (2-9-8A, Window 3)
2-HS-57-27, VOLTAGE REGULATOR MANUAL/AUTO SELECT, (no illuminated lights)

Learning Objective: OPL171.134, Obj. 3 (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 **X**
Comprehension or Analysis

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

Comments:

Excerpts from 2-ARP-9-8A:

BFN Unit 2	Panel 9-8 2-XA-55-8A	2-ARP-9-8A Rev. 0051 Page 7 of 47

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

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	

Excerpt from 2-OI-47:

Attachment 2
(Page 1 of 6)

Generator Voltage Regulator

1.0 SYSTEM DESCRIPTION

The Generator Voltage Regulator is an ABB Unitrol 5000 dual channel Automatic Voltage Regulator (AVR). The two channels are independent with one in active status and the other inactive (standby). Each channel consists of an automatic regulator (AUTO mode) and a manual regulator (MANUAL mode). Normal status for the voltage regulator includes having one channel active controlling in the AUTO mode with the other channel inactive and following the active channel. Should a transfer occur from the active channel to the inactive channel, the transfer will be "bump-less".

In addition to the two channels described above, the voltage regulator has two emergency backup regulators. They maintain field current like the main channels in the MANUAL mode. The emergency regulators act as backups to the main channels should they fail in the AUTO and MANUAL mode. Voltage Regulator control transfer to a emergency regulator is also a bump-less transfer.

AUTO MODE - In AUTO mode, the voltage regulator directly controls the generator voltage. The Voltage Regulator is in automatic mode when VOLTAGE REGULATOR MAN/AUTO SEL, 2-HS-57-27, on Panel 2-9-8, is selected to the auto position and returned to mid-position. The blue light above the handswitch will be illuminated when in automatic control. The Voltage Regulator has two automatic channels, primary and alternate. Each channel has its own power source. The primary automatic channel is always aligned to 480V Unit Board 2A while the alternate automatic channel is always aligned to 480V Common Board 2.

MANUAL MODE - In MANUAL mode, the field current is controlled manually. The Voltage Regulator is in the MANUAL mode when the VOLTAGE REGULATOR MAN/AUTO SEL, 2-HS-57-27, on Panel 2-9-8, is selected to the MAN position and returned to mid-position. The amber light above the handswitch will be illuminated when in the manual control. Transfer between Automatic and Manual modes is bump-less. The voltage regulator will automatically trip to manual mode when certain degraded conditions occur.

Excerpt from 0-AOI-57-1E:

Attachment 1
(Page 2 of 5)

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 -25 MVARs in automatic voltage regulator operation. (minimum reactive power is limited to -175 MVARs for any unit operating less than 1100MWe) Manual voltage regulator operation beyond this limit is allowed only as directed by the Load Coordinator during transmission system disruption contingencies.

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

Examination Outline Cross-reference:

218000 (SF3 ADS) Automatic Depressurization

G2.2.36 (10CFR 55.41.10)

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	218000G2.2.36	
Importance Rating	3.1	------

Proposed Question: **# 39**

A Unit 2 startup is in progress with the following conditions:

- Reactor Pressure is 135 psig
- 250 VDC RMOV Board 2A is lost
- ADS BLOWDOWN POWER FAILURE (2-9-3C, Window 32) alarms



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

In accordance with Tech Spec 3.5.1, ECCS, RPV WATER INVENTORY CONTROL, AND RCIC SYSTEM, Automatic Depressurization System (ADS) currently **___(1)___** required to be OPERABLE.

In accordance with the given ARP, **ONLY ___(2)___** ADS Valves have lost their **NORMAL** power supply.

A. (1) is
(2) 2

B. (1) is
(2) 3

C. (1) is NOT
(2) 2

D. (1) is NOT
(2) 3

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the candidate could confuse the 'above the line' LCO 3.5.1 exception for MODES 2 and 3 for ADS with LCO 3.4.3, which states the safety function of 12 SRVs shall be OPERABLE in MODES 1, 2, 3, no exceptions. 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 power supplies (both normal, alternate, alternate battery board) for 13 SRVs are easily and often confused by candidates.

C **CORRECT**: (*See attached*) In accordance with Unit 2 Tech Spec 3.5.1, ECCS – Operating, LCO 3.5.1 requires ADS function of 6 SRVs to be OPERABLE with the exception of when in MODES 2 and 3, ADS valves are not required to be OPERABLE with Reactor Steam Dome Pressure ≤ 150 psig. Second part, in accordance with the given ARP, only 2 ADS Valves have lost their normal power supply. The ARP automatic action states that 2-PCV-1-22 and 1-30, Main Steam Auto Relief Valves, auto transfer power supply from 250VDC RMOV BD 'A' to 250VDC RMOV BD 'B' (2-PCV-1-22) and 250VDC RMOV BD 'C' (2-PCV-1-30), respectively, on loss of normal power supply.

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 analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation of Automatic Depressurization Valves. 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.

Technical Reference(s): 2-ARP-9-3C, Rev. 29 (Attach if not previously provided)
U2 Tech Spec 1.0, Amend. 253
U2 Tech Spec 3.4.3, Amend. 253
U2 Tech Spec 3.5.1, Amend. 351
2-AOI-1-1, Rev. 30

Proposed references to be provided to applicants during examination: ADS BLOWDOWN POWER FAILURE (2-9-3C, Window 32)

Learning Objective: OPL171.009, Obj. 14e (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
Comprehension or Analysis **X**

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

Excerpt from U2 Tech Spec 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.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

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.

Excerpt from U2 Tech Spec 1.0:

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel(a) or Startup/Hot Standby	NA
3	Hot Shutdown(a)	Shutdown	> 212
4	Cold Shutdown(a)	Shutdown	≤ 212
5	Refueling(b)	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Excerpt from 2-AOI-1-1: Also supports Distractors B2, D2

Attachment 1
(Page 1 of 3)

UNIT 2 SRV Solenoid Power Breaker/Fuse Table, Panel 25-32

NOTE

UNIT 2 ADS valves that have more than one power supply will AUTO TRANSFER on a loss of power, and are NORMAL SEEKING.

UNIT 2 SRV Solenoid Power Breaker Table

SRV	NORMAL PWR 250V RMOV	BRKR	ALTERNATE 250V RMOV	BRKR	ALTERNATE BATT. BRD.	BRKR
1-4	2A	11B2				
1-5	* 2C	8A			1	727
1-18	2B	8C1				
1-19	* 2B	1B2	**			
1-22	* 2A	11C2	2B	1C1		
1-23	2C	1B1				
1-30	* 2A	9B1	2C	7A	1	727
1-31	* 2B	8B2	**			
1-34	* 2C	10A			1	727
1-41	2A	11C1				
1-42	2B	1C2				
1-179	2B	8C2	**			
1-180	2C	10B1				

*ADS Valves.

**All components have been removed from 2-PNL-25-32.

Excerpt from U2 Tech Spec 3.4.3: Supports Distractors A1, B1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

LCO 3.4.3 The safety function of 12 S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

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

Examination Outline Cross-reference:	Level	RO	SRO
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff	Tier #	2	------
K6.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF:	K/A #	223002K6.04	
• Nuclear boiler instrumentation	Importance Rating	3.7	------

Proposed Question: **# 40**

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

- Both 2-LI-3-203A and 2-LI-3-203D, REACTOR WATER LEVEL NORMAL RANGE, instruments fail **DOWNSCALE**
- **NO** Operator actions have been taken

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

An automatic Full SCRAM **(1)** occur and PCIS Groups **(2)** reach their isolation setpoints.

A. (1) will
(2) 2, 3, 6, 8

B. (1) will
(2) 2, 3, 4, 6, 8

C. (1) will NOT
(2) 2, 3, 6, 8

D. (1) will NOT
(2) 2, 3, 4, 6, 8

Proposed Answer**: A**

Explanation (Optional):

A **CORRECT:** (*See attached*) Given that 2-LI-3-203A (RPS A1) and 2-LI-3-203D (RPS B2) REACTOR WATER LEVEL NORMAL RANGE (0 - (+) 60 inches) both fail downscale, this will result in an automatic SCRAM since a channel in both RPS A and B fail. In accordance with the ARP (not given), REACTOR VESSEL WATER LEVEL LOW HALF SCRAM, (2-9-4A, Window 2), alarms from any 2-LIS-2-203A, B, C or D (reaches ≤ (+) 2 inches in which a manual SCRAM is required. This ARP references 2-OI-99, RPS, where Attachment 3 shows 2-LIS-2-203A-D RPS channel assignments, alarms, SCRAM condition, PCI actuation (if any).
Second part, in accordance with 2-OI-64, Primary Containment, Attachment 2, 2-LIS-2-203A-D are listed as they are in 2-OI-99 along with the specific PCIS Group Isolations. In this case PCIS Groups 2, 3, 6, 8 reach their isolation setpoints when the given instruments both fail downscale passing through Reactor Water Level ≤ (+) 2 inches.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that PCIS Group 4 is HPCI which initiates when Reactor Water Level reaches (-) 45 inches from LT-3-58A-D, EMERGENCY RANGE INSTRUMENTS, (-) 155 to (+) 60 inches.

C INCORRECT: First part is incorrect but plausible if 2-LI-3-203A (RPS A1) and 2-LI-3-203C (RPS C2) failed downscale, only a half SCRAM would occur in RPS A channel. 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 effect of Nuclear Boiler Instrumentation malfunctions, or component malfunctions, on Primary Containment Isolation System as it relates to Reactor Water Level Instrumentation. 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 complexity of Reactor Water Level Instrumentation and RPS.

Technical Reference(s): 2-OI-64, Rev. 132 (Attach if not previously provided)
2-OI-73, Rev. 107
2-OI-99, Rev. 101
2-ARP-9-4A, Rev. 52
PIP-95-71, Rev. 6
OPL171.003, Rev. 27U1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.003 Obj. 19 (As available)

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

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:

Excerpt from 2-ARP-9-4A:

BFN Unit 2	Panel 9-4 2-XA-55-4A	2-ARP-9-4A Rev. 0052 Page 5 of 47

RX VESSEL
WTR LEVEL LOW
HALF SCRAM

2

(Page 1 of 1)

Sensor/Trip Point:

2-LIS-3-203A,B,C,D — 2.0" above instrument zero (539" above vessel zero).

Sensor Location:
2-LIS-003-0203A, 2-PNLA-009-0083 U2 Aux Inst Room
2-LIS-003-0203B, 2-PNLA-009-0084 U2 Aux Inst Room
2-LIS-003-0203C, 2-PNLA-009-0085 U2 Aux Inst Room
2-LIS-003-0203D, 2-PNLA-009-0086 U2 Aux Inst Room

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

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 ≤ +2.0" AND reactor has **NOT** scrammed, **THEN** **PERFORM** the following:
   1. **MANUALLY SCRAM** the reactor.
   2. **ENTER** 2-EOI-1 FLOWCHART.

C. **DISPATCH** personnel to 2-LIS-3-203 A, B, C and D to check for abnormal condition.
D. **IF** alarm is **NOT** valid, **THEN** **PERFORM** the following:
   1. With UNIT SRO permission, **RESET** Half Scram.
   2. **REFER TO** 2-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:
GE 730E915-9 10 2-45N620-5
FSAR Sections 7.2.3.5 and 13.6.2 Tech Specs: 3.3.1.1, 3.3.6.1, 3.3.6.2, 3.3.7.1.

Excerpt from 2-OI-99:

Attachment 3
(Page 6 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
2-LIS-3-203A RX WATER LEVEL LOW (Level 3) A1 CHANNEL Function: 4	2-FU1-3-203AA (5AF6A)	2-RLY-099-05AK06A 2-RLY-099-5A-K25A 2-RLY-064-16AK5A 2-RLY-064-16AK6A	9-15	2-730E915-9 2-730E927-7 2-45E671-26	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203B RX WATER LEVEL LOW (Level 3) B1 CHANNEL Function: 4	2-FU1-3-203BA (5AF6B)	2-RLY-099-05AK06B 2-RLY-099-5A-K25B 2-RLY-064-16AK5B 2-RLY-064-16AK6B	9-17	2-730E915-10 2-730E927-8 2-45E671-38	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203C RX WATER LEVEL LOW (Level 3) A2 CHANNEL Function: 4	2-FU1-3-203CA (5AF6C)	2-RLY-099-05AK06C 2-RLY-099-5A-K25C 2-RLY-064-16AK5C 2-RLY-064-16AK6C	9-15	2-730E915-9 2-730E927-7 2-45E671-32	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203D RX WATER LEVEL LOW (Level 3) B2 CHANNEL Function: 4	2-FU1-3-203DA (5AF6D)	2-RLY-099-05AK06D 2-RLY-099-5A-K25D 2-RLY-064-16AK5D 2-RLY-064-16AK6D	9-17	2-730E915-10 2-730E927-8 2-45E671-44	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
NOTE: Device Function corresponds to the TS Table 3.3.1.1 Functions.						

Excerpt from 2-OI-64:

Attachment 2
(Page 1 of 10)

Actions to Place PCIS in Tripped Condition

NOTE

Water level designators (1-8)are listed for relationship to the applicable device only.

(T.S. Tables 3.3.6.1-1,3.3.6.2-1, & 3.3.7.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARM	REMARKS
2-LIS-3-203A RX WATER LEVEL LOW (Level 3)	2-FU1-3-203AA (5A-F6A)	5AK6A 5AK25A 16AK5A 16A6A	9-15	2-730E915-9 2-730E927-7 2-45E671-26	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203B RX WATER LEVEL LOW (Level 3)	2-FU1-3-203BA (5A-F6B)	5AK6B 5AK25B 16AK5B 16AK6B	9-17	2-730E915-10 2-730E927-8 2-45E671-38	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203C RX WATER LEVEL LOW (Level 3)	2-FU1-3-203CA (5A F6C)	5AK6C 5AK25C 16AK5C 16AK6C	9-15	2-730E915-9 2-730E927-7 2-45E671-32	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203D RX WATER LEVEL LOW (Level 3)	2-FU1-3-203DA (5A-F6D)	5AK6D 5AK25D 16AK5D 16AK6D	9-17	2-730E915-10 2-730E927-8 2-45E671-44	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
Table 3.3.6.1-1: Function 2a, 5h, and 6b Table 3.3.6.2-1: Function 1 Table 3.3.7.1-1: Function 1						

Excerpts from OPL171.003 Lesson Plan:

(b) low level alarm (+27" Level 4) are actuated as sensed by the point on the recorder. Defines lower end of normal operating region. If level is > +27 inches, transients such as a reactor feedpump trip from full power will not cause level to decrease to reactor scram point with a runback of the recirculation pumps.

2) Provides recirculation pump runback signal at +27" (low level alarm) with a loss of one RFP.

3) Density compensated

b) LT-3-208 A-D (0-60")

1) Provides main turbine and RFP turbine trips (+55" Level 8). Logic is the correct 2 out of 2 taken once.

2) Provides high level trip (+51" Level 8) signal for the RCIC and HPCI turbines. Logic is 2 out of 2
(a) LIS-3-208 A,C supply signal to RCIC trip logic. LI-3-208C located above RCIC on 9-3
(b) LIS-3-208 B,D supply signal to HPCI trip logic. LI-3-208B located above HPCI on 9-3

3) Provides level indication on Panel 9-5 and 9-3. LI-3-208A and D located on 9-5

b. LT-3-184 and 3-185 (0-60")
1) Provides low reactor vessel water level signal (+2" Level 3) to ADS initiation logic.
2) Provides level indication on the ATU cabinets (9-81 and 9-82).

c. LT-3-203 (A-D) (0-60")
1) Provides low reactor water level signal (+2" Level 3) for
(a) reactor scram,
(b) PCIS Groups 2, 3, 6, 8 isolation,
(c) SGT initiation,
(d) Reactor Building Ventilation System isolation, and
(e) CREV units' start.
(f) Provides level indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86).

2) Reactor scram at +2 inches, Level 3
   (a) Assures the reactor will not be operated without sufficient water above the reactor core.
   b) In conjunction with the containment isolation signal of +2 inches, prevents fuel cladding perforation
   c) Prevents operation with separator skirts uncovered, which could result in carry under.
   d) Set low enough to prevent spurious operation for normal operating transients.

3) Starting SGT establishes secondary containment.

2) Emergency Systems Range (Wide Range) instruments

(-155" to +60") (Referenced to instrument zero). Per Regulatory Guide 1.97 Emergency Range identified by Black Labels.

a) LT-3-56 (A-D)

1) Provides low reactor water level signal (-122") for PCIS isolation Level 1

2) Provide level indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86)

b) LT-3-58 (A-D)

1) PAM instrumentation

2) ATWS- RPT trip at -45 inches Level 2

(a) Reduces core flow and bleeds off CRD HCU air header in event that low level was due to a failure to scram.

(b) Also prevents operation of recirc pumps without adequate net positive suction head.

2) HPCI and RCIC system initiation at -45 inches Level 2

(a) HPCI initiates at -45 inches; is set to allow adequate core cooling for small line breaks.

(b) RCIC initiation at -45 inches; is set to provide adequate cooling under loss of feedwater flow and/or main steam isolation.

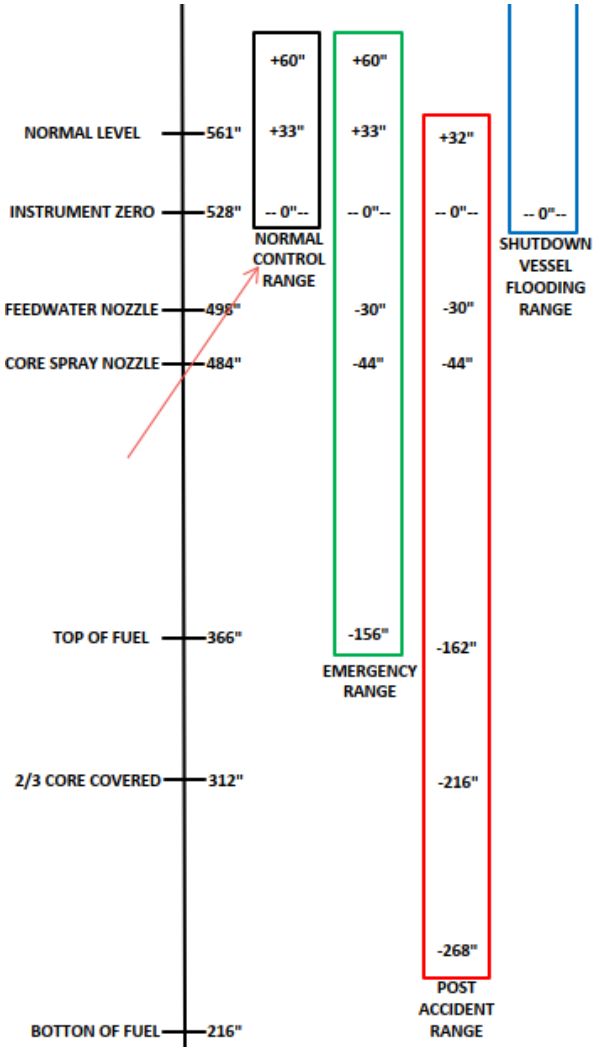
Excerpts from PIP-95-71:

CONDENSING POT A (3-821)
A NORMAL CONTROL RANGE INSTRUMENTS (0 TO +60)

LOCAL PANEL	INSTRUMENT NUMBER	MASTER TRIP	SLAVE TRIP	PANEL NUMBER	FUNCTION	POWER SUPPLY
25-5C	LT-3-203A			9-83	RX SCRAM, PCIS 2,3,6,8	PANEL 9-15 RPS A1
25-5C	LT-3-208A			9-81	MT,RFPT, RCIC TRIP	DIV I ECCS INV
PNL- 0426	LT-3-206			9-18	FW LEVEL CONTROL	UNIT PREF./ICS
PNL- 0426	PT-3-207				FW LEVEL CONTROL	UNIT PREF./ICS
25-5C	PT-3-204A			9-81	ATWS/ARI-RPT INIT	DIV I ECCS INV

CONDENSING POT D (3-822)
B NORMAL CONTROL RANGE INSTRUMENTS (0 TO +60)

LOCAL PANEL	INSTRUMENT NUMBER	MASTER TRIP	SLAVE TRIP	PANEL NUMBER	FUNCTION	POWER SUPPLY
25-6C	LT-3-203D	LIS-3-203D		9-86	RX SCRAM, PCIS 2,3,6,8	PANEL 9-17 RPS B2
25-6C	LT-3-208D	LIS-3-208D		9-82	MT, RFPT, HPCI TRIP	DIV II ECCS INV
25-6C	PT-3-204D	PIS-3-204D		9-82	ATWS/ARI-RPT INIT	DIV II ECCS INV
25-6D	PT-3-253			9-18	FW LEVEL CONTROL	UNIT PREF./ICS



PIP - 95 - 71
REACTOR LEVEL AND PRESSURE INSTRUMENTATION
BROWNS FERRY NUCLEAR PLANT
11/23/2022 REV. 6

Excerpt from 2-OI-73: Supports Distractors B2, D2

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.

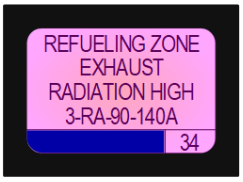
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
295023 (APE 23) Refueling Accidents / 8	Tier #	1	------
AK3.03 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses or actions as they apply to REFUELING ACCIDENTS:	K/A #	295023AK3.03	
• Ventilation isolation	Importance Rating	3.7	------

Proposed Question: **# 41**

Unit 3 is in a Refueling Outage when the following conditions occur:

- An irradiated fuel bundle is dropped onto the Reactor Core
- REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 34) alarms



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

An automatic isolation of the **(1)** occurs.

Standby Gas Treatment (SGT) System will automatically initiate with this automatic isolation to ensure the Secondary Containment **(2)**.

A. (1) Refuel Zone Ventilation **ONLY**
(2) air is filtered and discharged to the environment at an elevated release point

B. (1) Refuel Zone Ventilation **ONLY**
(2) radiation levels remain below 3-EOI-3, Secondary Containment Control, entry conditions

C. (1) Refuel Zone **AND** Unit 3 Reactor Building Ventilation
(2) air is filtered and discharged to the environment at an elevated release point

D. (1) Refuel Zone **AND** Unit 3 Reactor Building Ventilation
(2) radiation levels remain below 3-EOI-3, Secondary Containment Control, entry conditions

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: *(See attached)* In accordance with the given ARP, Control Room and Refuel Zone Ventilation automatically isolates on a PCIS Group 3 Isolation signal. The ARP directs 3-AOI-64-2D, which lists all of the automatic actions associated for when Refuel Zone Exhaust Radiation reaches 72 mr/hr. Second part, in accordance the Safety Design Basis in UFSAR 5.3, Secondary Containment, when isolated, the Secondary Containment atmosphere is filtered by SGT System and released from the plant stack.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that if the candidate fails to recognize that in this case, the given alarm's setpoint is also the same radiation setpoint required for all three SGT trains to initiate, which also meets the entry conditions for EOI-3.

C INCORRECT: First part is incorrect but plausible in that if REACTOR ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 21) alarms, then Refuel and Reactor Zone Ventilation both isolate. 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 reasons for ventilation isolation as it applies to Refueling Accidents. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to Secondary Containment isolations.

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-3, Rev. 18 (Attach if not previously provided)
3-AOI-64-2D, Rev. 21
0-OI-65, Rev. 57
3-ARP-9-3A, Rev. 60
3-AOI-64-2D, Rev. 21
UFSAR 5.3, Rev. 27

Proposed references to be provided to applicants during examination: REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 34)

Learning Objective: OPL171.067, Obj. 2c, e (As available)

Question Source:

Bank #	BFN 1404 NRC #10	
Modified Bank #		(Note changes or attach parent)
New		

Question History:

Last NRC Exam	2014

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:

QUESTION 10

Given the following conditions:

- Refueling is in progress on Unit 3
- An irradiated fuel bundle is dropped onto the top of the Unit 3 reactor core
- REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 34) is in alarm

Which ONE of the following completes both statements below?

This condition will cause an automatic isolation of the __(1)__.

The reason why Standby Gas Trains automatically start during this event is to ensure that __(2)__.

A. (1) Refuel Zone ventilation ONLY
(2) air is discharged at an elevated release point

B. (1) Refuel Zone ventilation ONLY
(2) the Secondary Containment radiation levels remain below 3-EOI-3,Secondary Containment Control, entry conditions

C. (1) Refuel Zone and Unit 3 Reactor zone ventilation
(2) air is discharged at an elevated release point

D. (1) Refuel Zone and Unit 3 Reactor zone ventilation
(2) the Secondary Containment radiation levels remain below 3-EOI-3,Secondary Containment Control, entry conditions

Answer is: **A**

Excerpts from 3-ARP-9-3A:

REFUELING ZONE EXHAUST RADIATION HIGH
3-RA-90-140A
34

(Page 1 of 2)

Sensor/Trip Point:

3-RE-90-140A	72 mR/HR	
3-RE-90-140B	72 mR/HR	
3-RE-90-141A	72 mR/HR	Required setting of ≤ 100 mR/HR.
3-RE-90-141B	72 mR/HR	

Sensor Location: Rx Bldg, El 664' (Refuel Floor), R-17 P-LINE

Probable Cause:
A. Radiation levels have risen above alarm setpoint.
B. Refueling accident.
C. Dry Cask loading/unloading activities in progress.
D. Loss of power to NUMAC drawer.

Automatic Action:
A. Control Room and Refuel Zone ventilation isolates.
B. SGTS initiates.
C. Control Room Emergency Pressurization units start.

Operator Action:
A. **CHECK** alarm condition on the following:
1. REACTOR & REFUEL ZONE EXHAUST RADIATION recorder, 3-RR-90-144 points 3 and 4 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.

C. **NOTIFY** Shift Manager, Unit 1 and Unit 2.

Continued on Next Page

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0060 Page 57 of 59

REFUELING ZONE EXHAUST RADIATION HIGH 3-RA-90-140A, Window 34
(Page 2 of 2)

Operator Action: (Continued)

D. **IF** the TSC is **NOT** manned, **THEN**
EVACUATE personnel from the refuel floor.

E. **IF** the TSC is manned, **THEN**
REQUEST the TSC to evacuate non-essential personnel from affected areas.

F. **ENTER** 3-EOI-3 Flowchart.

G. **REFER TO** 3-AOI-64-2d and, for loss of power to NUMAC drawer, to 3-OI-90.

H. **REFER TO** 3-AOI-79-1 or 3-AOI-79-2 as applicable.

I. **REFER TO** EPIP-1.

J. **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

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

Excerpt from 0-OI-65:

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0057 Page 10 of 44

3.0 PRECAUTIONS AND LIMITATIONS (continued)

T. [NRC/C] If any relays are ACTUATED, Site Engineering SHALL be contacted prior to energizing the circuit. The pull-to-lock logic will **NOT** inhibit the SGT Blower from starting when the SGT Blower breaker is racked in and the MCX relay is actuated (blue contact position indicator retracted). [NRC LER 88-017]

U. Start relays, MAX and MBX for Standby Gas Treatment trains "A" and "B" respectively, are of a different type than the MCX for train "C". However, the same problem exist for these relays as does for the MCX relay. If the contacts are closed (pulled up) prior to the breaker being closed, the standby gas treatment train will start when the breaker is closed. FAILURE to have the contacts **open** (dropped down position) will result in the associated Standby Gas Treatment train starting when the breaker is closed.

V. The following signals on any unit will start all three SGT trains when the respective control switches are in AUTO:

1. High drywell pressure (2.45 psig).
2. Low Reactor Water Level (LEVEL 3).
3. High Rx Zone Ventilation Radiation (72 MR/hr).
4. High Refuel Zone Ventilation Radiation (72 MR/hr).
5. One out of two taken twice trip logic for Reactor Zone Ventilation Radiation downscale.
6. One out of two taken twice trip logic for Refuel Zone Ventilation Radiation downscale.

W. When the control room handswitch for an SGT Fan is in PULL-TO-LOCK, the fan may still be operated locally.

X. The following system valves fail open upon a loss of power (all other system valves fail closed):

1. SGT FILTER BANK C OUTLET DAMPER, 0-DMP-065-0067
2. SGT FAN A INLET DAMPER, 0-DMP-065-0017
3. SGT FAN B INLET DAMPER, 0-DMP-065-0039

Y. The SGT FILTER BANK A & B BYPASS DAMPER, 0-DMP-065-0022, is normally fed power from 480V Diesel Aux Bd A. Power to 0-DMP-065-0022 is automatically transferred to 480V Diesel Aux Bd B upon a loss of power from Aux Bd A.

Excerpts from 3-AOI-64-2D:

BFN Unit 3	Group 6 Ventilation System Isolation	3-AOI-64-2D Rev. 0021 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 (LEVEL 3)
   - 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 2.

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

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

3.1 Refueling Zone Isolation (continued)

10. 3-FCV-064-0033, SUPPR CHBR VENT OUTBD ISOL VLV
11. 3-FCV-064-0034, SUPPR CHBR INBD ISOLATION VLV
12. 3-FCV-084-0020, PRI CTMT VENT TO SGT ISOL VALVE
13. 3-FSV-043-0050, RHR SAMPLE INLET INBD ISOL VALVE
14. 3-FSV-043-0056, RHR SAMPLE INLET OUTBD ISOL VALVE
15. 3-FSV-043-0040, PASS SAMPLE RETURN INBD ISOL VALVE
16. 3-FSV-043-0042, PASS SAMPLE RETURN OUTBD ISOL VALVE
17. 3-FCV-064-0019, SUPPR CHAMBER ATM SUPPLY INBD ISOL VLV
18. 3-FCV-064-0029, DRYWELL EXHAUST INBD ISOL VLV
19. 3-FCV-064-0140, DRYWELL DP CPRSR DISCH VLV
20. 3-FCV-064-0139, DRYWELL DP CPRSR SUCT VLV

C. Standby Gas Treatment System starts

D. 1-FCO-64-44, REFUEL ZONE EXH TO SGT CROSSTIE DMPR, OPENS

E. 3-FCO-64-44, REFUEL ZONE EXH TO SGT CROSSTIE DMPR, OPENS

F. 1-FCO-64-45, REFUEL ZONE EXH TO SGT CROSSTIE DMPR, OPENS

G. CREV Units start

Excerpt from UFSAR 5.3:

5.3 SECONDARY CONTAINMENT SYSTEM

5.3.1 Safety Objective

The safety objective of the Secondary Containment System is to limit the release of radioactivity to the environs after an accident so that the resulting exposures are kept to a practical minimum and are within the guideline values given in published regulations (10 CFR 20 and 10 CFR 50.67 as applicable).

5.3.2 Safety Design Basis

The Secondary Containment System provides secondary containment when the primary containments are intact. In the event of release of radioactivity to the Reactor Building atmosphere, the Secondary Containment System contains the necessary reliable, redundant components and subsystems to isolate, to contain, and to assure controlled filtered elevated release of Secondary Containment Building atmosphere.

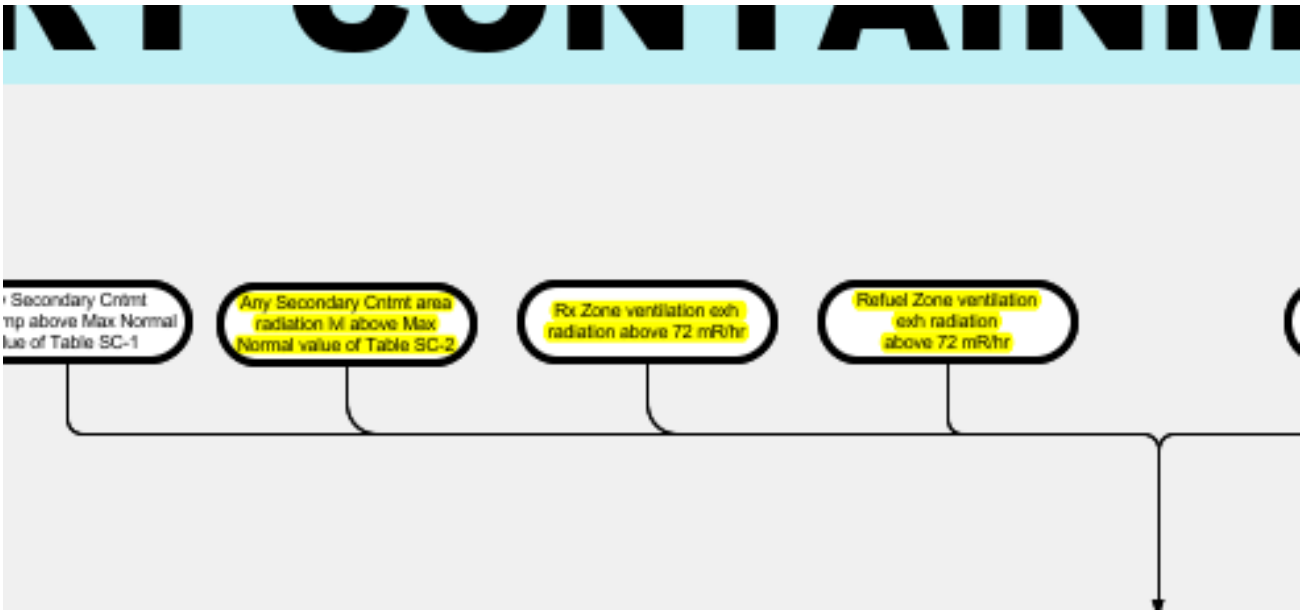
The Secondary Containment System can provide primary containment when any of the three Primary Containment Systems are open such as during refueling (MODE 5) and maintenance operations.

During normal operation and when isolated, the secondary containment is maintained at a negative pressure relative to the building exterior. When isolated, the secondary containment atmosphere is filtered by the Standby Gas Treatment System and released from the plant stack. Provision is made for removal of decay heat from activity deposited on filters.

The secondary containment inleakage rate is less than the SGTS capacity when the building is subjected to an internal negative pressure of 0.25 inch of water. Wind conditions are considered.

The Reactor Building superstructure siding is designed to withstand internal pressure in excess of 57.6 lb/ft^2 without structural failure. Pressures in excess of 50 lb/ft^2 will be relieved by blowoff panels in the siding. The above-grade exterior concrete walls are designed for pressures up to 250 lb/ft^2 without structural failure. The roof is designed for 50 lb/ft^2 live load and 25 lb/ft^2 dead load. The roof internal pressure design goal is 5-inches water gauge[1] (see page III-145 of Reference 1). The loads from fans, ducts, and tanks are carried directly to the building steel and do not load the deck.

Excerpt from 3-EOI-3:



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

Examination Outline Cross-reference:	Level	RO	SRO
262001 (SF6 AC) AC Electrical Distribution	Tier #	2	------
K4.08 (10CFR 55.41.7)	Group #	1	------
Knowledge of AC ELECTRICAL DISTRIBUTION design features and/or interlocks that provide for the following:	K/A #	262001K4.08	
• Alternate breaker control methods	Importance Rating	3.2	------

Proposed Question: **# 42**

Unit 1 is operating at 100% RTP with the following conditions:

Component:	**Status:**
• 4KV Unit Board **2A**	**Energized**
• 2-HS-57-3, 4KV UNIT BD **2A** NORM FDR BKR 1212, Control Switch	**Normal-After-Close**
• 2-HS-57-5, 4KV UNIT BD **2A** ALT FDR BKR 1428, Control Switch	**Normal-After-Open**
• 2-XS-57-4, 4KV UNIT BD **2A** MAN/AUTO SELECT Switch	**AUTO**

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

When 2-HS-57-3, 4KV UNIT BD **2A** NORM FDR BKR 1212, Control Switch is **MANUALLY** taken to TRIP**,** the Normal Feeder Breaker 1212 **(1)** and the Alternate Feeder Breaker 1428 **(2)**.

A. (1) OPENS
(2) CLOSES

B. (1) OPENS
(2) remains OPEN

C. (1) remains CLOSED
(2) CLOSES

D. (1) remains CLOSED
(2) remains OPEN

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Breaker 1212 would OPEN if the Alternate Feeder Breaker 1428 would have CLOSED after Breaker 1212 trips as sensed by the 52b contact finger. The 52b contact on the normal supply breaker (1212) shuts in the close circuit of the alternate feeder breaker (1428), indicating the normal breaker (1212) is OPEN.

Second part is incorrect but plausible in that the Alternate Feeder Breaker

1428 requires the 52b contact on the normal supply breaker (1212) to be shut in the close circuit of Breaker 1428, indicating the normal breaker (1212) is 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 0-OI-57A, Switchyard and 4KV AC Electrical System, and PIP 02-03-AC Electrical, since 4KV UNIT BD 2A NORM FDR BKR 1212 Control Switch, is in Normal-After-Close, it is supplied from USST 2B (normally), not it's alternate power supply from 4KV Start Bus 1A from BKR 1428. With the two given MAN/AUTO SELECT Switches in AUTO, when the BREAKER 1212 Control Switch is manually taken to TRIP, BREAKER 1212 remains CLOSED. This is specifically due to the MAN/AUTO SELECT '43' Switches must be in MANUAL in order to allow a MANUAL transfer to occur.
Second part, with the given conditions, when BREAKER 1212 is manually taken to TRIP, the Alternate Feeder BREAKER 1428 remains OPEN. This is also due the MAN/AUTO SELECT Switches in AUTO. A manual transfer from normal to alternate breaker power supplies would have occurred if the given MAN/AUTO SELECT Switches were in MANUAL, and the 'make-before-break' methodology was used where BREAKER 1428 was CLOSED first, then BREAKER 1212 was OPENED.

RO Level Justification: Tests the candidate's knowledge of AC Electrical Distribution design features and/or interlocks that provide alternate breaker control 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 a solution.

Technical Reference(s): 0-OI-57A, Rev. 177 (Attach if not previously provided)
0-OI-57A, Att, 2, Rev. 139
OPL171.036, Rev. 26U2
PIP-02-03, 04/06/2022

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.036, Obj 8d (As available)

Question Source:

	Bank #	
	Modified Bank #	OPL171.036-06 015 #1288
	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	
	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

1288. OPL171.036-06 015

The following conditions exist for the 4kV Unit Board 1A controls in the main control room:

- 4kV Unit Board 1A — Energized
- Norm Feeder Breaker Control Switch — Normal-After-Close
- Alt Feeder Breaker Control Switch — Normal-After-Open
- Unit Bd 1A Man/Auto Transfer Switch — Auto
- S/D Bus I Man/Auto Transfer Switch — Auto

Which ONE of the following describes the effect of placing the 4kV Unit Board 1A Norm Feeder Breaker Control Switch to **"TRIP"**?

A. Normal feeder OPENS AND alternate feeder CLOSES.
Shutdown Bus I does **NOT** auto transfers.

B. Normal feeder OPENS AND alternate feeder remains OPEN.
Shutdown Bus I auto transfers.

C. Normal feeder remains CLOSED AND alternate feeder remains OPEN.
Shutdown Bus I does **NOT** auto transfer.

D. Normal feeder remains CLOSED AND alternate feeder remains OPEN.
Shutdown Bus I auto transfers.

OBJ B.8

0-OI-57A Rev. 0156, & OPL171.036 Rev. 24

A. incorrect: For manual transfer, transfer switch required to be in MANUAL

B. incorrect: For manual transfer, transfer switch required to be in MANUAL

D. incorrect: 4kV UB BD 1A still normally supplying SD BS 1 therefore no auto transfer occurs

C. correct:

Excerpts from 0-OI-57A:

Attachment 1
(Page 3 of 7)

Auxiliary Power Supplies and Bus Transfer Schemes

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE	REMARKS
7	4kV Recirculation Pump Boards: (Unit 1,2,3)			
	A. Recirc VFD set A	Unit SS TR A (BKRs 1122,1222, 1322)	Start Bus 2A (BKRs 1436,1438, 1442)	Automatic high speed transfer from the normal to the alternate source is initiated by main generator unit trip relays. Automatic delayed transfer from the normal to the alternate source is initiated by high-speed voltage relay. (Breakers are listed in Unit 1, 2, 3 order.)
	B. Recirc VFD set B	Unit SS TR A (BKRs 1124,1224, 1324)	Start Bus 2B (BKRs 1534,1536, 1538)	

		NORMAL	ALTERNATE	ALTERNATE 2	REMARKS
8	4kV Unit Boards (Unit 1)				
	A. 4kV Unit Bd. 1A	Unit SS TR 1B (BKR 1112)	Start Bus 1A (BKR 1424)	Backfeed from shutdown buses	Automatic high-speed transfer from the normal to the alternate source is initiated by main transformer, main generator, or USST protective relays. Automatic delayed transfer from the normal to the alternate source has been defeated by DCN W14030A On 1A, 1B, 2A and 2B Unit Bds. Unit Bds 1C, 2C, 3A, 3B, & 3C will still auto delay transfer from normal to alternate source. To permit use of the condensers as heat sinks for the possible case of normal power outage at the plant, the plant design includes a mode of operation for one running condenser circulating water pump. The controls provide for backfeeding from the shutdown boards to the 4160V unit boards A and B on each unit. For Unit 1 and 2, each shutdown bus and 4160V unit boards A and B provide for trip and lockout of unit transformer and start bus sources to the selected 4160V unit boards, before closure of the selected 4160V shutdown bus breaker which feeds the diesel generator power from the shutdown bus to the 4160V unit board. For Unit 3 only, the backfeed to the Unit boards A and B is directly from the shutdown boards.
	B. 4kV Unit Bd. 1B	Unit SS TR 1B (BKR 1114)	Start Bus 1B (BKR 1524)	Backfeed from shutdown buses	
	C. 4kV Unit Bd. 1C	Unit SS TR 1A (BKR 1116)	Start Bus 1B (BKR 1532)		
	4kV Unit Boards (Unit 2)				
	D. 4kV Unit Bd. 2A	Unit SS TR 2B (BKR 1212)	Start Bus 1A (BKR 1428)	Backfeed from shutdown buses	
	E. 4kV Unit Bd. 2B	Unit SS TR 2B (BKR 1214)	Start Bus 1B (BKR 1526)	Backfeed from shutdown buses	
	F. 4kV Unit Bd. 2C	Unit SS TR 2A (BKR 1216)	Start Bus 1A (BKR 1426)		
	4kV Unit Boards (Unit 3)				
	G. 4kV Unit Board 3A	Unit SS TR 3B (BKR 1312)	Start Bus 1A (BKR 1432)	Backfeed from shutdown bds	
	H. 4kV Unit Bd. 3B	Unit SS TR 3B (BKR 1314)	Start Bus 1B (BKR 1528)	Backfeed from shutdown bds	
	I. 4kV Unit Bd. 3C	Unit SS TR 3A (BKR 1316)	Start Bus 1A (BKR 1434)		

4KV Unit Board 2A from normal to alternate power supply (4KV Start Bus 1A) listing 4KV Shutdown Bus 2 requirement to transfer to/from NORMAL to ALTERNATE and back (see 0-OI-57A, p. 120-121)

8.12 Control Room Transfer of 4kV Unit Board 2A Power Supplies

8.12.1 Transfer 4kV Unit Board 2A from USST to Start Bus

[1] **REVIEW** all Precautions and Limitations in Section 3.0.

[2] **NOTIFY** NSS of possible loss of power to Security Systems prior to transferring 4kV UNIT BD 2A.

NOTES

1) All procedural steps are performed from Control Room Panel 2-9-8, unless specified.
2) This procedure section contains actions necessary to ensure electrical loading restrictions are not exceeded when 4kV UNIT BD 2A is placed on Alternate Supply.

CAUTIONS

1) This board transfer result in a loss of Computer Rooms and Communication Battery Board ACU, Computer UPS ACU, and Communication rooms ACU.
2) Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161 source and large pumps are started. NUSOs should evaluate placing the Capacitor Banks in Manual prior to starting Condensate, CBP, RHR, CS or CCW pumps.
3) If 4kV Unit Board 2A is fed from the Alternate Power Supply, then Auto Transfer must be blocked for 4kV Unit Boards: 1A, 1B, 2B, 3A and 3B. (Ref. 2-45E721 OPL3 & 1-45E721 OPL5)
4) If 4kV Unit Board 2A is fed from the Alternate Power Supply (Start Bus), then Shutdown Bus 2 must be supplied from the Alternate 500kV source (UNIT BD 1B) and Shutdown Bus 2 Auto transfer must be blocked. (Ref. 2-45E721 OPL3)
5) If either 4kV UNIT BD 1A, 1B, 2B, 3A or 3B is aligned to a Start Bus, prior to aligning UNIT BD 2A to Start Bus, check Technical Specifications 3.8.1.a and 3.8.2.a to determine operability of qualified AC circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System.

[3] **ENSURE** 4kV Start Busses are aligned per Attachment 6.

[4] **ENSURE** 4kV SHUTDOWN BUS 2 is transferred to Alternate Supply (4kV UNIT BOARD 1B) per Section 8.19 of this procedure.

8.12.1 Transfer 4kV Unit Board 2A from USST to Start Bus (continued)

[5] **RE-ALIGN** 4kV Auto Transfers to met Load Restrictions.

[5.1] On Panel 1-9-8, **ENSURE** 1-XS-57-4, 4kV UNIT BD 1A MAN/AUTO SELECT switch to MAN.

[5.2] On Panel 1-9-8, **ENSURE** 1-XS-57-7, 4kV UNIT BD 1B MAN/AUTO SELECT switch to MAN.

[5.3] **ENSURE** 2-XS-57-4, 4kV UNIT BD 2A MAN/AUTO SELECT switch to MAN.

[5.4] **ENSURE** 2-XS-57-7, 4kV UNIT BD 2B MAN/AUTO SELECT 67switch to MAN.

[5.5] On Panel 3-9-8, **ENSURE** 3-XS-57-4, 4kV UNIT BD 3A MAN/AUTO SELECT switch to MAN.

[5.6] On Panel 3-9-8, **ENSURE** 3-XS-57-7, 4kV UNIT BD 3B MAN/AUTO SELECT switch to MAN.

[6] **TRANSFER** 4kv UNIT BD 2A to the ALT FDR, BKR 1428.

[6.1] **PLACE** 2-XS-202-1, 4kV BD/BUS/XFMR VOLTAGE SELECT switch to START BUS 1A.

[6.2] **CHECK** START BUS 1A Voltage on 2-EI-57-28 is between 3950 and 4400 Volts.

[6.3] **PLACE** and **HOLD** 2-HS-57-5, 4kV UNIT BD 2A ALT FDR BKR 1428 switch to CLOSE.

[6.4] **PLACE** 2-HS-57-3, 4kV UNIT BD 2A NORM FDR BKR 1212 switch to TRIP.

[6.5] **CHECK CLOSED** the 4kV UNIT BD 2A, ALT FDR BREAKER 1428.

[6.6] **CHECK OPEN** the 4kV UNIT BD 2A, NORM FDR BREAKER 1212.

[6.7] **RELEASE** BKRs 1428 and 1212 control switches.

[6.8] **PLACE** 2-XS-202-1, 4kV BD/BUS/XFMR VOLTAGE SELECT SWITCH TO UNIT BD 2A.

[6.9] **CHECK** 4kV UNIT BD 2A voltage is between 3950 and 4400 Volts.

8.19 Control Room Transfer of 4kV Shutdown Bus 2 Power Supplies (continued)

NOTE

Prior to Transferring Shutdown Bus 2 fed to 4kV UNIT BD 1B, the Auto Transfer for 4kV UNIT BD 1B must be blocked. (Ref. 1-45E721 OPL5)

[10] **PLACE** 1-XS-57-7, 4kV UNIT BD 1B MAN/AUTO SELECT switch to MAN. (Located on panel 1-9-8)

[11] To transfer 4kV SD BUS 2 to the ALT Supply, **PERFORM** the following:

[11.1] On Panel 1-9-8, **PLACE** 1-XS-202-1, 4kV BD/BUS/XFMR VOLTAGE SELECT switch to UNIT BD 1B.

[11.2] **CHECK** UNIT BD 1B voltage on 1-EI-57-28 is between 3950 and 4400 Volts.

[11.3] **CHECK CLOSED** 4kV SD BUS 2 ALT SUPPLY BKR 1132 (Panel 1-9-8).

[11.4] **PLACE** and **HOLD** 0-HS-211-B/22A, 4kV SD BUS 2 ALT FDR BKR 1712, to CLOSE.

[11.5] **PLACE** 0-HS-211-C/26A, 4kV SD BUS 2 NORM FDR BKR 1722 to TRIP.

[11.6] **CHECK CLOSED** the 4kV SD BUS 2 ALT FDR BKR 1712.

[11.7] **CHECK OPEN** the 4kV SD BUS 2 NORM FDR BKR 1722.

[11.8] **RELEASE** breakers 1712 and 1722 switches.

[11.9] **CHECK** 4kV SD BUS 2 AC volts between 3950 and 4400 Volts on 0-EI-211-C, 4kV SD BD C VOLTS.

[11.10] **CHECK** 0-II-211-B/22, 4kV SD BUS 2 ALT FDR BKR 1712 amps, less than 1200.

[11.11] **IF** 4kV UNIT BD 2A is to be aligned to the START BUS, **THEN** do NOT perform step 8.19[11.12] and continue to step 8.19[11.15].

[11.12] **PLACE** 0-43-211-CD2, 4kV SD BUS 2 AUTO/LOCKOUT RESET, to RESET and **CHECK** the following:

8.19 Control Room Transfer of 4kV Shutdown Bus 2 Power Supplies (continued)

[11.13] 0-43-211-CD2, 4kV SD BUS 2 AUTO/LOCKOUT RESET, black target present

[11.14] 0-HS-211-CD2, 4kV SD BUS 2 AUTO TO MANUAL TRIP push-button, amber light illuminated.

[11.15] **CHECK LOCALLY** NORM FDR BKR 1712 closing spring target indicates charged and the amber breaker spring charged light is on.

[11.16] **REFER TO** Precaution 3.3G regarding board voltages while 4kV SHUTDOWN BUS 2 is on the alternate feeder.

[12] To transfer 4kV SD BUS 2 to the NORM Supply, **PERFORM** the following:

[12.1] On Panel 2-9-8, **PLACE** 2-XS-202-1, 4kV BD/BUS/XFMR VOLTAGE SELECT switch to UNIT BD 2A.

[12.2] **CHECK** UNIT BD 2A voltage on 2-EI-57-28 is between 3950 and 4400 Volts.

[12.3] **CHECK CLOSED** 4kV SD BUS 2 NORM SUPPLY BKR 1226 (Panel 2-9-8).

[12.4] **ENSURE** 0-43-211-CD2, AUTO TO MANUAL TRIP switch in MANUAL.

[12.5] **PLACE** and **HOLD** 0-HS-211-C/26A, 4kV SD BUS 2 NORM FDR BKR 1722 to CLOSE.

[12.6] **PLACE** 0-HS-211-B/22A, 4kV SD BUS 2 ALT FDR BKR 1712 to TRIP.

[12.7] **CHECK CLOSED** 4kV SD BUS 2 NORM FDR BKR 1722.

[12.8] **CHECK OPEN** 4kV SD BUS 2 ALT FDR BKR 1712.

[12.9] **RELEASE** breakers 1722 and 1712 switches.

[12.10] **CHECK** 4kV SD BUS 2 AC volts between 3950 and 4400 Volts on 0-EI-211-C, 4kV SD BD C volts.

[12.11] **CHECK** 0-II-211-C/26, 4kV SD BUS 2 NORM FDR BKR 1722 amps, less than 1200.

4KV Unit Board 2A from alternate to normal power supply MAN/AUTO select/transfer switches referencing Attachment 5 (see pg. 186):

8.12.2 Transfer 4kV Unit Board 2A from Start Bus to USST (continued)

NOTES

1) Step 8.12.2[6] restores AUTO transfer to 4kV Boards where AUTO transfer was required to be blocked while UNIT BD 2A was supplied by the Start Bus. Performance of these steps may be delayed until all 4kV UNIT BDs are fed from USSTs.

2) Attachment 5 provides a reference for all restrictions on returning a Unit Board MAN/AUTO transfer switch to AUTO.

[5] For each 4kV UNIT BD addressed in Step 8.12.2[6], **REFER TO** Attachment 5 and **ENSURE** no restrictions on returning the 4kV UNIT BD MAN/AUTO SELECT switch to AUTO. N/A the step if restrictions do not allow return to AUTO.

[6] **RE-ALIGN** 4kV Board 2A Bus Auto Transfer Scheme.

[6.1] On Panel 1-9-8, **ENSURE** 1-XS-57-4, 4kV UNIT BD 1A MAN/AUTO SELECT switch to AUTO.

[6.2] On Panel 1-9-8, **ENSURE** 1-XS-57-7, 4kV UNIT BD 1B MAN/AUTO SELECT switch to AUTO.

[6.3] **ENSURE** 2-XS-57-4, 4kV UNIT BD 2A MAN/AUTO SELECT switch to AUTO.

[6.4] **ENSURE** 2-XS-57-7, 4kV UNIT BD 2B MAN/AUTO SELECT switch to AUTO.

[6.5] On Panel 3-9-8, **ENSURE** 3-XS-57-4, 4kV UNIT BD 3A MAN/AUTO SELECT switch to AUTO.

[6.6] On Panel 3-9-8, **ENSURE** 3-XS-57-7, 4kV UNIT BD 3B MAN/AUTO SELECT switch to AUTO.

[7] **RETURN** the Computer Rooms, Communication Battery Board, Computer UPS, and Communication rooms ACUs to service per 0-OI-31.

[8] **REMOVE** the Caution Order (if one was issued) placed in Step 8.12.1[6.11].

Attachment 5
(Page 1 of 2)

Unit Board Auto Transfer Return to Normal Matrix

A. To determine if a 4kV Unit Board's Auto/Man Transfer Switch can be placed in Auto:

1. On Row 1 of the table below, **FIND** the applicable 4kV Unit Board.
2. **IF** a cell in the column below the Unit Bd. is white, **THEN** the Limiting Condition in Column A of that row APPLIES.

   IF the cell is Black, **THEN** the Limiting Condition does NOT apply.
3. **IF** ALL the applicable Limiting Conditions are satisfied, **THEN** the board Auto/Man Transfer Switch MAY be returned to the **AUTO** position.

	A	B	C	D	E	F	G	H	I
	Limiting Conditions	**4kV Unit Boards**							
1	LIMIT	**1A**	**1B**	**1C**	**2A**	**2B**	**2C**	**3A**	**3B**
2	Unit Bd 1A NOT fed from Start Bus	N/A		N/A			N/A		
3	Unit Bd 1B NOT fed from Start Bus		N/A	N/A			N/A		
4	Unit Bd 1C NOT fed from Start Bus			N/A			N/A		
5	Unit Bd 2A NOT fed from Start Bus			N/A	N/A		N/A		
6	Unit Bd 2B NOT fed from Start Bus			N/A			N/A		
7	Unit Bd 2C NOT fed from Start Bus			N/A			N/A		
8	Unit Bd 3A NOT fed from Start Bus							N/A	N/A
9	Unit Bd 3B NOT fed from Start Bus							N/A	N/A
10	Unit Bd 3C NOT fed from Start Bus							N/A	N/A
11	Com. Bd A NOT fed from Start Bus			N/A			N/A		
12	Com. Bd B NOT fed from Start Bus			N/A			N/A		
13	Start Bus 1A & 2A fed from CSST A	N/A	N/A	(17)	N/A	N/A	(17)	(17)	N/A
14	Start Bus 1B & 2B fed from CSST B	N/A	N/A	(17)	N/A	N/A	(17)	(17)	N/A
15	Shutdown Bus 1 fed from Unit Bd 1A	N/A	N/A	N/A	N/A		N/A	N/A	N/A
16	Shutdown Bus 1 Auto Xfer Operable		N/A	N/A	N/A	N/A	N/A	N/A	N/A
17	Shutdown Bus 2 fed from Unit Bd 2A	N/A		N/A	N/A	N/A	N/A	N/A	N/A
18	Shutdown Bus 2 Auto Xfer Operable	N/A	N/A	N/A		N/A	N/A	N/A	N/A

4KV Shutdown Bus MANUAL/AUTO SELECT/TRANSFER ‘43’ switches:

BFN Unit 0	Attachment 2 Panel Lineup Checklist	0-OI-57A/ATT-2 Rev. 0139 Page 11 of 26

2.0 ATTACHMENT DATA (continued)

Performed On: _____

Component Number	Component Description	Required Position	Initials 1st/IV
Control Bay - Control Room (1 and 2) Panel 9-23-8 - El 617’			
0-HS-211-CD2	4KV SD BUS 2 AUTO TO MANUAL TRIP	Light Illuminated	____ ____
0-43-211-CD2	4KV SD BUS 2 AUTO/LOCKOUT RESET	RESET	____ ____
0-HS-211-CD	4KV SD BUS 2 BACKFEED	NORM	____ ____
0-HS-211-C	4KV SD BD C AUTO TO MANUAL TRIP	Light Extinguished	____ ____
0-HS-211-D	4KV SD BD D AUTO TO MANUAL TRIP	Light Extinguished	____ ____
0-43-211-C	4KV SD BD C AUTO/LOCKOUT RESET	TRIPPED TO MANUAL	____ ____
0-43-211-D	4KV SD BD D AUTO/LOCKOUT RESET	TRIPPED TO MANUAL	____ ____
0-HS-211-B/22A	4KV SD BUS 2 ALT FDR BKR 1712	NORMAL AFTER TRIP	____ ____
0-HS-211-C/26A	4KV SD BUS 2 NORM FDR BKR 1722	NORMAL AFTER CLOSE	____ ____
0-HS-211-C/2A	4KV SD BD C ALT FDR BKR 1624	NORMAL AFTER TRIP	____ ____
0-HS-211-C/4A	DG C BKR 1812	NORMAL AFTER TRIP	____ ____
0-HS-211-C/21A	4KV SD BD C EMER FDR BKR 1814	NORMAL AFTER TRIP	____ ____
0-HS-211-C/22A	4KV SD BD C NORM FDR BKR 1718	NORMAL AFTER CLOSE	____ ____
0-25-211-C/2A	4KV SD BD C BKR 1624 SYNC	OFF	____ ____

Excerpts from OPL171.036 Lesson Plan:

OPL171.036, AC Power Distribution, Rev# 26U2

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
2. There are nine 4kV Unit Boards - three per unit. They are located in the turbine building on Elev. 604 (A and C Boards) and Elev. 586 (B Boards). The USSTs are the normal supply and start buses are the alternate supplies.	
a) USST A is the normal supply to 4kV Unit Board C and USST B is the normal power supply to 4kV Unit Boards A and B. (All Units)	ILT/NLO Obj. 6.d LOR Obj.1.d NLOR Obj. 1.d
b) 4kV Start Bus 1A is the alternate power supply to 4kV Unit Boards 1A, 2A, 2C, 3A, and 3C.	
c) 4kV Start Bus 1B is the alternate power supply to 4kV Unit Boards 1B, 1C, 2B, and 3B	
3. U1 and U2 4kV Unit Boards A and B supply power to 4kV Shutdown Buses 1 and 2 thereby providing off-site power to the Standby AC Power System. 3A and 3B 4kV Unit Boards supply power directly to the U3 4kV Shutdown Boards.	ILT/NLO Obj. 6.a, 6.c, 7 LOR Obj.1.a, 1.c NLOR Obj. 1.a, 1.c
4. Control Room Indications Indication of the 4kV Unit Boards' voltages and amperages are available on Panel 9-8. In addition, each boards pump motor amps is also available (except CRD pumps).	
5. Transfer Schemes	45E763
a) General Operation The 4kV Unit Boards are normally fed from the Unit Station Service Transformers with an alternate feed from the 4kV Start Buses.	
Transfer to the Start Buses may be manual or automatic but transfer back to the USST is manual only. All manual transfers and transformer trip-actuated transfers are fast transfers. Undervoltage relay-actuated transfer is delayed until bus voltage has decreased to 30% normal. A voltage relay prevents automatic transfer to a dead bus. The breakers are electrically interlocked to prevent paralleling the Unit and Common transformers.	ILT/NLO Obj. 8.d LOR Obj. 2.d, OF-5 The 30% UV delayed transfer has been removed from 1A/1B, 2A/2B Unit Boards.

OPL171.036, AC Power Distribution, Rev# 26U2

Lesson Plan Content

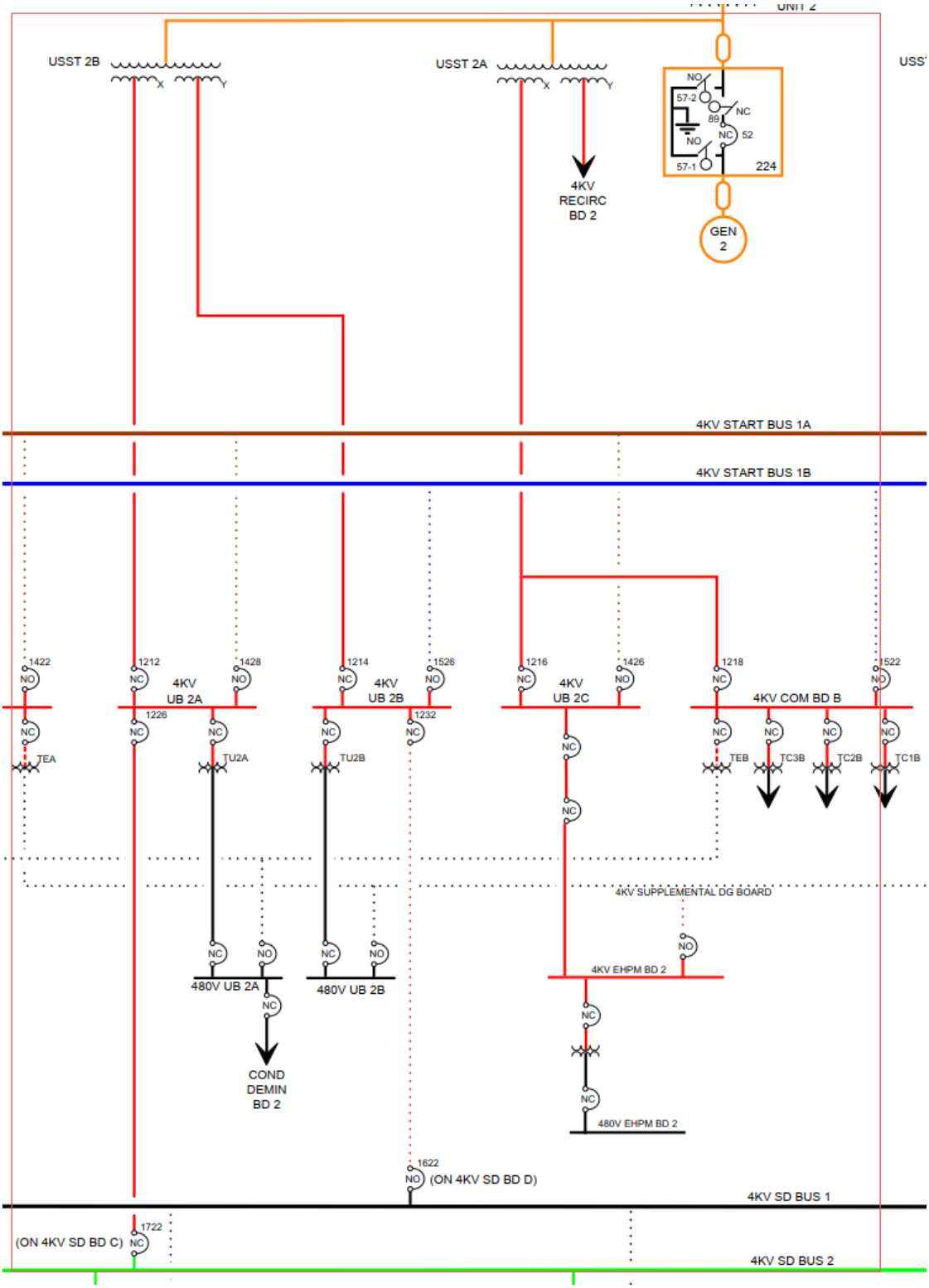
Outline of Instruction	Instructor Notes and Methods (Optional)
b) Automatic fast transfer of Unit Boards occurs on main transformer protective, main generator, or USST protective relaying. To automatically fast transfer from normal to alternate: (1) normal feed breaker tripped (2) 43 selector switch in AUTO (3) Alternate feed line-side voltage available 27SU(x) (4) Alternate feeder breaker closes, provided no lock-outs are present.	0-OI-57A Attachment 1 27SUA - A UB 27SUB - B UB 27SUC - C UB
c) To automatically transfer from normal to alternate on undervoltage: (1) 43 transfer switch in AUTO (2) Alternate voltage available (3) Undervoltage on bus (bus voltage < 30%) and the normal feeder breaker trips. (4) Alternate feeder breaker closes after normal feeder breaker trips (as sensed by 52B finger) provided no lockouts are present.	
d) Manual transfer Unit Board supplies (1) 43 switch in MAN (2) Alternate feeder breaker control switch in CLOSE (3) Open normal feeder breaker and the alternate feeder breaker closes when 52b finger contact from normal feeder breaker closes.	
6. Unit Board Control Power	
Unit Board / Normal / Alternate 1A / TB Distr. BD 1 / TB Distr. BD 2 1B / TB Distr. BD 1 / TB Distr. BD 2 1C / TB Distr. BD 1 / TB Distr. BD 2 2A / TB Distr. BD 2 / TB Distr. BD 3 2B / TB Distr. BD 2 / TB Distr. BD 3 2C / TB Distr. BD 2 / TB Distr. BD 3 3A / TB Distr. BD 3 / TB Distr. BD 1 3B / TB Distr. BD 3 / TB Distr. BD 1 3C / TB Distr. BD 3 / TB Distr. BD 1	0-OI-57A Attachment 2 ILT Obj. 15.c
e) Trip signals (1) 43 switch to manual and Control Switch to trip (2) 69-x to backfeed (3) Alternate feeder breaker closed (4) 86 on Unit Board (differential) 87UA, B, C	Generic list of trip signals for the 4kv unit boards. The specific trips are breaker and board specific.

OPL171.036, AC Power Distribution, Rev# 26U2

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
6. Control Power	
a) Normal 250VDC control power for the 4kv Shutdown Boards is from either a shutdown battery or a unit battery, depending on the board. All alternate supplies are from unit batteries. If control power is lost indication and control is lost. Breakers must be manually operated.	
Shutdown Board Normal Alternate A SB-A BB2 B SB-B BB2 C SB-C BB1 D SB-D BB3 3EA BB1 BB2 3EB SB-3EB BB3 3EC BB3 BB1 3ED BB2 BB3	0-45E765-8 (A 4KV SD BD) 0-45E765-3 (BKR 1614 &1716) ILT Obj. 15.b, 17 How does a loss of Control Power to "A" 4KV SD Board affect the "A" EDG?
b) All control power transfers are manual at the Shutdown Boards.	Answer: There will be no power to the Degraded Voltage relays or the Loss of Voltage relays. Therefore, the EDG would not auto start on either of these conditions.
c) Control power provides for the following: (1) Local and remote breaker operation capability. (2) Local and remote breaker indications. (3) Automatic breaker operation due to protective relays. (4) Board relay power.	Use 0-45E765-8 to show students the above.
7. Transfer Scheme (Units 1/2)	
Note: Capability only, ALL 4KV SD BD 43 switches are procedurally maintained in MANUAL.	ILT/NLO Obj. 8.c LOR Obj. 2.c, OF-5
a) The only automatic transfer of power on a U1/2 4kv shutdown board is a delayed (slow) transfer. In order for the transfer to take place, the bus transfer control relay 43S(x) must be in AUTOMATIC. (1) Undervoltage is sensed on the line side of the normal feeder breaker. (2) Voltage is available on the line side of the alternate feeder breaker. (3) The normal feeder breaker then receives a trip signal. (4) A 52b contact on the normal supply breaker shuts in the close circuit of the alternate feeder breaker, indicating that the normal breaker is open. (5) A residual voltage relay shuts in the close circuit of the alternate supply breaker, indicating that board	0-45E765-3 0-OI-57A Attachment 1 Unit 3 4kv shutdown boards have no immediately available alternate sources for automatic transfers.

Excerpt from PIP-02-03:

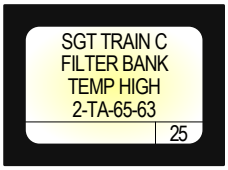


Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
261000 (SF9 SGTS) Standby Gas Treatment	Tier #	2	------
K4.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of STANDBY GAS TREATMENT SYSTEM design features and/or interlocks that provide for the following:	K/A #	261000K4.02	
• Charcoal bed decay heat removal	Importance Rating	3.0	------

Proposed Question: **# 43**

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

Which **ONE** of the following completes the statements below given the conditions above?

The SGT decay heat removal mode **(1)** initiated.

0-DMP-065-0052, Train C Decay Heat Damper, is powered from **(2)**.

A. (1) is automatically
(2) 'A' Diesel Aux Board

B. (1) is automatically
(2) 'A' 480V Common Board

C. (1) must be manually
(2) 'A' Diesel Aux Board

D. (1) must be manually
(2) 'A' 480V Common Board

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Standby Gas Trains automatically start when initiation setpoints are reached. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that Control Bay Ventilation Boards are provided alternate powered from 480V Common Boards. Additionally, both the AC and DC Electrical Distribution Systems at BFN are quite complex and easily and often confused.

C **CORRECT**: *(See attached)* In accordance with 2-ARP-9-3B, Window 25, and 0-OI-65, Standby Gas Treatment System, when filter bank temperature reaches 150 °F, the Operators must take action to initiate decay heat removal on Standby Gas Trains. Second part, in accordance with 0-OI-65, when power is lost to 'C' SGT train, the decay heat removal dampers necessary to achieve decay heat cooling is powered by the 'A' Diesel Aux Board (480V D/G Aux Bd A, Bkr 11A).

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 Standby Gas Treatment System design features and/or interlocks that provide for charcoal bed 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. 57 (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 #	BFN 2304 NRC #45	(Note changes or attach parent)
Modified Bank #		
New		

Question History: Last NRC Exam 2023

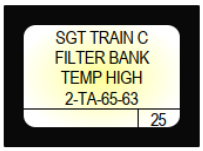
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

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

Copy of Bank Question:

Proposed Question: **# 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

Which **ONE** of the following completes the statements below given the conditions above?

The SGT decay heat removal mode ___**(1)**___ initiated.

0-DMP-065-0052, Train C Decay Heat Damper, is powered from ___**(2)**___.

A. (1) is automatically
(2) 'A' Diesel Aux Board

B. (1) is automatically
(2) 'A' 480V Common Board

C. (1) must be manually
(2) 'A' Diesel Aux Board

D. (1) must be manually
(2) 'A' 480V Common Board

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

Sensor/Trip Point:
TE-65-63B TS-65-63 150°F

(Page 1 of 1)

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. 0057 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.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Z. Although all three trains of the SGT System can be started from the Unit 3 Control Room, it is recommended that the trains be started from the Units 1 and 2 Control Rooms due to the availability of instrumentation and shutdown capability.

AA. In the event that both SGT Trains A and B malfunction, these two trains can be manually aligned and controlled to use either filter bank with either fan. One operating fan can draw suction from both filter banks together (50% through each), or full flow through either filter bank alone.

BB. All three SGT Trains are equipped with decay heat removal lines and dampers to allow a small cooling flow of air for filter decay heat removal through a shut down train. When power is lost to a train, the dampers necessary to achieve decay heat cooling are powered by the adjacent train of SGT as follows:

1. TRAIN A DECAY HEAT DMPR, 0-DMP-065-0004, is powered from the same source as Fan B (480V D/G Aux Bd B BKR 11A).

2. TRAIN B DECAY HEAT DMPR, 0-DMP-065-0026, is powered from the same source as Fan A (480V D/G Aux Bd A BKR 11A).

3. TRAIN C DECAY HEAT DMPR, 0-DMP-065-0052, is powered from the same source as Fan A (480V D/G Aux Bd A BKR 11A).

4. FSV-65-24(66)(2) have been replaced with manual valves 0-DMP-065-0024(0066)(0002) which are throttled for decay heat removal.

CC. When train temperature is greater than 180°F, the relative humidity heater turns off. The heater automatically starts when the SGT train starts.

DD. SGT Trains A and B trip on initiation of 480V load-shed logic, but will auto restart in 40 seconds when an initiation signal is present.

EE. SGT FILTER BANK A(B)(C) INLET DAMPERs, 0-DMP-065-0003(0025)(0051) auto open on automatic or manual initiation.

FF. RX ZONE EXH SGT XTIE DMPR OPR, 1-FCO-064-0041, REACTOR ZONE EXH TO SGT CROSSTIE DAMPER, 2(3)-FCO-64-41 and RFF SGT SUCT DMPR OPR, 1-FCO-064-0045 will **NOT** automatically open in the event that SGT Fan A fails to auto start on a PCIS Group 6 isolation signal.

GG. RX ZONE EXH SGT XTIE DMPR OPR, 1- FCO-064-0040, REACTOR ZONE EXH TO SGT CROSSTIE DAMPER, 2(3)-FCO-64-40 and RFF SGT SUCT DMPR OPR, 1-FCO-064-0044 will **NOT** automatically open in the event that SGT Fan B fails to auto start on a PCIS Group 6 isolation signal.

Supports Distractors A1, B1:

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0057 Page 15 of 44

NOTE

SGT System is now in standby readiness and ready for auto initiation.

5.0 STARTUP

5.1 Standby Gas Treatment System Automatic Initiation

NOTES

1) When normal power is available, the SGT Trains will start without any delay. Otherwise, the trains will start when diesel power is available.

2) All three trains of the SGT System will initiate on any of the following signals from any unit:

- High radiation in the Reactor Zone exhaust duct or one out of two taken twice downscale trip logic.
- High radiation in the Refueling Zone exhaust duct or one out of two taken twice downscale trip logic.
- Low reactor vessel water level (level 3).
- High drywell pressure.

[1] **ENSURE** the following automatic actions occur:

- SGT Trains A, B, and C start.
- SGT FILTER BANK A(B)(C) INLET DAMPER, 0-DMP-065-0003(0025)(0051), opens upon associated Fan Start.
- SGT Filter Banks A, B and C RH Control Heaters energize.
- RX ZONE EXH SGT XTIE DMPR OPR, 1-FCO-064-0040, -0041, REACTOR ZONE EXH TO SGT CROSSTIE DMPRS, 2(3)-FCO-64-40 & 41, and RFF SGT SUCT DMPR OPR, 1-FC0-064-0044 & 0045 (Unit 1 only) open.
- **CHECK** appropriate Reactor Zone or Refueling Zone Ventilation Systems have properly isolated. **REFER TO** 2-AOI-64-2d.

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
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.	ILT Objective 6e LOR/NLOR Objective 4e NLO Objective 2
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	------
K2.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following:	K/A #	262001K2.02	
• AC breaker control power	Importance Rating	3.7	------

Proposed Question: **# 44**

All Units are operating at 100% RTP when the following occurs:

- Battery Board 2 de-energized
- **NO** Operator Actions have been taken

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

4KV Shutdown Board _____ has lost **NORMAL** breaker control power.

A. B

B. C

C. 3EA

D. 3ED

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that both AC and DC Electrical Distribution Systems at BFN is quite complex and easily and often confused by candidate, especially as it relates to normal, alternate, and control power supply conventions. This makes any combination of battery boards plausible as shown where 4KV Shutdown Board 'B' normal breaker control power is 250V Battery Shutdown Board (SB)-B.

B INCORRECT: Incorrect but plausible in that 4KV Shutdown Board 'C' normal breaker control power is 250V Battery Shutdown Board (SB)-C.

C INCORRECT: Incorrect but plausible in that 4KV Shutdown Board 3EA normal breaker control power is 250V Battery Board 1.

D **CORRECT**: (*See attached*) In accordance with 0-OI-57D, DC Electrical System, 4KV Shutdown Board 3ED normal breaker control power is 250V Battery Board 2.

RO Level Justification: Tests the candidate's knowledge of electrical power supplies to the AC breaker control power as it relates to AC Electrical Distribution. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to AC Electrical Distribution.

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

Technical Reference(s): 0-OI-57D, Rev. 203 (Attach if not previously provided)
0-OI-57A, Rev. 177

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171. 036, Obj. 15b (As available)

Question Source:

Bank #	NRC BFN 1501#3	
Modified Bank #		(Note changes or attach parent)
New		

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

Copy of Bank Question:

QUESTION 3 Rev 4

All units are operating at 100% power.

- Battery Board 2 de-energized.

Which 4kV Shutdown Board has no breaker protection for any of its pump loads? (Assume no operator actions have been taken.)

A. 3EA

B. 3ED

C. B

D. C

Answer: B

Excerpts from 0-OI-57D: Also Supports Distractors A, B, C

3.0 PRECAUTIONS AND LIMITATIONS (continued)

FF. 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

GG. 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).

HH. 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."

3.0 PRECAUTIONS AND LIMITATIONS

A. In the event a Unit Battery System is removed from service or a 250VDC RMOV Board is transferred to the alternate supply, one or more of the limitations below may apply. If time permits, a Caution Order should be placed on the affected MOV hand switches prior to transfer of board to alternate to prevent violation of these safe shutdown restrictions.

1. In the event any 250VDC RMOV Board is on its alternate supply, the following restrictions apply to DC motor operated valves that are supplied from a battery that is feeding any RMOV board alternate supply:

   a. No DC MOV may be operated except as required to mitigate accident conditions, to obtain safe shutdown or to comply with Technical Specifications (i.e. to comply with LCO ACTIONS statements only).

   b. Testing (including SI/SRs) that requires DC motor operated valve operation is NOT allowed. [Ref. Dwgs. 1-45E701-3, 2-45E702-4, 3-45E703-3]

DC MOVs that may NOT be operated except as required to mitigate accident conditions or to obtain safe shutdown or to comply with Technical Specifications (i.e. to comply with LCO ACTIONS statements only) with RMOV boards on alternate supply.

RMOV BOARD ON ALTERNATE	NORMAL SUPPLY BATTERY	ALTERNATE SUPPLY BATTERY	MAY NOT OPERATE MOVs SUPPLIED FROM RMOV BD (i.e. supplied from the alternate battery)
1A	1	2	1C, 2A, 3C, 1A
1B	3	1	1A, 2C, 3B, 1B
1C	2	1	1A, 2C, 3B, 1C
2A	2	3	1B, 2B, 3A, 2A
2B	3	1	1A, 2C, 3B, 2B
2C	1	2	1C, 2A, 3C, 2C
3A	3	2	1C, 2A, 3C, 3A
3B	1	3	1B, 2B, 3A, 3B
3C	2	3	1B, 2B, 3A, 3C

Excerpts from 0-OI-57A: Also Supports Distractors A, B, C

Attachment 2
(Page 1 of 1)

4kV Control Power Supplies

BOARD	NORMAL CONTROL POWER SUPPLY	ALTERNATE CONTROL POWER SUPPLIES
4kV SHUTDOWN BOARD A	250V DC DISTRIBUTION PANEL SB-A	BATTERY BOARD 2
4kV SHUTDOWN BOARD B	250V DC DISTRIBUTION PANEL SB-B	BATTERY BOARD 2
4kV SHUTDOWN BOARD C	250V DC DISTRIBUTION PANEL SB-C	BATTERY BOARD 1
4kV SHUTDOWN BOARD D	250V DC DISTRIBUTION PANEL SB-D	BATTERY BOARD 3
4kV SHUTDOWN BOARD 3EA	BATTERY BOARD 1	BATTERY BOARD 2
4kV SHUTDOWN BOARD 3EB	250V DC DISTRIBUTION PANEL SB-3EB	BATTERY BOARD 3
4kV SHUTDOWN BOARD 3EC	BATTERY BOARD 3	BATTERY BOARD 1
4kV SHUTDOWN BOARD 3ED	BATTERY BOARD 2	BATTERY BOARD 3
COOLING TOWER BOARD A	250V DC DISTRIBUTION PANEL DCP-1	250V DC DISTRIBUTION PANEL DCP-2
COOLING TOWER BOARD B	250V DC DISTRIBUTION PANEL DCP-1	250V DC DISTRIBUTION PANEL DCP-2
COOLING TOWER BOARD C	250V DC DISTRIBUTION PANEL DCP-2	250V DC DISTRIBUTION PANEL DCP-1
COOLING TOWER BOARD D	250V DC DISTRIBUTION PANEL DCP-2	250V DC DISTRIBUTION PANEL DCP-1
COOLING TOWER BOARD E/F	CT 7 BATTERY BANK A	CT 7 BATTERY BANK B
RECIRC BOARD 1	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
RECIRC BOARD 2	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
RECIRC BOARD 3	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
UNIT BOARD 1A	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
UNIT BOARD 1B	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
UNIT BOARD 1C	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
UNIT BOARD 2A	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
UNIT BOARD 2B	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
UNIT BOARD 2C	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
UNIT BOARD 3A	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
UNIT BOARD 3B	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
UNIT BOARD 3C	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
COMMON BOARD A	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
COMMON BOARD B	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 1
BUS TIE BOARD	BATTERY BOARD 1	BATTERY BOARD 2
START BOARD 1	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
START BOARD 2	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 1

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

Examination Outline Cross-reference:

259002 (SF2 RWLCS) Reactor Water Level Control

A2.03 (10CFR 55.41.5)

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

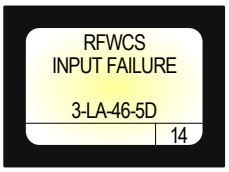
- Loss of reactor water level input

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	259002A2.03	

Importance Rating	3.8	

Proposed Question: **# 45**

Unit 3 is operating at 100% RTP when the following conditions occur:



- Reactor Water Level is (+) 33 inches
- RFWCS INPUT FAILURE (3-9-6C, Window 14) alarms
- OATC reports REACTOR WATER LEVEL NARROW RANGE instruments 3-LI-3-53 (LEVEL **A**) **AND** 3-LI-3-253 (LEVEL **D**) failed **DOWNSCALE**
- **NO** Operator Actions have been taken

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

In accordance with 3-OI-3, Reactor Feedwater System, the Operator is required to bypass the failed instruments on Panel **(1)**.

In the event 3-LI-3-60 (LEVEL **B**) and 3-LI-3-206 (LEVEL **C**), instruments fail, 3-LIC-46-5, REACTOR WATER LEVEL MASTER CONTROLLER PANEL DISPLAY (PDS), will **(2)**.

Note: 3-LI-3-60, REACTOR WATER LEVEL NARROW RANGE, LEVEL B
3-LI-3-206, REACTOR WATER LEVEL NARROW RANGE, LEVEL C

A. (1) 3-9-5
(2) trip to MANUAL

B. (1) 3-9-5
(2) remain in AUTOMATIC

C. (1) 3-9-81
(2) trip to MANUAL

D. (1) 3-9-81
(2) remain in AUTOMATIC

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 3-OI-3, the Operator is required to bypass the failed Narrow Range Reactor Level instruments on Panel 3-9-5 even though with the given ARP, RFWCS automatically bypasses a bad or invalid signal. Second part, 3-OI-3 states that RFWCS 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 already bypassed, 3-LIC-46-5, REACTOR WATER LEVEL MASTER CONTROLLER PANEL DISPLAY (PDS), will trip to MANUAL mode.

B INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that many other Reactor Water Level Instruments are more logical in nature as it relates to logic trip requirements. For example, LI-3-58A/B, LI-3-203A-B-C-D, LI-3-208A-B-C-D, however with the given instruments in the question, candidate must deduce that all 4 of these Narrow Range instruments are related to the same RFWCS trip logic.

C INCORRECT: First part is incorrect but plausible in that Panel 2-9-81/82 are located in the Auxiliary Instrument Room and used to bypass Reactor Water High Level Trip Channel A/B for LI-3-208A-D. Second part is correct (*See A*).

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

RO Justification: Tests the candidate's ability to predict the impacts of REACTOR WATER LEVEL input to Reactor Water Level Control System and use plant procedures to mitigate the consequences of the 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.

Technical Reference(s): 3-OI-3, Rev. 123 (Attach if not previously provided)
3-ARP-9-6C, Rev. 31

Proposed references to be provided to applicants during examination: RFWCS INPUT FAILURE (3-9-6C, Window 14)

Learning Objective: OPL171.003, Obj. 19, (As available)

Question Source:
Bank #
Modified Bank # BFN NRC 2304 #84 (Note changes or attach parent)
New

Question History:
Last NRC Exam 2023

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:

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)

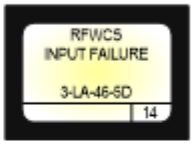
Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- 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 (+) 33 inches
- 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, which **ONE** of the following completes the statements below?

3-LIC-46-5, REACTOR WATER LEVEL MASTER CONTROLLER PANEL DISPLAY STATION (PDS), is in ___**(1)**___ mode.

Emergency Preparedness (EP) ___**(2)**___ required to be notified.

[REFERENCE PROVIDED]

A. (1) MANUAL
(2) is NOT

B. (1) MANUAL
(2) is

C. (1) AUTOMATIC
(2) is NOT

D. (1) AUTOMATIC
(2) is

Proposed Answer: **D**

Excerpt from 3-ARP-9-6C:

BFN Unit 3	Panel 9-6 3-XA-55-6C	3-ARP-9-6C Rev. 0031 Page 21 of 49

RFWCS
INPUT
FAILURE
3-LA-46-5D

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(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

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

BFN Unit 3	Panel 9-6 3-XA-55-6C	3-ARP-9-6C Rev. 0031 Page 33 of 49

RFWCS
FAILED TO
MANUAL
3-LA-46-5B

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(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:

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.

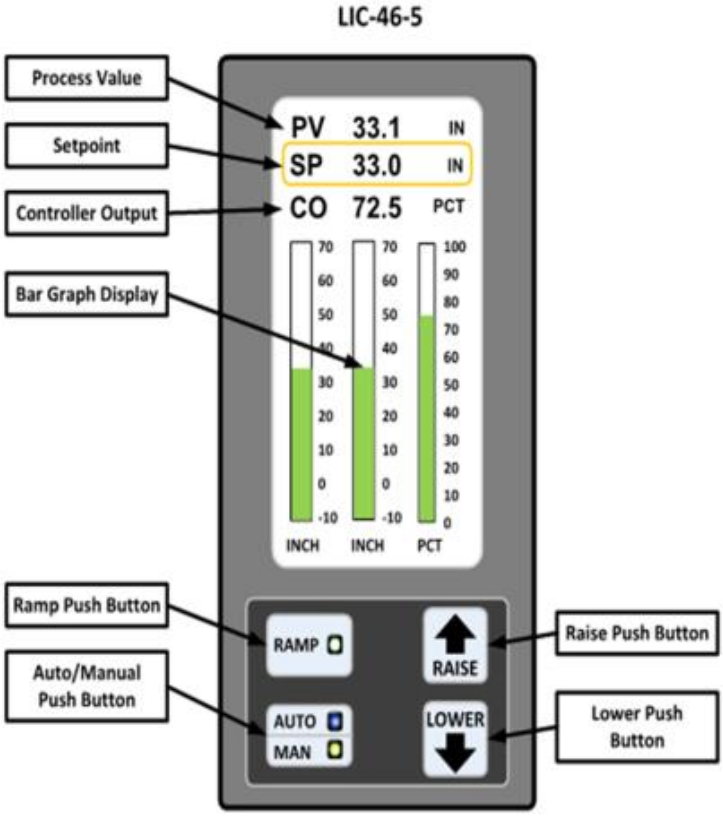
[6] **ENSURE** RFW Control System continues to maintain Reactor water level.

BFN Unit 3	Reactor Feedwater System	3-OI-3 Rev. 0123 Page 300 of 334

Attachment 3
(Page 1 of 4)

Reactor Water Level Master Controller Panel Display Station

1.0 FIGURE 3



Supports Distractors C1, D1:

8.41 Bypassing Reactor Water High Level Trip for Main Turbine and RFPTs for Outage Testing

8.41.1 Bypassing Reactor Water High Level Trip Channel A

[1] **ENSURE** Unit 3 is in Mode 4 or Mode 5.

[2] **OBTAIN** Unit SRO permission to bypass the Reactor Water High Level Trip Channel A.

[3] **NOTIFY** a Unit 3 Control Room Unit Operator that the annunciator for ECCS ANALOG TRIP UNIT TROUBLE 3-XA-71-60 (3-XA-55-3E, window 30) will alarm.

[4] **OBTAIN** key for 3-HS-003-0208AC, RX WTR LVL CH A HI RFPT/MAIN TURB TRIP BYP, from key cabinet in Unit 3 Control Room.

[5] **PLACE** 3-HS-3-208AC, RX WTR LVL CH A HI RFPT/MAIN TURB TRIP BYPASS, in BYPASS on Panel 3-9-81.

[6] **CHECK** Amber indicating light above 3-HS-003-0208AC, RX WTR LVL CH A HI RFPT/MAIN TURB TRIP BYP, is ILLUMINATED on Panel 3-9-81.

[7] **CHECK** Blue indicating light above 3-HS-003-0208AC, RX WTR LVL CH A HI RFPT/MAIN TURB TRIP BYP, is EXTINGUISHED on Panel 3-9-81.

[8] **CHECK** Green indicating light above 3-HS-3-208A, RX WTR LVL CH A HI RFPT/MT TRIP RESET, is ILLUMINATED on Panel 3-9-5.

[9] **CHECK** Red indicating light above 3-HS-3-208A, RX WTR LVL CH A HI RFPT/MT TRIP RESET, is ILLUMINATED on Panel 3-9-5. (N/A if water level was below 55 inches on 3-LI-3-208A and 3-LI-3-208C.)

[10] **DEPRESS** 3-HS-3-208A, RX WTR LVL CH A HI RFPT/MT TRIP RESET, on Panel 3-9-5.

[11] **CHECK** Green indicating light above 3-HS-3-208A, RX WTR LVL CH A HI RFPT/MT TRIP RESET, is ILLUMINATED on Panel 3-9-5.

[12] **CHECK** Red indicating light above 3-HS-3-208A, RX WTR LVL CH A HI RFPT/MT TRIP RESET, is EXTINGUISHED on Panel 3-9-5.

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

Examination Outline Cross-reference:	Level	RO	SRO
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)	Tier #	2	------
A2.05 (10CFR 55.41.5)	Group #	1	------
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:	K/A #	262002A2.05	
• Loss of UPS	Importance Rating	3.8	------

Proposed Question: **# 46**

Unit 1 is operating at 100% RTP with the following conditions:

- Unit Preferred power to Battery Board 1 Panel 11 is lost due to the 1001 breaker tripping
- Panel 1-9-9 Cabinet 5, Non-Preferred, and Cabinet 6, Unit Preferred, are on their respective **ALTERNATE** power supplies in accordance with 1-AOI-57-4, Loss of Unit Preferred

The Unit Operator is now ready to re-energize Battery Board 1, Panel 11 using the 1002 breaker in accordance with 1-AOI-57-4, Loss of Unit Preferred.

Which **ONE** of the following completes the statement below?

Upon re-energization of the Unit Preferred Bus (Battery Board 1 Panel 11), Panel 1-9-9 Cabinet 5 **(1)** automatically transfer to normal power, and Panel 1-9-9 Cabinet 6 **(2)** automatically transfer to normal power.

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: **B**

Explanation (Optional):

A INCORRECT: The BFN AC and DC Electrical Distribution System is quite complex and easily and often confused by candidates especially as it relates to 120VAC and Battery Board breaker and transfer schemes along with the Unit differences. First part is incorrect but plausible in that, ONLY Cabinets 1, 2, 3 and 6 all AUTO transfer back to normal power, however Cabinet 5 does not (*See B*). Second part is correct (*See B*).

B **CORRECT**: (*See attached*) Panel 1-9-9 Cabinet 5 (Unit Non-Preferred) normal power supply is from Unit 1 Battery Board 1 Panel 11 Unit Preferred Bus via breaker 1104 from 1001. Given that Unit Preferred power to Battery Board 1 Panel 11 is lost due to the 1001 breaker tripping open, the Unit Preferred Bus is no longer powered via the inverter being powered from 480V RMOV 1A. Therefore, Panel 1-9-9 Cabinet 5 (Unit Non-Preferred) is de-energized. Alternate power to Cabinet 5 can be provided from a MANUAL transfer to the Plant Non-Preferred Battery Board 2 Panel 14. Once power is restored to the Unit Preferred Bus via 1002 breaker, Cabinet 5 can be MANUALLY transferred back.
Second part, Panel 1-9-9 Cabinet 6, Unit Preferred will AUTO transfer to Battery Board 2 Panel 11 on a loss of normal supply voltage as an Uninterruptable Power Supply. Upon re-energizing the Unit Preferred Bus (Battery Board 1 Panel 11) via breaker 1002 through its regulating transformer (since 1001 tripped), Panel 1-9-9 Cabinet 6 Unit Preferred will AUTO transfer from Battery Board 2 Panel 11 back to the normal source (480V RMOV 1A).

C INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that 9-9 Cabinets 4 and 5 must be manually transferred (*See B*).

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

RO Level Justification: Tests the candidate's ability to predict the impacts of a Loss of UPS based on specific plant conditions and determine the required transfer scheme of the Vital 120V AC Distribution 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.

Technical Reference(s): 0-OI-57C, Rev. 137 (Attach if not previously provided)
1-AOI-57-4, Rev. 40
1-ARP-9-7A, Rev. 34
1-ARP-9-8B, Rev. 18
1-ARP-9-8C, Rev. 21
OPL171.102, Rev. 12
1-45E1647-5 Rev. 24
1-45E1647-6 Rev. 25

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.102 Obj. 2b, 5c (As available)

Question Source:

Bank #	BFN 1804 #46
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2018

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: **# 46**

Unit 1 is operating at 100% RTP with the following conditions:

- Unit Preferred power to Battery Board 1 Panel 11 is lost due to the 1001 breaker tripping
- Panel 1-9-9 cabinet 5, Non-Preferred, and cabinet 6, Unit Preferred, are on their respective alternate power supplies in accordance with 1-AOI-57-4, Loss of Unit Preferred

The Unit Operator is now ready to re-energize Battery Board 1, Panel 11 using the 1002 breaker in accordance with 1-AOI-57-4, Loss of Unit Preferred.

Which ONE of the following completes the statement below?

Upon re-energization of the Unit Preferred Bus (Battery Board 1 Panel 11), Panel 1-9-9 Cabinet 5 **(1)** automatically transfer to normal power and Panel 1-9-9 Cabinet 6 **(2)** automatically transfer to normal power.

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: B

Excerpt from 1-ARP-9-8B:

BFN Unit 1	Panel 1-9-8 1-XA-55-8B	1-ARP-9-8B Rev. 0018 Page 42 of 42

UNIT PFD SUPPLY ABNORMAL

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(Page 1 of 1)

Sensor/Trip Point:

Relay SE - loss of normal DC power source.
Relay TS - DC Xfer switch transfers to Emergency DC Power Source.
Regulating Transformer Common Alarm.
1-INV-252-001, INVT-1 System Common Alarm.

Sensor Location: EL 593' 250V DC Battery Board 2

Probable Cause:

A. Loss of normal DC power source
B. DC power transfer.
C. Relay failure
D. INVT-1 System Common Alarms
   1. Fan Failure Rectifier
   2. Over temperature Rectifier
   3. AC Power Failure to Rectifier
   4. Low DC Voltage
   5. High DC Voltage
   6. Low DC Disconnect
   7. Fan Failure Inverter
   8. Alternate Source Failure
   9. :Low AC Output Voltage
   10. High Output Voltage
   11. Inverter Fuse Blown
   12. Static Switch Fuse Blown
   13. Over Temperature Inverter

E. PFD Regulating XFMR Common Alarms
   1. Transformer Over temperature
   2. Fan Failure
   3. CB1 Breaker Trip
   4. CB2 Breaker Trip

Automatic Action:

A. Auto transfer to DC Power Source on Rectifier failure.
B. Auto transfer to Alternate AC supply (Regulated Transformer) on Inverter failure.

Operator Action:

A. **IF** 120V AC Unit Preferred is lost, **THEN** **REFER TO** 1-AOI-57-4.

B. **REFER TO** appropriate portion of 0-OI-57C.

References:

0-45E641-2	1-45E620-11	1-3300D15A4585-1
10-100467	0-20-100756	20-110437

Excerpt from 1-ARP-9-7A:

PANEL 9-9
CABINET 1,2,3 OR 6
CONTROL POWER
TRANSFER

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(Page 1 of 1)

Sensor/Trip Point:

Transfer switch 83E — Alarm comes in on normal to alternate transfer of power supply to cabinets 1, 2, 3 or 6 on Panel 9-9.

Sensor Location: Panel 9-9
Elev. 617
Control Bay

Probable Cause:
A. Manual transfer of Cabinet 1 250V from Battery Bd. 1.
B. Auto transfer of the following power supplies:
1. Cabinet 2 Inst. & Control Sys. A 120/208V AC
2. Cabinet 3 Inst. & Control Sys. B 120/208V AC
3. Cabinet 6 Unit Preferred Bus 120/240V AC

Automatic Action: None

Operator Action:
A. **ENSURE** or place unit in stable condition.
B. **CHECK** Panel 9-9 for manual or auto transfer of above listed power supplies.
C. **RESTORE** normal power supplies per 0-OI-57C and 0-OI-57D.

References:

1-45E620-10-1	1-45E1647-1 thru 6	0-45E701-1 and 2
0-731E700	0-731E723-1	0-731E753

Excerpt from 1-ARP-9-8C:

BFN Unit 1	Panel 1-9-8 1-XA-55-8C	1-ARP-9-8C Rev. 0021 Page 11 of 48

PNL 9-9 PFD
OR NON-PFD
BKR TRIPOUT
1-EA-57-96

6

(Page 1 of 1)

Sensor/Trip Point:

Breaker tripout.

Sensor Location:
Unit 1 Control Room
EL 617'
Panel 9-9, Cabinets 4, 5, and 6

Probable Cause:
A. Breaker overload or fault tripout (thermal or magnetic trip elements).
B. Manual tripping of breaker on cabinet 6, Panel 9-9.

Automatic Action:
A. None
B. Potential loss of any of the following equipment fed from cabinets 4, 5, and 6.
   1. RFPT oil coolers temperature control valves.
   2. RBCCW heat exchanger temperature control valve.

Operator Action:
A. **CHECK** Panel 9-9, cabinets 4, 5, and 6 for tripped breaker.

B. **CHECK** condition of equipment fed from the tripped breaker.

C. **CLOSE** breaker as directed by Unit SRO.

D. **REFER TO** appropriate OI.

E. **IF** alarm is sealed in, **THEN**
PERFORM the following:
   1. **REFER TO** OPDP-4.
   2. Frequently **MONITOR** the other cabinets that input to this alarm until the alarm is reset.

References:
45E620-11 1-45E1647-4, 5 and 6 45N702-3
GE-731E753

Excerpts from 1-AOI-57-4:

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0040 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)

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.

3.0 AUTOMATIC ACTIONS

A. Unit 1 Panel 1-9-9 Cabinet 6 Unit Preferred automatically transfers to the alternate power source.

B. Reactor Feed Water Control System Panel Display Stations are disabled and the system continues controlling on the last known signal.

C. CRD SYS FLOW CONTROL VLV 1A / B, 1-FCV-85-11 A / B CLOSES on loss of Cabinet 6 and OPENS if only Cabinet 5 is de-energized.

D. UPSTREAM MSL DRAIN TO CONDENSER, 1-FCV-1-58 and BYPASS VLV UPSTREAM MSL DRAIN TO CONDENSER, 1-FCV-1-185 auto open.

E. The following systems swap to the alternate auctioneered ICS power source:

1. EHC Control System (This includes the PLU)
2. Reactor Feed Water Control System
3. Recirc Flow Control System
4. RFPT 1C Woodward Governor.

F. TSI Panel 1-9-46 is powered through auctioneered power supply from both Unit Preferred Panel 1-9-9 Cabinet 6 and Unit Non-preferred Panel 1-9-9 Cabinet 5. On loss of either supply the other will automatically provide power to TSI instruments.

G. Power may be interrupted to various power outlets in the MCR which could result in computer and/or internet access being affected.

4.2 Subsequent Actions (continued)

[5] To reenergize Panel 1-9-9 Cabinet 5, **PERFORM** the following from Panel 1-9-9 Cabinet 5 Non-preferred:

[5.1] **VISUALLY CHECK** XI-57-500A, UNIT NONPFD 120V-AC NORMAL SUPPLY light.

[5.2] **VISUALLY CHECK** XI-57-500B, UNIT NONPFD 120V-AC ALTERNATE SUPPLY light.

[5.3] **OPEN** the supply breaker (normal or alternate) for the light that is extinguished.

[5.4] **CLOSE** the supply breaker (normal or alternate) for the light that is illuminated.

[5.5] **ENSURE** the Off Gas System is lined up and operating properly in accordance with 1-OI-66.

[6] **IF** Unit Preferred Panel 1-9-9 Cabinet 6 has not auto-transferred, **THEN**

PERFORM the following:

[6.1] **ENSURE CLOSED** UNIT 1 CKT BKR BD PNL 1-9-9 UPS (ALT FDR) BKR 1172, at Battery Board 2 (Panel 11).

[6.2] **CHECK** the following fuses in rear of Panel 1-9-9, Cabinet 6 (to the right and under the Auto Transfer switch):

- 1-FU1-252-0600AA, CAB 6 Unit PFD NOR IND ATMR3,
- 1-FU1-252-0600BA, CAB 6 Unit PFD ALT IND ATMR3.

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0040 Page 10 of 32

4.2 Subsequent Actions (continued)

[7] **DISPATCH** personnel to determine status of UNIT PFD SYSTEM INVERTER AND RECTIFIER, 1-INV-252-0001.

NOTE

Upon reenergization of the Unit Preferred Bus (Battery Board 1 Panel 11) Panel 1-9-9 Cabinet 6 Unit Preferred will auto-transfer back to the normal source.

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0040 Page 12 of 32

4.2 Subsequent Actions (continued)

[14] **WHEN** the normal power supply (Unit Preferred) is available to the Non-preferred System, **THEN**

TRANSFER Panel 1-9-9 Cabinet 5 Non-preferred to the normal source as follows:

[14.1] **CHECK** XI-57-500A and XI-57-500B Normal and Alternate Supply lights illuminated.

[14.2] **OPEN** Non-preferred 120V AC Alternate Supply breaker.

[14.3] **CLOSE** Non-preferred 120V AC Normal Supply breaker.

5.5 Plant Drawings

0-45E701-1, 2, Wiring Diagram Battery Board 1

0-45E702-1 & 3, Wiring Diagram Battery Board 2

7-45E702-2 & 4, Wiring Diagram Battery Board 2

0-45E703-1, 2, Wiring Diagram Battery Board 3

0-45E704-5, Wiring Diagram ICS UPS Power Distribution

1-45E1647-5, Wiring Diagram Panel 1-9-9 Cabinet 5 Unit 1

1-45E1647-6, Wiring Diagram Panel 1-9-9 Cabinet 6 Unit 1

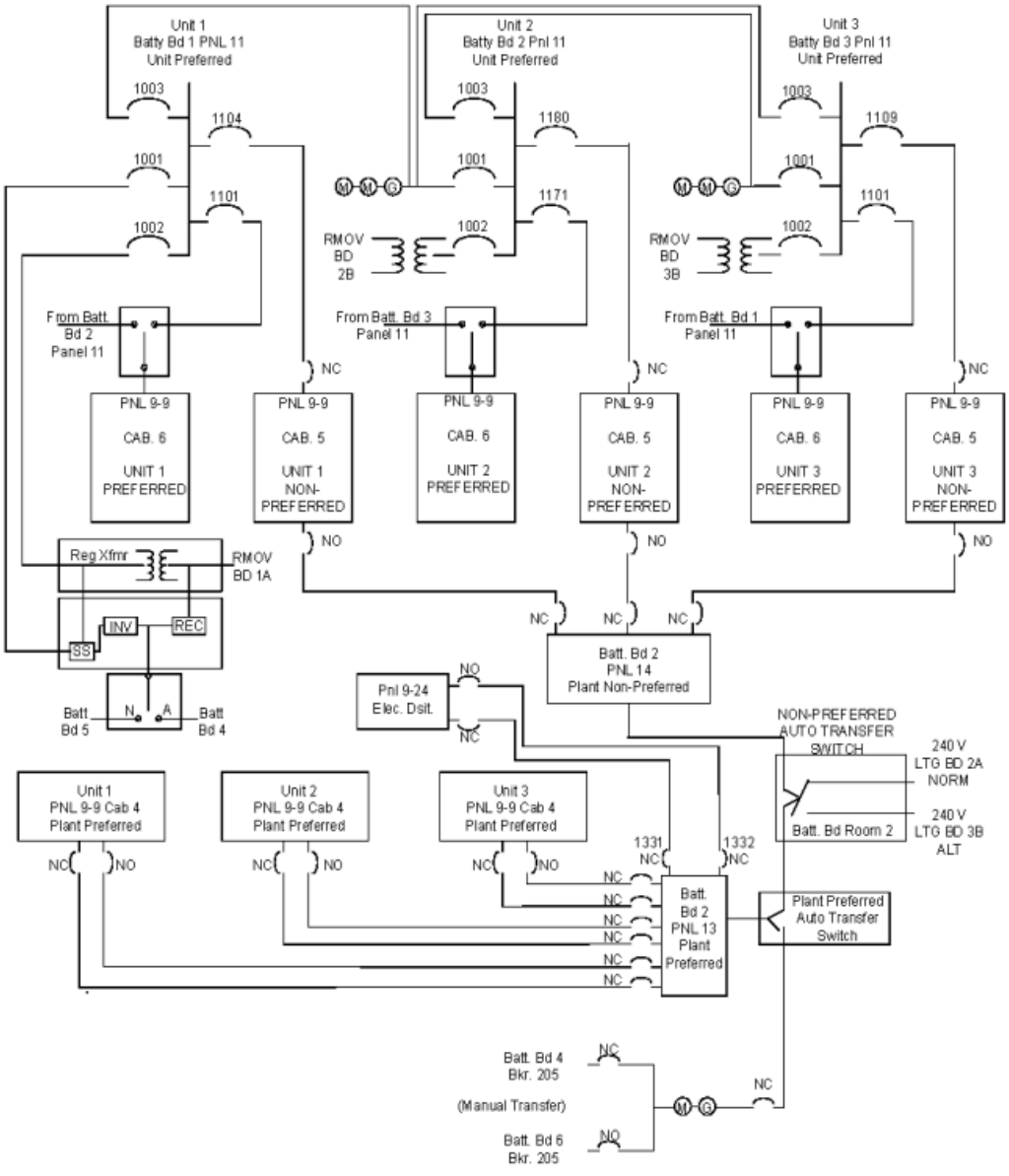
6.0 RECORDS

Records are to be maintained per NPG-SPP-31.2, Records Management.

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0040 Page 17 of 32

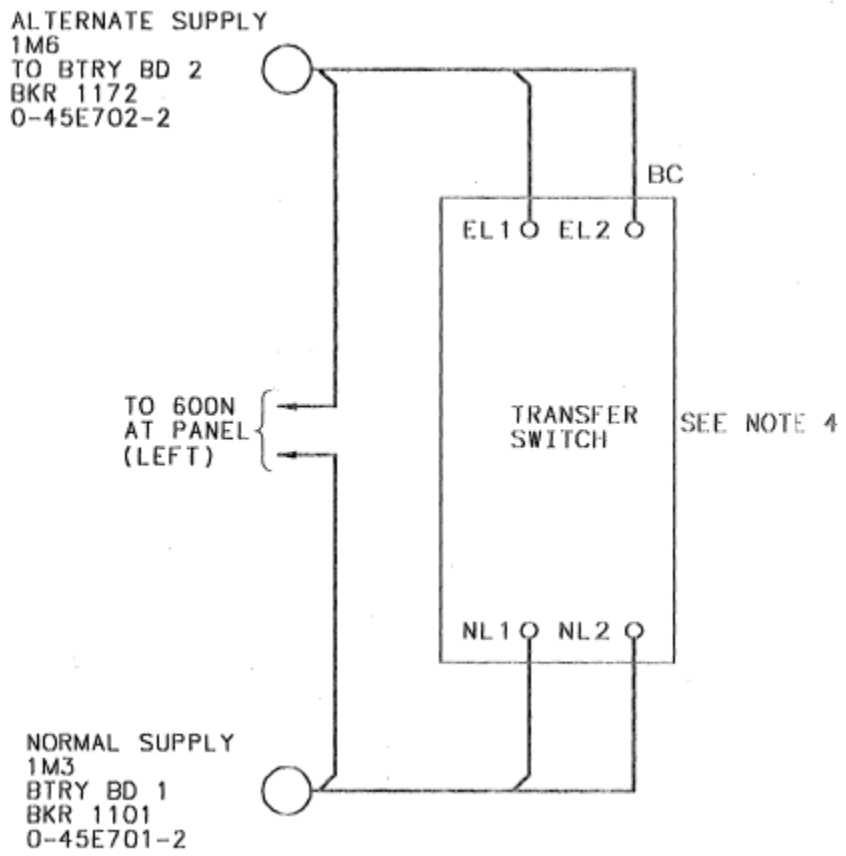
Attachment 1
(Page 1 of 1)

Vital 120V AC Distribution



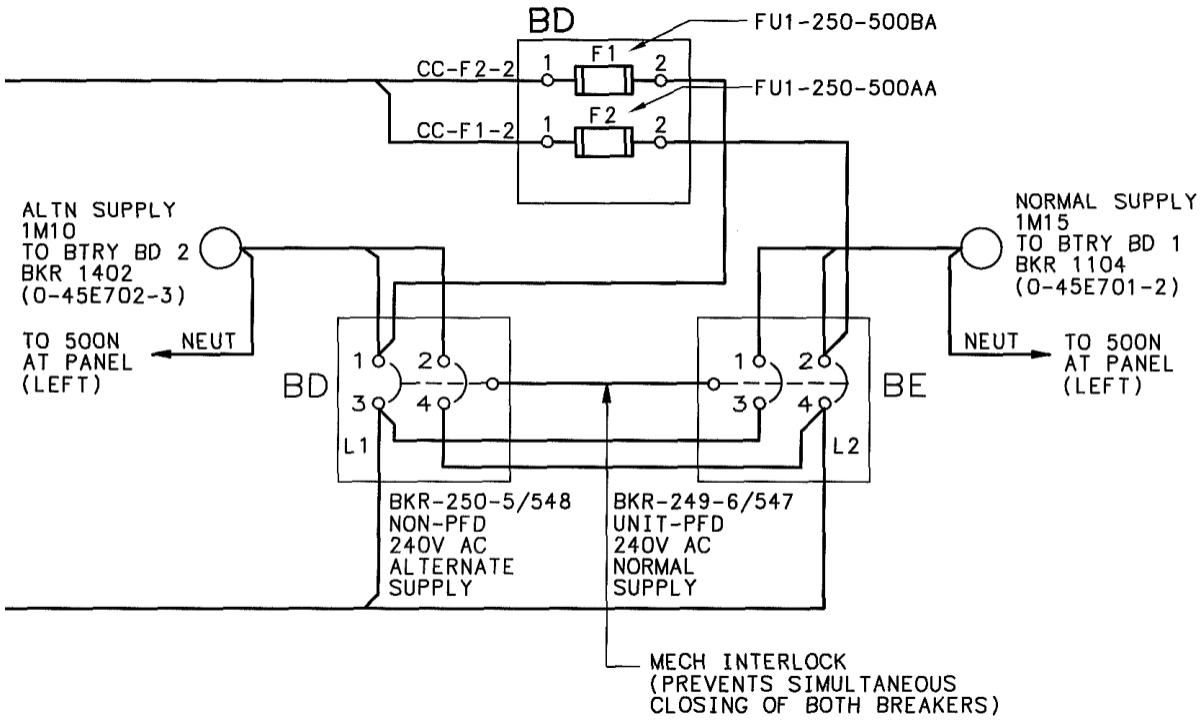
Excerpts from 1-45E1647-6 for Cabinet 6:

PANEL 9-9, CABINET 6 UNIT PREFERRED BUS 120/240V
AND AC ICS UPS DISTRIBUTION 120/208V AC



Excerpts from 1-45E1647-5 for Cabinet 5:

PANEL 9-9, CABINET 5 UNIT PREFERRED BUS,
NORMAL SUPPLY PLANT NON-PREFERRED BUS,
ALTERNATE SUPPLY 120/240V AC



Excerpts from 0-OI-57C:

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Q. A transfer of power supplies for Unit Preferred, Unit Non-Preferred, and Plant Preferred should not affect the EHC Workstations (HMI) and the EHC Control System Power Load Unbalance (PLU) channels. For Unit 1 and 2, uninterruptible power supplies have been installed to keep the PLU and HMI screens energized for approximately 15 minutes. A sustained loss of these power supplies (>15 minutes) would result in the applicable PLUs bypassing and HMI screens going blank. The status of PLUs, whether in Normal or Bypassed, can be seen on the Panel 9-7 (Lower) screen on the EHC Workstation (HMI). System Engineering should be contacted whenever a PLU needs to be unbypassed or when an EHC Workstation (HMI) needs to be restarted.

Plant experience has shown that PLU Uninterruptible Power Supplies (UPS), 1(2)(3)-UPX-047-0089/1(/2) located in 1(2)(3) PNL-9-47 Cabinet 3 for the affected unit should be verified energized with no alarm indications PRIOR to transfer of power supplies for Unit Preferred, Unit Non-Preferred, and Plant Preferred

R. Unit 1 Unit Preferred normal supply to Unit Preferred Battery Bus 1 Cabinet 11 is using inverter 1-INV-252-0001. The alternate supply is through Unit Preferred Regulating Transformer XFMR1. To prevent possible inverter or transformer damage, all operation with Unit 2 MMG (Alternate Emergency) source, must be performed by a dead bus transfer.

S. UNIT PFD SYSTEM INVER AND RECTIFIER, 1-INV-252-0001 can only operate in parallel with the Alternate AC source, XFMR1, for a short period of time. Damage to the inverter/rectifier or transformer could occur if operated in parallel for an extended period.

T. Transfer of DC power from Normal Feeder Battery Board 5 to Alternate Feeder Battery Board 4 is limited by the availability of the Battery Board 4 changer. If the charger is not available, the load is not to be transferred.

U. 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)

8.2 Placing the Unit Preferred Transformer in Service with the Alternate (Emerg) MMG Set Supplying Power to the Bus (continued)

[3] **ENSURE** the following initial conditions are satisfied:

A. The Unit Preferred 2nd Alternate (Emerg) MMG is supplying power to the bus (see Table 2).

TABLE 2

BATTERY BOARD	NORMAL POWER	1st ALTERNATE POWER	2nd ALTERNATE (EMERG POWER)
1	1-INV-252-0001	1-XFA-252-0001	MMG Set 2
2	MMG Set 2	XFMR 2	MMG Set 3
3	MMG Set 3	XFMR 3	MMG Set 2

B. The Unit Preferred 1st Alternate Power Transformer primary side is energized (480V RMOV Bd 1A Compartment 19E or (480V RMOV BD 2B(3B) Compartment 5C1(17D)).

C. The Unit Preferred Transformer is not in service to RPS.

NOTE

The following step applies to Unit 3 only.

D. UNIT PREFERRED TRANSFORMER TUP3 FUSED DISCONNECT SWITCH, 3-FUDS-252-0003/1, located in Battery Board Room 3 ON.

Excerpts from OPL171.102 Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
interruption of power. IF normal AC returns, then the diode will become reversed bias and the normal AC will supply the inverter output voltage.	
(c) On a loss of inverter output (inverter failure) the AC load will automatically shift to the alternate supply via regulating transformer supply with no loss of loads. When the inverter output returns, the supply to the loads must be manually returned to the inverter output via the static switch.	0-OI-57C Sect 8.6 and 8.7
(d) The supply to the BAT BD 1 PNL11 can be manually transferred from the static switch to the regulating transformer and back again. The feed from the regulating transformer should be minimized due to lack of battery backup. (This is used to shutdown the inverter for maintenance)	See OI-57C P&L's
d) Distribution	IL-3,4,5,6
The UPS Distribution Bus for each unit is located on Panels 10 and 11 on the Unit 250V Battery Board.	NLO Obj. 2.h.
(1) Panel 10 contains the 1001, 1002, and 1003 breakers for that unit. These breakers are controlled from the Synchronizing and Control Panel on Batt Bd 2, Panel 11.	
(2) Panel 11 contains the UPS distribution bus and breakers. From Panel 11, power is sent to Panel 9-9 cabinet 6 in each unit's control room, Panel 25-32 for each unit, and computer (unit 1), and to Panel 9-9 Cabinet 5 (Unit Non-Preferred).	ILT Obj. 2.a HU Tool Touch STAR, controls for all 3 units are on same panel. NLO Obj. 2.h.iii
(3) Panel 9-9 Cabinet 6 will auto transfer to another unit's battery board panel 11 on a loss of normal supply voltage. When normal voltage is restored, it will auto transfer back to the normal source. The normal and alternate supplies for each unit are:	ILT Obj. 2.c LOR Obj. 3.a

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)

	NORM	ALT
U1	BB1 Panel 11	BB2 Panel 11
U2	BB2 Panel 11	BB3 Panel 11
U3	BB3 Panel 11	BB1 Panel 11

NLO/NLOR Obj. 2.f

(4) Unit 1 25-32 panel has a normal and an alternate Unit Preferred supply, which can be manually transferred. The normal and alternate supplies are:

IL-3

	NORM	ALT
U1 25-32	BB1 Panel 11	BB3 Panel 11

(5) Unit 2 & 3 normal and alternate supplies for 25-32 Unit Preferred loads are:

IL-4,5,7

	NORM	ALT
U2 25-32	BB2 Panel 11	BB1 Panel 11
U3 25-32	BB3 Panel 11	BB2 Panel 11

(6) U-1/2/3 25-32 Panels ATU bus normal supply is from Div I ECCS ATU Inverter, and have an ALT power supply from:

IL-7,8,9,12

U1	U2	U3
BB3 1106	BB1 1106	BB2 1176

(7) The Unit 1, computer is normally supplied from 480V Common Board 1 compt. 9C with backup DC to the inverter from Battery Bd 5-breaker 316. The Unit 2 computer is fed from 480V Common Bd 2 compt. 9A. Backup DC is from Battery Bd 4 bkr 316. The Unit 3 computer is normally supplied from the 480V Common Board 3 with DC backup from battery Board 6 breaker 316.

e) Controls

Normally, Unit 2 & 3 Unit Preferred MMG sets are controlled from Battery Bd 2 panel 11.

(1) A Local/Remote switch for each MMG set will also allow for local MMG set control.

ILT Obj. 2.b, 2.h

(2) An Off/Auto/Manual switch for each MMG set controls transfers from the AC to DC MMG set motors. In Auto, if the AC motor loses

UF or UV on AC motor supply will stop AC motor and start DC motor.

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
lighting Board 3B will occur. When normal power is restored, the system will automatically transfer back.	
b) Distribution	
(1) Battery Board 2, panel 14 contains the distribution breakers for the Plant Non-Preferred system.	ILT Obj. 4.a NLO Obj. 4.f
(2) From Battery Board 2, power is sent to each unit's panel 9-9 cabinet 5, as an alternate power supply. The normal supply to panel 9-9, cabinet 5 is that unit's unit preferred system. The power supplies to 9-9 cabinet are manually transferred if necessary.	NLO/NLOR Obj. 4.c ILT Obj. 4.b NLO/NLOR Obj. 4.e
c) The alarm "Panel 9-9 PFD or Non-PFD Bkr Tripout" could indicate trip of load or supply breaker on Panel 9-9, cabinet 5.	ILT Obj. 4.b, 4.c. LOR Obj. 4
(1) Since 9-9 cabinet 5 is normally powered from UPS, loss of Non-Preferred will not affect it. If 9-9 cabinet 5 is de-energized, off-gas panel 9-53 will auto transfer to 9-9 cabinet 4. If 9-9 cabinet 4 is also unavailable, an off-gas system isolation will occur causing a loss of condenser vacuum.	NLO/NLOR Obj. 4.d LOR Obj. 5
(2) Refer to AOI 57-4 for other indications.	
5. Unit Non-Preferred system	
a) Purpose and design	
(1) The Unit Non-Preferred system supplies loads which are not required to safely shutdown the plant following a loss of AC power.	ILT/NLO Obj. 5.b
(2) The Unit Non-Preferred system Panel 9-9-5 is normally powered from its respective Unit Preferred System (UPS) via Bat Bd 1(2)(3) Panel 11. (Control building 1C in respective Bat Board rooms)	ILT/NLO/NLOR Obj. 5.a ILT/NLO Obj. 5.e
(3) The backup power supply is from the Plant Non-Preferred system, Batt Bd panel 14.	
b) Distribution	
(1) Battery Board 1(2)(3), Panel 11 has feeder bkr (1180) that feeds 9-9-5 (located in the respective control room). A loss of Unit Preferred System (UPS) will result in a loss of Unit Non-Preferred until power is manually transferred.	ILT/NLO Obj. 5.c NLOR Obj. 5.b

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods (Optional)
(2) Battery Board 2, panel 14, (Located in the battery Bd. 2, elev. 1C) Plant Non-preferred power contains the distribution breakers that provide alternate feed the Unit Non-Preferred system. Plant non-preferred power has an automatic bus transfer to maintain its power. Note: Transfer of power to the Unit Non-Preferred from UPS to Plant Non-Preferred is done manually.	ILT/NLO Obj. 5.e
(3) From Battery Board 2, power is sent to each unit's panel 9-9, cabinet 5 as an alternate power supply. The normal supply to panel 9-9, cabinet 5 is that unit's unit preferred system.	ILT/NLO Obj. 5.e
c) The alarm "Panel 9-9 PFD or Non-PFD Bkr Tripout" (annunciator XA-55-8C, window 6) could indicate trip of load or supply breaker on Panel 9-9, cabinet 5. AOI-57-4 “Loss of Unit Preferred” has actions to transfer Unit Non-Preferred to its backup supply.	ILT/NLO Obj. 5.d NLOR Obj. 5.c
(1) Since panel 9-9, cabinet 5 is normally powered from UPS, loss of Non-Preferred will not affect it. If panel 9-9, cabinet 5 is de-energized, off-gas panel 9-53 will auto transfer to 9-9, cabinet 4. If panel 9-9, cabinet 4 is also unavailable, an off-gas system isolation will occur causing a loss of condenser vacuum.	ILT/NLO Obj. 5.d LOR Obj. 6 NLOR Obj. 5.c
(2) Other impacts with loss of Unit Non-preferred: (a) Loss of TIP channel components (b) Loss of various Control Rod indications and relays/permissives (c) Loss of power to various turbine supervisory components (d) Loss of jet pump instrumentation (e) Reactor recirculation protective relays (f) Loss of RB flooding level switches in RHR/HPCI/core spray pump rooms (g) Refer to AOI-57-4 for a complete list	ILT/NLO Obj. 5.d
(3) Refer to AOI 57-4 for other indications.	
6. ECCS ATU Inverters	
a) Purpose and design	

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

Examination Outline Cross-reference:

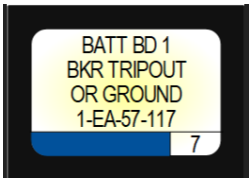
263000 (SF6 DC) DC Electrical Distribution
A4.04 (10CFR 55.41.7)
Ability to manually operate and/or monitor DC ELECTRICAL DISTRIBUTION in the control room:

- Ground detection circuit

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	263000A4.04	
Importance Rating	2.8	------

Proposed Question: **# 47**

Unit 1 is operating at 100% RTP when the following alarm is received:



- BATTERY BOARD 1 BREAKER TRIPOUT OR GROUND (1-9-8C, Window 7)

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

In accordance with 1-ARP-9-8C, Window 7, 250V DC ground indication **(1)** available in Battery Board Room 1.

In accordance with 0-OI-57D, DC Electrical System, within **(2)** **AFTER** the loss of the normal charger to a 250V Unit Battery, another charger shall be placed in service to that battery.

A. (1) is
(2) 30 minutes

B. (1) is
(2) 4 hours

C. (1) is NOT
(2) 30 minutes

D. (1) is NOT
(2) 4 hours

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 1-ARP-9-8C, Window 7, a probable cause is that a ground fault exists on 250V DC Battery Board 1. The Operator is required to check Battery Board 1 volts (EI-57-37) and amps (EI-57-38) on Panel 1-9-8 in the Main Control Room to determine if load is on the battery or the charger. Also required to check Battery Board Room 1 for 250V DC ground indication on the board itself on Panel 10 in the Control Bay. Second part, in accordance with 0-OI-57D, within 30 minutes after the loss of the normal charger to a 250V Unit Battery (Battery 1, 2, 3), another charger shall be placed in service to that battery.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that 0-AOI-57-1A, states that 250V Battery Charger for Unit Battery and Board related to common HVAC equipment operation has 4 hour qualified Station Blackout Requirements.

C INCORRECT: First part is incorrect but plausible in that the BFN DC Systems consist of 250 VDC, 125 VDC, 48 VDC, +/- 24 VDC, countless batteries/chargers/panels and Unit differences which could easily confuse candidates on specific parameter indications/locations both inside/outside of the MCR. 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 DC ELECTRICAL DISTRIBUTION in the control room as it relates to an electrical ground and/or fault determination. This question is rated as Memory due to the requirement to strictly recall required Operator actions and component locations pertaining to DC grounds.

Technical Reference(s): 0-OI-57D, Rev. 203 (Attach if not previously provided)
1-ARP-9-8C, Rev. 21
0-AOI-57-1A, Rev. 121

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.037 Obj. 15 (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 1-ARP-9-8C:

BFN Unit 1	Panel 1-9-8 1-XA-55-8C	1-ARP-9-8C Rev. 0021 Page 12 of 48

BATT BD 1
BKR TRIPOUT
OR GROUND
1-EA-57-117
7

(Page 1 of 2)

Sensor/Trip Point:

Breaker tripout.
Ground on Battery Board 1

Sensor Location: Battery Board Room 1, El 593'

Probable Cause:
A. Breaker overload or fault (thermal and magnetic trip) on Panels 1-14
B. A ground fault exists on 250V DC Battery Board 1.
C. Fuse failure to instrumentation circuit, Panel 1.
D. Blown light bulb in ground detector indicator photo cell.
E. A ground fault exists on 48V DC,(Battery Board 1 PNL 10R)

Automatic Action: Potential loss of any of the following equipment fed from Battery Board 1.
- 250V DC RMOV Board 1A, 2C and 3B
- Feedwater inverter
- 250V DC Cabinet 1, Panel 1-9-9

Operator Action:
A. **CHECK** Battery Board 1 volts (EI-57-37) and amps (EI-57-38) on Panel 1-9-8 to determine if load is on the battery or the charger.

B. **CHECK** Battery Board 1 for abnormal conditions:
- Loss of volt and amp indication.
- Breaker position.
- Ground indication, 250V DC and 48V DC.
- Fuse failure.
- Check ground detector photo cell light bulb (requires electrical maintenance).

C. **CHECK** condition of equipment fed from the tripped breaker and attempt reclosure as directed by Unit SRO.

Continued on Next Page

BFN Unit 1	Panel 1-9-8 1-XA-55-8C	1-ARP-9-8C Rev. 0021 Page 13 of 48

BATT BD 1 BKR TRIPOUT OR GROUND 1-EA-57-117, Window 7
(Page 2 of 2)

Operator Action: (continued)

D. **REFER TO** TS Section 3.8.7.

E. **REFER TO** GOI-300-2 for grounds and location of D-C grounds.

F. **IF** this alarm is invalid, **THEN**
REFER TO OPDP-4.

G. **MONITOR** other breakers on the Battery Board until alarm is reset.

References: 1-45E620-11 0-45E701-1,2 TS Section 3.8.7

Excerpts from 0-OI-57D:

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0203 Page 20 of 399

3.0 PRECAUTIONS AND LIMITATIONS (continued)

F. 250V Unit Battery Charger 1, 2A, 2B and 3 LOAD SHED SWITCH bypasses battery charger emergency load shed contacts. Placing the LOAD SHED SWITCH in EMERG 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 LOAD SHED SWITCH is in the EMERG Position.

G. The Autostart mode will be enabled for Main Bank Battery 1, 2A, 2B, 3, 4, 5 and 6 and Shutdown Board Battery SB-A, SB-B, 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. The polarity of the Shutdown Board 250V Distribution Panels may not be the same for all Shutdown Board 250V Distribution Panels, therefore the polarity and pin configuration must be checked at the SPARE CHARGER INPUT RECEPTACLE on the distribution board. The discharge cable that goes between the distribution board and portable spare battery charger is to be connected in the configuration needed to place the proper polarity DC Voltage to the distribution board.

I. [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]

J. 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.

K. [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]

L. 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.

Table listing Battery Board Room 1 location for ground indication:

6.1 Normal Operations (continued)

TABLE 3
48 VOLT DC UNIT BATTERY SYSTEM

LOCATION	PARAMETER	NORMAL RANGE
Battery Board Room No. 1 (3) 48V Charger A (B, Spare) Control Bay 593'	DC Volts	Greater than 48 Volts
	DC Amperes	Greater than zero less than 110 amps
	Power on Light	Illuminated
	Overvoltage DC Light	Extinguished
	Undervoltage DC Light	Extinguished
	Undervoltage AC Light	Extinguished
Battery Board Room No. 1(3) Panel 10 Rear Control Bay 593'	DC Volts	Greater than 48 Volts
	DC Amperes	Zero amps
	Bus GRD Indication(1)	Zero Volts
Control Room Panel 9-09 48V DC Distribution	48V DC Annunciator Supply Failure Light X1-55-129A	Extinguished
	48V DC Annunciator Supervisory Supply Light X1-55-129B	Illuminated
	Unit 1 - 48V DC SPLY AVAILABLE BATTERY BD 1, IL-251-0129A Unit 2(3) - 48V DC ANN Normal Supply Light X1-55-129C	Illuminated
	Unit 1 - 48V DC SPLY AVAILABLE BATTERY BD 3, IL-251-0129B Unit 2(3) - 48V DC ANN Alternate Supply Light X1-55-129D	Illuminated
Control Room Panel 9-24 Bat 1/Bat 3/Bus	48V DC Voltmeter	Greater than 48 Volts
	48V DC Ammeter	Zero amps
	48V DC Ground Indicator(1)	Zero Volts

(1) **IF** a ground of an absolute valve greater than or equal to 10 volts is indicated, **THEN**

REFER TO 0-GOI-300-2.

Excerpt from 0-AOI-57-1A: Supports Distractors B2, D2

Attachment 5
(Page 4 of 4)

Common HVAC Equipment Operation

1.0 COMMON HVAC EQUIPMENT OPERATION (continued)

Table 5b

Load	SBO Requirements	KW	Source/BKR All 480V Loads	Source 2/BKR All 480V Loads
SD BD Rm 3EA & 3EB Chiller 3A-1/3B-1	Only required if either 4KV SD Bd 3EA or 3EB is in use	20.33	RMOV Bd 3A (R9A)	RMOV Bd 3A (R9B1)
SD BD Rm 3EC & 3ED Chiller 3A-2/3B-2	Only required if either 4KV SD Bd 3EC or 3ED is in use	20.33	RMOV Bd 3B (R9A)	RMOV Bd 3B (R9B1)
SD BD Rm 3EA & 3EB CW Pump 3A-1/3B-1	Only required if either 4KV SD Bd 3EA or 3EB is in use	4.70	RMOV Bd 3A (R9C)	RMOV Bd 3A (R9D)
SD BD Rm 3EC & 3ED CW Pump 3A-2/3B-2	Only required if either 4KV SD Bd 3EC or 3ED is in use	4.79	RMOV Bd 3B (R9C)	RMOV Bd 3B (R9D)
SD BD Rm 3EA & 3EB AHU Pump 3A-1/3B-1	Only required if either 4KV SD Bd 3EA or 3EB is in use	4.30	RMOV Bd 3A (R9E)	RMOV Bd 3A (R9F)
SD BD Rm 3EC & 3ED AHU 3A-2/3B-2	Only required if either 4KV SD Bd 3EC or 3ED is in use	4.79	RMOV Bd 3B (R9E)	RMOV Bd 3B (R9F)
Unit 1 Electric Board Rm AHU 1A/1B	One unit is required	25.43	RMOV Bd 1A (9A)	RMOV Bd 1B (13D1)
Unit 2 Electric Board Rm AHU 2A/2B	One unit is required	18.50	RMOV Bd 2A (1B)	RMOV Bd 2B (8A1)
Unit 3 Electric Board Rm ACU 3A/3B	One unit is required	66.86	RMOV Bd 3A (9A)	RMOV Bd 3B (10B)
250V Batt Charger for Unit Batt & Bd	Not required 4 hour qualified	93.79	Multiple sources	Multiple sources
4KV SD BD Batt Charger A-D, 3EB	Only battery chargers associated with the in-use 4KV and 480-V SD BDs are required	14.86	Note 3	Note 3
DG Battery Charger A-D, 3A-3D	Only battery chargers associated with the in-use DGs are required	5.6	Note 1	Note 2

1 From Diesel Aux Board A or 3EA according to DG involved.

2 From Diesel Aux Board B or 3EB according to DG involved.

3 Shutdown Board Battery Chargers are located on RMOV 1A (16C1), RMOV 1B (17B1), RMOV 2A (5A2), RMOV 2B (4C), and DG Aux Board 3EA (4C1).

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

Examination Outline Cross-reference:

264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG
K2.02 (10CFR 55.41.7)
Knowledge of electrical power supplies to the following:

- Fuel oil pumps

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	264000K2.02	
Importance Rating	2.9	------

Proposed Question: **# 48**

Unit 3 is operating at 100% RTP with the following conditions:

- 3-SR-3.8.1.1(3A), Diesel Generator (EDG) '3A' Monthly Operability Test, is in progress
- '3A' EDG is loaded with DG '3A' MODE SELECT switch in PARALLEL WITH SYSTEM

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

The power supply for '3A' EDG electric fuel oil priming pump is **(1)** and if the power supply is lost, '3A' EDG **(2)** automatically trip.

A. (1) DC
(2) will

B. (1) DC
(2) will NOT

C. (1) AC
(2) will

D. (1) AC
(2) will NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in the event the EDG requires emergency stopping, the DC fuel oil priming pump is required to be stopped. This can be performed by depressing both ENGINE STOP push buttons simultaneously on the 3A Engine Control Cabinet to initiate the shutdown sequence. By emergency stopping the EDG, the DC fuel oil priming pump will be stopped.

B **CORRECT**: (*See attached*) The power supply for '3A' EDG electric fuel oil priming pump is on 0-BKR-254-000A/04, 3A DG DC OIL PUMPS CONTROL PANEL, 125VDC DSL SYS BATTERY BOARD A.
Second part, the EDG normally operates with the engine-driven fuel oil priming pump providing fuel and the DC fuel oil priming pump running as a backup. The DC priming pump is manually started for EDG manual slow starts and manual fast starts and can be stopped with local EDG stop signal with DC logic breaker open. In the event the EDG requires emergency stopping, the DC fuel oil priming pump is required to be stopped. This can be performed by depressing both ENGINE STOP push buttons simultaneously on the 3A Engine Control Cabinet to initiate the shutdown sequence. However if the power is lost to the 3A DC priming fuel oil pump, '3A' EDG will not automatically trip.

C INCORRECT: First part is incorrect but plausible in that 480V Shutdown Board 3A is the normal AC power supply to 480V Diesel Auxiliary Board 3EA. 480V Diesel Auxiliary Boards provide power to each respective Diesel's AC soakback pump and AC circulation pump.
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 Emergency Diesel Generator fuel oil pump electrical power supplies. 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): 3-OI-82, Rev. 166 (Attach if not previously provided)
0-OI-57B, Rev. 213
3-OI-82, Att. 3A, Rev. 101
OPL171.038, Rev. 27U3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.038, Obj. 11 (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 0-OI-82/Att. 3A:

BFN Unit 0	Attachment 3A Standby Diesel Generator A Electrical Lineup Checklist	0-OI-82/ATT-3A Rev. 0101 Page 6 of 8

4.0 ATTACHMENT DATA (continued)

Performed On: _____

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
Diesel Generator Bldg - 480V Diesel Aux Bd. B - El 583'			
9A	0-BKR-086-0668A/AC DG A LEFT BANK AIR COMP MTR BKR	ON	____ ____
9B	0-BKR-030-0065 DIESEL GENERATOR A EXH FAN B	ON	____ ____
9F	0-BKR-254-0000AB DSL GEN A BAT CHGR B	ON	____ ____
Diesel Generator Bldg - Central Diesel Information Center - El 583'			
	48V POWER DISCONNECT DGA	ON	____ ____
Diesel Generator Room A - 0-BDGG-254-0000A, 125V DC DSL SYS BAT BOARD A			
0-BKR-254-000A/01A	NORM SUPPLY BAT CHARGER A	ON	____ ____
0-BKR-254-000A/01B	ALT SUPPLY BAT CHARGER B	OFF	____ ____
0-BKR-254-000A/02	FIELD FLASHING	ON	____ ____
0-BKR-254-000A/03	A Electrical Control Cabinet FDR Breaker	ON	____ ____
0-BKR-254-000A/04	DG A DC OIL PUMPS CONTROL PANEL	ON	____ ____
0-BKR-254-000A/05	DSL GEN A PROTECTION RELAY PANEL	ON	____ ____
0-BKR-254-000A/06	DSL GEN A LOGIC RELAY PANEL (LOGIC BREAKER)	ON	____ ____

Excerpts from 3-OI-82:

7.5 Emergency Shutdown at Diesel Engine

NOTES

1) This section should only be performed during an emergency condition which requires the Diesel Generator to be stopped immediately.
2) When time permits, all manipulations of the Diesel Generator Logic Breaker are required to be logged in the Narrative Log.

CAUTION

If diesel engine speed rises to greater than 1035 RPM, the Diesel Generator room should be evacuated immediately due to the potential for personnel injury.

[1] **PULL OUT and HOLD** the Injector Control Lever in the No Fuel position until the diesel engine comes to a complete stop.

[2] **IF** the diesel engine does **NOT** stop, **THEN**

PERFORM the following:

[2.1] **PLACE** 3-BKR-254-000A(B,C,D)/06, DSL GEN 3A(3B,3C,3D) LOGIC RELAY PANEL (LOGIC BREAKER), in OFF on 3-BDGG-254-0003A(3B,3C,3D), 125 VDC DSL SYS BAT BOARD A(B,C,D).

[2.2] **DEPRESS both** ENGINE STOP push-buttons simultaneously on the Engine Control Cabinet to initiate the shutdown sequence (this will also stop the Priming Fuel Pump).

[2.3] **CLOSE** ENGINE DRIVEN FUEL PMP SUCT, 0-SHV-018-0664-3A (3B, 3C, 3D).

[2.4] **PLACE** both DG 3A(3B,3C,3D) START CIRCUIT 1 CONT POWER BKR, 3-BKR-082-003A(B,C,D) 35W1 and DG 3A(3B,3C,3D) START CIRCUIT 2 CONT POWER BKR, 3-BKR-082-003A (3B,3C,3D) 35W2 in the OFF position at the Electrical Control Cabinet.

[3] **ENSURE** DG 3A (3B, 3C, 3D) Output Bkr 1838 (1842, 1832, 1836) is OPEN.

5.5 Manual Fast Start at the Diesel Engine Control Cabinet (continued)

[4] **PERFORM** DIESEL GENERATOR ROLLING per Section 5.7.

NOTE

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).

[5] **ENSURE** the presence of EECW cooling water flow to the diesel.

[6] **DEPRESS** the FUEL PRIMING PUMP push-button, 3-HS-18-65A(B,C,D), until fuel pressure exceeds 20 psig as indicated on PRIMING FUEL PRESS, 3-PI-18-66A(B,C,D).

[7] **DEPRESS** the DIESEL GENERATOR 3A(3B,3C,3D) LOCAL FAST START PUSH BUTTON.

Diesel	Handswitch Name	Handswitch No.
3A	ENGINE FAST START	3-HS-082-0544
3B	ENGINE FAST START	3-HS-082-0583
3C	ENGINE FAST START	3-HS-082-0507
3D	ENGINE FAST START	3-HS-082-0520

[8] **DIRECT** the operator in the Diesel Generator room to **CHECK** the following:

A. Lube oil pressure rises to greater than 20 psig as indicated on the following.

- MAIN BEARINGS LUBE OIL PRESS, 3-PI-82-10A(B,C,D)
- DG-3A(3B,3C,3D) TURBOCHARGER COMP BEARING LUBE OIL PRESS INDR 3-PI-082-1001A(B,C,D)

[9] **CHECK** Diesel Generator starts and RPM stabilizes at a nominal 900 RPM.

5.6 Manual Slow Start at Diesel Engine Control Cabinet (continued)

NOTE

All manipulations of the Diesel Generator Logic Breaker are required to be logged in the Narrative Log.

[1.5] **IF** Diesel Generator 3A(3B,3C,3D), does not need to be rolled **OR** has been operated or rolled in the last 24 hours, **THEN**

OPEN 3-BKR-254-000A(B,C,D)/06, DSL GEN 3A(3B,3C,3D), LOGIC RELAY PANEL (LOGIC BREAKER), **AND DO NOT PERFORM** Step 5.6[2].

[2] **PERFORM** DIESEL GENERATOR ROLLING per Section 5.7.

NOTE

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] **ENSURE** the presence of EECW cooling water flow to the diesel.

[4] **DEPRESS** the FUEL PRIMING PUMP push-button, 3-HS-18-65A(B,C,D), until fuel pressure exceeds 20 psig as indicated on PRIMING FUEL PRESS, 3-PI-18-66A(B,C,D).

[5] **DEPRESS** the DG-3A(3B,3C,3D) IDLE START, 3-HS-082-003A(B,C,D)/8, **AND CHECK** the following:

A. Diesel engine starts, accelerates and stabilizes at an engine speed of 440 RPM to 460 RPM.

B. Lube oil pressure rises to greater than 20 psig as indicated on the following:

- MAIN BEARINGS LUBE OIL PRESS, 3-PI-82-10A(B,C,D).
- DG-3A(3B,3C,3D) TURBOCHARGER COMP BEARING LUBE OIL PRESS INDR 3-PI-082-1001A(B,C,D).

7.5 Emergency Shutdown at Diesel Engine

NOTES

1) This section should only be performed during an emergency condition which requires the Diesel Generator to be stopped immediately.
2) When time permits, all manipulations of the Diesel Generator Logic Breaker are required to be logged in the Narrative Log.

CAUTION

If diesel engine speed rises to greater than 1035 RPM, the Diesel Generator room should be evacuated immediately due to the potential for personnel injury.

[1] **PULL OUT and HOLD** the Injector Control Lever in the No Fuel position until the diesel engine comes to a complete stop.

[2] **IF** the diesel engine does **NOT** stop, **THEN**

PERFORM the following:

[2.1] **PLACE** 3-BKR-254-000A(B,C,D)/06, DSL GEN 3A(3B,3C,3D) LOGIC RELAY PANEL (LOGIC BREAKER), in OFF on 3-BDGG-254-0003A(3B,3C,3D), 125 VDC DSL SYS BAT BOARD A(B,C,D).

[2.2] **DEPRESS both** ENGINE STOP push-buttons simultaneously on the Engine Control Cabinet to initiate the shutdown sequence (this will also stop the Priming Fuel Pump).

[2.3] **CLOSE** ENGINE DRIVEN FUEL PMP SUCT, 0-SHV-018-0664-3A (3B, 3C, 3D).

[2.4] **PLACE** both DG 3A(3B,3C,3D) START CIRCUIT 1 CONT POWER BKR, 3-BKR-082-003A(B,C,D) 35W1 and DG 3A(3B,3C,3D) START CIRCUIT 2 CONT POWER BKR, 3-BKR-082-003A (3B,3C,3D) 35W2 in the OFF position at the Electrical Control Cabinet.

[3] **ENSURE** DG 3A (3B, 3C, 3D) Output Bkr 1838 (1842, 1832, 1836) is OPEN.

Excerpt from 0-OI-57B: Supports Distractors C1, D1

Attachment 1
(Page 9 of 9)

Auxiliary Power Supplies and Bus Transfer Schemes

15	480V Diesel Auxiliary Boards			
	A. 480V Diesel Aux. BD A	4kV Shutdown Board A (Via TDA)	4kV Shutdown Board B (Via TDE)	Transfer from normal to alternate source is manual. Interlocks prevent transferring a fault from one source to another and paralleling sources.
	B. 480V Diesel Aux. BD B	4kV Shutdown Board D (Via TDB)	4kV Shutdown Board B (Via TDE)	
	C. 480V Diesel Aux. BD 3EA	480V Shutdown Board 3A	480V Shutdown Board 3B	
	D. 480V Diesel Aux. BD 3EB	480V Shutdown Board 3B	480V Shutdown Board 3A	
16	480V Offgas MCC A	4kV Common Board A (Via TDGA)	4kV Common Board B (Via TDGA)	Transfer from normal to alternate source is manual. Mechanically interlocked.
17	480V Offgas MCC B	4kV Common Board B (Via TDGB)	4kV Common Board A (Via TDGA)	
18	480V Cooling Water MOG. BD and Aux. Raw Cooling Water Pumping Station	480V Offgas MCC A	480V Offgas MCC B	Automatic transfer from normal to alternate source is initiated by time-undervoltage on the normal source. Return to normal source is accomplished by manual transfer only.
19	480V HVAC BOARDS			
	A. 480V HVAC BOARD B	4KV SHUTDOWN BD 3ED	480V SHUTDOWN BD 3B	Transfer from normal to alternate source is manual. Interlocks are provided to prevent manual transferring to a faulted board and to prevent paralleling two sources

Excerpts from OPL171.038 Lesson Plan:

OPL171.038 Diesel Generator and Auxiliary Power System Rev 27 U3

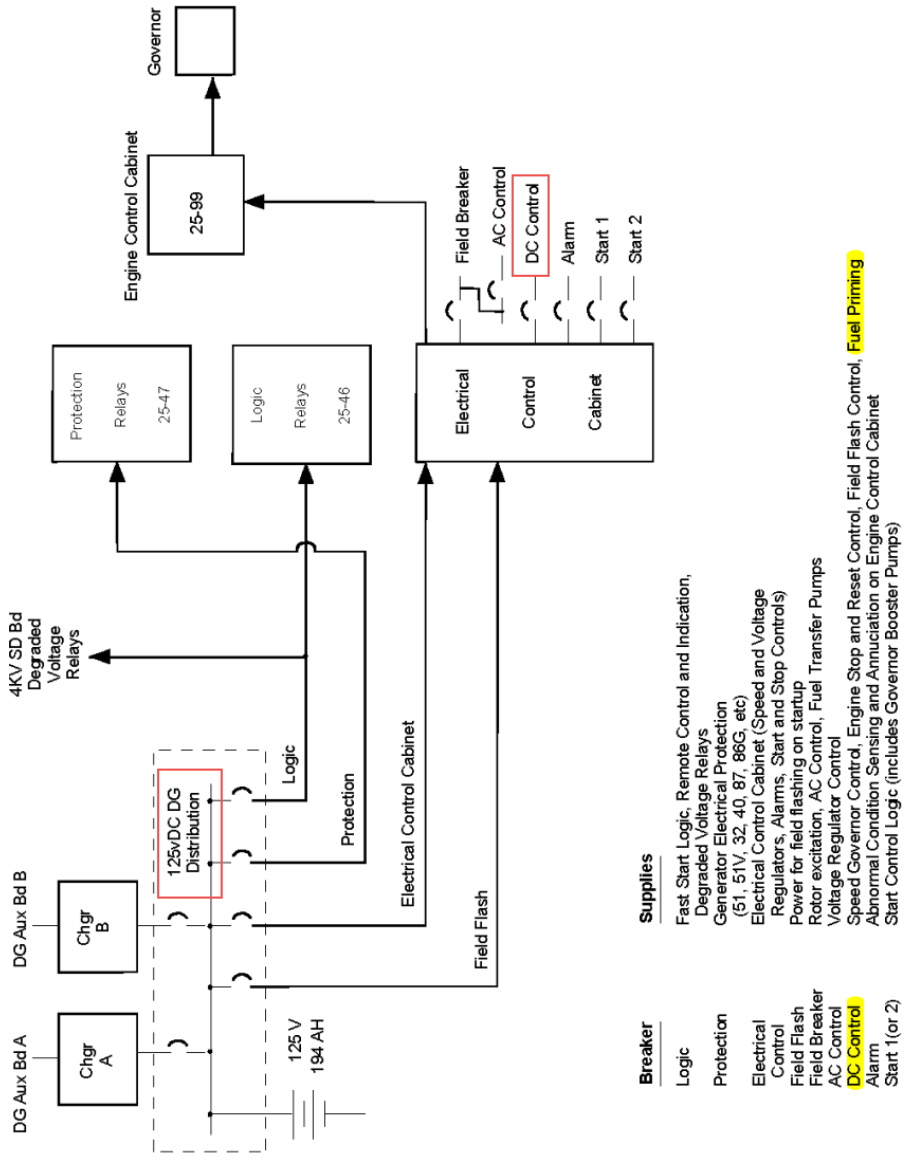


Fig.5: DIESEL GENERATOR 125VDC DISTRIBUTION

OPL171.038 Diesel Generator and Auxiliary Power System Rev 27 U3

Outline of Instruction	Instructor Notes and Methods (optional)
system via a 30-psig check valve and keeps the engine in a constant state of readiness for an immediate start	
(11) Pressure switches stop/start the circulation and soak back pumps when the diesel running and bypass the existing alarms to prevent nuisance alarms for the pumps when they are desired to be turned off. Pumps stop at 72 psig rising and start ~44 psig lowering	
b) Fuel oil system	
(1) During DG fast start, an electric priming pump starts and supplies fuel oil to the injection subsystem. The electric priming pump is manually started for slow start.	FIG 2 ILT OBJ 11
(2) Normally the engine-driven fuel oil subsystem supplies the injectors, with the electrically driven one as backup. Both subsystems operate simultaneously with the electrically driven system bypassing the engine fuel injectors and returning fuel to the day tank.	LOR OBJ 9 NLOR OBJ 3 NLO OBJ 3
(3) The 550-gallon day tank provides fuel for~ 2 1/2 hours at full load (260-500 gal) with fuel pump pressure> 57 psig	
(4) Fuel must be transferred from the 40,000-gallon 7-day diesel storage tank with the diesel oil transfer pump to continue operation. This is accomplished automatically by level switches on the day tank. The 7-day storage tank capacity is required by Technical Specifications. (ARP low level alarm = 83%) Tech. Spec. requires 35,280 gallons per 7-day tank (~81% per OI-18 P&L.).	
(5) An emergency fuel cut-off valve is provided in event of a fire or other emergency. It should not be used indiscriminately because the fuel oil also provides cooling to the injectors, and they may be damaged. This valve will not stop the diesel unless the electric driven priming pump has been stopped.	
(6) A transfer pump is located at the fuel oil storage tanks which can supply fuel oil from the two 71,000-gallon auxiliary boiler fuel oil tanks to the 7-day storage tanks.	ILT OBJ.10 LOR OBJ.8
(7) Another transfer pump is in each diesel building which can transfer fuel oil from any seven-day tank in the building to another seven-day tank, recirculate back to the same seven-day tank, or pump fuel oil back to the yard storage tanks.	
c) Starting air system [TS/R low air press >165 psig]	
(1) Two fully redundant air-start systems per diesel are provided, either of which is capable of diesel starting. When air compressor is turned off, it brings in a Trouble alarm	NLOR OBJ. 4 NLO OBJ. 4
(2) Air, supplied by 5 twenty cubic foot air receiver units for	Fig. 3

OPL171.038 Diesel Generator and Auxiliary Power System Rev 27 U3

Outline of Instruction	Instructor Notes and Methods (optional)
e) Diesel generator control power	
(a) 125VDC battery system consisting of a 60-cell battery with two chargers per diesel	
(b) 194 amp-hour battery in DG room	Tech. Specs. DG Battery
(c) Battery charger A outside DG room is powered from DG Aux BD A	
(d) Battery charger B inside DG room is powered from DG Aux BD B	NLOR OBJ. 8 NLO OBJ. 8
(2) 125VDC loads	Fig. 5
(a) Generator Protective Relay Cabinet PNL 25-47	
(b) Fast Start Logic and Control PNL 25-46	
(c) Degraded Voltage Relays	ILT OBJ 14 LOR OBJ 12
(d) Engine Start circuits	
(e) Engine Stop, Reset, and Alarm circuits	
(f) Speed governor control	
(g) Governor booster pumps	
(h) Fuel priming pump	
(i) Field flashing	
(j) DC Soak back lube oil pump	
(k) DC Circulating lube oil pump	
f)Turbocharger	
(1) The diesel is turbocharged to force more air into the engine and raise its power	
(2) The turbocharger is driven by the engine gear train at low loads and by exhaust gas at high loads (>70%). Mechanically from the DG crankshaft gear through an assembly of gears (Sun gear, ring gear, & planetary gears).	NLOR OBJ. 7 NLO OBJ. 7
(3) An "Overrun" clutch is supplied to engage/disengage gear assembly driven by engine shaft until turbo/fan "overspeeds" the gear assembly to clutch the mechanism into exhaust drive mode	
(4) An after-cooler reduces the temperature of the air, increasing its density and allowing more oxygen to be forced into the engine	
(5) The aftercooler is cooled by engine coolant circulating through the heat exchanger	
(6) Provides "Scavenging Air" to cylinder intake points via crankcase air space	
3. Instrumentation	
a) Control Room (For Each Diesel and Diesel Control Panels) Panel 9-23	
(1) Synchronizing equipment	

OPL171.038 Diesel Generator and Auxiliary Power System Rev 27 U3

Outline of Instruction	Instructor Notes and Methods (optional)
lube oil and circulating oil systems The main oil pump, piston oil pump, and scavenging oil pump are engine-driven. The soak back and circulating oil pumps are driven by electrical motors with one AC motor driven pump and a second DC motor driven pump.	
(5) The main lubricating oil subsystem supplies oil at 80 to 90 psig to the various moving parts of the engine. The main lube oil pump takes oil from the strainer housing and discharges it to the main oil manifold, for the main bearings, gear train, turbocharger, cam shaft and other moving part	ILT OBJ 2
(6) The piston cooling oil subsystem supplies oil for piston cooling and lubrication of the piston pin bearing surface. The pump receives its oil from a common suction with the main lube oil pump and delivers oil to the two piston oil cooling manifolds, which supply the pistons.	
(7) The scavenging oil subsystem supplies the other subsystems with cooled, filtered oil. The scavenging oil pump takes suction from the engine oil pan and discharges the oil through the oil filter and the oil cooler to the strainer housing to supply the main oil pump and the piston oil pump.	DCN 71541 DCN 71425
(8) The soak back and circulating oil system keep the diesel pre-lubed and warm for standby conditions. One AC electric soak back pumps run continuously to pre-lubricate the turbocharger bearing area One AC electric circulating oil pump maintains the engine lube oil warm when the diesel is in standby by circulating oil from the lube oil cooler to the main oil gallery.	SOER 83-1
(9) . Each engine has one 3 gpm AC soak back pumps (ACSP) and one 6 gpm AC circulation pump (ACCP). The ACSP and ACCP are normally in service with DC pumps (DCSP and DCCP) available if needed.	OI-82 & ARP 1/2-ARP-9-23 A, B, C, D window 9 DIESEL GEN ABCD FUEL PRESSURE ABNORMAL or (0-ARP-25-41)
(a) A diesel ACSP and ACCP are supplied from the same 480V Diesel Aux. board.	
(b) DCSP and DCCP are supplied from the 125V battery for its respective diesel. The DC pumps The DCSP starts at 15psig decreasing of AC pump discharge pressure.	
(c) Pressure switches at the discharge of the AC pumps monitor for 15 psig decreasing pressure. DCSP starts immediately at 15 psig, the DCCP starts after a 45 second time delay. The time delay prevents overloading the 125VDC system by starting both DC motors simultaneously.	
(10) The soak back pump removes residual heat from the turbocharger bearing area upon shutdown of the engine. AC circulation pump circulates warm oil through the oil	

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

Examination Outline Cross-reference:	Level	RO	SRO
295025 (EPE 2) High Reactor Pressure / 3	Tier #	1	------
EK1.07 (10CFR 55.41.8)	Group #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to HIGH REACTOR PRESSURE:	K/A #	295025EK1.07	
• Pressure control strategies	Importance Rating	4.2	------

Proposed Question: **# 49**

Unit 2 is operating at 100% RTP when the following conditions occur:

- A Loss of Offsite Power
- ATWS actions have been completed
- Reactor Power is 12%
- **NO** other Operator actions have been taken

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

In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, a **MINIMUM** of two (2) **(1)** are required to be OPEN to control Reactor Pressure in the normally required band.

The Reactor Pressure SCRAM initiation signal setpoint **(2)** be bypassed.

A. (1) Main Turbine Bypass Valves
(2) can

B. (1) Main Turbine Bypass Valves
(2) can NOT

C. (1) Main Steam Relief Valves
(2) can

D. (1) Main Steam Relief Valves
(2) can NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that Main Turbine Bypass Valves can be used for Reactor Pressure Control if the MSIVs are OPEN, however the candidate is required to recognize that upon a Loss of Offsite Power occurring, all MSIVs CLOSE. Regardless of the Bypass Valve capacity, the candidate must realize they are not available. Second part is incorrect but plausible in that 2-OI-99, Reactor Protection System (RPS), lists all of the Reactor SCRAM initiation signals with numerous ones capable of being bypassed.

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 BFN-ODM-4.20, one Main Steam Relief Valve (MSRV) is equivalent to approximately 5 – 7% Reactor Power near rated Reactor Pressure. With the given ATWS Reactor Power remaining at 12%, a minimum of 2 MSRVs are required to be OPEN for Reactor Pressure Control to be in the normally required band of 800-1000 psig. Second part, in accordance with 2-OI-99, the Reactor Pressure SCRAM initiation signal setpoint can NOT be bypassed.

RO Level Justification: Tests the candidate's knowledge of the operational implications and/or cause and effect relationships of pressure control strategies 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-AOI-100-1, Rev. 123 (Attach if not previously provided)
2-OI-99, Rev. 101
0-TI-394, Rev. 14
BFN-ODM-4.20, Rev. 9
OPL171.009, Rev. 17U2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009 Obj. 10 (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	

Excerpt from 2-AOI-100-1:

BFN Unit 2	Reactor Scram	2-AOI-100-1 Rev. 0123 Page 13 of 72

4.2 Subsequent Actions (continued)

NOTE

Suppression Pool cooling should be placed in service as soon as practicable following MSRV, HPCI and / or RCIC operation regardless of indicated Suppression Pool temperatures to verify that thermal stratification does not exist.

[12] **MONITOR** and **CONTROL** RPV pressure to keep below 1073 psig and stable, or as directed by Unit SRO.

[12.1] **IF** RPV pressure is lowering rapidly, **THEN**

CLOSE MSIVs. (Otherwise N/A)

[12.2] **IF** MSRVs are cycling and bypass valves are available, **THEN**

MANUALLY OPEN MSRVs on Panel 2-9-3 to maintain RPV Pressure below 1073 psig while maximizing use of the bypass valves. (Otherwise N/A)

[12.3] **IF** MSRVs are cycling and bypass valves are **NOT** available, **THEN**

MANUALLY OPEN MSRVs on Panel 2-9-3 until RPV pressure is controlled between 800 and 1000 psig. (Otherwise N/A)

[13] **IF** any PCIS isolation signal is received, **THEN**

ENSURE PCIS isolations using any of the following: (Otherwise N/A)

- Containment Isolation Status System on Panel 2-9-4
- PCIS Mimic and individual control switch indications
- ICS
- 2-OI-64

[14] **ENSURE** all available drywell cooling coils and blowers in service.

Excerpt from 2-OI-99:

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0101 Page 101 of 112

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 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 (9.9 Unit 2)	2,200 (2,400 unit 2)	1,100,000 (1,140,000 unit 2)
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

Excerpts from OPL171.009 Lesson Plan:

5. Main Steam Isolation Valves

a. Each steam line has two isolation valves - one inside and one outside the primary containment.

b. Purpose

(1) Prevent exceeding radiation release rates in excess of *10CFR50.67* guidelines in the event of a steam line break outside of primary containment.

(2) Limits inventory loss during a steam line break accident. Helps maintain clad integrity by preventing core from uncovering.

(3) Prevent exceeding radiation release rates in excess of *10CFR50.67* guidelines by closing the primary containment barrier in the event of a steam leak inside of primary containment.

c. Valve description — Figure-6

(1) Major components

(a) air cylinder
(b) hydraulic dash pot
(c) speed control valve
(d) closing springs
(e) valve seat
(f) pilot and pilot valve seat

(2) Open valve operation — Figure 5

(a) Air is supplied to the underside of the air operator piston which causes the piston and stem to move upward against spring tension.
(b) Stem lifts pilot valve first - 1/2 inch
(c) Equalizes upstream and downstream steam pressures through the balancing orifice.
(d) Upper portion of pilot makes contact with main valve body, and lifts the valve off its seat to the full open position.
(e) Valves are designed to open against 200 psid differential pressure (≤50 psid by procedure)

OPL171.009 Main Steam System Rev.# 17U2

(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.

(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.

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

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

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

Examination Outline Cross-reference:	Level	RO	SRO
400000 (SF8 CCW) Component Cooling Water **291004 K1.05** (10CFR 55.41.7) Pumps • Discuss relationships among head, flow, and power, as related to pump speed	Tier #	2	------
	Group #	1	------
	K/A #	291004K1.05	
	Importance Rating	2.9	------

Proposed Question: **# 50**

Which **ONE** of the following completes the statements below?

The RBCCW Pumps operate where _____.

A. Pump Volumetric Flow Rate/capacity ($\dot{V}$ in gpm) is directly proportional to the Pump Speed (n).

B. Pump Head/discharge head (H_P in psid or feet) is directly proportional to the cube (n^3) of Pump Speed (n).

C. Power required by the pump motor (P in kW) is directly proportional to square (n^2) of Pump Speed (n).

D. Current draw of the pump (I in amps) is directly proportional to Pump Speed (n).

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In a centrifugal pump like RBCCW Pumps, Pump Volumetric Flow Rate/capacity ($\dot{V}$ in gpm) is directly proportional to the Pump Speed (n).

B INCORRECT: Incorrect but plausible in that, in a centrifugal pump, Pump Head/discharge head (H_P in psid or feet) is directly proportional to the square (n^2) of Pump Speed (n).

C INCORRECT: Incorrect but plausible in that, in a centrifugal pump, Power required by the pump motor (P in kW) is directly proportional to the cube (n^3) of Pump Speed (n).

D INCORRECT: Incorrect but plausible in that, in a centrifugal pump, Current draw of the pump (I in amps) is directly proportional to the cube (n^3) of Pump Speed (n).

RO Level Justification: Tests the candidate's knowledge of relationships among head, flow, and power, as related to pump speed pertaining to component cooling water. This question is rated as Memory due to the requirement to strictly recall general fundamental facts.

Technical Reference(s): GFE Components-Pumps Ch. 2, Rev. 4 (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: GFE Components-Pumps Ch. 2, Obj. 11 (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 GFE Components-Pumps Chapter 2:

cylinder by the number of discharge strokes the pump makes in an hour.

A positive displacement pump is started with both the suction and discharge valves open. Since this type of pump has almost no slip, the discharge valve must be open as the pump is started. Starting or operating a positive displacement pump with the discharge valve closed can result in severe damage, including potential rupture of the pump casing and discharge piping. A blocked discharge line has the same potential for damage, and should be verified as clear before starting the pump.

CENTRIFUGAL PUMP LAWS

Centrifugal pumps generally obey what are known as the pump laws. These laws apply only to centrifugal pumps in closed systems, the laws do not apply to open systems discussed earlier in the calculation of the suction lift, suction head, and discharge head equations. These laws state that the volume flow rate or capacity is directly proportional to the pump speed, the discharge head is directly proportional to the square of the pump speed, and the power required by the pump motor is directly proportional to the cube of the pump speed. These laws are summarized in the following equations:

$$\dot{V} \propto N$$

$$H_p \propto N^2$$

$$P \propto N^3$$

Where:

$\dot{V}$ = pump volumetric flow rate (gpm)

$\propto$ = proportional

N = pump speed (rpm)

H_p = pump discharge head (psi)

P = pump power (kW)

Equation 2-16

Excerpt from GFE Components-Pumps:

4.6 Pump Laws

Centrifugal pumps generally obey what are known as the pump laws. These laws state the following:

- Flow rate or capacity is directly proportional to the pump speed;
- Discharge head is directly proportional to the square of the pump speed;
- Power required by the pump motor is directly proportional to the cube of the pump speed.
- Current draw of the pump is directly proportional to the cube of the pump speed.

These laws are summarized in the following proportionalities:

$$\dot{V} \propto n$$

$$H_p \propto n^2$$

$$P \propto n^3$$

$$I \propto n^3$$

Where:
N = speed of pump impeller (rpm)
V = volumetric flow rate of pump (gpm)
H_p = head developed (psid or feet)
P = pump power (kW)
I = current

$$\dot{V}_1 \frac{n_2}{n_1} = \dot{V}_2$$

$$H_{p1} \frac{n_2}{n_1}^2 = H_{p2}$$

$$P_1 \frac{n_2}{n_1}^3 = P_2$$

$$I_1 \frac{n_2}{n_1}^3 = I_2$$

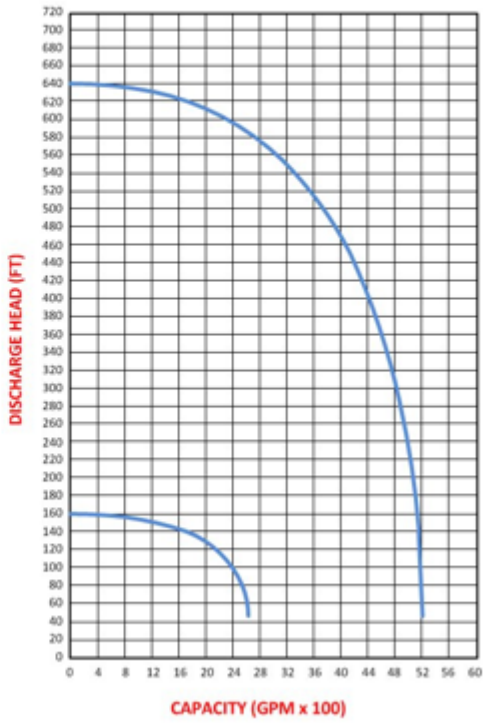


Figure 4-20 – Doubling Pump Speed

Knowledge 1.05
Discuss relationships among head, flow, speed, and power

4.6.1 Example

A multi-speed centrifugal pump is operating at 1800 rpm, providing a flow of 400 gpm with a pump head of 20 psid. If the pump speed is increased to 3600 rpm, the new pump head will be…

A. 160 psid

B. 80 psid

C. 60 psid

D. 40 psid

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

Examination Outline Cross-reference:	Level	RO	SRO
510000 (SF4 SWS*) Service Water (Normal and Emergency) **A1.02** (10CFR 55.41.5)	Tier #	2	------
Ability to predict and/or monitor changes in parameters associated with operation of the SERVICE WATER SYSTEM including: • Temperature control valve position	Group #	1	------
	K/A #	510000A1.02	
	Importance Rating	3.3	------

Proposed Question: **# 51**

Unit 1 is operating at 100% RTP when a Loss of Control Air occurs.

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

The Raw Cooling Water (RCW) six (6) inch temperature control valves, 1-TCV-24-80A, 85A, 90A, RBCCW HEAT EXCHANGER, will fail **(1)**.

If RCW Pressure lowers to 15 psig supporting RBCCW Heat Exchangers, RHRSW/EECW Pumps A3, B3, C3, and D3 **(2)** receive an automatic start signal.

A. (1) OPEN
(2) will

B. (1) OPEN
(2) will NOT

C. (1) CLOSED
(2) will

D. (1) CLOSED
(2) will NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 1-OI-70, on loss of Control Air, the 6-inch valves, 1-TCV-024-0080A, 0085A, 0090A fail open. Second part, in accordance with 1-OI-24, RHRSW Pumps A3, B3, C3, and D3 will automatically start when Raw Cooling Water pressure at RBCCW heat exchanger is less than15 psig.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that BFN always maintains two EECW pumps in service, one supplying the North EECW Header and one supplying the South EECW Header, therefore, EECW Header pressure may be confused as readily able to provide backup to the RCW supply and never reach a low pressure setpoint of 15 psig.

C INCORRECT: First part is incorrect but plausible in that, in accordance with 1-OI-70, on loss of Control Air, the 10-inch valves, 1-TCV-024-0080B, 0085B, 0090B fail closed. 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 predict and/or monitor changes in parameters associated with operation of the Service Water System including temperature control valve positions. This question is rated as Memory due to the requirement to strictly recall procedural facts.

Technical Reference(s): 1-AOI-32-2, Rev. 9 (Attach if not previously provided)
1-OI-70, Rev. 64
1-OI-24, Rev. 65

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.047, 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 **X**
Comprehension or Analysis

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

Comments:

Excerpt from 1-AOI-32-2:

Attachment 1
(Page 3 of 7)

Expected System Responses

4.0 HEATER DRAINS AND VENTS (continued)

J. Moisture Separator level control reservoir drains to Number 2 High Pressure Heaters shell side fail CLOSED.

K. CONDENSATE INJECTION ISOLATION, 1-FCV-002-0144, to Moisture Separator Drain System fails CLOSED.

5.0 RAW SERVICE WATER

A. RSW STRG TNK ISOLATION, 0-FCV-25-32, fails CLOSED. A High Pressure Fire Pump should be started immediately to maintain fire system pressure.

6.0 RCW

A. All RCW temperature control valves fail OPEN EXCEPT for 1-TCV-24-80B and 1-TCV-24-85B on 1A and 1B RBCCW heat exchangers and 1-TCV-024-0075B on the Main Turbine Oil Coolers (4" line) which fail CLOSED.

7.0 HIGH PRESSURE FIRE PROTECTION

HP FIRE PROT BACK PRESSURE 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

SERVICE AIR TO CONTROL AIR ISOLATION VALVE, 0-PCV-33-1, opens at 85 psig control air pressure but will fail CLOSED on loss of control air at 30 psig.

9.0 CONTROL AIR

A. U-1 TO U-2 CONT AIR CROSSTIE, 1-PCV-032-3901, will CLOSE when Control Air Header pressure reaches 65 psig lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

B. UNIT 2 TO UNIT 3 CONTROL AIR CROSSTIE, 2-PCV-032-3901, will CLOSE when Control Air Header pressure reaches 65 psig lowering at the valve. The valve will reopen automatically when Control Air Header pressure rises above 85 psig at the valve.

Excerpt from 1-OI-70:

BFN Unit 1	**Reactor Building Closed Cooling Water System**	**1-OI-70 Rev. 0064 Page 9 of 86**

3.0 PRECAUTIONS AND LIMITATIONS (continued)

K. Two RBCCW temperature control valves 6 inches and 10 inches and a manual bypass are located on the RCW discharge line of each RBCCW heat exchanger to automatically maintain RBCCW temperature. The Controllers are operated in accordance with Attachment 1 and are normally set at 27% (80°F). On loss of control air, the 6 inch valve (1-TCV-024-0080A, 0085A, 0090A (spare)) fail open while the 10 inch valve (1-TCV-024-0080B, 0085B, 0090B (spare)) fail closed. The TCVs are sequenced so that the 6 inch valve opens first on rising demand, and when it is full open, the 10 inch valve will begin to open. Caution should be exercised when operating the TCVs to change RBCCW temperature. This operation has resulted in the auto start of the EECW pumps due to opening the TCVs to the point of dropping RCW to less than 15 psig. The TCVs may be set in a predetermined position from the Backup Control Center via override control switches, HS-24-80A(B) 85A(B) and 90A(B), to provide minimum cooling for critical equipment.

L. When in cold shutdown, RBCCW heat exchanger (HX) outlet temperature be reduced to maximize fuel pool cooling heat removal capability. Lower RBCCW temperatures may result in SFSP temperatures below design analysis if inadequate decay heat exists.

M. Failure to maintain RBCCW HX outlet temperature at or above the following temperature limits may result in the below component damage:

1. 50°F, RBCCW piping component integrity damage.

2. 60°F, RWCU pump bearing damage due to improper oil viscosity. While 60°F is the optimum lower limit, Site Engineering evaluations have indicated that with RBCCW temperatures of 45°F will not result in RWCU Pump oil temperatures such that damage could occur to the bearings for a running RWCU Pump. However, the limitation of 50°F for the RBCCW piping is still binding.

3. 70°F, Reactor recirculation pump motor bearing damage due to improper oil viscosity.

Excerpt from 1-OI-24:

BFN Unit 1	Raw Cooling Water System	1-OI-24 Rev. 0065 Page 7 of 45

3.0 PRECAUTIONS AND LIMITATIONS

3.1 RCW System

A. Coordination of RCW operation between Units 1, 2, and 3 is required to ensure proper system operation.

B. To provide a source of strained water to the RCW Pumps, a Unit 1 or 2 CCW pump discharge valve must be open.

C. [NRC/C] When the RCW 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. [NRC LER 259/88010 & 296/88006]

D. The number of RCW Pumps in service should be equally divided between Unit 3 and Units 1 and 2 to ensure adequate suction header pressure to the RCW Pumps.

E. RHRSW Pumps A3, B3, C3, and D3 as well as A1, B1, C1, and D1, when lined up for EECW operation, will auto-start when either:

1. Low Raw Cooling Water header pressure at control air compressor (less than 30 psig).

2. Low Raw Cooling Water pressure at RBCCW heat exchanger (less than 15 psig).

F. To avoid RCW Pump cavitation, no more than one RCW Pump per unit may be started until a CCW pump is operating supplying RCW Pump suction.

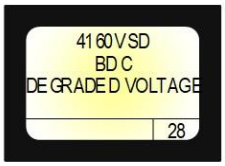
G. Raw Cooling Water Pumps are equipped with Rains-Flo type seals which do not require the seal water to be valved in service. Seal water valves to these pumps are to remain closed.

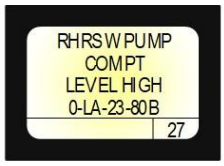
Form 4.2-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
510000 (SF4 SWS*) Service Water (Normal and Emergency)	Tier #	2	------
K5.03 (10CFR 55.41.4/7)	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 #	510000K5.03	
• Flood prevention	Importance Rating	3.1	------

Proposed Question: **# 52**

Unit 1 is operating at 100% RTP when the following occurs:





- 4160V SHUTDOWN BOARD 'C' DEGRADED VOLTAGE (1/2-9-23C, Window 28) alarms
- The supporting EDG fails to start
- RHRSW PUMP COMPARTMENT LEVEL HIGH (1-9-22A, Window 27) alarms
- AUO reports 'A' and 'B' sump pumps in RHRSW Pump Room 'C' will **NOT** operate
- **NO** Operator actions have been taken

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

(1) RHRSW/EECW Pumps in RHRSW Pump Room 'C' are now INOPERABLE.

(2) RHRSW/EECW Pumps have lost their associated power supply.

A. (1) **ONLY** C1, C2
(2) B2 and B3

B. (1) **ONLY** C1, C2
(2) C2 and C3

C. (1) C1, C2, **AND** C3
(2) B2 and B3

D. (1) C1, C2, **AND** C3
(2) C2 and C3

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the candidate may incorrectly conclude that only C1, C2 RHRSW pumps, powered from Unit 1/2 are inoperable given that that C3 RHRSW/EECW is powered from Unit 3. Second part is correct (*See C*).

B INCORRECT: First part is incorrect (*See A*). Second part is incorrect but plausible in that the A(D) 4kV Shutdown Boards power RHRSW pumps A1(D2) and A2(D3). The B(C) 4kV Shutdown Boards power RHRSW/EECW pumps C1(B2) and C2(B3).

C **CORRECT**: (*See attached*) RHRSW/EECW pumps C1, C2 and C3 are housed in RHRSW Pump Room 'C' With no sump pump capability to support the RHRSW/EECW pumps' operability, the RHRSW/EECW pumps are inoperable. Second part, the 4kV Shutdown Board 'C' power supply has been lost, which is the power supply to B2 and B3 RHRSW/EECW pumps.

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

RO Level Justification: Tests the candidate's knowledge of the operational implications or cause and effect relationships of flood prevention as it applies to the Service Water 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.

Technical Reference(s): 1/2-ARP-9-23C, Rev. 37 (Attach if not previously provided)
1-ARP-9-22A, Rev. 12
0-OI-67, Rev. 130
OPL171.046, Rev. 20

Proposed references to be provided to applicants during examination: 4160 SHUTDOWN BOARD 'C' DEGRADED VOLTAGE (1/2-9-23C, Window 28)
RHRSW PUMP COMPARTMENT LEVEL HIGH (1-9-22A, Window 27)

Learning Objective: OPL171.046, Obj. 7d (As available)

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

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**

Comments:

Excerpt from 1/2-ARP-9-23C:

BFN Unit 1 & 2	Panel 0-9-23-8, 0-XA-55-23C	1/2-ARP-9-23C Rev. 0037 Page 34 of 52

4160V SD
BD C
DEGRADED VOLTAGE
28

(Page 1 of 1)

Sensor/Trip Point:

0-27-211-000C/11D

A. Energized by relays 0-27-211-000C/25A,25B,25C a two out of three logic at 3920V lowering after ~4.3 sec delay.

or

0-27-211-000C/11C

B. Energized by relays 0-27-211-000C/11A & 11B at 0 Volts after ~1.5 sec delay.

Sensor Location:
4kV Shutdown Bd C
Elevation 621
Electric Bd Rm 2A

Probable Cause:
A. Supply voltage low.
B. Fuse failure.
C. Sensor malfunction.
D. Voltage Unbalance condition.

Automatic Action:
A. Diesel generator starts.
B. Approximately 2.9 seconds later, load shed logic relay picks up (2-211-3C).
C. Approximately 1.3 seconds after load shed logic, diesel generator breaker logic picks up (relay 2-211-4C).

Operator Action:
A. **ENSURE** unit in stable condition by checking:
1. Auto actions have occurred and diesel at rated frequency and voltage.
2. Diesel connects properly to assigned 4160V Shutdown Bd.
3. Diesel reaches proper loading when supplying equipment fed from board.

References: 0-45E765-24 0-45E765-8 0-45E724-3

Excerpt from 1-ARP-9-22A:

RHRSW PUMP COMPT LEVEL HIGH
0-LA-23-80B

27

(Page 1 of 1)

Sensor/Trip Point:

Any of the following compartment level switches:

LS-23-76B	565.5 elevation
LS-23-77B	
LS-23-78B	
LS-23-79B	

Sensor Location:

RHRSW Pumping Station

LS-23-76B	RHRSW Pump Room A
LS-23-77B	RHRSW Pump Room B
LS-23-78B	RHRSW Pump Room C
LS-23-79B	RHRSW Pump Room D

Probable Cause:

A. Inleakage due to any of the following:
- Rain.
- Air release valve failed open.
- Relief valve failure.
- Pipe break.

B. Sump pump or level switch failure.
C. Alarm level switch malfunction.
D. Loss of power to level switches (Pnl 1-9-9 bkr 327)
- Bkr 327 also powers sump pump level switches. (consider operability)

Automatic Action:

None

Operator Action:

A. **DISPATCH** personnel to PERFORM the following:
- **OBSERVE** compartment sump level to determine which sump is in alarm.
- **ENSURE** affected sump pump is running.
- **DETERMINE** source of inleakage.

B. **IF** system breach is the source of inleakage, **THEN** **SWAP** pumps to isolate leak in accordance with 0-OI-23.

References:

1-45E620-13-1 0-47E610-23-3 2-45E779-18

Excerpt from 0-OI-67:

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0130 Page 13 of 106

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Q. [NRC/C] RHRSW Pump Motor nameplate full load current is 53 amps, except for C3 RHRSW Pump Motor. The maximum allowable continuous running current is 61 amps (based on full load amps multiplied by the motor service factor of 1.15). C3 RHRSW Pump Motor nameplate full load current is 52 amps and the maximum allowable continuous running current is 59.8 amps (based on full load amps multiplied by the motor service factor of 1.15). [SLT 861087005]

R. To prevent possible RHRSW (EECW) pump motor damage, the following start limitations should be observed for the pump motor:

1. Two starts per hour from ambient temperature.
2. One start per hour from rated temperature.
3. When the motor has operated at no load for a minimum of 30 minutes, it may be restarted immediately.
4. The motor should be allowed to stand idle for at least 60 minutes before each additional restart is attempted.

S. The throttle valves that control the flow rate to each equipment cooler in the EECW system are preset and locked. Whenever the valve position is changed, it will have to be reset to the appropriate number of turns from full open in accordance with the valve checklist. Verify the outlet valve is open before setting the inlet valve for the DG engine coolers.

T. Whenever the discharge valve on a RHRSW (EECW) pump is closed, the power must be removed from that pump to prevent it from auto-start.

U. [NRC/C] When the EECW 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. [NRC LER 259/88010]

V. If chemical treatment is aligned to RHRSW or EECW, then chemical treatment to the RHRSW (EECW) pump discharge line should be secured prior to securing the associated RHRSW (EECW) pump in accordance with 0-OI-50.

W. At least one RHRSW pump room sump pump must be operable or the RHRSW/EECW pumps in that room must be declared inoperable.

X. Placing EECW pump control room handswitch in Normal After Start position, as soon as possible after EECW pump auto start, will enable handswitch disagreement light and pump tripped annunciator to function as designed.

Excerpts from OPL171.046 Lesson Plan:

at ~120 psig. Minimum flow is 1700 gpm per pump. Per DCN 71313 RHRSW Pumps have been replaced with new pumps that require a higher minimum flow rate of 1700 gpm. RHRSW pumps are located at the intake structure in their respective pump rooms A-D. (Pumps A1, A2, & A3 in 'A' room, Pumps B1, B2, & B3 in 'B' room, Pumps C1, C2, & C3 in 'C' room, and Pumps D1, D2, & D3 in 'D' room).

2. Each unit's Panel 9-3 has a control switch (CS) for each pump (12 CSs per unit).

3. The RHRSW pump motors are powered from the 4kV Shutdown Boards. Power supplies are as follows:

TP-3
Obj. 7.d ILT
Obj. 6.d LOR
Obj. 10.a NLO

Pwr Supply	Pump	Pump	Pwr Supply
A 4kV SD BD	A1	C1	B 4kV SD BD
	A2	C2	
3EA 4kV SD BD	A3	C3	3EB 4kV SD BD
3EC 4kV SD BD	B1	D1	3ED 4kV SD BD
C 4kV SD BD	B2	D2	D 4kV SD BD
	B3	D3	

Note: An aid to remembering power supplies is to build a box based on plant knowledge and remembering which pumps are supplied by Unit 3 boards. The plant knowledge is that safety systems labeled A, B, C, D are powered from board A, C, B, D in order. Unit 3 powers pumps A3, B1, C3, D1 (EECW capable) or 3, 1, 3, 1 in order per the A, C, B, D board scheme. Unit 1 & 2 boards power the remaining pumps A, B, C, D per the A, C, B, D scheme.

OPL171.046, Residual Heat Removal Service Water (RHRSW), Rev.#20

4. RHRSW pump assignments are as follows:

RHRSW	RHRSW/EECW	EECW	Division/Header
A2	A1	A3	Div I N EECW Header
C2	C1	C3	
B2	B1	B3	Div II S EECW Header
D2	D1	D3	

Obj. 10.b,c NLO

5. Each pump room contains a sump with 2 sump pumps. At least one sump pump must be operable or all the pumps in the room must be declared inoperable.

TP-3
Obj. 5, 8 ILT
Obj. 4 LOR
Obj. 6 NLOR
Obj. 8 NLO

6. RHRSW valves are powered as follows:

- Loop 1 Hx Outlet Valves – 480V RMOV BD A (on the given unit)

Obj. 7.d ILT
Obj. 6.d LOR

- Loop 2 Hx Outlet Valves – 480V RMOV BD B(on the given unit)
- RHR HTX 1D, U1-U2 STANDBY COOLANT VALVE FROM RHRSW 1-FCV-23-57 from 480V RMOV BD 1B

U1/U2

- STANDBY COOLAND VALVE 2-FCV-23-57 from 480V RMOV BD 3B

U2/U3

7. RHR heat exchangers are vertical shell-and tube 2-pass HX's (4 per Unit), located between Elev. 565 and Elev. 593 in each southern quadrant of the Reactor Building (pair of HX's located above a corresponding pair of RHR pumps).

a) Shell-side fluid is the water from the discharge of the RHR pump (i.e. either reactor water or suppression pool water). Tube-side fluid is RHR Service Water (reservoir water). RHRSW enters the top portion of the HX at the upper head of the tube bundle. The tube bundle and upper head is divided in half at the upper tube sheet by baffling inside the head. RHRSW passes through inlet half of tubes in a downward direction; the water then enters a lower "floating" - head plenum and returns, upward through the remaining half of the tubes, into the outlet side of the upper head.

b) Shell-side (RHR pump discharge) design flow is 10,000 gpm.

Obj. 5 ILT

c) Rated RHRSW flow through an RHR Heat Exchanger is 4500 gpm. To prevent damage to RHRSW pumps, minimum flow is 1700 gpm per pump in service. Per DCN 71313 RHRSW Pumps

What are the consequences of not

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

Examination Outline Cross-reference:	Level	RO	SRO
201001 (SF1 CRDH) CRD Hydraulic	Tier #	2	------
K2.05 (10CFR 55.41.7)	Group #	2	------
Knowledge of electrical power supplies to the following:	K/A #	201001K2.05	
• Alternate rod insertion valve solenoids	Importance Rating	3.9	------

Proposed Question: **# 53**

Unit 3 is operating at 100% RTP when an event results in the following conditions:

- A manual SCRAM is inserted
- Reactor Water Level lowers to (-) 60 inches
- All 4 SCRAM solenoid group 'A' lights are **ILLUMINATED**
- All 4 SCRAM solenoid group 'B' lights are **EXTINGUISHED**
- **ALL** Control Rods inserted to '00'

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

The Control Rods inserted from the operation of **(1)** that energized from **(2)** to vent the SCRAM air header.

A. (1) Alternate Rod Insertion (ARI) valves
(2) 120 VAC

B. (1) Alternate Rod Insertion (ARI) valves
(2) 250 VDC

C. (1) SCRAM pilot solenoid valves
(2) 120 VAC

D. (1) SCRAM pilot solenoid valves
(2) 250 VDC

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible, in that various power supplies differ between the numerous RPS, PCIS and ECCS logic and control power systems, these power supplies are easily and often confused.

B **CORRECT**: (*See attached*) Control Rods inserted due automatic initiation of ARI when Reactor Water level lowered below (-45) inches. Second part, ARI valves are 250 VDC powered, controlled through ATU (ECCS) powered logic independent of the RPS system.

C INCORRECT: First part is incorrect but plausible in that SCRAM pilot solenoid valves will actuate on a SCRAM. The stem states that a SCRAM was inserted, and Reactor Water level is (-) 60 inches, well below the RPS trip setpoint of (+) 2 inches. The candidate may incorrectly conclude that Control Rods inserted due to the SCRAM. However, SCRAM pilot solenoid valves did not actuate because RPS A did not deenergize as indicated in the stem due to all 4 SCRAM solenoid group 'A' lights remaining illuminated. 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 electrical power supplies to the alternate rod insertion valve solenoids. 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): 3-OI-85, Rev. 104 (Attach if not previously provided)
OPL171.005, Rev. 24U3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.005 Obj. 30e (As available)

Question Source:

Bank #	
Modified Bank #	OPL171.005-23 005 #334
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:

QUESTIONS REPORT
for ILT EXAM BANK 08 22 2018

334. OPL171.005-23 005

Unit 3 is operating at 100% RTP.

Subsequently;

- The Reactor SCRAMs and Reactor Water Level lowers to (-) 60 inches
- All 4 SCRAM Solenoid Group 'A' Reset lights are **illuminated**
- All 4 SCRAM Solenoid Group 'B' Reset lights are **extinguished**
- All Control Rods inserted to 00

(Assume no Operator actions have been taken)

Which one of the following completes the statement below?

The Control Rods inserted due to operation of the ___**(1)**___ which ___**(2)**___ to vent the air header.

A. (1) ARI valves
(2) energize

B. (1) ARI valves
(2) de-energize

C. (1) SCRAM pilot solenoid valves
(2) energize

D. (1) SCRAM pilot solenoid valves
(2) de-energize

Excerpt from 3-OI-85:

3.2 ATWS/ARI/RPT

A. The ARI System auto initiation can be reset after a 30 sec time delay, only when all initiation signals are reset.

B. The ATWS/ARI/RPT is activated by either two low levels (≤ -45 in) or two high pressures 1162 psig, or manual initiation pushbutton.

1. An automatic signal from either A or B trip channel causes two actions:

a. it opens one of the two RPT breakers on each of the two recirculation pumps,

AND

b. It energizes one of the two identical sets of four ATWS/ARI/RPT valves.

2. Manual initiation from either A or B trip channel only initiates the ARI portion of the system. The RPT will not trip from manual initiation.

3.3 Tech Specs

A. When maintenance is performed on a control rod that could affect the control rod's scram time, scram time testing is required on the rod prior to declaring the rod operable.

B. Control rod scram time testing is required prior to declaring a rod operable after maintenance which could affect its scram time. (T.S. SR-3.1.4.3)

C. If rod insertion to position 00 is required and core thermal power is less than or equal to 10%, entry into LCO 3.1.6 may be required.

D. CRDs that do not insert with normal system drive pressures may be inoperable in accordance with Tech Spec 3.1.3. [NER/C] [GE SIL 139 and 538]

However, if rod motion is observed, but the CRD fails to notch, drive water pressure may be raised to compensate for excessive under-piston leakage.

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.

Excerpts from OPL171.005 Lesson Plan:

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 24U3

o. ATWS/ARI (Alternate Rod Insertion) System	Obj. ILT 23
1) Purpose - to provide an alternate method of shutting down the reactor should control rods fail to insert on a scram.	Obj. LOR 13
2) This assumes an ATWS occurs with or without RPS de-energization.	ATWS/ARI implementation was required by 10 CFR 50.62 and DCR 3125, Rev. 1
3) The associated valves are controlled through ATU (ECCS) powered logic, independent of the RPS System.	
a) ARI Valve FSV-85-730 - 250VDC, Pnl A - 68-25/418	Obj. ILT 30.e
b) ARI Valve FSV-85-731 - 250VDC, Pnl A - 68-25/419	
4) Two 3-way ARI scram valves were "added" to the HCU air supply header, to both isolate the header and also vent the scram air header.	These are actually one pair of the old back-up scram valves that were re-powered from the ECCS ATU cabinet
5) Six 2-way air vent valves were added to the HCU banks and SDV air supply for fast dumping of air to atmosphere, ensuring rod insertion begins within 15 seconds and completes within 25 seconds.	
6) A coincident trip of either 2 low levels or 2 high pressures in the same trip channel causes ATWS/ARI/RPT; trip values are 1162 psig or level 2, ≤ - 45" RWL. Both channels must trip to trip both recirc pumps and realign all the ARI valves.	
7) Manual ARI pushbuttons A/B on pnl. 9-5 is supplied to actuate either trip channel. These PB will only initiate the ARI system. The RPT won't manually initiate.	
p. ATWS/ARI Logic - Manual Initiation	
1) Turn collar on HS-68-119A to arm ARI.	Obj. ILT 31
2) RLY-68-119A1 energizes and the white "ARMED" light on panel. 9-5 energizes and RPT is blocked.	Obj. LOR 16 Figure-5
3) Depress HS-68-119A to initiate CH. A ARI.	ARI cannot be manually initiated unless armed

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 24U3

I. Scram pilot solenoid valves	
1) Purpose - to route air normally to the diaphragm operators of the scram inlet and outlet valves which are held closed by instrument air at 73 psig.	Obj. ILT 22 Obj. LOR 12 NLO/NLOR 11
2) These are solenoid-operated 3-way valves.	
3) Originally there was a pair of solenoid-operated valves for each of the 185 HCUs, both of which had to realign to vent the scram valves. They are being replaced with single valve bodies with 2 solenoid valves internally operating a third internal 3 way valve. Either solenoid being energized keeps the 3 way valve aligned to hold the scram valves closed.	New valves do not require rebuild as often as the old valves. Performance is identical.
4) The power supplied to each solenoid is from a separate Reactor Protection System bus. Both solenoids are normally energized.	Deenergize RPS A and RPS B solenoids to scram a rod. DCN 64502.
5) FSV-85-39A for all HCU's - 120v, 60Hz, Batt BD 2, Panel 9, Bkr. 903, RPS Bus A.	Obj. ILT 30.b (OF5) 2-730E915-1, 13
6) FSV-85-39B for all HCU's - 120v, 60Hz, Batt BD 2, Pnl 9, Bkr. 953, RPS Bus B.	
7) When both solenoid-operated scram pilot solenoids are de-energized by the Reactor Protection System, air is vented to atmosphere and the scram inlet and discharge valves are opened by internal spring pressure, resulting in the drive scramming. A stronger spring on the exhaust valve ensures it opens first providing a vent path for the high pressure discharge.	Piston size difference amplifies the applied scram pressure such that a scram with the discharge isolated would damage the mechanism due to the hydraulic amplification of the applied pressure.
8) If only one scram pilot valve is de-energized, the scram valves will remain closed.	

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

Examination Outline Cross-reference:	Level	RO	SRO
214000 (SF7 RPIS) Rod Position Information **K5.01** (10CFR 55.41.5)	Tier #	2	------
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the ROD POSITION INFORMATION SYSTEM:	Group #	2	------
	K/A #	214000K5.01	
• Rod position indication failures	Importance Rating	3.5	------

Proposed Question: **# 54**

Unit 3 was operating at 100% RTP when the following plant conditions occur:

- Control Rod 30-31 lost its **ONES** digit position indication
- 3-AOI-85-4, Loss of RPIS, is entered

Given the conditions above, which **ONE** of the following completes the statements below in accordance with 3-AOI-85-4?

Control Rod 30-31 **(1)** be moved to an Operable Position Indication as a means of position verification.

If it is determined that Operators must SCRAM Control Rod 30-31, this will be conducted from the **(2)** .

A. (1) can
(2) Aux Instrument Room

B. (1) can
(2) Battery Board Room 3

C. (1) can NOT
(2) Aux Instrument Room

D. (1) can NOT
(2) Battery Board Room 3

Proposed Answer: A

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with NOTE in 3-AOI-85-4, Loss of RPIS, Control Rods may be moved to an Operable Position Indication as a means of position verification. As a minimum, rod position will be verified, preferably with an independent position indication or other method (i.e., Traversing Incore Probe (TIP) trace). Given only the ONES digit position indication has been lost, Control Rod may be moved to verify its position indication. Second part, in accordance with 3-AOI-85-4, Loss of RPIS, if verification of indication position has failed, Control Rod 30-31 must be fully inserted by individually SCRAM the Control Rod at 3-PNLA-009-0016 (Aux Instrument Room).

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that both RPS Bus 'A' and/or 'B' are de-energized from Battery Board 3 which can cause a half or full SCRAM.

C INCORRECT: First part is incorrect but plausible in that if the candidate believes only other methods would be used for position indication verification such as ICS display or Rod Worth Minimizer. Additionally, if candidate confuses the given loss of 'ONES' digit with the 'TENS' digit criteria, then they could easily believe that Control Rod 30-31 is assured to be in its required/intended position. 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 effect from a loss of Control Rod position indication available from the Main Control Room and actions that can be taken to determine actual Control Rod position. This question is rated as Memory due to the requirement to recall facts.

Technical Reference(s): 3-AOI-85-4, Rev. 16 (Attach if not previously provided)

3-OI-99, Rev. 68

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.029 Obj. 6 (As available)

Question Source:

Bank #	BFN NRC 1909 #61
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:

Unit 3 was operating at 100% RTP when the following plant conditions occur:

- Control Rod 30-31 lost its **ONES** digit position indication
- 3-AOI-85-4, Loss of RPIS is entered

Given the conditions above, which **ONE** of the following completes the statements below in accordance with 3-AOI-85-4, Loss of RPIS?

Control Rod 30-31 **(1)** be moved to an Operable Position Indication as a means of position verification.

If it is determined that Operators must SCRAM Control Rod 30-31, this will be conducted from the **(2)** .

A. (1) can
(2) Aux Instrument Room

B. (1) can
(2) Battery Board Room 3

C. (1) can NOT
(2) Aux Instrument Room

D. (1) can NOT
(2) Battery Board Room 3

Proposed Answer: **A**

Excerpts from 3-AOI-85-4:

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a total or partial loss of Rod Position Information System (RPIS) or an individual rod position indication.

Control Rod Position may be determined by the use of operable position indicators, by moving a Control Rod to a position with an operable indicator, or by the use of other approved methods. Refer to Tech Spec Bases SR 3.1.3.1 for further guidance. Control Rods need **NOT** be declared immediately inoperable upon discovery of inoperable position indication. The Tech Spec Bases may be referred to for Control Rod Position verification methods to satisfy the requirements of TS SR 3.1.3.1.

Operable position indicators include, but are **NOT** limited to, the following:

- Full Core Digital Display
- Four Rod display
- Full Core Display green background lighting (Full In)
- Full Core Display red background lighting (Full Out)
- Full Core Display double green dashes (Full In Overtravel)
- Full Core Display double red dashes (Full Out Overtravel)
- ICS display
- RWM

2.0 SYMPTOMS

A. Symptoms of a total loss of the Rod Position Information System.

1. RPIS INOPERABLE annunciator, (3-XA-55-5A, Window 35) in alarm.
2. INSERT BLOCK/WITHDRAW BLOCK indicating lights illuminated on RWM Switch Panel, 3-XS-85-9025, on Panel 3-9-5.
3. All the RWM blocks are applied in the event of a system hardware or software failure, when power is below the LPAP, (27% as sensed by Total Steam Flow). At any Rx power, when a loss of ICS 3A occurs, a select block occurs due to the loss of power and cannot be bypassed using the RWM Bypass key.
4. ICS indicates unknown control rod positions when new scan initiated.

4.2 Subsequent Actions (continued)

NOTE

The following are indications of a failed reed switch position indicator:

- Even switch failed closed gives continuous indication superimposed on present position indication.
- Even switch failed open gives blank indication when control rod is at that notch and gives false DRIFT alarm when control rod is latched at that notch.
- Odd switch failed closed gives continuous DRIFT alarm.
- Odd switch failed open will **NOT** show any effect, or produce any indication that the switch has failed.

[8] **IF** an individual control rod position indication is off on Full Core Display or 4-Rod Display, **THEN**

CHECK individual circuit cards in Panel 3-9-27 for proper operation.

[9] **IF** an individual reed switch position indicator is out-of-service on Full Core Display or 4-Rod Display, **THEN**

RECORD control rod coordinates and position in Unit Operators Log.

NOTE

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). As a minimum, rod position will be verified, preferably with an independent position indication or other method (i.e., TIP trace).
[INPO SOER 84-002]

[10] [NER/C] **IF** unable to restore position indication for an individual control rod or rods, **THEN**

CHECK with Reactor Engineer and **DETERMINE** additional corrective action.

4.4 Alternate Methods of Determining Control Rod Position of A Single Control Rod (continued)

[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[1.2] and/or 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 Control Rod from [48] to [00] is @ 60 seconds.

B. **WHEN** the above indications show the Control Rod is fully inserted and RPIS has not restored, Individually SCRAM the Control Rod at 3-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.

C. **IF** RPIS indication has not been restored, **THEN**

DISARM electrically the CRD HCU Amphenols by disconnecting and tagging them with a hold order.

Excerpt from 3-OI-99: Supports Distractors B2, D2

BFN Unit 3	Reactor Protection System	3-OI-99 Rev. 0068 Page 27 of 108

7.0 SHUTDOWN

7.1 De-energization of RPS Bus 3A

[1] **REVIEW** all Precautions and Limitations in Section 3.0.

[2] **PERFORM** Scram Solenoid Temperature Checks per Section 8.14.

[3] **REVIEW** Attachment 1 to determine affected systems and components.

NOTE

The following Steps 7.1[6] or 7.1[7] de-energize RPS Bus A.

CAUTIONS

1) De-energization of an RPS Bus causes a half-scram and associated isolations. Operation of RHR Shutdown Cooling and Reactor Water Cleanup System is NOT possible while an RPS Bus is de-energized.

2) If any MSIV has a failed solenoid circuitry, deenergizing the opposite RPS BUS may cause a fast closure of the MSIV. Fast closure of a MSIV at 100% Power results in a Reactor Scram.

[4] **CHECK** an inadvertent fast closure of an MSIV is **NOT** expected due to the RPS bus de-energization.

[5] **IF** a fast closure is expected with the unit at power, **THEN**

REFER TO 3-OI-1. (Otherwise N/A)

[6] **IF** De-energizing the "A" RPS Bus, **THEN**

At Battery Board 3, **OPEN** RPS MAIN BREAKER BUS A U-3, 0-BKR-280-0003/902.

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

Examination Outline Cross-reference:	Level	RO	SRO
216000 (SF7 NBI) Nuclear Boiler Instrumentation	Tier #	2	------
K1.07 (10CFR 55.41.7/8)	Group #	2	------
Knowledge of the physical connections and/or cause and effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following systems:	K/A #	216000K1.07	
• ADS	Importance Rating	4.1	------

Proposed Question: **# 55**

Unit 2 is operating at 100% RTP when an event results in the following:

- 2-AOI-100-2, Control Room Abandonment, IMMEDIATE ACTIONS completed
- **BOTH** Reactor Water Level and Reactor Pressure Control established at **Backup Control Panel 2-25-32**

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

Currently, 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE, indication ___(1)___ available and ADS function ___(2)___ available.

A. (1) is
(2) is

B. (1) is
(2) is NOT

C. (1) is NOT
(2) is

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

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that 2-AOI-1-1, Relief Valve Stuck Open, IMMEDIATE ACTIONs requires the stuck open MSRV to be identified by observing 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE, on Panel 2-9-47 or SRV Tailpipe Flow Monitor indications on Panel 2-9-3 in the Main Control Room. Second part is incorrect but plausible given that the same control power supplies exist for an ADS MSRV when transfer to Panel 2-25-32 is complete. The candidate can incorrectly conclude that with the same ADS SRV power supplies, regardless of controlling location, that all normal logic and control function remains when transfer to Panel 2-25-32 is complete.

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*) The 13 SRV temperature indications are normally provided by 2-TR-1-1, MSRV DISCHARGE TAILPIPE TEMPERATURE, on Panel 2-9-47 in the Main Control Room, not the Backup Control Panel 2-25-32. From Panel 2-25-32, only 4 SRVs can be operated. Also, 2-FMT-1-4, SRV Tailpipe Flow Monitor, indications are only available on Panel 2-9-3 in the Main Control Room.
Second part, in accordance with 2-AOI-100-2, operation from Panel 2-25-32 bypasses logic and interlocks normally associated with the components. In this case, once the MSRV disconnect switches are operated to transfer control to panel 2-25-32, ADS logic is bypassed and Main Control room handswitch indication is lost.

RO Level Justification: Tests the candidate's knowledge of the physical connections and/or cause and effect relationships between Nuclear Boiler Instrumentation and 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.

Technical Reference(s): OPL171.009, Rev. 17U2 (Attach if not previously provided)
2-AOI-100-2, Rev. 62
2-AOI-1-1. Rev 30

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009, Obj 12 (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:

Excerpts from 2-AOI-100-2:

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0062 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.

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0062 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	☐

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0062 Page 9 of 95

4.2 Unit 2 Subsequent Actions (continued)

[3] **PLACE** the following MSRV disconnect switches in DISCT at Panel 2-25-32:

Switch No.	Description	
2-XS-1-18	MAIN STM LINE B RELIEF VALVE DISCT	☐
2-XS-1-4	MAIN STM LINE A RELIEF VALVE DISCT	☐
2-XS-1-42	MAIN STM LINE D RELIEF VALVE DISCT	☐
2-XS-1-23	MAIN STM LINE B RELIEF VALVE DISCT	☐
2-XS-1-41	MAIN STM LINE D RELIEF VALVE DISCT	☐
2-XS-1-180	MAIN STM LINE D RELIEF VALVE DISCT	☐

[4] **PLACE** the following MSRV transfer switches in EMERG at Panel 2-25-32:

Switch No.	Description	
2-XS-1-22	MAIN STM LINE B RELIEF VALVE XFR	☐
2-XS-1-5	MAIN STM LINE A RELIEF VALVE XFR	☐
2-XS-1-30	MAIN STM LINE C RELIEF VALVE XFR	☐
2-XS-1-34	MAIN STM LINE C RELIEF VALVE XFR	☐

NOTE

Use of the following sequence when opening MSRVs should distribute heat evenly in the Suppression Pool.

[5] **MAINTAIN** Reactor Pressure between 800 and 1000 psig using the following sequence at Panel 2-25-32:

A. 2-HS-1-22C, MAIN STM LINE B RELIEF VALVE

B. 2-HS-1-5C, MAIN STM LINE A RELIEF VALVE

C. 2-HS-1-30C, MAIN STM LINE C RELIEF VALVE

D. 2-HS-1-34C, MAIN STM LINE C RELIEF VALVE

Excerpts 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;

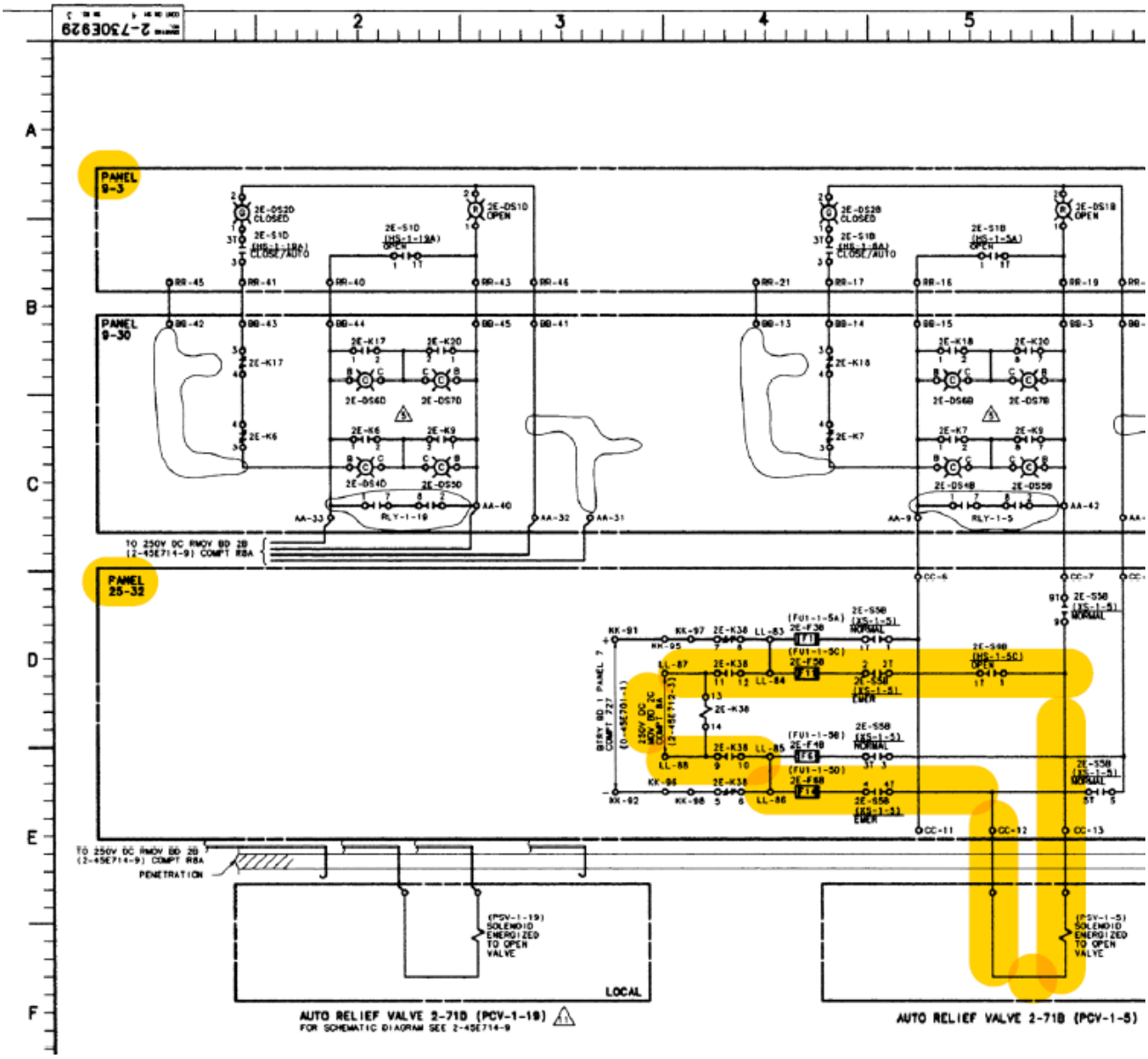
CYCLE the affected relief valve control switch as required up to three times:

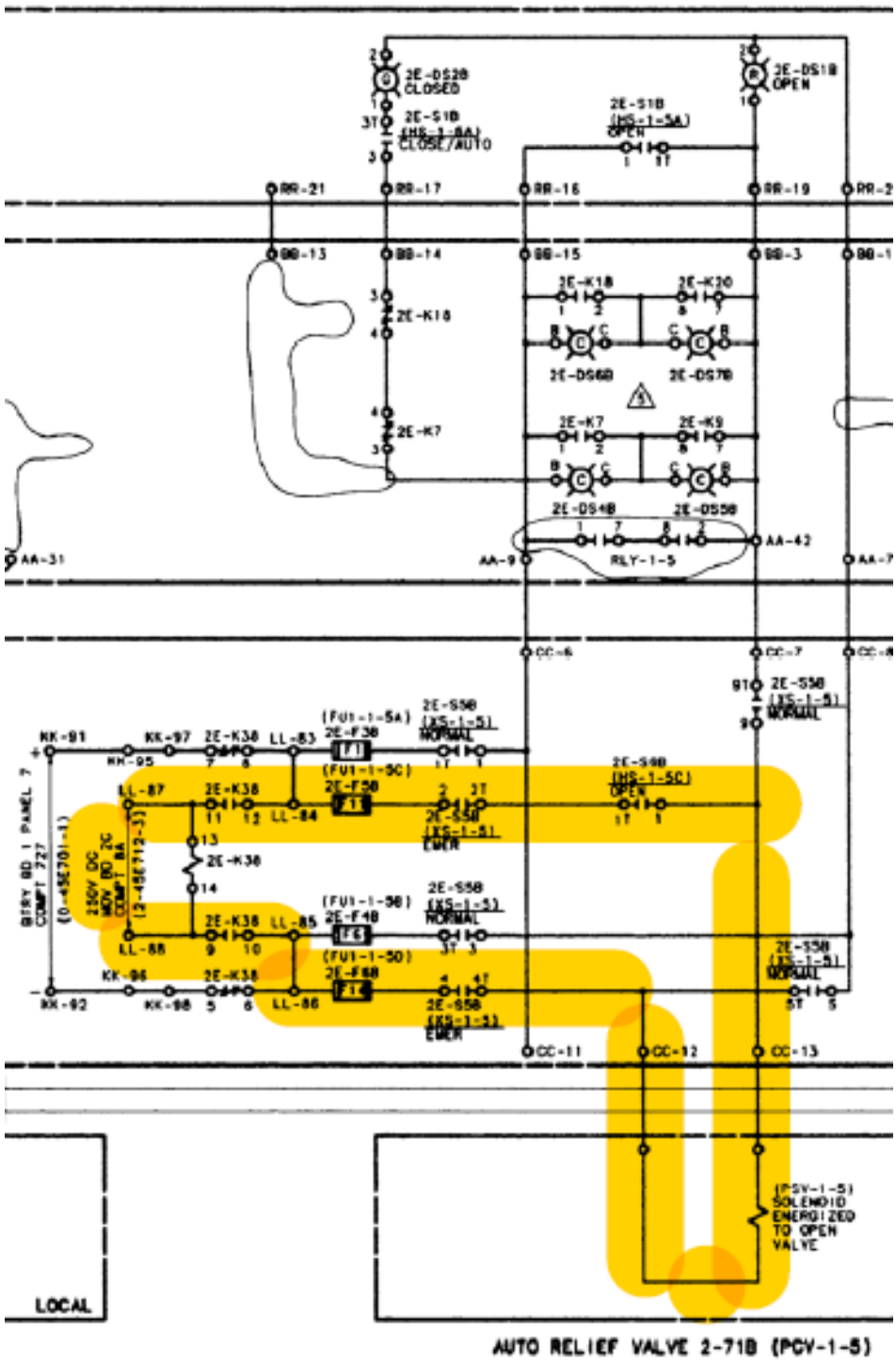
- CLOSE to OPEN to CLOSE positions ☐

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

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

Excerpt from 2-730E929 (continue to subsequent pages for larger area of detail): Illustrates power differences for Panel 9-3 in MCR and Panel 25-32




2E-DS2B CLOSED
2E-S1B CLOSE/AUTO
2E-DS1B OPEN
2E-S1B OPEN
RR-21
RR-17
RR-16
RR-19
BB-13
BB-14
BB-15
BB-3
2E-K18
2E-K7
2E-K20
2E-K9
2E-DS6B
2E-DS7B
2E-DS4B
2E-DS5B
AA-31
AA-9
AA-42
RLY-1-5
CC-6
CC-7
2E-S5B
2E-F3B
2E-F5B
2E-F4B
2E-F6B
2E-K38
KK-91
KK-97
KK-95
KK-92
KK-98
KK-96
LL-83
LL-84
LL-85
LL-86
LL-87
LL-88
CC-11
CC-12
CC-13
(PSV-1-5) SOLENOID ENERGIZED TO OPEN VALVE
LOCAL
AUTO RELIEF VALVE 2-71B (PCV-1-5)

Excerpt from OPL171.009 Lesson Plan:

(h) On Unit 1, four of the SRVs are electrically isolated at 1-LPNL-925-0658 (these four can be operated from LPNL-925-0658) Of the remaining nine all can be isolated at Panel 25-32 and four can be controlled from there

(i) On Unit 2, three of the SRVs are electrically isolated from Panel 25-32. Of the remaining 10, all have control transfer switches which isolate their circuits from the control room, and 4 have control switches on Panel 25-32 to allow manual actuation from Panel 25-32. Transfer to Panel 25-32 has no effect on the valve's safety function, but will prevent automatic operation of the ADS valves.

(j) On Unit 3, four of the SRVs are electrically isolated from Panel 25-32.

(k) When the disconnect switches are repositioned at Panel 25-32 or Panel 925-0658, manual control from Panel 9-3 is lost, including position indication and the ADS function.

Table 1
Obj. ILT 12
Obj. LOR 6
Obj. NLOR 4

(OF-5 Solid understanding of plant design)

Describe the effect on pant operation if these switches are inadvertently operated

4. Steam Line Flow Restrictors

a. Purpose

(1) Limit flow during steam line rupture to <200% of rated line flow.

(2) Limit the loss of coolant from the Reactor Vessel following a steam line rupture outside the primary containment to the extent the vessel level does not fall below the top of the core within the MSIV closure time.

(3) Limit the ΔP across core internals by restricting flow.

(4) Provides mechanism for measuring steam flow.

(a) Indication

(b) Provides input to reactor level control system

(c) Provides input to the primary containment isolation system.

b. Venturi-type restrictor. Pressure drop ~10 psid across the restrictor at full flow.

Figure-5
Obj. ILT 7
Obj. LOR 5
Obj. NLOR 3
Obj. NLO 3

Form 4.3-1 **Written Examination Question Worksheet**

Examination Outline Cross-reference:	Level	RO	SRO
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode	Tier #	2	------
A3.01 (10CFR 55.41.7)	Group #	2	------
Ability to monitor automatic operation of the RHR/LPCI TORUS/SUPPRESSION POOL COOLING MODE including:	K/A #	219000A3.01	
• Valve operation	Importance Rating	3.9	------

Proposed Question: **# 56**

Unit 3 is operating at 100% RTP with the following conditions:

- A HPCI SR is in progress
- RHR Loop I Suppression Pool Cooling is in service

Subsequently, a LOCA occurs resulting in the following:

- Drywell Pressure 3 psig
- Reactor Pressure 400 psig
- Reactor Water Level (-) 80 inches

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

2-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING / TEST VALVE,

(1) receive an automatic closure signal.

The RHR System **(2)** receive an automatic LPCI initiation signal.

A. (1) will
(2) will NOT

B. (1) will
(2) will

C. (1) will NOT
(2) will NOT

D. (1) will NOT
(2) will

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that the given Reactor Water Level of (-) 80 inches is above the LPCI initiation signal of (-) 122 inches. There are numerous ECCS, PCIS and RPS trip setpoints and or interlocks, these values are easily and often confused.

B **CORRECT**: (*See attached)* In accordance with 3-OI-74, the RHR torus cooling/test valves, 3-FCV-74-59(73), receive an auto closure signal if a LPCI initiation signal is present. Second part, in accordance with 3-OI-74, LPCI will initiate on the following signal, High Drywell Pressure with a low Reactor Vessel Pressure, 2.45 psig and 450 psig, respectively (Accident signal).

C INCORRECT: First part is incorrect but plausible in that the RHR torus cooling/test valves may be opened to align a Suppression Pool cooling flow path anytime a LPCI signal is NOT present, whereas RHR torus spray and containment spray valves have an additional permissive of Drywell Pressure ≥ 1.96 psig before their respective spray flow paths can be aligned. 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 automatic operation of the RHR/LPCI Torus/Suppression Pool Cooling Mode pertaining to valve 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.

Technical Reference(s): 3-OI-74, Rev. 147 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044, Obj. 3i (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 3-OI-74:

3.6 Interlocks (continued)

9. The RHR spray/cooling valves, 3-FCV-74-57(71), receive an auto-closure signal in the presence of a LPCI initiation signal and are interlocked to prevent opening if the in-line torus spray valve, 3-FCV-74-58(72), is not fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

   a. Reactor level is >2/3 core height **AND**

   b. LPCI initiation signal is present **AND**

   c. The select reset switch is in the SELECT position.

   The requirements for >2/3 core height and a LPCI initiation signal may be bypassed using the Keylock Bypass Switch, 3-XS-74-122/30.

10. If primary containment cooling is desired with reactor level at <2/3 core height, the keylock bypass switch must be placed in BYPASS before the select reset switch is placed in SELECT to ensure relay logic is made up.

11. The RHR torus spray valves, 3-FCV-74-58(72), have the same in-line valve interlocks as those outlined in Step 3.6A.10 for the torus spray/cooling valves. Additionally these valves have an interlock preventing opening unless drywell pressure is ≥ 1.96 psig which cannot be bypassed.

12. The RHR torus cooling/test valves, 3-FCV-74-59(73), receive an auto closure signal in the presence of a LPCI initiation signal. Auto-closure may be bypassed by the same conditions/actions outlined in Step 3.6A.10

13. The RHR containment spray valves, 3-FCV-74-60(74) and 61(75), have in-line valve interlocks similar to these described in Step 3.6A.10 through 3.6A.12 for the RHR torus spray valves 3-FCV-74-57(58) and 71(72).

14. If 3-FCV-74-59(73) LOCA CLOSURE TIME Light (3-IL-74-59Y Loop I; 3-IL-74-73Y Loop II) on Panel 3-9-3 is extinguished due to its associated valve being opened, that Loop is inoperable for LPCI.

15. If RHR Pumps 3A(B) and 3C(D) SD COOLING SUCT VLV, 3-FCV-74-2(25) and 13(36) are not fully closed, the RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 3-FCV-74-57(71) cannot be opened.

16. If RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 3-FCV-74-57(71) is not fully closed, the RHR Pumps 3A(B) and 3C(D) SD COOLING SUCT VLV, 3-FCV-74-2(25) and 13(36) cannot be opened.

3.2 RHR Pumps (continued)

B. [NRC/C] The RHR pumps are considered to be operable without the seal cooler under the following conditions:

1. Always operable in the LPCI and Containment Cooling Mode.
2. During Shutdown Cooling, operable up to a suction temperature of 215°F.
3. Operable for an emergency with suction temperatures between 215°F and 400°F. Operation in this condition for more than two days will require an inspection of the seal surfaces. [NRC LER 296/83047 R1]]

C. To prevent Recirculation Pump operation at shutoff head, RHR pumps must not be operated in parallel with Recirculation Pumps.

3.3 LPCI

A. LPCI will initiate on any of the following signals:

1. Reactor Vessel low low low water level (-122 inches)(level 1).
2. High Drywell Pressure (2.45 psig) with low Reactor Vessel Pressure (450 psig).

B. Manually stopping an RHR pump after LPCI initiation disables automatic restart of that pump until the initiation signal is reset. The affected RHR pump can still be started manually.

C. Upon an automatic LPCI initiation with normal power available, RHR Pump 3A will start immediately, THEN 3B, 3C, 3D will then sequentially start at 7 second intervals. Otherwise, all RHR pumps will start immediately once diesel power is available (and normal power unavailable).

D. As soon as practicable after an RHR pump(s) auto start, the corresponding control room handswitch should be placed in normal-after-start position to ensure the handswitch disagreement light(s) and pump tripped annunciator(s) function as designed.

E. If RECIRC PUMP 3A(3B) DISCHARGE VALVE, 3-FCV-068-0003(0079), is declared inoperable while the valve is OPEN, the associated LPCI subsystem is required to be declared INOPERABLE. (Tech Spec BASES SR 3.5.1.5)

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

Examination Outline Cross-reference:	Level	RO	SRO
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup	Tier #	2	------
G2.4.4 (10CFR 55.41.10)	Group #	2	------
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures	K/A #	233000G2.4.4	
	Importance Rating	4.5	------

Proposed Question: **# 57**

Unit 2 is operating at 100% RTP when the following conditions occur:

- 2B Fuel Pool Cooling (FPC) Pump is tagged
- Fuel Pool Temperature is 90 °F
- In service 2A FPC Pump trips
- **NO** Operator actions have been taken

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

In accordance with 2-AOI-78-1, Fuel Pool Cleanup System Failure, inputs to 2-TR-74-80, RHR/FUEL POOL COOLING TEMPERATURE, on Panel 2-9-21 **(1)** accurate.

Subsequently, if Fuel Pool Temperature rises to 115 °F, 2-EOI-3, SECONDARY CONTAIMENT CONTROL, entry **(2)** required.

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: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if candidate incorrectly determines that the location of fuel pool sensors are not dependent on Fuel Pool cooling flow. Second part is incorrect but plausible, in that candidate may incorrectly determine that Fuel Pool temperature of 115 °F is the entry condition of 2-EOI-3 vice 125 °F.

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**: In accordance with 2-AOI-78-1, loss of Fuel Pool Cooling System flow will cause Fuel Pool Cooling System temperature inputs on 2-TR-74-80 to be inaccurate until Fuel Pool Cooling flow is reestablished. Second part, in accordance with 2-EOI-3, Spent Fuel Pool temperature entry condition is above 125 °F.

RO Level Justification: Tests the candidate's ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures as it relates to Fuel Pool Cooling/Cleanup. 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.

Technical Reference(s): 2-AOI-78-1, Rev. 36 (Attach if not previously provided)
2-EOI-3, Rev. 20

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.052 Obj. 7 (As available)

Question Source:
Bank # _____
Modified Bank # BFN NRC 2204 #58 (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:

Proposed Question: **# 58**

A trip of 4KV Shutdown Board **(1)** would cause a loss of 1B Fuel Pool Cooling Pump.

The **MINIMUM** Fuel Pool Temperature which requires entry into 1-EOI-3, Secondary Containment Control, is **(2)** .

A. (1) B
(2) 125 ºF

B. (1) B
(2) 150 ºF

C. (1) C
(2) 125 ºF

D. (1) C
(2) 150 ºF

Proposed Answer: **C**

Excerpts from 2-AOI-78-1:

BFN Unit 2	Fuel Pool Cleanup System Failure	2-AOI-78-1 Rev. 0036 Page 5 of 25

2.2 Loss of Level (continued)

B. FUEL POOL HIGH/LOW LEVEL, 2-IL-078-0002, white indicating light illuminated on 2-LPNL-925-0015.

C. FUEL POOL GATE LEAKAGE EXCESSIVE, 2-IL-078-0005, white indicating lamp illuminated on 2-LPNL-925-0015.

D. REFUEL BELLOWS LEAKAGE EXCESSIVE, 2-IL-078-0051, white indicating lamp illuminated on 2-LPNL-925-0015.

E. FUEL POOL SYSTEM ABNORMAL (2-XA-55-4C, Window 1) on Panel 2-9-4 in alarm.

F. Fuel Pool Cooling Pump TRIPPED.

G. Lowering SFSP level as Indicated by either of the following:

1. 2-LI-078-0042, WIDE RANGE LEVEL INDICATOR LOOP 1, on 2-LPNL-925-0242

2. 2-LI-078-0043, WIDE RANGE LEVEL INDICATOR LOOP 2, on 2-LPNL-925-0243

4.2 Subsequent Actions (continued)

NOTE

The Fuel Pool Cooling System may be used as desired as an alternate method of decay heat removal, when available: (Refer to applicable Tech Spec Bases B 3.4.7, B 3.4.8, B 3.9.7,.B 3.9.8

CAUTION

Loss of Fuel Pool Cooling System flow will cause Fuel Pool Cooling System Temperature inputs on 2-TR-74-80 to be inaccurate until Fuel Pool Cooling flow is reestablished.

[3] **IF** Fuel Pool Cooling System failure is from loss of cooling, **THEN**

PERFORM the following, otherwise, **MARK** steps N/A:

[3.1] **START** idle Fuel Pool Cooling Pump 2B(2A).

[3.2] **ATTEMPT** to restart the tripped Fuel Pool Cooling Pump 2A(2B) as follows:

[3.2.1] **INSPECT** the tripped Fuel Pool Cooling Pump and Breaker for damage or abnormal conditions.

[3.2.2] **IF** no damage or abnormal conditions are found, **THEN**

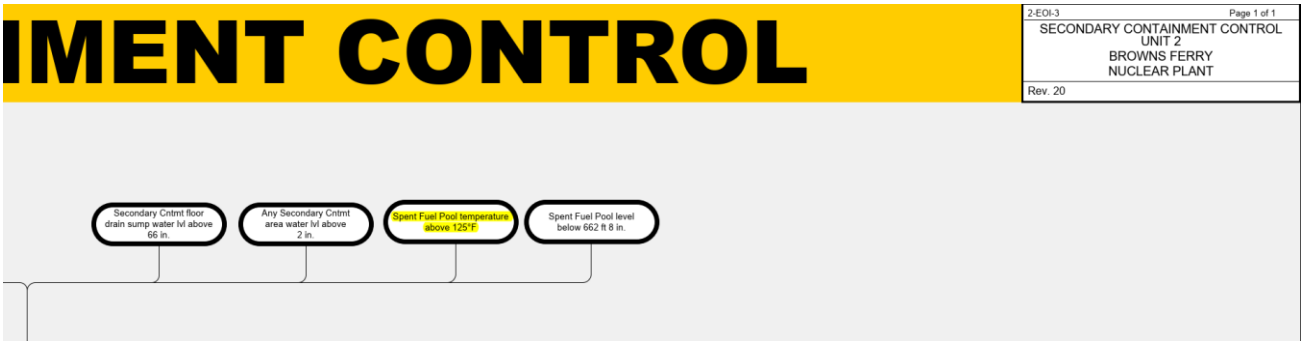
START the tripped Fuel Pool Cooling Pump.

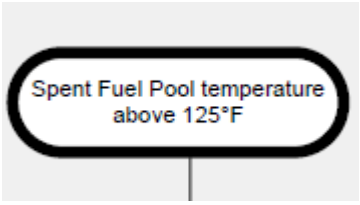
[3.3] **ENSURE** RBCCW System is operating and **REFER TO** 2-OI-70

[3.4] **IF** Fuel Pool Cooling flow cannot be established such that Fuel Pool Cooling temperature input 2-TE-78-20 on 2-TR-74-80 is available, **THEN:**

NOTIFY MIG to install a temporary Fuel Pool temperature monitoring device to determine Fuel Pool temperature. Otherwise, **MARK** N/A.

Excerpt from 2-EOI-3:





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	------
EA1.08 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:	K/A #	295026EA1.08	
• LPCS	Importance Rating	3.8	------

Proposed Question: **# 58**

Unit 3 has experienced a LOCA resulting in the following conditions:

- Suppression Pool Temperature is stable at 200 ºF
- Suppression Chamber Pressure is stable at 5 psig
- Suppression Pool Level is (-) 5.5 inches and lowering
- Drywell Pressure is stable at 10 psig
- 3A RHR Pump flow is 11,500 gpm
- 3B/3D Core Spray Pump combined flow is 4,000 gpm
- **NO** other ECCS Pumps are running

Given the current conditions above, which **ONE** of the following identifies the ECCS Pump(s), if any, that has/have sufficient NPSH for continued operation?

[REFERENCE PROVIDED]

A. **NONE**

B. 3A RHR Pump **ONLY**

C. 3B/3D Core Spray Pumps **ONLY**

D. 3B/3D Core Spray Pumps **AND** 3A RHR Pump

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that Suppression Pool Level is at the low set point for Suppression Pool Level abnormal alarm, but above the Tech Spec minimum. Also, if Core Spray is plotted for the **loop** flow and not the **pump** flows, it would be in the unsafe region of Curve 1. As referenced in EOI Tables L-1 and L-3 to EOI-5 Caution #2, states: Operating pumps with suction from the Suppression Pool above the NPSH Limit (Curve 1,2,…..) may cause equipment damage. Given flows for both RHR Pump 3A and Core Spray Loop II (3B/3D), Curve #1 and Curve #2 must be used to determine if sufficient NPSH exists or not.

B INCORRECT: Incorrect but plausible in that RHR Pump 3A is **above** the safe region of NPSH Limits Curve #2. If Drywell Pressure is used to plot Curve #2, Pump would be operating in the safe region of Curve #2 and if Core Spray plotted for **loop** flow, it would then be in the unsafe region of Curve #1.

C **CORRECT**: *(See attached)* Curve #1 and #2 are to be used to determine Core Spray and RHR NPSH Limits. Both of these Curves are on a per **pump** basis NOT per loop. Given Core Spray **Loop II (3B/3D)** flow is 4000 gpm, 2000 gpm **per** pump would be used with Curve #1. Therefore, the Curve #1 intersecting operating point of Suppression Chamber Pressure of 5 psig and Suppression Pool Temperature of 200 ºF **IS** within the safe NPSH region (below the 5 psig Safe * line). Given RHR **Pump** 3A flow is 11,500 gpm, the Curve #2 intersecting operating point is **NOT** within the safe NPSH region (above the 5 psig Safe * line).

D INCORRECT: Incorrect plausible in that Core Spray Loop II (3B/3D) Pumps DO have adequate NPSH *(See C).* However, RHR 3A does NOT *(See B/C).*

RO Level Justification: Tests the candidate's knowledge and ability to use given plant conditions to determine operational implications related to NPSH Limit for ECCS pumps, specifically Core Spray. 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 related to EOIs for CS/RHR operations. This requires mentally using specific knowledge and its meaning to predict the correct outcome. Graphs must be read and data interpolated to derive an answer relative to given 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-1, Rev. 16 (Attach if not previously provided)
3-EOI-1A, Rev. 6
3-EOI-5, Rev. 5
OPL171.201, Rev. 10U2
EOIPM Sect:
0-V(B), Rev. 3
0-II(G), Rev. 0
0-V(G), Rev. 2
3-ARP-9-3B, Rev. 25
Unit 3 Tech Spec 3.6.2.2, Amend. 212

Proposed references to be provided to applicants during examination: 3-EOI-5 ONLY Curves #1 (CS NPSH Limits) and #2 (RHR NPSH Limits) – pictures only

Learning Objective: OPL171.201, Obj. 11, 12 (As available)

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

Question Source:

	Bank #	BFN NRC 1804 #15
	Modified Bank #	
	New	

(Note changes or attach parent)

Question History: Last NRC Exam 2018

Question Cognitive Level: Memory or Fundamental Knowledge

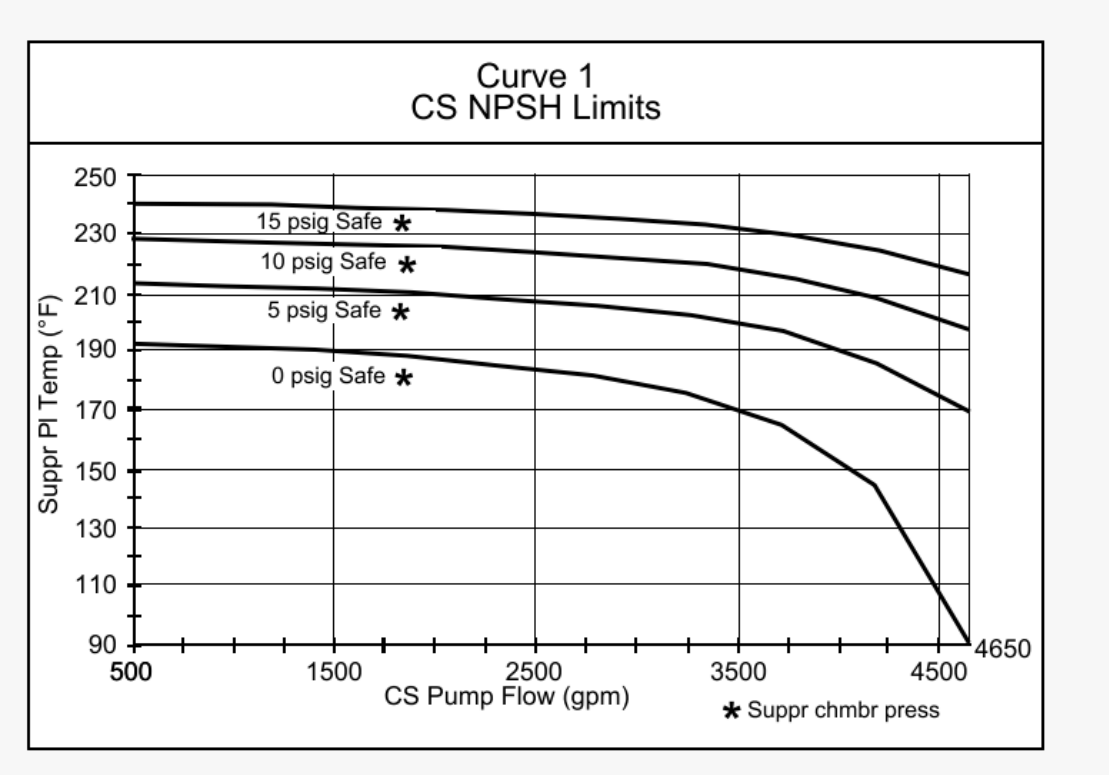
Comprehension or Analysis **X**

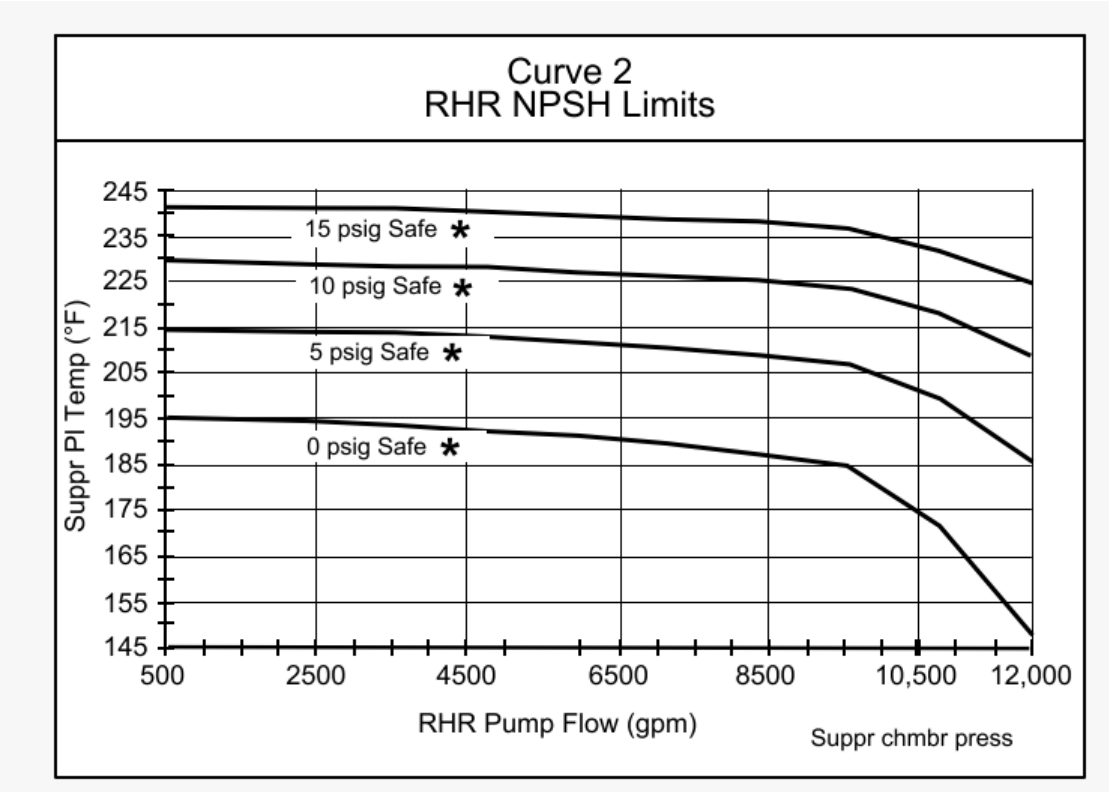
10 CFR Part 55 Content: 55.41 **X**

55.43

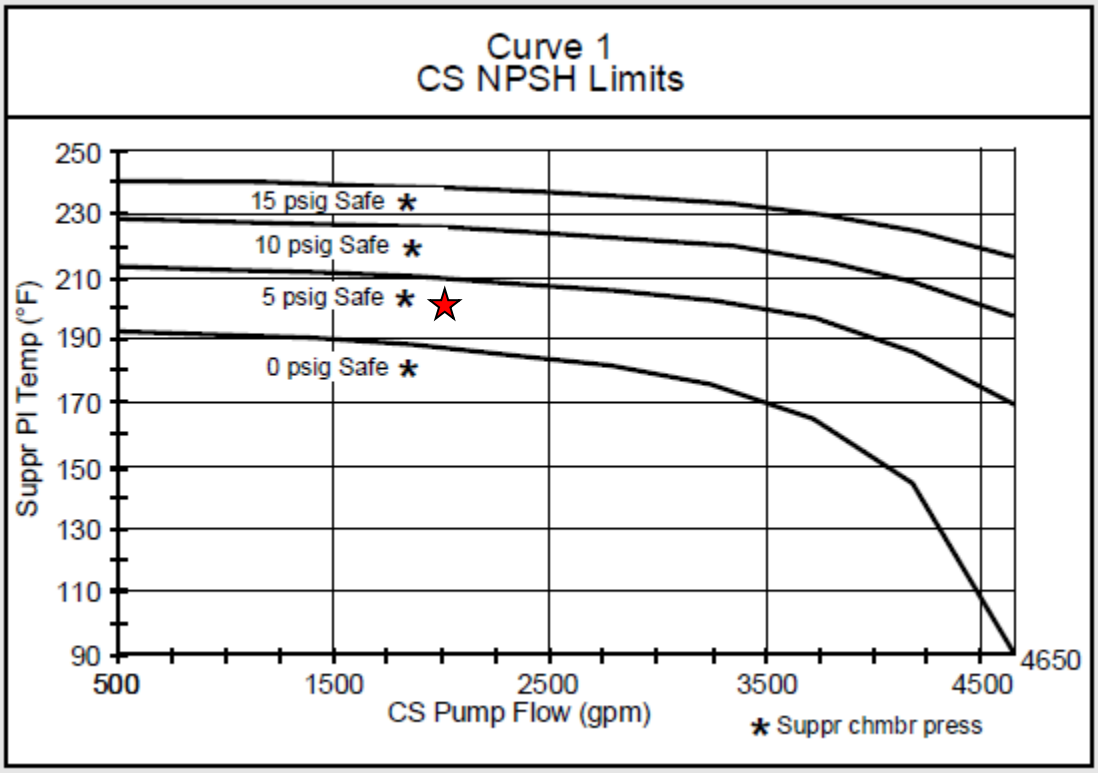
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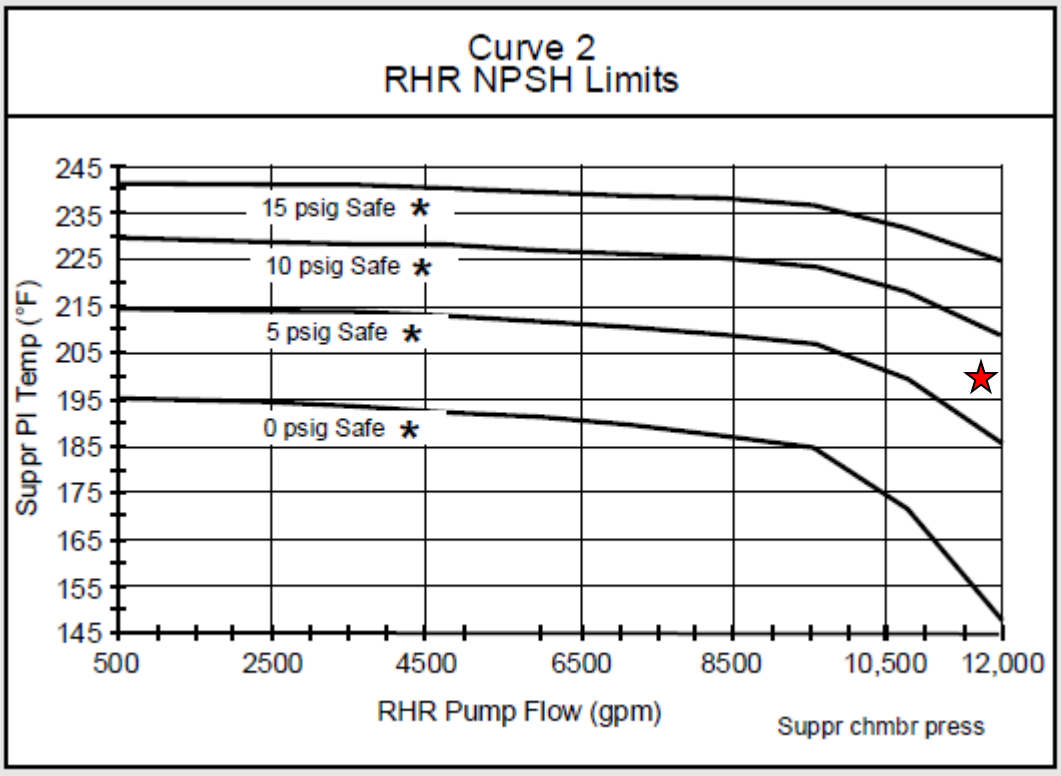
REFERENCE PROVIDED to candidate:





REFERENCE PROVIDED (KEY):





Copy of Bank Question:

HLT 0810 Written Exam

Question: **# 15**

Unit 3 has experienced a LOCA **AND** the following conditions exist:

- Suppression Pool Level is (-) 5.5 inches
- Suppression Chamber Pressure is 5 psig
- Drywell Pressure is 10 psig
- Suppression Pool Temperature is 200° F
- RHR Pump 2A flow is 11,500 gpm
- Core Spray Loop II flow is 4,000 gpm
- **NO** other ECCS Pumps are running

Based on the above conditions, which ONE of the following identifies the ECCS Pump(s), if any, that has (have) sufficient NPSH for continued operation?

[REFERENCE PROVIDED]

A. **NONE**

B. RHR Pump 2A **ONLY**

C. Core Spray Loop II Pumps **ONLY**

D. Core Spray Loop II Pumps **AND** RHR Pump 2A

Proposed Answer: C

Excerpt from 3-ARP-9-3B:

BFN Unit 3	Panel 9-3 3-XA-55-3B	3-ARP-9-3B Rev. 0025 Page 18 of 38

SUPPR CHAMBER
WATER LEVEL
ABNORMAL
3-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. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** 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** 3-TR-64-161 and -162.

D. **REFER TO** 3-OI-74, 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** (otherwise N/A)
ENTER 3-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 3-EOI-2 entry is appropriate.
2. **RECORD** actions in NOMS log.

References: 3-45E620-3 3-47E610-64-1 GE 730E933-1
Technical Specifications 3.6.2.2

Excerpt from Unit 3 Tech Spec 3.6.2.2:

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ -6.25 inches with and -7.25 inches without differential pressure control and ≤ -1.0 inches.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Suppression pool water level not within limits.	A.1	Restore suppression pool water level to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

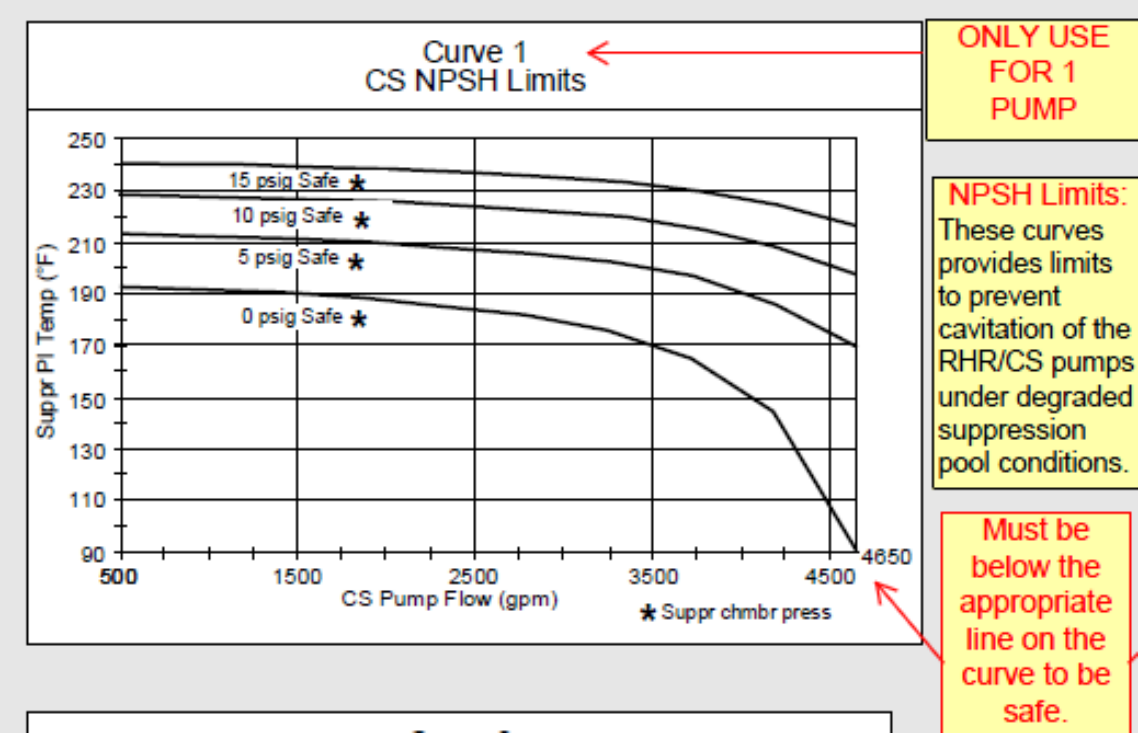
Excerpts from EOI-1 and EOI-1A Tables referencing Caution #2 in 3-EOI-5:

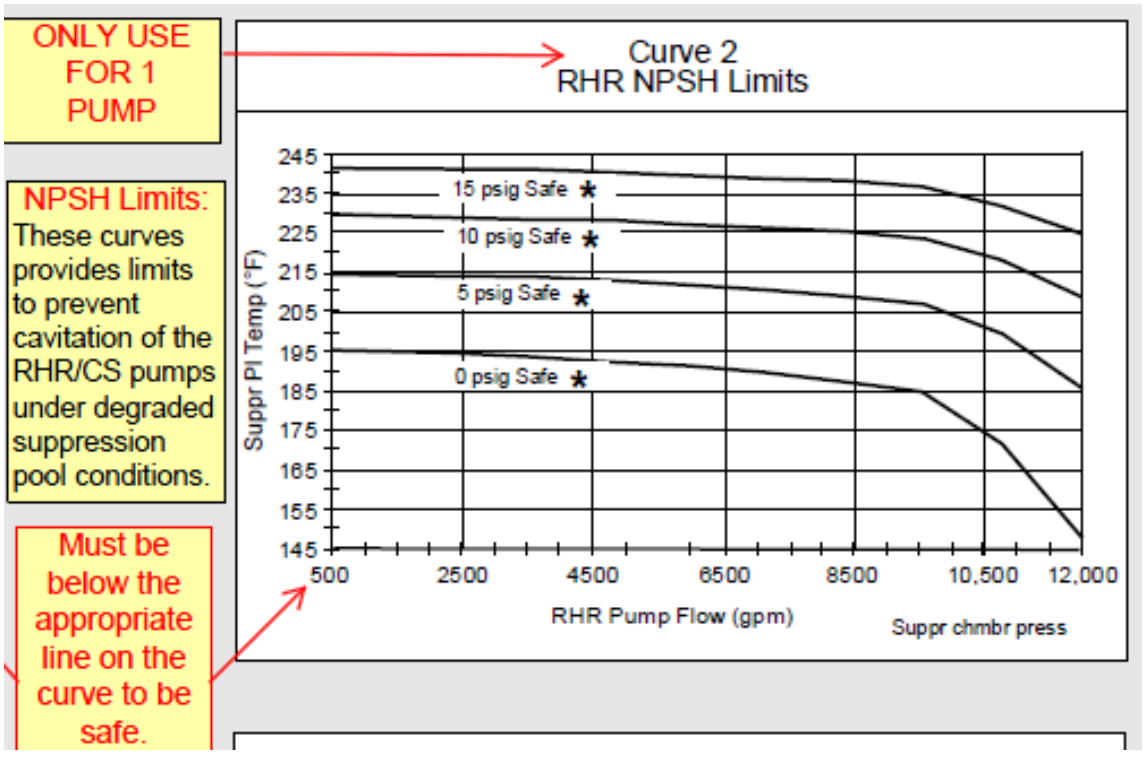
Table L-3
Preferred ATWS Injection Systems

SOURCES	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available 2 3 6	5C, 20M	1200 psig
HPCI with CST suction if available 2 6 7	5D, 20N	1200 psig
CNDS	6A	410 psig
LPCI 2	6B, 6C	320 psig
SLC (boron tank)	7B	1450 psig
Table L-2 systems or CS ONLY IF Step ARC/L-19 has been previously performed	----	----

2 Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1, 2, 9 or 10) or with suppression pool water level below 10 ft (Vortex limit) may cause equipment damage

Excerpt from 3-EOI-5 (Training Mark-Up):





Excerpt from EOIPM for Cautions Bases:

BFN Unit 0	EOI Curves, Cautions, Tables and Variables Bases	EOIPM Section 0-V(B) Rev. 0003 Page 28 of 149

3.2 EOI Caution #2



Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1, 2, 9 or 10) or with suppression pool water level below 10 ft (Vortex limit) may cause equipment damage

The NPSH (Net Positive Suction Head) limits are defined to be the highest suppression pool temperature which provides adequate net positive suction head for pumps taking suction on the pool. The NPSH Limits are functions of pump flow and suppression chamber overpressure (airspace pressure plus the hydrostatic head of water over the pump suction). It is utilized to preclude pump damage from cavitation.

Excerpt from EOIPM for Curves Bases:

BFN Unit 0	**PSTG/SATG Introduction and References Bases**	**EOIPM Section 0-II(G) Rev. 0000 Page 156 of 179**

8.25 NET POSITIVE SUCTION HEAD LIMIT

The Net Positive Suction Head (NPSH) Limit is the highest suppression pool temperature which provides adequate net positive suction head for RCIC or an ECCS pump taking suction on the pool. The NPSH Limit is a function of pump flow and suppression chamber overpressure (airspace pressure plus hydrostatic head over the RCIC or ECCS suction). Separate limits are provided for RCIC and each ECCS.

Excerpts from EOIPM for Primary Containment Bases:

BFN Unit 0	**EOI-2, Primary Containment Control Bases**	**EOIPM Section 0-V(G) Rev. 0002 Page 95 of 129**

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SP/T Flowpath

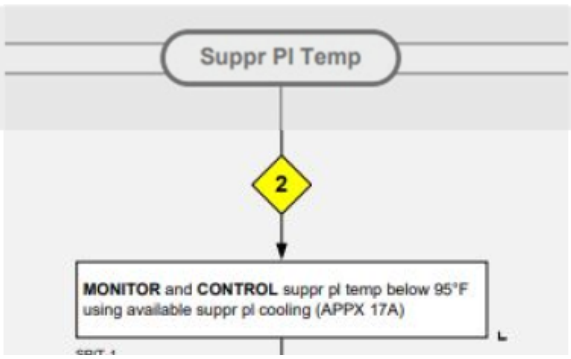
The Suppression Pool Temperature Control flowpath specifies operator action to control and maintain suppression pool temperature. Failure to maintain suppression pool temperature within prescribed limits could result in any of the following:

- Loss of pressure suppression function of primary containment
- Exceeding suppression chamber design temperature limits
- Insufficient NPSH limits for pumps that take suction on the suppression pool

BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0002 Page 96 of 129

1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

Caution #2 and SP/T-1



Excerpts from OPL171.201 Lesson Plan:

instrument run temperatures with actual RPV water level at the variable leg tap. The Maximum Run Temperature (MRT) is the instrument run temperature that would produce an on-scale indication with actual water level at the variable leg RPV tap. The MIL and MRT are utilized in establishing the conditions under which an RPV water level instrument may be used to determine RPV water level. Separate levels and temperatures are provided for each RPV water level instrument.

e) The second bullet of Caution #1 identifies the limiting conditions beyond which boiling of the water in the instrument legs may occur.

f) The RPV Saturation Temperature is a plot of the saturation temperature of water as a function of pressure. If the temperature of the water in an RPV water level instrument run exceeds this temperature, the water may start to boil, resulting in unreliable level indication.

g) Boiling is of concern in all instrument runs—horizontal and vertical, reference and variable, drywell and secondary containment. Boil off from the reference leg reduces the height of water in the leg. This decreases the pressure on the reference leg side of the differential pressure transmitter and increases the indicated level. Boiling in the variable leg increases the pressure on the variable leg side of the transmitter, likewise increasing the indicated level. Continued boiling would produce an increasing level trend.

h) Boiling and loss of valid level indication may not occur immediately when drywell or secondary containment area temperature exceeds Figure 8, RPV Saturation Temperature. As drywell or secondary containment area temperature increases, the temperatures in the instrument runs will lag the temperature in the surrounding area. The temperature of the water in the instrument run may therefore be lower than the indicated air temperatures.

2. Caution 2

Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1, 2, 9, or 10) or with suppression pool water level below 10 ft (vortex limit) may cause equipment damage.

a) The NPSH (Net Positive Suction Head) limits are defined to be the highest suppression pool temperature which provides adequate net positive suction head for pumps taking suction on the pool. The NPSH Limits are functions of pump flow and suppression chamber overpressure (airspace pressure plus the hydrostatic head of water over the pump suction).

b) The vortex limit is defined to be the lowest suppression pool water level above which air entrainment is not expected to occur in pumps taking suction on the pool. Exceeding the limit can lead to air entrainment at the pump suction strainers.

Vortex Limit = 10 ft.

containment, transferring water from the suppression pool may be unnecessary in the short term.

- If suppression pool water level is approaching the Maximum Pressure Suppression Primary Containment Water Level, it may be necessary to operate suppression pool pump down systems even if such action may increase area radiation levels, require personnel exposure, or result in radioactivity release outside the primary containment.

G. EOI Curves

Located on EOI-5

Objective 12

Objective 13

Various curves are used throughout the EOIs to define regions of safe operation or to define limits for specific equipment operation. Regions of safe operation are cross-hatched in green, while unsafe regions (those where action is required) are cross-hatched in red.

Each place a curve is used in the EOIs, it will be referenced from the step which it applies.

Each curve is numbered sequentially (1 through 10), retaining the same number throughout its use in the EOIs. For example, Curve #3 in EOI-1, RPV Control, is the same as Curve #3 in EOI-2, Primary Containment Control.

1. **Curve 1**: CS NPSH Limits

Curves 1 and 2 are on a per pump basis, NOT per loop.

a) This curve provides limits to prevent cavitation of the Core Spray pumps under degraded suppression pool conditions.

b) The NPSH limit curves show the maximum pump flow that maintains adequate NPSH for a range of suppression chamber overpressure conditions with elevated suppression pool temperatures.

c) At flows above the curve the pump will lose NPSH. If cavitation is observed, reduce flow. For overpressures between curves, estimate the correct limits and monitor for cavitation.

2. **Curve 2**: RHR NPSH Limits

Same as Curve 1 except applied to RHR pumps.

3. **Curve 3**: Heat Capacity Temp Limit

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which emergency RPV depressurization will not raise Suppression Chamber pressure above the Primary Containment Pressure Limit (56 psig) while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

a) For RPV pressures above the Minimum Decay Heat Removal Pressure, the HCTL ensures that there is sufficient heat capacity in the suppression pool to sustain a blowdown. If a blowdown is initiated from any point below a specific RPV pressure curve. RPV pressure will drop below the minimum decay Heat Removal Pressure before suppression

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

Examination Outline Cross-reference:

256000 (SF2 CDS) Condensate

A1.08 (10CFR 55.41.5)

Ability to predict and/or monitor changes in parameters associated with operation of the CONDENSATE SYSTEM including:

- System water quality

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	256000A1.08	
Importance Rating	2.8	------

Proposed Question: **# 59**

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



- CONDENSATE DEMIN ABNORMAL (2-9-6B, WINDOW 6) alarms
- Operator reports that 2-FCV-2-130, CONDENSATE DEMINERALIZER BYPASS VALVE, is OPEN
- **NO** Operator actions have been taken

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

In accordance with 2-OI-2, Condensate System, 2-FCV-2-130 automatically opens on high **(1)**.

Given that 2-FCV-2-130 is OPEN, Condensate effluent conductivity will **(2)**.

A. (1) differential pressure **ONLY**
(2) rise

B. (1) differential pressure **ONLY**
(2) lower

C. (1) differential pressure **AND** conductivity
(2) rise

D. (1) differential pressure **AND** conductivity
(2) lower

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 2-OI-2, 2-FCV-2-130 automatically opens when the demineralizer system dP reaches 50 psid. Second part, when Condensate Filter/Demineralizers are bypassed, condensate conductivity will rise due to lack of filtration. Normally the Condensate Filter/Demineralizers treat the full flow of condensate from the Main Condenser hotwell to remove dissolved and suspended solids from corrosion in the condenser and associated piping systems.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible given that when 2-FCV-2-130 is open the differential pressure across the condensate demineralizers will lower, and condensate system conductivity will rise.

C INCORRECT: First part is incorrect but plausible in that 2-ARP-9-6B, Window 6, CONDENSATE DEMIN ABNORMAL, states that the probable cause is any alarm condition at condensate demineralizers such as high DP or conductivity. 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 predict and/or monitor changes in parameters associated with operation of the Condensate System including system water quality. This question is rated as C/A due to the integrated aspects of the question to predict an outcome when given specific conditions. This requires mentally using specific knowledge with parameter and system monitor and controls and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-2, Rev. 123 (Attach if not previously provided)
2-ARP-9-6B, Rev. 22
OPL171.001, Rev. 19U2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.001 Obj. 9 (As available)

Question Source: Bank # _____
Modified Bank # QC NRC 2020 #62 (Note changes or attach parent)
New _____

Question History: Last NRC Exam 2020

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:

Test Answer Key

2020 QDC NRC Exam RO Section
Test ID: 331006

62 **ID: 2121296** **Points: 1.00**

Unit 1 is at 100% power.

MO 1-3303, COND DEMIN BYPASS VLV, is opened.

How will the following condensate system parameters respond?

1. PI 1-3340-49, COND BOOST PMP SUCT PRESS, on the 901-6 panel.
2. CR 1-3340-8, CONDENSATE CONDUCTIVITY, on the 901-7 panel.

A. (1) Rise
(2) Rise

B. (1) Rise
(2) Lower

C. (1) Lower
(2) Rise

D. (1) Lower
(2) Lower

Answer: A

Test Answer Key

2020 QDC NRC Exam RO Section
Test ID: 331006

Answer Explanation
Answer Explanation: If the Condensate Demin bypass valve is opened, d/p across the demineralizers will drop. Since there is less d/p across the demins, Condensate Booster pump suction pressure will rise. When the demins are bypassed, total dissolvable solids (TDS) will rise and conductivity will increase. Distractor 1: Plausible but incorrect because the presence of TDS would not raise the resistivity of a solution. Therefore, conductivity would actually Rise. Distractor 2: Plausible but incorrect because the bypass valve is NOT downstream of the Condensate Booster pumps. Distractor 3: Plausible but incorrect, Combination of Distractor 1 and 2. Reference: QCAN 901(2)-7 H-12 Rev 4 Reference provided during examination: None. Cognitive level: Higher Level (RO/SRO): RO Tier: 2 Group: 2 Question Source: Quad Cities Exam Bank (QDC.ILT.686417) Question History: 2009 ILT NRC Exam 10 CFR Part 55 Content: 41.5/45.5 KA: 256000, A1.08 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: System water quality. (RO=2.7) Comments:

Excerpt from 2-OI-2:

BFN Unit 2	Condensate System	2-OI-2 Rev. 0123 Page 12 of 138

3.2 Condensate Demins (continued)

E. CONDENSATE DEMIN ABNORMAL alarm will annunciate in the main control room when Condensate Demineralizer System differential pressure rises to approximately 34 psid. The setpoint for this alarm will alert the operator that the plant is in a condition that could result in a scram if they subsequently lose an operating Condensate pump. (DCN W26496)

F. CONDENSATE DEMIN BYPASS VALVE, 2-FCV-2-130, automatically opens when the demineralizer system dP reaches 50 psid, and fails open on loss of control air (≤ 50 psig).

G. If flow is lost through any Condensate Demineralizer and the holding pump fails to start, it should NOT be placed back in service until the Condensate Demineralizer is backwashed and precoated.

H. If the Condensate Demins are to be left out of service for extended periods of time, they should be left backwashed and filled.

3.3 Condensate Pumps

A. Prior to starting a Condensate Pump, a minimum of three Condensate Demins should be backwashed, precoated, and ready for service per pump.

B. During normal steady state operations, the normal maximum line current to Condensate Pump Motors should not exceed 161 amps (bottom of yellow band).

This limit may be exceeded during short term operation under the following conditions:

1. Operation with two Condensate Pumps at high power or
2. During periods of high river temperature.

Do not exceed 184 amps (lower end of red band). During the time that the Condensate Pump Motors are operating above the normal maximum line current values (in the yellow band); the operators should request assistance from Site Engineering and monitor pump motor parameters more frequently.

Excerpt from 2-ARP-9-6B:

BFN Unit 2	Panel 9-6 2-XA-55-6B	2-ARP-9-6B Rev. 0022 Page 9 of 38

CONDENSATE DEMIN ABNORMAL
2-XA-2-255
6

(Page 1 of 1)

Sensor/Trip Point:

TMR 24030	Normally energized, de-energize on loss of PLC power.
TMR 24033	(re-flash) normally de-energized, energize for re-flash.
CR 24004	Normally energized, de-energize to alarm.

Sensor Location: Condensate Demin Control Panel
Elevation 557'
Turbine building

Probable Cause: Any alarm condition at condensate demin such as high DP or conductivity.

Automatic Action: None

Operator Action:
A. **CHECK** system flow conductivity, and system DP, on Panel 2-9-6 for abnormalities.
B. **REQUEST** personnel determine problem with condensate demineralizers.
C. **IF** High Conductivity exists, **THEN** **REFER TO** 2-AOI-2-1.

References: 2-45E620-8 Graver T51046 Series

Excerpt from OPL171.001 Lesson Plan:

b) Silica in reactor water is regularly monitored because silica offers a convenient parameter for monitoring demineralizer break-through.
   (1) The chemical characteristics of anion demineralizer resins allow silica to break through before any other anions break through.

7. Condensate and Feedwater

ILT Objective 9
NLO Objective 6

a) Impurities that were present in the Feedwater remain behind in the reactor water when it is turned to steam.

ILT Objective 11
NLO Objective 7

b) The only removal methods are as follows:
   (1) Carryover (negligible)
   (2) The Reactor Water Cleanup System (flow is 0.9% of the Feedwater flow).

OF-5, Understanding Plant Design - Corrosion products tend to plate out on low-flow area piping and will accumulate there. Noble Metals and Zinc injection are designed to help maintain some of these corrosion products in solution, however, this is why RWCU is not effective at removing corrosion products

c) Corrosion products (Fe and Cu) do not follow this same rule because they do not stay suspended or dissolved in the reactor water.
   (1) Corrosion products disappear from the reactor water quickly.
   (2) About 90% deposit on surfaces, such as pipe walls, fuel, other components, and crud traps
   (3) Only about 10% of the input remains suspended long enough to be removed by the RWCU System.
   (4) The importance of this is that the levels of ionic materials can be measured by conductivity and are related to the input rate, but the corrosion products being put into the system (crud) cannot be measured in the reactor water. The input of metals into the reactor must be measured in the Feedwater System.

d) To meet the normal operating limit of 200 ppb for chlorides in the reactor, the Feedwater must contain 2 ppb or less.

e) Metals analysis done on the Feedwater is carried out through the use of ion exchange / particulate filter paper.

f) Good Practice Values
   (1) < 0.05 ppb for Cu
   (2) $\leq$ 1.0 ppb for Fe

g) Action Level Limits
   (1) >0.2 ppb for Cu
   (2) > 5.0 ppb for Fe

h) Oxygen concentrations are measured by grab measurements or on-line instrumentation.

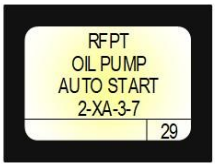
i) The object of the Condensate Filter/Demineralizers is to treat the full flow of condensate from the Main Condenser hotwell to remove dissolved and suspended solids which result from corrosion in the condenser and associated piping systems and from leakage of cooling water into the Main Condenser

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

Examination Outline Cross-reference:	Level	RO	SRO
259001 (SF2 FWS) Feedwater	Tier #	2	------
K6.09 (10CFR 55.41.7)	Group #	2	------
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the FEEDWATER SYSTEM:	K/A #	259001K6.09	
• Reactor feedwater pump lube oil system	Importance Rating	3.1	------

Proposed Question: **# 60**

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



- RFPT OIL PUMP AUTO START (2-9-6B, Window 29) alarms
- Operator reports 2A RFPT Emergency Bearing Oil Pump (EBOP) just started

In accordance with the given ARP, which **ONE** of the following completes the statements below?

The **(1)** powered 2A RFPT EBOP started due to low RFP Turbine Bearing Oil Pressure **(2)**.

A. (1) AC
(2) from both companion Main Oil Pumps tripping **ONLY**

B. (1) AC
(2) ≤ 36 psig **OR** both companion Main Oil Pumps tripping

C. (1) DC
(2) from both companion Main Oil Pumps tripping **ONLY**

D. (1) DC
(2) ≤ 36 psig **OR** both companion Main Oil Pumps tripping

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that numerous power supplies to multitudes of plant equipment are easily and often confused. Each RFPT has 2 AC motor driven lube and control oil pumps (MOP), example-1A1,1A2 MOPs. Second part is incorrect but plausible in that **one** start signal for RFPT EBOP is both main oil pumps trip. This is not the only start signal to the EBOP.

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*) The 2A RFPT EBOP is DC powered from Battery Board 5. Second part, in accordance with 2-ARP-9-6B, Window 29, Standby EBOP will auto start if Turbine Bearing Oil pressure is ≤ 36 psig **OR** if both companion Main Oil Pumps trip.

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

RO Level Justification: Tests the candidate's knowledge of the effect Reactor Feedwater Pump lube oil system or component malfunctions on the FEEDWATER SYSTEM. This question is rated as Memory due to the requirement to strictly recall procedural facts related to RFWPT System.

Technical Reference(s): 2-ARP-9-6B, Rev. 22 (Attach if not previously provided)
OPL171.026, Rev. 25U1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.026, Obj.3g (As available)

Question Source:

Bank #	
Modified Bank #	OPL171.026-09 005 #1072
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:

1072. OPL171.026-09 005

Unit 2 is at 100% power.

The RFPT Emergency Bearing Oil Pump will auto start with its handswitch in AUTO when _____.

A. ONLY one companion Main Oil Pump trips OR Turbine Bearing Oil pressure lowers to ≤ 36 psig

B✔ both companion Main Oil Pumps trip OR Turbine Bearing Oil pressure lowers to ≤ 36 psig

C. ONLY one companion Main Oil Pump trips OR Turbine Bearing Oil pressure lowers to ≤ 175 psig

D. both companion Main Oil Pumps trip OR Turbine Bearing Oil pressure lowers to ≤ 175 psig

Excerpts from OPL171.026 Lesson Plan:

operation of the 86 PGA (fail to engage relay which trips and locks out the TG until reset).

(5) ON-INJ BYPASS - This switch position will start the turning gear motor without the requirement of>17 psid injection water pressure. Only pump and turbine bearing oil pressure >7 psig are needed to start the motor.

c) Turning Gear Indicating Lights

(1) GREEN - The green light will be lit if the hand-switch is not in OFF and ALL of the above permissives for TG operation in the selected switch position are met.

(2) RED - The red light will be lit when the TG motor is running

(3) AMBER - The amber light will be lit when the TG is engaged.

(4) BLUE - The blue light will be lit if the turbine speed is <4 rpm as sensed by the Low Speed Magnetic Pickup Unit. (MPU)

8. Control and Lubricating Oil System

Figure-4

Obj. ILT-3.e
Obj. ILT-2.c
Obj. ILT-3.f
Obj. ILT-3.g
Obj. NLOR-4
Obj. NLO-5

a) Each RFPT has two 100% capacity AC motor driven lube and control oil pumps (MOP) along with a DC motor driven emergency bearing lube oil pump (EOP).

(1) Power Supplies:

AC Pump 1A1 - 480V Unit Bd. 1A
AC Pump 1A2 - 480V TMOV Bd 1B
DC Pump 1A3 - Battery Bd 5

AC Pump 1B1 - 480V Unit Bd.
1B AC Pump 1B2 - 480 V TMOV Bd. 1A
DC Pump 1B3 - Battery Bd 4

AC Pump 1C1 - 480V TMOV Bd. 1A

AC Pump 1C2 - 480V TMOV Bd. 1B
DC Pump 1C3 - Battery Bd 6

AC Pump 2A1 - 480V Unit Bd. 2A
AC Pump 2A2 - 480V TMOV Bd 2B
DC Pump 2A3 - Battery Bd 5

AC Pump 2B1 - 480V Unit Bd. 2B
AC Pump 2B2 - 480 V TMOV Bd. 2A

DC Pump 2B3 - Battery Bd 4

AC Pump 2C1 - 480V TMOV Bd. 2A
AC Pump 2C2 - 480V TMOV Bd. 2B
DC Pump 2C3 - Battery Bd 6

AC Pump 3A1 - 480V Unit Bd. 3A
AC Pump 3A2 - 480V TMOV Bd 3B
DC Pump 3A3 - Battery Bd 5

AC Pump 3B1 - 480V Unit Bd. 3B
AC Pump 3B2 - 480V TMOV Bd. 3A
DC Pump 3B3 - Battery Bd 4

AC Pump 3C1 - 480V TMOV Bd. 3A
AC Pump 3C2 - 480V TMOV Bd. 3B
DC Pump 3C3 - Battery Bd 6

(2) Any lube oil pump AC driven (MOP) or DC (EOP) driven automatically starting will annunciate a single RFPT alarm; (XA-55-6B 'RFPT Oil Pump Auto Start'). This alarm will reset after the affected pump motor control room handswitch is placed in 'Auto after Start'.

Obj. ILT-4.b/c
Obj. NLOR-5.a/b
Obj. NLOR-6.a/b

(3) The AC oil pumps and DC oil pump handswitches will spring return to 'Auto' from the 'Start' and 'Stop' positions.

Obj. NLO-6.a/b

(4) Each switch will have a 'Pull to Lock' position, but ONLY the DC oil pump switch will alarm when placed in that position.

(5) Each oil pump has a 'Test' push-button with 'Amber & Red' lights above. The test push- button will simulate a low pressure condition in the common discharge header. The amber light will come on provided the pump has developed sufficient discharge pressure AND the main handswitch has not been taken to 'START'.

(6) As soon as the main handswitch is taken to 'START', the light will go out. The red light will be on when pump discharge pressure is above its setpoint of 125 psig.

b) RFPT oil tanks are 1000 gal. Capacity each, filtered by duplex strainers located vertically against the tank wall in the pit. HP fire protection is provided for each of the oil tanks.

c) Two 100% capacity oil coolers are located in the pit under the cat-walk. These straight-tube type coolers are aligned for

Excerpts from 2-ARP-9-6B:

BFN Unit 2	Panel 9-6 2-XA-55-6B	2-ARP-9-6B Rev. 0022 Page 32 of 38

RFPT
OIL PUMP
AUTO START
2-XA-3-7
29

(Page 1 of 1)

Sensor/Trip Point:
2-RLY-003-1007(2A1 PUMP)
2-RLY-003-1008(2A2 PUMP)
2-STR-003-0102(2A3 PUMP)
2-RLY-003-1037(2B1 PUMP)
2-RLY-003-1038(2B2 PUMP)
2-STR-003-0128(2B3 PUMP)
2-RLY-003-1067(2C1-PUMP)
2-RLY-003-1068(2C2 PUMP)
2-STR-003-0153(2C3 PUMP)

Any RFPT Lube Oil Pump automatically starting (pump motor breaker closed with handswitch in Auto after Stop).

Sensor Location: Main Control Room
Panel 2-9-6

Probable Cause:
A. RFPT AC Oil Pump auto started and/or RFPT DC Oil Pump auto started

Automatic Action:
A. Amber and Red Lights illuminate above auto started oil pump(s) test pushbutton. The amber light indicates a pump that auto started.
B. Standby Main Oil Pump will auto start with handswitch in AUTO when any of the following occur:
- Low discharge pressure ≤ 175 psig.
- Companion running Main Oil Pump tripped.
- Electrical Board undervoltage for companion running Main Oil Pump.

C. Standby EBOP will auto start with handswitch in AUTO when any of the following occur:
- Turbine Bearing Oil pressure low ≤ 36 psig.
- Both companion Main Oil Pumps tripped.

Operator Action:
A. **ENSURE** standby oil pump(s) running.
B. **PLACE** handswitch(es) for oil pump(s) that auto started to START AND **RETURN** to AUTO after START,
C. **CHECK** annunciation resets.
D. **DISPATCH** personnel to the affected RFPT to PERFORM the following:
- **CHECK** Bearing Oil pressure ≥ 125 psig.
- **CHECK** for possible leaks in RFPT oil system.
- **INSPECT** tripped oil pump (if a pump has tripped).

E. **REFER TO** 2-OI-3.

References: 2-47E803, 2-45E612-2,8, 2-45E620-8, 0-509E603AD-1, 2-SIMI-3B

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

Examination Outline Cross-reference:	Level	RO	SRO
268000 (SF9 RW) Radwaste	Tier #	2	------
K3.04 (10CFR 55.41.5)	Group #	2	------
Knowledge of the effect that a loss or malfunction of the RADWASTE SYSTEM will have on the following systems or system parameters:	K/A #	268000K3.04	
• Primary containment drain sumps	Importance Rating	3.0	------

Proposed Question: **# 61**

Unit 1 is operating at 100% RTP with the following conditions:

- At 0800, in accordance with 1-OI-64, Primary Containment System, the crew is manually pumping down Drywell Floor Drain Sump to Radwaste
- Operator reports 1-HS-77-2B, DW FLOOR DRAIN OUTBOARD ISOLATION VALVE, failed CLOSED and will **NOT** OPEN

NOTE: CONSIDER EACH STATEMENT BELOW SEPARATELY.

Which **ONE** of the following completes the statements below?

Given the conditions above, the Operator is unable to calculate the **(1)** leakage.

If OPEN, 1-FCV-77-2B **(2)** close on a PCIS Group signal.

A. (1) identified
(2) will

B. (1) identified
(2) will NOT

C. (1) unidentified
(2) will

D. (1) unidentified
(2) will NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that numerous sources other than the Drywell floor drains are captured by the Drywell floor drain sump such as leakage from Control Rod drives, valve flanges or packings, RBCCW and Drywell coolers. The Drywell floor drains and equipment drains relation to identified or unidentified leakage is often and easily confused among candidates. 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 normal position of 1-FCV-77-2B is closed, therefore the candidate does not see this valve change state during a PCIS Group 2 isolation.

C **CORRECT**: (*See attached*) Unidentified leakage consists of the Drywell floor drains and other Drywell sources that are not collected into the Drywell equipment drain sumps that is sent to Radwaste. In accordance with 1-OI-64, Primary Containment System, 6.4-Manually Pumping Down Drywell Floor Drain Sump, 1-HS-77-2B, DW FLOOR DRAIN OUTBOARD ISOLATION VALVE, is required to be OPEN. Second part, 1-FCV-77-2B closes on a Group 2 PCIS signal.

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 that a loss or malfunction of the RADWASTE SYSTEM will have on primary containment drain sumps. This question is rated as Memory due to the requirement to strictly recall procedural facts in relation to containment isolations.

Technical Reference(s): 1-OI-64, Rev. 32 (Attach if not previously provided)
OPL171.016, Rev 27
OPL171.017, Rev 22U1
TSB 3.4.5, Rev 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.016, Obj. 8 (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 1-OI-64:

6.4 Manually Pumping Down Drywell Floor Drain Sump

NOTE

This section is performed at Panel 1-9-4 in Unit 1 Control Room.

[1] **REVIEW** Precautions and Limitations. Reference Section 3.0.

[2] **ENSURE OPEN** the following valves:

- DW FLOOR DRAIN OUTBD ISOL VALVE, 1-HS-77-2B
- DW FLOOR DRAIN INBD ISOL VALVE, 1-HS-77-2A

[3] **PLACE** in START RETURN TO AUTO one of the following pumps:

- DW FLOOR DRAIN SUMP PUMP 1A, 1-HS-77-1A-A

OR

- DW FLOOR DRAIN SUMP PUMP 1B, 1-HS-77-1B-A

[4] **ENSURE** pump started in Step 6.4[3] auto stops on sump low level.

[5] **CLOSE** DW FLOOR DRAIN OUTBD ISOL VALVE, 1-HS-77-2B

Excerpt from OPL171.016 Lesson Plan:

(e) The sump pump flow rates are adjusted to 30-49 gpm by throttling local manual valves in the sump pump discharge lines. These manual valves are located on top of the torus in the NE quadrant.

(f) The equipment and floor drain sump pumps A/B receive power from 480V RMOV Board A/B respectively.

(6) DW FD & DW ED sumps isolate on PCIS signals.

(a) D/W floor drain inboard/outboard isolation valve FCV-77-2A/B

(i) Inboard Normally Open. Outboard Closed.

(ii) Fails closed

(iii) Closes on group 2 isolation

(b) D/W equipment drain inboard/outboard isolation valve FCV-77-15A/B

(i) Inboard Normally Open. Outboard Closed. NLO-16

(ii) Fails closed

(iii) Closes on group 2 isolation

Note: DCN 70958 changed the position of the Outboard Valves to CLOSED due to siphoning of the sumps. The alarm normally associated with these valves being closed only alarms now when a PCIS Group 2 isolation occurs. (1/2/3-ARP-9-4C Windows 22 and 23)

(c) D/W equipment drain recirc/discharge valve FCV-77-14A/B

(i) Normally open

(ii) Fails open

(iii) Controlled by hand-switch 2-HS-77-14C on panel 9-4

(7) Flow from either sump is recorded on flow recorder FR-77-6/16. Equipment drain flow is indicated on a Westronics. Range is 0 to 150 gpm.

(8) Sump Pump operation

(a) Manually operated from control room panel 9-4 or from the Rad-waste control panel.

(i) The hand switches have a red running light, a green stopped light, and an amber light in the control room and Rad-waste.

(ii) If placed in the pull to lock position the amber light for the sump pump hand-switch will be illuminated.

(iii) If either hand-switch for a sump pump is placed in the pull to lock and the automatic control is inhibited. However, the manual operating capability is still available from the control room.

Excerpt from OPL171.017 Lesson Plan:

OPL171.017, PRIMARY CONTAINMENT ISOLATION SYSTEM, Rev 22U1

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

Excerpt from TS Bases 3.4.5:

BASES

BACKGROUND (continued)

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Reactor Building Closed Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell floor drain sump has transmitters that supply level indications locally.

The floor drain sump level indicators have switches that start and stop the sump pumps when required. A timer starts each time the sump is pumped down to the low level setpoint. If the sump fills to the high level setpoint before the timer ends, an alarm sounds in the control room, indicating a LEAKAGE rate into the sump in excess of a preset limit.

A flow transmitter in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room. The pumps can also be started from the control room.

The primary containment air monitoring systems continuously monitor the primary containment atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The primary containment atmosphere particulate and gaseous radioactivity monitoring systems are not capable of quantifying LEAKAGE rates, but are sensitive enough to indicate increased LEAKAGE rates. This system is capable of detecting radiation levels in containment atmosphere of three times background (Ref. 2).

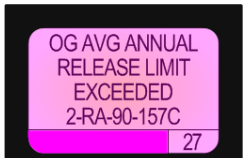
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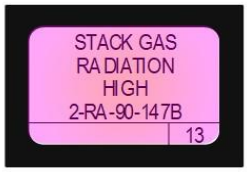
Form 4.2-1 **Written Examination Question Worksheet**

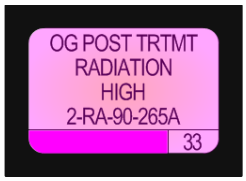
Examination Outline Cross-reference:	Level	RO	SRO
272000 (SF7, SF9 RMS) Radiation Monitoring	Tier #	2	------
A2.01 (10CFR 55.41.7)	Group #	2	------
Ability to (a) predict the impacts of the following on the RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:	K/A #	272000A2.01	
• Fuel element failure	Importance Rating	4.1	------

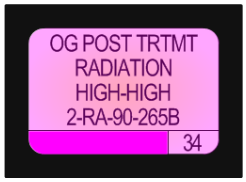
Proposed Question: **# 62**

U2 is operating at 100% RTP when a transient results in the following conditions:









- OG AVERAGE ANNUAL RELEASE LIMIT EXCEEDED (2-9-4C, WINDOW 27)
- STACK GAS RADIATION HIGH (2-9-3A, WINDOW 13)
- OG POST TREATMENT RADIATION HIGH (2-9-4C, WINDOW 33)
- OG POST TREATMENT RADIATION HIGH-HIGH (2-9-4C, WINDOW 34)

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

In accordance with the given ARPs, a manual SCRAM **CURRENTLY** **(1)** required.

2-RM-90-266A(265A), OG POST-TREATMENT CHANNEL A(B) RADIATION MONITORs, will be monitored on Panel **(2)**.

A. (1) is
(2) 2-9-4

B. (1) is
(2) 2-9-10

C. (1) is NOT
(2) 2-9-4

D. (1) is NOT
(2) 2-9-10

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that with escalating off gas post treatment radiation, a symptom of fuel damage, the candidate can easily confuse which radiation alarm(s) direct AOI entry or require a manual SCRAM, or even result in an automatic SCRAM being reached. Second part is incorrect but plausible given that Panel 2-9-4 is the location of 2-ARP-9-4C, Windows 27, 33 and 34. The candidate can be easily confused about the location of Control Room indicators.

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-9-3A, Window 13, or 2-ARP-9-4C, Windows 27, 33 and 34, a manual or automatic SCRAM is not required. Second part, as directed by 2-ARP-9-4C, Windows 33 and 34, 2-RM-90-266A(265A), OG POST TREATMENT CHANNEL A(B) RADIATION, is monitored at Panel 2-9-10.

RO Level Justification: Tests the candidate's ability to (a) predict the impacts of fuel element failure on the RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal 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): 2-ARP-3A, Rev. 59 (Attach if not previously provided)
2-ARP-4C, Rev. 39

Proposed references to be provided to applicants during examination:
OG AVERAGE ANNUAL RELEASE LIMIT EXCEEDED (2-9-4C, WINDOW 27)
STACK GAS RADIATION HIGH (2-9-3A, WINDOW 13)
OG POST TREATMENT RADIATION HIGH (2-9-4C, WINDOW 33)
OG POST TREATMENT RADIATION HIGH-HIGH (2-9-4C, WINDOW 34)

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

Learning Objective: OPL171.030, Obj. 10 (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:

Excerpts from 2-ARP-9-3A:

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0059 Page 24 of 60

STACK GAS RADIATION HIGH

2-RA-90-147B

13

(Page 1 of 2)

Sensor/Trip Point:

	HI
RM-90-147B	11,948 CPS
RM-90-148B	11,948 CPS
0-RM-90-306	As listed in 2-AOI-90-2

Sensor Location: RE-90-147 and El 599'6", Pnl 25-39 RE-90-148 inside stack

Probable Cause:
A. Source check.
B. Resin trap failure (RWCU or Cond Demin).
C. Possible fuel element failure.
D. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **CHECK** alarm condition on following:
1. WIDE RANGE GASEOUS EFFLUENT RADIATION MONITOR, 0-RM-90-306 on Panel 2-9-10.
2. STACK GAS/CONT RM RADIATION, 0-RR-90-147 on Panel 1-9-2.

B. **IF** alarm is from 0-RM-90-306, **THEN REFER TO** 2-AOI-90-2.

C. **CHECK** following radiation recorder on Panel 2-9-2 and associated radiation monitors on Panel 2-9-10:
1. OFFGAS RADIATION, 2-RR-90-266.

D. **ENSURE** dilution fan running and damper open by checking red light illuminated above STACK DILUTION FAN 2A (2B), 2-HS-66-29A (31A) on Panel 2-9-8.

E. **ENSURE** Charcoal Adsorbers in service.

F. **NOTIFY** RAD PRO, Unit 1, Unit 3 and Unit SRO

G. **REQUEST** Chemistry perform radiochemical analysis to determine source.

Continued on Next Page

BFN Unit 2	**Panel 9-3 2-XA-55-3A**	**2-ARP-9-3A Rev. 0059 Page 25 of 60**

STACK GAS RADIATION HIGH 2-RA-90-147B, Window 13
(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: 2-47E620-3 0-47E610-90-4 & 20 GE-2-729E814RF-5 0-SIMI-90B

Excerpts from 2-ARP-9-4C:

OG AVG ANNUAL RELEASE LIMIT EXCEEDED 2-RA-90-157C

27

(Page 1 of 2)

Sensor/Trip Point:

2-RE-90-157 1.37 E+4 mR/hr (Alarm from recorder)

Sensor Location: Elevation 565'
Turbine Building
Column T-7 B-LINE
Recorder is on Panel 2-9-2.

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. To determine if the Off Gas Annual Release Rate Limit is exceeded, **PERFORM** the following:
1. **CHECK** alarm condition on the following:
   - OFFGAS RADIATION recorder, 2-RR-90-266, Panel 2-9-2.
   - OG PRETREATMENT RAD MON RTMR monitor, 2-RM-90-157, Panel 2-9-10.

NOTE
High off-gas flow can sweep settled particulates into flow stream causing momentary rise in monitor parameters. Low off-gas flow can result in improper dilution causing rise in monitor parameters.

2. **CHECK** off-gas flow and monitor sample flow normal.
3. **NOTIFY** Radiation Protection.
4. **REQUEST** Chemistry perform radiochemical analysis to determine source.
5. With OPS MGT and Shift Manager's permission, **PLACE** charcoal beds in parallel with another unit. **REFER TO** 2-OI-66.
6. **IF** fuel damage is suspected, **THEN**
REFER TO 2-SR-3.4.6.1 for dose equivalent iodine - 131 determination.

Continued on Next Page

BFN Unit 2	**Panel 9-4 2-XA-55-4C**	**2-ARP-9-4C Rev. 0039 Page 35 of 44**

OG AVG ANNUAL RELEASE LIMIT EXCEEDED 2-RA-90-157C, Window 27
(Page 2 of 2)

Operator Action: (Continued)

7. **REFER TO** 0-SI-4.8.B.1.a.1 and 2-SR-3.4.6.1-a for ODCM compliance and to determine if power level reduction is required.
8. **IF** directed by Shift Manager or Unit Supervisor/SRO, **THEN REDUCE** reactor power to maintain off-gas radiation within ODCM limits.
9. **REFER TO** EPIP-1.

References:

2-45E620-4	2-47E610-90-1	GE 2-729E814 series
2-47E610-55-14	FSAR Sections 1.6.4.4.6, 7.12.2.2, and 13.6.2	Technical Specifications 3.4.6.1-a.

Technical Requirements Manual Sections 3.3.5.1, 3.3.9.1, & 3.7.2.1

OG POST TRTMT
RADIATION
HIGH
2-RA-90-265A

33

(Page 1 of 1)

Sensor/Trip Point:

2-RM-90-265A	1.55E+4 CPS
2-RM-90-266A	1.55E+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: Close signal to ADSORBER BYPASS VALVE 2-FCV-66-113B. Open signal to ADSORBER INLET VALVE 2-FCV-66-113A.

Operator Action:
A. **CHECK** alarm condition and **MONITOR** 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 ODCM 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.

References: 2-45E620-4 2-47E610-90-2 GE 2-729E814-6
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
Technical Specifications Section 3.4.6.1-a.
Technical Requirements Manual Sections 3.3.5.1, 3.3.9.1, & 3.7.2.1

OG POST TRTMT
RADIATION
HIGH-HIGH
2-RA-90-265B

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(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

OG POST TRTMT RADIATION HIGH-HIGH 2-RA-90-265B, Window 34
(Page 2 of 2)

References: 2-45E620-4 2-47E610-90-2 GE 2-729E814-6
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
Technical Specifications Section 3.4.6.1-a.
Technical Requirements Manual Sections 3.3.5, 3.3.9, & 3.7.2

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

Examination Outline Cross-reference:	Level	RO	SRO
Equipment Control	Tier #	3	------
G2.2.40 (10CFR 55.41.10)	Group #	------	------
Ability to apply technical specifications with action statements of less than or equal to 1 hour	K/A #	G2.2.40	
	Importance Rating	3.4	------

Proposed Question: **# 63**

Which **ONE** of the following completes the statement below in accordance with Technical Specification 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air?

If EDG starting air receiver unit pressure is < (less than) **(1)**, then declare the associated EDG INOPERABLE **(2)**.

A. (1) 165 psig
(2) immediately

B. (1) 165 psig
(2) within one (1) hour

C. (1) 200 psig
(2) immediately

D. (1) 200 psig
(2) within one (1) hour

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with Technical Specification 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air: LCO 3.8.3, Condition E: One or more DGs with the required starting air receiver unit pressure < 165 psig, immediately declare the associated DG INOPERABLE.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that there are numerous “one hour or less”, or “immediately” required completion times throughout the Technical Specifications.

C INCORRECT: First part is incorrect but plausible in that the multiple associated values exist for EDG starting air, including expected compressor start and stop values, and these associated values are easily and often confused. 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 the candidate's ability to apply Technical Specifications with action statements of less than or equal to 1 hour. This question is rated as Memory due to the requirement to strictly recall procedural facts.

Technical Reference(s): Tech Spec 3.8.3, Amend. 317 (Attach if not previously provided)
1/2-ARP-9-23A, Rev. 37
OPL171.038, Rev 27U3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.038, Obj. 10 (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 Tech Spec 3.8.3:

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with fuel oil level less than a 7-day supply and greater than a 6-day supply in storage tank.	A.1 Restore fuel oil level to within limits.	48 hours
B. One or more DGs with lube oil inventory less than a 7-day supply and greater than a 6-day supply.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One or more DGs with stored fuel oil total particulates not within limits.	C.1 Restore fuel oil total particulates to within limit.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with the required starting air receiver unit pressure < 165 psig.	E.1 Declare associated DG inoperable.	Immediately

Excerpts from 1/2-ARP-9-23A:

BFN Unit 1 & 2	Panel 0-9-23-7, 0-XA-55-23A	1/2-ARP-9-23A Rev. 0037 Page 5 of 52

DIESEL GEN A TROUBLE

2

(Page 1 of 2)

Sensor/Trip Point:

(Relay ALM) 0-RLY-082-A/ALM	Any diesel trouble condition(s).
Crankcase pressure	• 1.3 inches of H_2O.
Low water pressure	• 20 psig (DG Closed Cooling Water).
No field	• No field 6 sec. after fast start. • Field flashing relay (EXCX) failure.
Air pressure	• 167 psig, except during engine startup.
Fuel transfer	• DG day tank ≤ 190 gals. or ≥497 gals.
Engine hot	• 190°F
Lube oil level	• Low oil level
Lube oil temp. low	• 85°F with DG speed <125 rpm.
Overspeed trip	• 1035
Raw water (EECW)	• 7 psig.
Fuel pressure	• 67 psig.
Circuit malfunction	A. Relays **NOT** reset, 12 min. after stop signal. B. DG >100 rpm with voltage <75%, 6 sec. after fast start.
Oil pressure	A. 20 psig if DG >125 rpm but ≤870 rpm. B. 44 psig if DG >870 rpm.
Start failure	A. Fast start relay **NOT** picked up within 1 sec. after start signal. B. DG <40 rpm 3 sec. after start signal. C. DG >40 rpm but <125 rpm 4 sec. after start signal.
NOT auto	A. Either air compressor sws. **NOT** in auto. B. Field breaker **NOT** closed. C. Overspeed trip lever **NOT** reset.

Sensor Location: DG Engine Control Cab. Elevation 565 Diesel Generator Building

Probable Cause:
A. See SENSOR/TRIP POINT above.
B. Sensor(s) malfunction.

Continued on Next Page

DIESEL GEN A TROUBLE, Window 2
(Page 2 of 2)

Automatic Action: None

Operator Action:
A. **CHECK** panel 0-9-23-7 alarm panel for other alarms.
B. **DISPATCH** personnel to diesel generator room to check alarm.

NOTES

1) Either of the Diesel Generators A Left Bank motor or Right Bank motor control switches (0-HS-086-0668A or 0-HS-086-669A) in the OFF position will prevent a MANUAL field flash.
2) A Low Compressor Oil Level may cause the Diesel Generator Air Compressor not to start either automatically or manually.

C. **IF** receiver air pressure is at or below 167 psig, **THEN**

PERFORM the following: (Otherwise N/A)
1. **START** a D/G Air Compressor by performing the following as required:
   - Left Bank by placing DG A LEFT BANK AIR COMP MOTOR, 0-HS-086-0668A, to MAN.
   - Right Bank by placing DG A RIGHT BANK AIR COMP AC MOTOR, 0-HS-086-0669A, to MAN.
2. **WHEN** receiver air pressure reaches 200 psig, **THEN**

Excerpt from OPL 171.038 Lesson Plan:

OPL171.038 Diesel Generator and Auxiliary Power System Rev 27 U3

Outline of Instruction	Instructor Notes and Methods (optional)
each air start system, cranks the starting motor (3) Receiver pressure in each bank of units is maintained at 165 psig to 200 psig by air compressors in automatic operation per OI-82. (See applicable OI-82 for any unit differences) (4) Air compressor AC motors are powered from the 480VAC Diesel Auxiliary Boards (a) Air compressors are piston type air-cooled Ingersoll Rand	OI-82 P&L Tech Specs Tech. Specs does not require compressor, only air bank pressure above TS limit in either bank for operability

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

Examination Outline Cross-reference:	Level	RO	SRO
Conduct of Operations **G2.1.37** (10CFR 55.41.10) Knowledge of procedures, guidelines, or limitations associated with reactivity management	Tier #	3	------
	Group #	------	------
	K/A #	G2.1.37	
	Importance Rating	4.3	------

Proposed Question: **# 64**

Which **ONE** of the following completes the statements below pertaining to Reactivity Management in accordance with OPDP-1, Conduct of Operations?

In order to operate the controls of the facility including reactivity manipulations, verification of **(1)** is required.

The first Control Rod movement or Recirc Flow adjustment **(2)** require place keeping techniques.

A. (1) a successful GFES evaluation **ONLY**
(2) does NOT

B. (1) a successful GFES evaluation **ONLY**
(2) does

C. (1) a successful GFES evaluation **AND** all Systems training completed
(2) does NOT

D. (1) a successful GFES evaluation **AND** all Systems training completed
(2) does

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible given that a candidate may believe that once GFES/Reactor Theory is complete, a license candidate may be eligible to maneuver Reactor Power while under instruction without the completion of Systems training. Second part is incorrect but plausible in that for subsequent reactivity manipulations, no place keeping is required.

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*) OPDP-1 states that a candidate under instruction is not allowed to operate the controls of the facility, including reactivity manipulations, until verification of a successful GFES evaluation and all Systems training has been completed. Second part, OPDP-1, states that for Browns Ferry, the first control rod movements or flow adjustments shall require place keeping techniques.

RO Level Justification: Tests the candidate's knowledge of procedures, guidelines, or limitations associated with reactivity management. This question is rated as Memory due to the requirement to strictly recall procedural facts.

Technical Reference(s): OPDP-1, Rev. 58 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.071, Obj. 2 (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 OPDP-1:

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0058 Page 52 of 88

3.16.1 Overview (continued)

5. Provide feedback to adjust course content, emphasis, and content quality, as necessary.

F. Worker Qualifications (including technical qualifications, medical qualifications, and collateral qualification such as Respiratory Training):

1. Each worker shall understand their qualifications and ensure that they are kept up to date.

2. The work assigning supervisor shall verify qualification before any task requiring special qualifications.

3. Worker qualifications are to be addressed during pre-job briefs.

4. Stop When Unsure. Qualification discrepancies or concerns are to be addressed prior to task performance and documented in a condition report.

G. For initial licensed operator training programs, advisors/mentors are assigned for the class and individual students for the duration of the training. It is desired to assign class advisors. If not assigned, the Operations Director ensures that the class advisor responsibilities are fulfilled through other individuals or methods.

3.16.2 Oversight of Operations Trainees

A. Operations management will designate which candidates are allowed to operate the facility controls while under instruction in Standing/Night Orders. A candidate under instruction is not allowed to operate the controls of the facility including reactivity manipulations until:

1. Verification of a successful GFES evaluation.

2. All systems training completed.

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0058 Page 39 of 88

3.5.1 Reactivity Management (continued)

8. Perform no actions except those related to the reactivity manipulation, monitoring for expected changes. This is a dedicated action for the UO performing the reactivity change. Peer checks are expected to be performed for all reactivity management evolutions. Reactivity controls are placed in a safe and stable condition before attending to any emergent conditions.

9. For Browns Ferry - The first control rod movements or flow adjustments shall require place keeping techniques. Subsequent BWR control rod movements or recirculation flow adjustments, during the sequence of the evolution, will be at the discretion of the NUSO. The NUSO should use judgement as to when the place keeping rule may be waived, considering the repetitive number of times the rod movements or flow adjustments will be made. Any changes to personnel or positions, or pause in the evolution, will require resequencing using place keeping rules and practices.

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

Examination Outline Cross-reference:

Conduct of Operations

G2.1.44 (10CFR 55.41.10)

Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with fuel handling personnel, operating systems from the control room in support of fueling operations, or supporting instrumentation

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.1.44	
Importance Rating	3.9	------

Proposed Question: **# 65**

Which **ONE** of the following completes the statements below in accordance with 0-GOI-100-3C, Fuel Movement Operation During Refueling?

The delegated Unit Operator that is monitoring core reactivity from the Main Control Room is allowed to use **(1)**.

At least **(2)** independent communication links between the Main Control Room and the Refueling Platform Bridge are required.

Note: Integrated Computer System (ICS)

A. (1) Control Room instrumentation **ONLY**
(2) two

B. (1) Control Room instrumentation **ONLY**
(2) three

C. (1) Control Room instrumentation **AND** ICS
(2) two

D. (1) Control Room instrumentation **AND** ICS
(2) three

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that the Safety Parameter Display System (SPDS) component of the Integrated Computer System (ICS) is NOT qualified instrumentation, and cannot be used as the sole guide in operating the plant. The examinee may incorrectly assume, as a conservative measure, that only the Control Room instrumentation and not ICS alone can be used to monitor core reactivity during fuel movement. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*).
Second part is incorrect but plausible in that multiple forms of communication exist in the Control Room such as, sound powered phones, cordless phones, radios and PAX phones.

C **CORRECT**: *(See attached)* 0-GOI-100-3C states that a Unit Operator will monitor core reactivity utilizing ICS and/or Control Room instrumentation. Second part, the prerequisites of 0-GOI-3C require that two independent communications links between the Control Room and the Refueling Platform Bridge must be established.

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 RO duties in the Control Room during fuel handling, such as responding to alarms from the fuel handling area, communication with fuel handling personnel, operating systems from the control room in support of fueling operations, or supporting instrumentation. This question is rated as Memory due to the requirement to strictly recall procedural facts.

Technical Reference(s): 0-GOI-100-3C, Rev. 110 (Attach if not previously provided)
0-OI-48, Rev. 56

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.060 Obj. 3a (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 0-GOI-100-3C:

3.5 Fuel or Fuel Related Component (FRC) Handling (continued)

T. The Shift Manager or Refuel Floor SRO will delegate the responsibility for recording SRM/FLC data from ICS to a Unit Operator, STA, or Reactor Engineer.

U. The Shift Manager or Refuel Floor SRO will delegate a Unit Operator to monitor RFF operations by radio and also monitor core reactivity (utilizing ICS and/or CR instrumentation). The delegated Unit Operator will notify the RFF SRO of any potential abnormalities that might affect refuel operations.

4.0 PREREQUISITES (continued)

[1] Two independent communication links between the Control Room and the Refueling Platform Bridge have been established.

- Sound powered phones.
- PAX Phone (active comm. link)
- Cordless phone
- Radios (different frequencies may be considered different comm. links.)

(8) _____ _____ _____
Initials Date Time

Excerpt from 0-OI-48:

BFN Unit 0	Integrated Computer System	0-OI-48 Rev. 0056 Page 7 of 44

3.0 PRECAUTIONS AND LIMITATIONS

A. The Safety Parameter Display System (SPDS) component of the Integrated Computer System (ICS) is **NOT** qualified instrumentation, and therefore cannot be used as the sole guide in operating the plant. It is a highly reliable operator's aid. Installed, qualified plant instruments are to be checked to back up any information shown by the SPDS before any actual plant manipulations are performed.

B. Loss of ICS or ICS Multiplexer, 1(2)(3)-MUX-261-0017, for any reason (INOP, NIS, OOS) will cause the following to be inoperable:

1. 1(2)(3)-XA-55-1, Turbine Tripped Alarms on the first out annunciator
2. 1(2)(3)-XA-55-6C, window #29, RFPT TRIPPED
3. 1(2)(3)-XA-55-7B, window 5, EHC POWER ABNORMAL
4. 1(3)-XA-55-7B, window 6, EHC/TSI SYSTEM TROUBLE
5. 2-XA-55-7B, window 6, EHC SYSTEM TROUBLE
6. 1(3)-XA-55-7A, window #18, MAIN TURB OIL TANK LEVEL LOW-LOW
7. 1-XA-55-6A, window # 26, EXHAUST HOOD A, B, OR C TEMP HIGH HIGH 1-TA-2-113A

C. The time the ICS Computer Power Supply is on the maintenance feed should be minimized due to the automatic battery backup being inoperative for the power supply.

D. The battery backup to the ICS Computer power supply only picks up when the normal power supply (480V Common Board supply) is lost.

E. The Operations Duty Specialist (ODS) should be notified anytime the ICS Computer is going to be removed from service or down powered. The ODS will lose the computer inter-link from BFN during this process [BFN PER 01-003158-000].

F. Operation of the ICS Inverter without the DC source is **NOT** recommended by the Vendor. Operation without the DC source can result in failure of the static switch control components and the total loss of power to equipment normally supplied from the ICS Inverter. (BFPER 01-012718-000)

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

Examination Outline Cross-reference:

290002 (SF4 RVI) Reactor Vessel Internals

A2.03 (10CFR 55.41.5)

Ability to (a) predict the impacts of the following on the (SF4 RVI) REACTOR VESSEL INTERNALS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

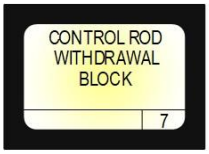
- Control rod drop accident

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	290002A2.03	
Importance Rating	4.4	------

Proposed Question: **# 66**

Unit 1 is in MODE 2 performing a startup, Control Rod 22-27 is being withdrawn from position 24 to 48 when the following occurs:

- RWM is **NOT** bypassed
- CONTROL ROD WITHDRAWAL BLOCK (9-5A, Window 7) alarms
- LPRM HI status lights illuminate around Control Rod 22-27 on Full Core Display
- IRM channels are rising
- **NO** Operator Actions have been taken



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

Indications are Control Rod 22-27 has **___(1)___** and will be limited by the velocity limiter.

In accordance with the associated AOI, the Operators are required to **___(2)___**.

A. (1) drifted
(2) individually SCRAM Control Rod 22-27

B. (1) drifted
(2) continuously insert Control Rod 22-27 to '00' using EMERG ROD IN switch

C. (1) dropped
(2) individually SCRAM Control Rod 22-27

D. (1) dropped
(2) continuously insert Control Rod 22-27 to '00' using EMERG ROD IN switch

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT (*See attached*) First part is incorrect but plausible in that the symptoms of a drifting Rod are at times, easily and often confused by candidates along with the concept of the velocity limiter to limit the velocity of a drifting coupled Rod or assist in normal Rod withdraw. Second part is incorrect but plausible in that the numerous methods to insert Rods, such as normal insertion, EMERG ROD IN, and individually SRAMMING a rod is directed among various OI, AOI and EOI procedures.

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

C First part is correct (*See D*). Second part is incorrect (*See A*).

D **CORRECT**: (*See attached*) In accordance with 1-AOI-85-1, Rod Drop Accident, symptoms of dropped Rod are: Reactor Power rise, LPRM HI status lights around dropping Rod in alarm on full core display, and alarm Window 7. With the given conditions, all must be integrated to determine that Control Rod 22-27 has dropped. The purpose of the velocity limiter is to limit the free fall rate of dropping Rod to less than 3.11 ft/sec. Second part, in accordance with 1-AOI-85-1, Rod Drop Accident, a dropped control rod is required to be continuously inserted to 00 using the EMERG ROD IN switch.

RO Level Justification: Tests the candidate's ability to predict the impacts of Control Rod drop accident on Reactor Vessel Internals and use the correct procedural/steps to mitigate the consequences of those abnormal 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): OPL171.006, Rev. 11U5 (Attach if not previously provided)
1-AOI-85-1, Rev. 2
1-AOI-85-6, Rev. 7
1-ARP-9-5A, Rev. 35

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.006 Obj. 4 (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 1-ARP-9-5A:

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0035 Page 11 of 48

CONTROL ROD WITHDRAWAL BLOCK

7

(Page 1 of 2)

Sensor/Trip Point:

Relays	Nuclear Instrumentation
3A-K1	Refuel Equipment in Use
3A-K2	High Level In Scram Discharge Volume
	Scram Discharge Volume High Water Level Bypass
	Rx. Mode Switch in Shutdown
	PRNM (ANY APRM, OPRM or RBM)

Sensor Location:
Panel 1-9-28
Elevation 593'
Aux. Inst. Room.

Probable Cause:
A. One or more sensors at or above set point.
B. Malfunction of sensor.
C. Control rod drop accident.

Automatic Action:
Rod withdrawal block.

Operator Action:
A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s). **REFER TO** operator action for alarm(s).

B. **IF** alarm due to inadvertent criticality during incore fuel movements, **THEN**
REFER TO 1-AOI-79-2.

C. **IF** alarm is from a control rod drop, **THEN**
REFER TO 1-AOI-85-1.

D. **IF** NO corresponding alarm exists, **THEN**
PERFORM the following:
1. AT ICS console, **DETERMINE** if there is a refuel rod block by selecting Single Point Menu, Single Value Display, and typing DIG090, Return.
2. **IF** rod block was from Refuel Floor, **THEN**
NOTIFY Refuel Floor Operator to have dummy plug (Refuel floor between cavity and pool, southside) checked and check jumpers in U-1 Aux. Inst. Room Panel 9-28 Bay 3, if installed per 1-OI-85 Section 8.34.

Continued on Next Page

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0035 Page 12 of 48

CONTROL ROD WITHDRAWAL BLOCK, Window 7
(Page 2 of 2)

3. **WHEN** IRM switches are below Range 3 with REACTOR MODE SWITCH NOT in RUN, **THEN**
   CHECK SRM detectors NOT FULL IN.
4. **WHEN** REACTOR MODE SWITCH is in START-UP position, **THEN**
   CHECK IRM detectors NOT FULL IN.

E. **REFER TO** Tech Spec Table 3.3.2.1-1, TRM Table 3.3.4-1.

References: 1-45E620-6-1 1-730E321-3, and 5

Excerpts from 1-AOI-85-1:

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a control rod drop accident.

2.0 SYMPTOMS

A. Reactor power rise.

B. LPRM HI status lights around dropping control rod in alarm on Full Core Display.

C. Reactor high neutron flux scram with any of the following annunciators on Panel 1-9-4 and 1-9-5 in alarm:

1. IRM CHAN A,C,E,G HI-HI/INOP (1-XA-55-5A, Window 33)
2. IRM CHAN B,D,F,H HI-HI/INOP (1-XA-55-5A, Window 34)
3. APRM HIGH/INOP OR OPRM TRIP (1-XA-55-5A, Window 25)
4. REACTOR CHANNEL A AUTO SCRAM (1-XA-55-5B, Window 1)
5. REACTOR CHANNEL B AUTO SCRAM (1-XA-55-5B, Window 2)
6. NEUTRON MONITORING SYS HALF SCRAM (1-XA-55-4A, Window 16)
7. CONTROL ROD WITHDRAWAL BLOCK (1-XA-55-5A, Window 7)

3.0 AUTOMATIC ACTIONS

NOTE

The extent to which automatic actions may occur is entirely dependent on amount of reactivity inserted and Reactor K_{eff} at the time of rod drop.

A. Control rod block.

B. Reactor high neutron flux scram.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **CHECK** automatic actions.

[2] **IF** NO scram occurs, **THEN**

REDUCE Reactor power by 10% from the power prior to event.

[3] **OBTAIN** concurrence of SM and **CONTINUOUSLY INSERT** dropped control rod to position 00 using EMERG ROD IN switch.

[4] **WHEN** control rod is fully inserted, **THEN**

REMOVE associated HCU from service. **REFER TO** 1-OI-85.

[5] **ADJUST** reactor power and flow as directed by Reactor Engineer/Unit SRO to stay within required thermal and feedwater temperature limits. **REFER TO** 1-GOI-100-12 or 1-GOI-100-12A for the power reduction.

NOTES

1) Reactor coolant samples and radiation recorders are analyzed and monitored to determine if fuel element failure has occurred.
2) Based on recent On-Shift Analysis from the License Amendment , chemistry sampling may be delayed following plant events. In the case of a REP event, Chemistry will be required to assist with dose assessments. Dose assessments will have precedence over sampling actions in REP events.

[6] **REQUEST** Chemistry personnel take Reactor coolant samples.

Excerpts from 1-AOI-85-6: Supports Distractors A1, B1

BFN Unit 1	Rod Drift Out	1-AOI-85-6 Rev. 0007 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 1-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 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, 1-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 1-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.

4.2 Subsequent Actions (continued)

[3] **IF** a Control Rod Block occurs during rod insertion due to Rod Worth Minimizer, **THEN**

BYPASS the RWM per step 4.2[1] above. (Otherwise N/A)

[4] **IF** Control Rod Drive does NOT respond to INSERT signal, **THEN**

PERFORM the following: (Otherwise N/A)

[4.1] **REDUCE** Total Core Flow, as indicated on TOTAL CORE FLOW/CORE PRESS DROP, 1-XR-68-50 on Panel 1-9-5, by approximately 10% to control possible power increase.

[4.2] **PLACE** the appropriate Rod Scram Switch, on 1-PNLA-009-0016 in the Auxiliary Instrument Room, in the DOWN position.

[4.3] [NER/C] **IF** drifting control rod is causing Reactor power to rapidly rise at a rate which can NOT be controlled by reducing recirculation flow, **THEN**

MANUALLY SCRAM the Reactor. (Otherwise N/A) [INPO SER 90-015]

[5] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern.

[6] **IF** another Control Rod Drift occurs before Reactor Engineering completes the evaluation, **THEN**

MANUALLY SCRAM Reactor and enter 1-AOI-100-1.

BFN Unit 1	Rod Drift Out	1-AOI-85-6 Rev. 0007 Page 7 of 10

4.2 Subsequent Actions (continued)

NOTES

1) A control rod that will NOT latch could be indicative of either the following:
   - A failed open withdraw directional control valve (1-FCV-085-40C).
   - 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 1-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 1-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 1-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 1-AOI-100-1.

[8] **IF** the control rod is latched into position "00", **THEN**

REMOVE associated HCU from service per 1-OI-85
(N/A if Control Rod will not latch at "00")

Excerpt from OPL171.006 Lesson Plan:

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
i. As temperature rises, slowing down length and thermal diffusion length increase. ii. Rod worth increases with as moderator temperature increases b) Void effects on rod worth i. As voids increase, average neutron flux energy increases. ii. U238 and Pu240 will capture more epithermal neutrons through resonance. iii. This produces lower localized neutron flux in the area of the voids. iv. As void fraction increases, control rod worth decreases.	
c. Velocity limiter description	FIG.-1, -4, -6
1) A stainless steel casting located below the poison section is an integral part of the blade assembly.	
2) Purpose is to limit the free fall rate of the blade to 3.11 ft/sec. (Technical Specification requirement) in the event the rod is uncoupled. The velocity limiter, in conjunction with the Rod Worth Minimizer, limits peak fuel enthalpy to < 280 calories per gram on a rod drop accident.	OBJ. ILT/NLO-4,NLOR-3
a) It is a one-way device which does not affect scram insertion.	
b) It is necessary to limit the free-fall rate in order to limit the rate at which reactivity is added to the core in order to assure that fuel damage does not occur during a rod drop accident.	
3) Principle of operation	
a) The falling action of the blade creates a large pressure drop across the velocity limiter due to the multi-directional change of flow.	
b) The small radial clearances between the cylindrical portion of the velocity limiter and the CRD guide tube also help to restrict the rate of fall.	
d. Blade alignment	OBJ. ILT/NLO-5, NLOR-4
1) Upper end is laterally supported by four spherical rollers that bear on the fuel channels (or blade guides, when fuel is removed). Unit 3 replaced ~ 120 control rod rollers with "Slider Pads" which ride on the channel (reduces radiation associated with stellite balls).	
2) Lower end is laterally supported by four rollers that bear on the inner surface of the CRD guide tubes.	

Figure 2: Marathon Control Rod Blade

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

Examination Outline Cross-reference:

Equipment Control

G2.2.43 (10CFR 55.41.10)

Knowledge of the process used to track inoperable alarms

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.43	
Importance Rating	3.0	------

Proposed Question: **# 67**

Which **ONE** of the following completes the statements below in accordance with OPDP-4, Annunciator Disablement?

Disabled annunciators are identified by a **(1)** border placed on the disabled annunciator window.

Each unit **(2)** required to carry over the list of disabled annunciators each shift in the eSOMS Narrative Log.

A. (1) blue
(2) is

B. (1) blue
(2) is NOT

C. (1) white
(2) is

D. (1) white
(2) is NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part incorrect but plausible in that a narrative log entry is made at the time of disablement, however, this log entry is not required to be carried over.

B **CORRECT**: (*See attached*) In accordance with OPDP-4, Annunciator Disablement, a blue magnetic border labeled, “Disabled Alarm Input,” is required to be placed on each affected disabled annunciator alarm window to indicate an input to that alarm has been disabled. Second part, a narrative log entry is required at the time of disablement, but is not required to be carried over.

C INCORRECT: Incorrect but plausible in that for testing and maintenance, an Out-of-Service Indicator is defined as a white magnetic border labeled “Testing/Maintenance.” Second part incorrect but plausible (*See A*).

D First part is incorrect (*See C*). Second part is correct (*See B*).

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

RO Level Justification: Tests the candidate's knowledge of the process used to track inoperable alarms. This question is rated as Memory due to the requirement to strictly recall procedural facts.

Technical Reference(s): OPDP-4, Rev. 10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.271 Obj. 4 (As available)

Question Source:

Bank #	BFN 1703 NRC #69
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2017

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:

QUESTION 69 Rev 2

Which one of the following completes the statement below, in accordance with OPDP-4, Annunciator Disablement?

Disabled annunciators are identified by a __ (1) __ placed on the disabled annunciator window.

Each unit __ (2) __ required to carry over the list of disabled annunciators each shift in the eSOMS Narrative Log.

A. (1) White border
(2) is

B. (1) White border
(2) is **NOT**

C. (1) Blue border
(2) is

D. (1) Blue border
(2) is **NOT**

Answer: D

Excerpts from OPDP-4:

3.5 Review and Audit (continued)

2. Monitoring plan

3. Length of time the alarm is disabled

4.0 RECORDS

4.1 QA-Records

A. Disabled Alarm Checklist OPDP-4-1

B. Annunciator Disablement Technical Evaluation OPDP-4-5

4.2 Non-QA Records

A. Disabled Alarm Index Sheet OPDP-4-2

B. Disabled Alarm Compensatory Measures OPDP-4-3

5.0 DEFINITIONS

Disabled Input Indicator -

- BFN - A blue magnetic border labeled “Disabled Alarm Input.”
- SQN- A blue dot (sticker) or numbered square attached to the window with the applicable details listed in the MCR Turnover Package.
- WBN- An orange plastic lens cover labeled “Disabled Alarm” which snaps over the affected window and a blue plastic lens cover labeled “Disabled Input.”

Out-of-Service Indicator -

- BFN - A white magnetic border labeled “Testing/Maintenance.”

3.2.4 Designated Operator (RO/SRO)

A. When all required approvals for disabling alarm are obtained, perform the following and complete Form OPDP-4-1, Item 9:

1. Review disabling steps.

2. Initiate Compensatory Monitoring as described in Forms OPDP-4-3 and/or OPDP-4-5, if applicable.

3. Designate a qualified individual to disable alarm(s) using method in Form OPDP-4-1, and another individual to perform verification.

3.2.4 Designated Operator (RO/SRO) (continued)

4. Place a disabled input indicator on each affected alarm window to indicate an input to that alarm has been disabled.

5. Sign Form OPDP-4-1, Item 9 "Performed By" and submit to a qualified individual to sign verification requirements.

NOTE

A Form OPDP-4-2 is to be maintained in each Disabled Annunciator Book. More than one entry per sheet is allowed.

6. Log alarm disablement on Form OPDP-4-2, "Disabled Alarm Index Sheet" in the associated Disabled Annunciator Book. Initiate a new sheet if required.

7. Log alarm disablement in the narrative log for the affected unit(s) when alarm disabled. The narrative log entry shall include alarm location, method used to disable alarm, date and time removed, justification for disablement, and Tech Spec action requirements, if applicable.

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: **# 68**

Determine which **ONE** of the following indications would be observed for an Area Radiation Monitor (ARM) when the detector is **SATURATED** in a radiation field.

A. The indicator remains upscale **BUT** all trips drop out.

B. The indicator drops downscale **AND** the trips drop out except for the downscale trip.

C. The indicator remains upscale **AND** the upscale trip remains in due to the pegging circuit.

D. The indicator drops downscale **BUT** the upscale trip remains in due to the pegging circuit.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that the upscale alarm function remains active, the candidate could assume that the pegging circuit only retains an upscale indication.

B INCORRECT: Incorrect but plausible in that If the candidate doesn't realize the pegging circuit is an actual component that functions. Neither downscale or a downscale trip are expected in a saturated field.

C **CORRECT:** The system is equipped with a protective electronic pegging circuit installed for the adverse GM detector characteristic known as saturation. Once the counts appear to be a solid count to the point it appears that there is no count, the detector would naturally indicate no counts. The installed pegging circuit prevents this false downscale indication by providing the indication as upscale with the associated alarm and indication response. A transistor current amplifier (Holding Circuit), controlled by the GM tube DC output, furnishes a meter pegging current when the radiation intensity exceeds the point of saturation.

D INCORRECT: Incorrect but plausible in that if the candidate assumes that the pegging circuit only retains an upscale indication while a downscale is not expected in a saturated field.

RO Level Justification: Tests the candidate's ability to use radiation monitoring systems, specifically Area Radiation Monitors and interpret their indications. This question is rated as Memory due to the requirement to strictly recall procedural facts.

Technical Reference(s): OPL171.034, Rev. 12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.034 Obj. 4 , (As available)

Question Source:

	Bank #	BFN NRC 2104 #72
	Modified Bank #	
	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	**X**
55.43	

Comments:

Copy of Bank Question:

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

Examination Outline Cross-reference:

G2.3.5 (10CFR 55.41.11)
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.3.5	
Importance Rating	2.9	------

Proposed Question: **# 72**

Determine which **ONE** of the following indications would be observed for an Area Radiation Monitor (ARM) when the detector is **SATURATED** in a radiation field.

A. The indicator remains upscale **BUT** all trips drop out.

B. The indicator drops downscale **AND** the trips drop out except for the downscale trip.

C. The indicator remains upscale **AND** the upscale trip remains in due to the pegging circuit.

D. The indicator drops downscale **BUT** the upscale trip remains in due to the pegging circuit.

Proposed Answer: **C**

Correct Answer: A

Excerpt from OPL171.034 Lesson Plan:

1. Geiger-Mueller Tube (Sensor)

a) The GM tube has an internal electric field which draws ions toward the electrodes. ~600V DC is applied across the anode and cathode from the external power supply — ILT/LOR –Obj 2 NLO/NLOR-Obj 2

b) Gamma radiation entering a GM tube interacts with the atoms of the detector gas, releasing ions to the detector volume.

c) The ions then cause secondary ionization in the contained gas.

d) As radiation intensity rises, the number of ionizations and ion pairs increases. Nominal sensitivity is $3.7x10^{-10}$ amps/R/hr.

e) The positive and negative ions, created by incident radiation, move toward the anode (+) and cathode (-) of the GM tube.

f) The charge collection on the anode causes a voltage change in the circuit, and a current flow (pulse) results.

g) If the interval between pulse generating events becomes short enough, the GM tube will go into steady DC conduction, "saturation," and stop producing output pulses — ILT/LOR –Obj 3 NLO/NLOR-Obj 3

h) When this happens, the external circuit that is counting and integrating pulses responds as though the events were only one pulse which would cause a decreased meter reading at the point of saturation

i) This possible source of error is overcome by using a high-pass filter in the GM tube circuit which detects this condition and provides a saturation signal which represents the degree of ionization of the GM tube

j) A transistor current amplifier (Holding Circuit), controlled by the GM tube DC output, furnishes a meter pegging current when the radiation intensity exceeds the point of "saturation. — ILT/LOR –Obj 4 NLO/NLOR-Obj 4

k) This assures proper indication. Without pegging circuit, saturation would indicate downscale. With pegging circuit, saturation provides upscale indication.

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

Examination Outline Cross-reference:	Level	RO	SRO
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1&4	Tier #	1	------
AK1.03 (10CFR 55.41.10)	Group #	1	------
Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Thermal limits	K/A #	295001AK1.03	
	Importance Rating	4.1	------

Proposed Question: **# 69**

Unit 1 is operating at 100% Rated Thermal Power (RTP) when the following occurs:

- 1A Recirc Pump trips
- IMMEDIATE ACTIONS complete

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

ONCE the APRM Flow-Biased Simulated Thermal Power - High Value for Single Loop Operation (SLO) reset occurs **(1)**, Unit 1 can be operated **(2)**.

A. (1) manually
(2) for **ONLY** 24 hours

B. (1) manually
(2) indefinitely

C. (1) automatically
(2) for **ONLY** 24 hours

D. (1) automatically
(2) indefinitely

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that in accordance with Tech Spec 3.4.1, Condition A, the SLO SR is required to be implemented within 24 hours (*See attached*).

B **CORRECT**: *(See attached)* In accordance with 1-AOI-68-1, Recirc Pump Trip/Core Flow Decrease, the Reactor Engineer is required to manually perform 1-SR-3.4.1(SLO) along with support from Operations and I&C Maintenance to place each APRM in SLO Mode. Second part, in accordance with 1-OI-68, Reactor Recirculation System, SLO per Technical Specifications, the Reactor can be operated indefinitely with one Reactor Loop out of service, provided the requirements of Tech Spec 3.4.1 are implemented within 24 hours of entering Single Loop Operations.

C INCORRECT: First part is incorrect but plausible in that the Recirc System contains a number of runbacks that occur automatically lowering Recirc Pump speed. 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 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-OI-68, Rev. 55 (Attach if not previously provided)
1-AOI-68-1, Rev. 3
1-SR-3.4.1 (SLO) Rev 14
U1 Tech Spec 3.4.1, Amend. 310

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.007 Obj. 22 (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 1-OI-68:

BFN Unit 1	Reactor Recirculation System	1-OI-68 Rev. 0055 Page 13 of 221

3.3 Tech Specs

A. 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** 1-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 1A(1B) DISCHARGE VALVE 1-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, 1-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.

Excerpts from 1-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 MELLLA+ operating domain. **REFER TO** Tech Spec 3.4.1.

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

CLOSE tripped Recirc Pump 1A(1B) discharge valve 1-FCV-068-0003(0079). (Otherwise **MARK** N/A)

[2.1] **IF** Recirc Pump was tripped due to dual seal failure, **THEN** (Otherwise **MARK** N/A)

[2.1.1] **ENSURE** TRIPPED, RECIRC DRIVE 1A(1B) NORMAL FEEDER, 1-HS-57-17(14).

[2.1.2] **ENSURE** TRIPPED, RECIRC DRIVE 1A(1B) ALTERNATE FEEDER, 1-HS-57-15(12).

[2.1.3] **CLOSE** tripped recirc pump suction valve using, RECIRC PUMP 1A(1B) SUCTION VALVE, 1-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 1-GOI-100-12A.

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

Excerpts from 1-SR-3.4.1 (SLO):

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

None

3.2 Limitations

A. APRM channels are bypassed one at a time to allow entry of the STP Upscale Trip and Alarm Flow-Bias Setpoint Slope, Offset, and ΔW values and placement of each channel in SLO mode. No LCOs are required to be entered as a result of this procedure performance.

B. When the INOP/OPER keylock switch on any APRM instrument is placed in INOP position, the APRM UPSC/INOP, OPRM DIDA/INOP, and OPRM CDA/INOP INPUT STATUS TRIP LEDs will ILLUMINATE on each 2-OUT-OF-4 VOTER for the APRM channel placed in INOP. Any other annunciation which occurs is required to be reviewed for its effect on proper unit operation.

C. This procedure is intended to be initiated by Reactor Engineering personnel. However, on occasions where a Reactor Engineer is NOT immediately available, Operations personnel may commence procedure performance provided Lead Performer is cognizant of this condition.

D. There are required to be three operable APRM channels for each trip system. Each APRM channel provides input to both Reactor Protection System (RPS) trip divisions.

E. One APRM channel per trip system may be bypassed at any one time for calibration or testing purposes.

F. All 2-OUT-OF-4 VOTER trips are required to be reset and affected APRM channel returned to operable status before commencing any work activities (setpoint changes) on another APRM channel.

G. The NUMAC APRM chassis front panel display will automatically turn off after 30 minutes whenever in OPERATE mode with no operator action, alarms, faults or trips (including past trips in TRIP MEMORY). When this occurs during performance of this procedure, depressing any key will turn display back on.

H. Each display also shows active alarms and bypassed conditions in inverse video. Text displayed in normal video indicates a past alarm or bypassed condition being stored in memory. The past alarms or bypassed conditions may be reset at any time during performance of this procedure by selecting TRIP STATUS display and next depressing the RESET MEMORY softkey.

Date _____

3.2 Limitations (continued)

I. The 2-OUT-OF-4 VOTER TRIP MEM LEDs may be reset at any time during performance of this procedure by depressing TRIP MEMORY RESET pushbutton on applicable voter.

J. At any time during performance of this procedure when an incorrect softkey or pushbutton is depressed, the incorrect display or mode may be exited by depressing EXIT or STOP/RETURN softkey as required to return to next or previous menu. The correct display or mode may next be entered by repeating the applicable instructions in this procedure without affecting outcome of this procedure.

K. Neutron noise seen by APRMs has a greater amplitude in SLO than in Dual Loop Operation (DLO). The increased noise has caused another plant to receive an Oscillation Power Range Monitoring (OPRM) trip, even on occasions well outside the stability regions and the reactor stable. Heightened awareness and increased monitoring should occur during SLO.

4.0 PREREQUISITE ACTIONS

[1] **NOTIFY** I&C Maintenance that their support will be needed to place APRM(s) in Single Loop Operation Mode. _____

[2] **IF** 1A Recirculation Pump is Tripped, **THEN**

NOTIFY I&C Maintenance that their support will be needed to isolate 1A Recirculation Loop flow transmitters, **OTHERWISE** N/A this step.

- 1-FT-68-5A
- 1-FT-68-5B
- 1-FT-68-5C
- 1-FT-68-5D _____

BFN Unit 1	Reactor Recirculation System Single Loop Operation	1-SR-3.4.1(SLO) Rev. 0014 Page 9 of 35

Date _____

4.0 PREREQUISITE ACTIONS (continued)

[3] **IF** 1B Recirculation Pump is Tripped, **THEN**

NOTIFY I&C Maintenance that their support will be needed to isolate 1B Recirculation Loop flow transmitters, **OTHERWISE** N/A this step.

- 1-FT-68-81A
- 1-FT-68-81B
- 1-FT-68-81C
- 1-FT-68-81D _____

[4] **ENSURE** the following Training Requirements are met prior to performing this procedure:

- IMP564 _____

5.0 ACCEPTANCE CRITERIA

Responses which fail to satisfy the Acceptance Criteria (AC) constitute unsatisfactory procedure results and require immediate notification of UNIT SRO at time of the failure.

A. The following acceptance criteria are to be satisfied in order to verify unit has been properly placed into SLO mode:

1. Each APRM chassis is verified to contain a ΔW value of 0.0%.
2. Each APRM chassis is verified to contain a slope value of 0.55.
3. Each APRM chassis is verified to contain a trip offset value of 53.0.
4. Each APRM chassis is verified to contain a rod block offset value of 47.0.
5. Each APRM channel has been placed in SLO mode.
6. Core Performance Log is calculated based on correct Thermal Limits Set.

B. Steps which determine the above criteria are designated by (AC) next to the signoff blank.

Excerpts from U1 Tech Spec 3.4.1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

----------NOTE----------
Single recirculation loop operation is prohibited in the MELLLA+ operating domain.

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Operating in the MELLLA+ operating domain with a single recirculation loop in operation.	B.1 Initiate action to exit the MELLLA+ operating domain.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met. OR No recirculation loops in operation.	C.1 Be in MODE 3.	12 hours

Excerpt from 1-OI-68: Supports Distractors C1, D1

BFN Unit 1	Reactor Recirculation System	1-OI-68 Rev. 0055 Page 199 of 221

Attachment 2
(Page 4 of 5)

Recirc Flow Control System Manual Runback Controls

An automatic runback always takes precedence over a manual runback. If an automatic runback occurs while a manual runback is in progress, the automatic runback signal will be sent to the Recirc Drive. The manual runback will be locked out and the manual runback push-button backlight will stop blinking. After a 30 second time delay, the manual runback may be reinitiated by depressing the applicable push-button a second time. Automatic runback setpoints to the 28% and 75% Limiters are as follows:

AUTOMATIC RUNBACKS		
Limiter	**Runback Setpoint**	**PUMP Speed After Runback**
28%	Total Feedwater Flowless than 16% for 15 seconds OR recirc pump discharge valve less than 90% open	480 RPM
75%	Individual RFP discharge flow less than16% and Reactor Water Level less than 27 inches OR Reactor scram	1130 RPM

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

Examination Outline Cross-reference:

Reactor Theory - NEUTRONS
292001(K1.04) (10CFR55.41.1)
Describe neutron moderation

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292001K1.04	
Importance Rating	3.2	------

Proposed Question: **# 70**

Unit 2 is operating at 80% RTP with the following conditions:

- OATC **RAISES** Recirc Flow using 2-HS-96-32, RECIRC MASTER CONTROL RAISE MEDIUM, pushbutton in accordance with the Reactivity Control Plan

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

The amount of neutron moderation **(1)** due to the void fraction/content **(2)**.

A. (1) rises
(2) rising

B. (1) rises
(2) lowering

C. (1) lowers
(2) rising

D. (1) lowers
(2) lowering

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part, incorrect but plausible in that lowering Recirculation flow lowers the amount of water flowing through the core. Lowering Recirculation raises the core void fraction (more steam/water mixture like), which will lower the amount of moderation.

B **CORRECT**: *(See attached)* From the given in the stem, raising Recirculation flow sweeps voids within the Reactor coolant thereby lowering the core void fraction, and raises the density of the moderator. This directly results in the amount of neutron moderation rising.

C INCORRECT: First part is incorrect but plausible in that in that lowering Recirculation flow lowers the amount of water flowing through the core. Lowering Recirculation raises the core void fraction (more steam/water mixture like), which will lower the amount of moderation.
Second part is incorrect but plausible (*See A*).

D INCORRECT: 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 knowledge of Generic Fundamentals, specifically of how neutron moderation changes in response to core flow. This question is rated as Memory due to the requirement to strictly recall facts in relation to neutron moderation.

Technical Reference(s): GFES Reactor Theory, Ch. 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: GFES Reactor Theory, Ch. 8 Obj. 21

Question Source:

Bank #	
Modified Bank #	GFES Reactor Theory #B445 (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:

0 / 0 QID: B445 (BWR ONLY) Neutron moderation	
BWR TOPIC: 292001	PWR TOPIC: 192001
KNOWLEDGE: BK1.04 [3.2/3.2]	KNOWLEDGE: BWR ONLY as worded
Similar questions	
Recent BWR exams	Recent PWR exams
Not used in 2001 through 2007	This is a BWR only question.

Which one of the following conditions will increase the amount of neutron moderation in a reactor operating at saturated conditions?

A. Increasing moderator temperature

B. Reducing feedwater inlet temperature

C. Reducing reactor pressure vessel pressure

D. Reducing reactor recirculation system flow

BWR ONLY

Modified 'D' to be to be the correct response, RAISING RECIRC FLOW.

0 / 0 QID: B445 (BWR ONLY) Neutron moderation

Explanation

Which one of the following conditions will increase the amount of neutron moderation in a reactor operating at saturated conditions?

To increase moderation the density of the moderator must increase. A temperature decrease, pressure increase, or decrease in voids would be required to affect this change. As the water molecules become closer together, either because the temperature is reduced (lowering the average kinetic energy of the water molecules allowing the molecules to move closer together), the pressure is increases (pushing the molecules closer together), or a decrease in voids (less heat added to the moderator or more moderator available to carry off the same amount of heat), there are more hydrogen molecules to serve as targets for the fast neutrons. This allows more moderation to occur.

A. Increasing moderator temperature
   Incorrect – this would cause moderator density to decrease, which would DECREASE the moderation.

B. Reducing feedwater inlet temperature
   CORRECT – reducing feedwater temperature would decrease moderator temperature, raising moderator density, increasing moderation of neutrons.

C. Reducing reactor pressure vessel pressure
   Incorrect - this would cause voids to increase, thus decreasing moderator density which would DECREASE the moderation.

D. Reducing reactor recirculation system flow
   Incorrect – Reducing reactor recirculation system flow decreases the amount of water flowing through the core. The water has to remove the same amount of heat, so the temperature of the water will increase; in an extreme reduction in recirculation flow voids will increase. An increase in moderator temperature resulting increase in the core void fraction will cause the moderator density to decrease, REDUCING the amount of moderation.

Modified 'D' to be to be the correct response, RAISING RECIRC FLOW.

The correct answer is ANSWER B. Reducing feedwater inlet temperature

B W R O N L Y

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Excerpt from BWR GFE Reactor Theory, Chapter 8, Reactor Operational Physics:

POWER INCREASE WITH RECIRCULATION FLOW

Another means of changing the reactor's power level is through the use of the recirculation system. Once a significant void fraction is established in the core, any change in the void fraction will cause a net change in the core reactivity. The following discussion studies the effects of making a step increase in recirculation flow in the core.

As core flow increases, the subcooled water from the downcomer area is forced into the bottom of the core at a higher velocity. This causes each water molecule to spend less time in contact with the fuel clad for a given area than prior to the flow increase. At the same time, more of the water molecules are in contact with the fuel clad per unit time. The net result is an increase in heat removal rate from the fuel rods with the coolant having to travel farther up the core to gain the same amount of heat gained at the lower flow. This effectively raises the point in the core where the coolant reaches saturation, resulting in the core "boiling boundary" being higher in the fuel zone. This increase in core subcooling causes a small decrease in core average moderator temperature and a significant decrease in the core void fraction. The resulting increase of moderator density inserts positive reactivity to the core. This positive reactivity insertion causes the power level of the reactor to increase. As the reactor power level (fission rate) increases, both the fuel pellets and the cladding begin to heatup. Figure 8-5 shows plant response to an increase in recirculation flow.

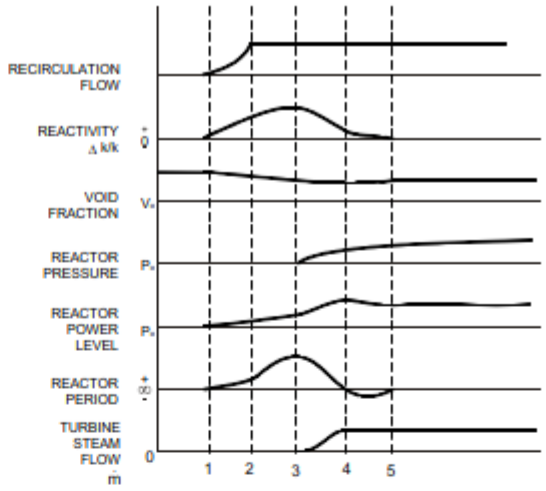


Figure 8-5 Plant Response to Increased Recirculation Flow in the Power Range

Points 1→2:

- As recirculation flow increases, subcooled water velocity from the downcomer area increases, thus reducing the core void fraction.
- A positive reactivity insertion results, causing reactor power level to increase.
- Because the reactor period is a measure of the rate of change of the power level, a decreasing (moving away from infinity) positive reactor period is observed.
- The increased recirculation flow stops at point 2 and is held constant at this new, higher value.

Points 2→3:

- Between points 2 and 3, positive reactivity continues to be inserted as a result of the void fraction continuing to drop due to a higher recirculation flow rate.
- Reactor power level continues to rise. This will gradually insert negative reactivity as a result of the fuel temperature increasing and negative Doppler coefficient.

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

Examination Outline Cross-reference:

Reactor Theory – FISSION PRODUCT POISONS
(292006K1.04) (10CFR55.41.1)
Describe the removal of xenon-135

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292006K1.04	
Importance Rating	2.9	------

Proposed Question: **# 71**

Unit 3 was operating at 75% RTP for the last 5 days when Reactor Power is returned to 100% RTP.

Which **ONE** of the following completes the statements below pertaining to the change in Xenon-135 concentration **IMMEDIATELY** following the return to 100% RTP over the course of the next 4 hours?

Xe-135 initially **(1)** due to the **(2)**.

A. (1) rises
(2) **HIGHER** rate of Xe-135 production from fission

B. (1) rises
(2) **LOWER** rate of Xe-135 burnout

C. (1) lowers
(2) **LOWER** rate of Xe-135 production from fission

D. (1) lowers
(2) **HIGHER** rate of Xe-135 burnout

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that production of Xe-135 takes time due to the decay of iodine. Second part incorrect but plausible in that eventually the production method of Xe-135 will overcome the removal method.

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that when Reactor Power is lowered, the rate of Xe-135 burnout lowers.

C INCORRECT: First part is correct (*See D*). Second part is incorrect but plausible if candidate confuses Xenon addition/removal processes.

D **CORRECT**: (*See attached*) The removal mechanism of Xe-135 is immediate, and thusly Xe-135 concentration will lower when flux is raised. Second part, production of Xe-135 takes time due to the decay of I-135, but eventually the production method will overcome the removal method and concentration will reach a new higher equilibrium value as it relates to burnout. In approximately 5 hours, the concentration of Xe-135 will stop lowering as it builds back into equilibrium over the next 25 hours.

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

RO Level Justification: Tests the candidate's knowledge of Generic Fundamentals, specifically of how a fission product poison such as Xe-135 responds following a Reactor Power change. This question is rated as C/A due to the requirement to correctly assemble given parameters from plant conditions. This requires mentally using this knowledge and its meaning to predict the correct outcome as it relates to fission product poisons and the removal of Xe-135.

Technical Reference(s): GFE Reactor Theory- Fission Product Poisons, Ch. 6, Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: GFES Reactor Theory, Ch. 6, Obj 4 (As available)

Question Source:

Bank #	
Modified Bank #	GFE Reactor Theory QIDB128
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:

BPR06Gr4_Fission_Products 2011May

1 / 0 QID: B128 Removal of xenon-135	
BWR TOPIC: 292006	PWR TOPIC: 192006
KNOWLEDGE: BK1.04 [2.9/2.9]	KNOWLEDGE: N/A
Similar questions	
Recent BWR exams	Recent PWR exams
Oct2001	Not used in 2001 through 2009Dec

Which one of the following describes the change in core xenon-135 concentration immediately following a power increase from equilibrium conditions?

A. Initially decrease due to the decreased rate of xenon-135 production from fission.

B. Initially decrease due to the increased rate of thermal neutron absorption by xenon-135.

C. Initially increase due to the increased rate of xenon-135 production from fission.

D. Initially increase due to the decreased rate of thermal neutron absorption by xenon-135.

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1 / 0 QID: B128 Removal of xenon-135

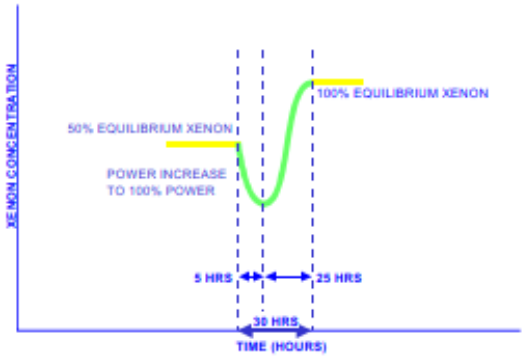
Explanation

Which one of the following describes the change in core xenon-135 concentration immediately following a power increase from equilibrium conditions?

$$\gamma_{Xe}\,\phi\Sigma_f^{fuel} + \gamma_I\,\phi\Sigma_f^{fuel} = \sigma_a^{Xe}\,N_{Xe}\,\phi + \lambda_{Xe}\,N_{Xe}$$
$$(5\%) \quad (95\%) \quad (80\%) \quad (20\%)$$

A. Initially decrease due to the decreased rate of xenon-135 production from fission.
Incorrect - since there is an increase in the amount of Xe-135 produced directly from fission. However it is initially offset by the large amount of Xe-135 experiencing burnout due to increased flux levels.

B. Initially decrease due to the increased rate of thermal neutron absorption by xenon-135.
CORRECT – even though both production of and removal of XE-135 are flux dependent, production takes time due to the decay of iodine. The removal mechanism is immediate, and thusly Xe-135 concentration will drop. Since flux was raised, eventually the production method will overcome the removal method and concentration will reach a new higher equilibrium value.



C. Initially increase due to the increased rate of xenon-135 production from fission.
Incorrect - there is an increase in the amount of Xe-135 produced directly from fission. However it is initially offset by the large amount of Xe-135 experiencing burnout due to increased flux levels

D. Initially increase due to the decreased rate of thermal neutron absorption by xenon-135.
Incorrect - since a power increase causes neutron flux to rise, which in turn caused neutron absorption by Xe-135 to rise.

The correct answer is ANSWER: B. Initially decrease due to the increased rate of thermal neutron absorption by xenon-135.

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Excerpt from BWR GFE Reactor Theory, Chapter 6, Fission Product Poisons:

UP-POWER TRANSIENT

If reactor power increases, an immediate increase in the burnout rate occurs due to the increased neutron flux level. An immediate increase in the direct Xe-135 production term also occurs. Since direct Xe-135 only accounts for about 5% of the Xe-135 produced, the burnout term predominates and causes the concentration of Xe-135 to decrease (Figure 6-5). As the concentration of the I-135 increases, more Xe-135 is produced, and in about five hours the concentration of Xe-135 stops decreasing. To maintain constant power during periods of xenon burnout, the operator must add negative reactivity. When Xe-135 concentration turns and begins increasing, the operator must begin adding positive reactivity. The higher power level eventually establishes a higher equilibrium level. The total time for the transient is 20 to 30 hours. The greater the power level changes, the greater the time required to re-establish the equilibrium.

Describe the Xe-135 concentration behavior in a startup from a xenon-free condition.

Example 6-7

A reactor has just started power operation and is ramped to full power in six hours. How long after the reactor reaches full power will it take to achieve an equilibrium xenon condition?

a. 8 to 10 hours

b. 20 to 30 hours

c. 40 to 50 hours

d. 70 to 80 hours

Example 6-8

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

Examination Outline Cross-reference:

293009 Thermodynamics – Core Thermal Limits
K1.10 (10CFR 55.41.14)
(MAPLHGR) Define APLHGR

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	293009K1.10	
Importance Rating	3.7	------

Proposed Question: **# 72**

Which **ONE** of the following completes the statement below pertaining to Thermal Limits?

Unit 1 was operating at 100% RTP, when a SCRAM occurs. Prior to the SCRAM, the amount of heat/power level stored in the fuel node resulting from the operating kW/foot (power level in a node divided by length of fuel rod in the node), is measured by the _____**.**

A. APLHGR-to-MAPLHGR ratio

B. average planar linear heat generation rate (APLHGR)

C. core fraction of limiting power density (MFLPD)

D. linear heat generation rate (LHGR) multiplied by the total peaking factor

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that numerous “ratios” pertaining to three thermal limits exist, such as, local peaking factor, FLCPR, MAPRAT.

B **CORRECT**: (*See attached*) The average planar linear heat generation rate (APLHGR) is average LHGR across particular node. The value of APLHGR is heat deposited in fuel divided by number of feet of fuel rod in node.

C INCORRECT: Incorrect but plausible in that MFLPD is defined as the Maximum LHGR divided by the LHGR limit.

D INCORRECT: Incorrect but plausible in that the linear heat generation rate (LHGR) is number of kilowatts of power produced per foot of fuel rod.

RO Level Justification: Tests the candidate’s knowledge of each of the three thermal limits (MCPR, LHGR, and APLHGR). This question is rated as Memory due to the requirement to strictly recall GFE Thermodynamic facts.

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

Technical Reference(s): GFE Thermodynamics, Core Thermal Limits, Ch. 9, Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: GFES Thermodynamics, Chapter 9, Obj. 8

Question Source:

Bank #	GFES Thermodynamics QID B297
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:

BPT09Gr4_Thermal_Limits 2011May.doc

2 / 0 QID: B297 APLHGR	
BWR TOPIC: 293009	PWR TOPIC: 193009
KNOWLEDGE: BK1.10 [3.3/3.7]	KNOWLEDGE:
Similar questions	
Recent BWR exams	Recent PWR exams
Dec2005, Oct2001	Not used in 2001 through 2009Dec

The amount of heat stored in the fuel, resulting from the operating kW/foot existing in the fuel prior to a scram, is measured by the...

A. average planar linear heat generation rate (APLHGR).

B. linear heat generation rate (LHGR) multiplied by the total peaking factor.

C. core fraction of limiting power density.

D. APLHGR-to-MAPLHGR ratio.

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BPT09Gr4_Thermal_Limits 2011May.doc

2 / 0 QID: B297 APLHGR
Explanation
The amount of heat stored in the fuel, resulting from the operating kW/foot existing in the fuel prior to a scram, is measured by the... The average planar linear heat generation rate (APLHGR) is the average LHGR across a particular node. The value of APLHGR is the heat deposited in the fuel divided by the number of feet of fuel rod in the node. A. average planar linear heat generation rate (APLHGR). CORRECT - B. linear heat generation rate (LHGR) multiplied by the total peaking factor. Incorrect – linear heat generation rate (LHGR) multiplied by the total peaking factor is the maximum local linear power density C. core fraction of limiting power density. Incorrect – this is just a distracter D. APLHGR-to-MAPLHGR ratio. Incorrect – no such limit, this is just a distracter The correct answer is ANSWER: A. average planar linear heat generation rate (APLHGR).

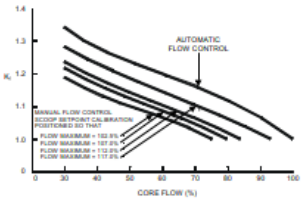
Excerpts from GFE Thermodynamics, Core Thermal Limits, Ch. 9:

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
	e. Therefore, it is average heat generation rate in node, rather than peak heat generation rate in rod, that more accurately reflects heatup characteristics of fuel during LOCA
Objective 8	3. The average planar linear heat generation rate (APLHGR) is average LHGR across particular node
	a. The value of APLHGR is heat deposited in fuel divided by number of feet of fuel rod in node
Objective 11	b. The maximum average planar heat generation rate (MAPLHGR) (pronounced mapple-hugger) in core is area of concern
	c. During design basis LOCA, this node would have most difficult time in dissipating heat that node has stored and that is generated due to decay heat
Equation 9-13 / TP 9-23 $MAPLHGR = \frac{P_n(1-f_c)}{N \times L}$ Where: P_n = nodal power (kW); f_c = fraction of power deposited directly in coolant (decimal); N = number of fuel pins per bundle; L = length of node (ft)	4. To determine MAPLHGR for each node following formula is used:
Example 9-5 / TP 9-24 Calculate MAPLHGR, given node generating 0.3177 MW in 8 × 8R fuel bundle with 6% of power deposited directly in coolant. Note:	

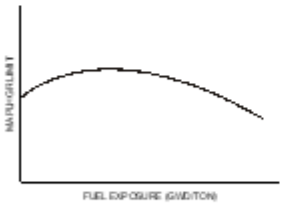
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KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Objective 20.d	5. Local Peaking Factor a. For given power distribution and given plane in bundle, highest quality is expected to occur around hottest pin b. As we have seen, ratio of hottest pin power, compared to average pin power, in plane is known as local peaking factor c. As local peaking factor increases at given power level, it is expected that hottest pin is closer to OTB d. Therefore, critical power is expected to decrease as local peaking increases
Objective 20.e	6. Axial Peaking Factor a. The critical power of bundle also depends on axial power distribution b. In bundle with bottom power peak, most rapid enthalpy rise in coolant occurs in areas of low quality or even subcooled liquid, thus providing large margin to OTB c. In top-peaked bundle, largest coolant enthalpy rise occurs in high quality region that is closer to OTB d. Therefore, critical power is expected to be lower for top-peaked power distribution

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KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
	c. Feedwater controller failure (maximum demand) L. Maximum Fraction Of Limiting Critical Power Ratio (MFLCPR) a. The process computer calculates CPR data evaluating core conditions to ensure limits are not exceeded.
Objective 23	b. One of most useful forms of this data output is ratio called "fraction of limiting critical power ratio" (FLCPR)
Equation 9-18 / TP 9-36 $FLCPR = \frac{CPR_{Limit}}{CPR}$	c. This ratio compares operating (steady-state) maximum CPR for fuel bundle to actual bundle CPR
	d. From this, maximum fraction of limiting critical power ratio (MFLCPR - pronounced "miffle-sipper"), which is maximum fraction of limiting critical power ratio (MFLCPR) and is ratio of flow-adjusted CPR operating limit for that fuel type to bundle CPR, is developed
Equation 9-19 / TP 9-36 $MFLCPR = \frac{CPR_{Limit} \times K_f}{CPR}$	e. For most nuclear plants MFLCPR ratio takes following form:
Objective 24	2. Here adjustment factor is called flow biasing correction factor (K_f) a. It is shown in Figure 9-8
Figure 9-8 / TP 9-37 	b. It adjusts MFLCPR at core flows less than rated c. The upper curve is to be used when operating in automatic flow control mode, to prevent violation of operating limit MCPR if flow increases to maximum flowrate allowed by Recirculation System

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KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
An 8 × 8R fuel bundle incorporates two water rods. $MAPLHGR = \frac{317.7\,kW(1-0.06)}{62\times 0.5\,ft}$ $MAPLHGR = 9.63\,kW/ft$	
Equation 9-14 / TP 9-25 $MAPRAT = \left(\frac{MAPLHGR_{actual}}{MAPLHGR_{LCO}}\right) < 1.0$	E. Maximum Average Planar Ratio (MAPRAT) 1. The limiting value of MAPLHGR is found in Technical Specifications and is function of fuel type and exposure (core age) a. The ratio of actual MAPLHGR value to limiting condition for operation (LCO) value of MAPLHGR is maximum average planar ratio (MAPRAT) b. The value of MAPRAT must always be less than 1.0
Objective 12 **Figure 9-2 / TP 9-26** 	2. Maintaining MAPRAT less than 1.0 ensures that peak cladding temperature does not exceed 2,200°F during design basis LOCA 3. In Siemens-fueled reactors, MAPLHGR limit is adjusted for exposure and off-rated conditions of power and flow in same fashion as is MLHGR limit
Objective 14 **Equation 9-15 / TP 9-27** $\dot{Q} \propto K(T_{CL} - T_{SURFACE})$	4. The mathematical relationships between pellet centerline temperature, heat transfer rate, and thermal conductivity are complex because of pellet geometry and fact that heat is generated throughout, and transferred through, pellet a. This relationship can be simplified by following:

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KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Equation 9-11 / TP 9-21 $LHGR_{limit} = LHGR_{design} \times \left[1 - \left(\frac{\Delta P}{P}\right)_{max}\right] \times \left(\frac{L}{LT}\right)$ Where: $LHGR_{design}$ = 17.5 kW/ft for 7 × 7 fuel $\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty (0.037 for 7 × 7 fuel) L = Axial position in feet above bottom of core LT = Total active core length (12 ft) **Objective 7** **Equation 9-12 / TP 9-22** $FLPD = \frac{LHGR}{LHGR_{limit}}$ $MFLPD = \frac{Maximum\ LHGR}{LHGR_{limit}}$ $MFLPD = \frac{MLHGR}{LHGR_{limit}}$	c. This phenomenon is known as power spiking 3. This results in higher heat flux generated in that region of affected bundle(s) a. To compensate for this effect, LHGR is adjusted for elevation of node as follows: C. Maximum Fraction Of Limiting Power Density (MFLPD) 1. The process computer compares LHGR in node to that node's limit a. This ratio is called fraction of limiting power density (FLPD) b. The fraction of limiting power density is defined as: 2. As long as largest FLPD is less than one, we are assured that LHGR limit is not exceeded 3. Similarly, process computer determines maximum fraction of limiting power density (MFLPD) (pronounced "muffle-pud") which is defined as:

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
	c. This limit satisfactorily protected core from damage, but placed unnecessarily restrictive limit on heat flux when operating at less than rated power 8. Current Technical Specifications use linear heat generation rate (LHGR) as replacement limit for MTPF limit a. LHGR is related to MTPF as follows: 9. The linear heat generation rate is discussed in more detail in next section
Equation 9-8 / TP 9-16 $MTPF = \frac{\text{Maximum LHGR (MLHGR)}}{\text{Core Average LHGR}}$	IV. THERMAL LIMITS A. Linear Heat Generation Rate
Objective 4	1. The linear heat generation rate (LHGR) is number of kilowatts of power produced per foot of fuel rod a. For example, if maximum thermal output (100% power) was divided by total number of fuel assemblies, number of fuel rods per fuel assembly, and length of each fuel rod, value of average LHGR would be obtained b. Said another way, each foot of fuel rod would have to produce this average LHGR so that total core would produce its maximum thermal power
Equation 9-9 / TP 9-16 $\text{Average LHGR} = \frac{\text{Maximum Thermal Output}}{\left(\begin{array}{c}\text{Number}\\ \text{of Fuel}\\ \text{Assemblies}\end{array}\right)\left(\begin{array}{c}\text{Number of}\\ \text{Fuel Rods per}\\ \text{Fuel Assembly}\end{array}\right)\left(\begin{array}{c}\text{Length}\\ \text{of Each}\\ \text{Fuel Rod}\end{array}\right)}$	c. The average LHGR would be equal to:
Objective 5	2. The amount of power generated by reactor core is limited by material temperature constraints

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SUMMARY QUESTIONS

35. Q: How do iodine and cadmium affect PCI?

A: Iodine and cadmium are embrittling agents which can cause stress corrosion cracking in cladding under high stress.

36. Q: What is the purpose of Pre-Conditioning Interim Operating Management Recommendations (PCIOMR)?

A: PCIOMR are intended to reduce cladding failures caused to PCI by limiting the rate of power increases above certain power levels and by holding power at certain levels to allow the clad to expand elastically.

37. Q: How does operation within the PCIOMR minimize PCI?

A: By adhering to PCIOMR limitations on rates of power increase and fuel "soak" time, the operator reduces the stress imposed on the clad, minimizing PCI effects.

38. Q: For each of the three thermal limits (MCPR, LHGR, and APLHGR), state the fuel parameters associated with each.

A: MCPR – total power in a fuel bundle

LHGR – total fuel pin power in a node

APLHGR-average fuel pin power in a node

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

Examination Outline Cross-reference:	Level	RO	SRO
Thermodynamics - STEAM	Tier #	4	------
293003 K1.22 (10CFR 55.41.14)	Group #	------	------
Explain the usefulness of steam tables to the control room operator.	K/A #	293003K1.22	
	Importance Rating	3.2	------

Proposed Question: **# 73**

Unit 2 is performing a shutdown in accordance with 2-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations, with the following conditions:

- NUSO has directed the Operator to commence a cooldown using EHC
- Reactor Pressure is 965 **psig**

Which **ONE** of the following completes the statements below?

Given the above conditions, the corresponding Moderator Temperature is currently **(1)**.

In accordance with the given GOI, the **MAXIMUM** cooldown rate allowed is **(2)**. per hour.

[REFERENCE PROVIDED]

A. (1) 540 °F
(2) 80 °F

B. (1) 540 °F
(2) 90 °F

C. (1) 542 °F
(2) 80 °F

D. (1) 542 °F
(2) 90 °F

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible if the given 965 psig is not properly converted to 980 psiA to determine the corresponding Saturation Temperature. Table 1, Saturated Temperature, page 9: Using 962.23 psiA incorrectly for 965 psiG in an attempt to locate 540 °F. Additionally, Steam Tables contain Superheated Steam Tables which adds another obstacle to overcome for candidates. Wrong table, wrong answer. Second part is incorrect but plausible in that the examinee may not know the maximum cooldown rate allowed IAW 2-GOI-100-12A.

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

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

D **CORRECT**: (*See attached*) For a saturated steam-water mixture, the temperature and pressure do not change under the vapor dome. Knowing the temperature of a saturated steam-water mixture allows the applicable pressure to be located using the steam tables. And vice versa, knowing the pressure of a saturated steam-water mixture allows the applicable temperature to be located using the steam tables. The given Reactor Pressure 965 psig must first be converted to absolute pressure (Pabs) using an equation not given from GFE handout.
Pabs=Patm (14.7 or 15) + 965 psig, results in 979.7 or 980 psiA. Using Steam Tables, Table 2, Saturated Steam: Pressure, page 12, locate 980 psiA down the left hand column. To the right in the Temp Fahr, (t) column, locate 542.206 which is the corresponding Saturation Temperature for Reactor Pressure 965 psig (980 psiA). Second part, maximum cooldown rate is 90°F per hour in accordance with 2-GOI-100-12A.

RO Level Justification: Tests the candidate's knowledge of Generic Fundamentals, specifically the usefulness of steam tables to a Control Room Operator as it relates to Thermodynamics topic for steam. With references provided, 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.

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

Technical Reference(s): GFE Thermodynamics-Steam, Ch. 3, Rev. 4 (Attach if not previously provided)

2-GOI-100-12A, Rev. 127

Proposed references to be provided to applicants during examination: Steam Tables

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

Question Source:

Bank #	
Modified Bank #	GFES Thermodynamics QID B139 (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:

0 / 0 QID: B139 T_{SAT} / P_{SAT} / Use of Saturated and Superheated Steam Tables	
BWR TOPIC: 293003	PWR TOPIC: 193003
KNOWLEDGE: BK1.23 [2.8/3.1]	KNOWLEDGE:
Similar questions	
Recent BWR exams	Recent PWR exams
Not used in 2001 through 2007	Not used in 2001 through 2007

The saturation temperature for steam at a pressure of 785 psig is approximately...

A. 510°F.

B. 513°F.

C. 515°F.

D. 518°F.

0 / 0 QID: B139 T_{SAT} / P_{SAT} / Use of Saturated and Superheated Steam Tables
Explanation
The saturation temperature for steam at a pressure of 785 psig is approximately... CAUTION – to use steam tables pressures must be in psia Pabs = Patm + Pgauge 785 psig Pabs = Patm + Pgauge Pabs = 15 + 785 Pabs = 800 psia Look up from steam table 518.21°F The correct answer is ANSWER: D. 518°F.

Excerpts from BWR Thermodynamics, Ch. 3:

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Objective 7	3. Properties of steam have been extensively measured over most ranges of interest. Steam tables were developed to contain this experimentally derived information 4. Operators use steam tables to verify plant instrumentation and examine equipment performance 5. Steam tables allow operator to have additional information of plant conditions, for example, determining exact steam temperature from pressure readings 6. These tables give the various properties of steam at a given state point, which is usually fixed by the measured properties of temperature or pressure. B. Steam tables list property values of temperature, pressure, specific volume, specific enthalpy, and specific entropy 1. To establish values of these properties, it was necessary to start from base temperature 2. The base of steam tables was selected as triple point of water (32.02°F) 3. This choice was purely arbitrary, as any other temperature, such as 0°F, might have been selected 4. However, the property values of steam are not ordinarily needed at such low temperatures; and 32.02°F, which is near the temperature of freezing water at standard atmospheric pressure, is a temperature that is conveniently reproducible as a starting point. C. One property not included in steam tables is specific heat (c_p) 1. Specific heat (c_p) is the quantity of heat required to raise the temperature of one pound mass of a substance one degree (Btu/lbm °F).

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KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Table 3-1 / TP 3-21 through TP 3-22 T = temperature (°F) P = pressure (psia) υ = specific volume (ft^3/lb_m) υ_f = specific volume of saturated liquid (ft^3/lb_m) υ_{fg} = specific volume change due to vaporization (ft^3/lb_m) υ_g = specific volume of saturated vapor (ft^3/lb_m) h = specific enthalpy (Btu/lb_m) h_f = specific enthalpy of saturated liquid (Btu/lb_m) h_{fg} = specific enthalpy change due to vaporization (Btu/lb_m) h_g = specific enthalpy of saturated vapor (Btu/lb_m) s = specific entropy (Btu/lb_m °R) s_f = specific entropy of saturated liquid (Btu/lb_m °R) s_{fg} = specific entropy change due to vaporization (Btu/lb_m °R) s_g = specific entropy of saturated vapor (Btu/lb_m °R) Sh = number of degrees of superheat (°F) u = specific internal energy (Btu)	2. The ASME Steam Tables "Thermodynamic and Transport Properties of Steam", published by American Society of Mechanical Engineers (ASME). E. The information in two sets of tables agrees closely 1. Highly accurate information on properties of subcooled water, saturated steam, and superheated steam may be obtained from either source 2. The formats of tables differ so you should become familiar with both sets of tables. F. Steam tables consist of two sets of tables for energy transfer property values of water and steam: (1) saturated steam tables and (2) superheated steam tables 1. Both sets of tables are tabulations of values for pressure, temperature, specific volume, specific enthalpy, and specific entropy 2. The following notations are used in steam tables: 3. Note: Some steam tables use lower case letter "v" to denote specific volume instead of Greek letter "upsilon" (υ), since there is little possibility of confusing it with velocity 4. Enthalpy (h) on some tables is replaced with internal energy (u). As with any reference, refer to associated legend to confirm symbols used.

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KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
Objective 11.i **Table 3-2 / TP 3-23** PORTION OF A TYPICAL SATURATED STEAM: TEMPERATURE TABLE Specific Volume / Enthalpy / Entropy Temp. (°F) t Abs. Press. (lbf/sq in.) p Sat. Liquid v_f Evap. v_{fg} Sat. Vapor v_g Sat. Liquid h_f Evap. h_{fg} Sat. Vapor h_g Sat. Liquid s_f Evap. s_{fg} Sat. Vapor s_g Temp. (°F) t PORTION OF A TYPICAL SATURATED STEAM: PRESSURE TABLE Specific Volume / Enthalpy / Entropy Abs. Press. (lbf/sq in.) p Temp. (°F) t Sat. Liquid v_f Evap. v_{fg} Sat. Vapor v_g Sat. Liquid h_f Evap. h_{fg} Sat. Vapor h_g Sat. Liquid s_f Evap. s_{fg} Sat. Vapor s_g Abs. Press. (lbf/sq in.) p	G. Use of Saturated Steam Tables 1. The saturated steam tables give energy transfer property values of saturated water and saturated steam for temperatures from 32°F to 705.47°F (critical temperature) and for corresponding absolute pressures of 0.08865 psia to 3,208.2 psia (critical pressure) 2. Normally, saturated steam tables are divided into two parts: a. Temperature tables, which list property values according to saturation temperature (T_{sat}) b. Pressure tables, which list property values according to absolute saturation pressure (P_{sat}) 3. Some tables refer to saturation pressure as "vapor pressure" a. Table 3-2 shows portion of typical saturated steam temperature table and portion of typical saturated steam pressure table 4. The values of specific enthalpy and specific entropy provided in these tables are measured relative to properties of saturated liquid at 32.02°F a. Hence, specific enthalpy of saturated liquid (h_f) and specific entropy of saturated liquid (s_f) have values of near zero at 32°F

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Excerpts from Steam Tables, Table 2, Saturated Steam: Pressure, page 12:

Table 2: Saturated Steam: Pressure

Abs Press. Lb/Sq in. p	Temp Fahr t	Specific Volume Sat. Liquid v_f	Specific Volume Evap v_{fg}	Specific Volume Sat Vapor v_g	Enthalpy Sat. Liquid h_f	Enthalpy Evap h_{fg}
0.1	35.005	0.016020	2945.0	2945.0	3.009	107
0.2	53.132	0.016027	1525.9	1525.9	21.204	106
0.4	72.834	0.016057	791.84	791.86	40.911	105
[illegible]	[illegible]	0.016087	539.89	539.90	53.242	104
[illegible]	532.022		0.02124	0.4798	0.5011	
940	537.197		0.02138	0.4563	0.4777	
980	542.206		0.02153	0.4347	0.4562	
1020	547.061		0.02167	0.4146	0.4363	
1060	551.773		0.02182	0.3960	0.4178	

Table 1, page 9: Using 962.23 psiA incorrectly for 965 psiG in an attempt to locate 540: Supports Distractors A1, B1

Table 1: Saturated Steam: Temperature Table--*Co*

Temp Fahr t	Abs Press. Lb per Sq in. p	Specific Volume Sat. Liquid v_f	Evap v_{fg}	Sat Vapor v_g	Sat. Liquid h_f	En E
460	466.71	0.019619	0.9755	0.9952	441.50	
464	485.39	0.019694	0.9368	0.9565	446.05	
468	504.64	0.019771	0.8997	0.9195	450.62	
472	524.47	0.019849	0.8643	0.8841	455.21	
476	544.89	0.019928	0.8304	0.8503	459.82	
480	565.92	0.02001	0.7980	0.8180	464.44	
484	587.56	0.02009	0.7669	0.7870	469.09	
488	609.83	0.02018	0.7372	0.7573	473.75	
492	632.74	0.02026	0.7087	0.7289	478.44	
496	656.30	0.02035	0.6813	0.7017	483.15	
500	680.53	0.02044	0.6551	0.6756	487.89	
504	705.43	0.02053	0.6300	0.6506	492.65	
508	731.03	0.02063	0.6059	0.6265	497.43	
512	757.32	0.02072	0.5828	0.6035	502.24	
516	784.34	0.02082	0.5605	0.5814	507.07	
520	812.08	0.02092	0.5392	0.5601	511.93	
524	840.57	0.02103	0.5186	0.5397	516.82	
528	869.81	0.02113	0.4989	0.5200	521.74	
532	899.83	0.02124	0.4799	0.5012	526.69	
536	930.63	0.02135	0.4617	0.4830	531.67	
540	962.23	0.02146	0.4441	0.4656	536.69	
544	994.64	0.02158	0.4272	0.4487	541.73	
548	1027.9	0.02170	0.4109	0.4326	546.82	

Excerpt from 2-GOI-100-12A:

3.0 PRECAUTIONS AND LIMITATIONS

3.1 General:

A. Unit SRO's permission is required to reject water to main condenser from Reactor Water Cleanup (RWCU) System without RWCU filter in service.

B. Maximum cooldown rate is 90°F per hour.

C. One method of Reactor decay heat removal should not be stopped prior to establishing another method.

D. [II/F] Prior to initiating any event that adds or has potential to add heat energy to the Suppression Chamber, the Unit SRO shall evaluate the necessity of placing Suppression Pool Cooling in service. This is due to the potential of developing thermal stagnation during sustained heat additions. [II-B-91-129]

E. Prior to opening any MSIV or MSIV drain isolation valve following Mode 4 or 5, operators are to verify secondary containment integrity will be maintained. Both an inboard and outboard MSIV on the same main steam line OR an inboard MSIV and both main steam line drain isolation valves are NOT to be opened at the same time unless Unit 2 Outage SRO or Secondary Containment System engineer confirm no work in progress which could violate Secondary Containment.

   Work is not to commence on an outboard MSIV unless 2-OI-64 has been addressed for the requirement to use extended secondary containment.

F. Operators are to identify and use Multiple and Diverse Indication of Reactor Water Level (RWL) during significant changes in RWL inventory. [INPO IER: 3-12-19]

G. 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.

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

Examination Outline Cross-reference:	Level	RO	SRO
THERMODYNAMIC CYCLES	Tier #	4	------
(293005K1.05) (10CFR55.41.14)	Group #	------	------
State the advantages of moisture separators/reheaters and feedwater heaters for a typical steam cycle	K/A #	293005K1.05	
	Importance Rating	2.8	------

Proposed Question: **# 74**

Unit 1 is operating at 100% RTP when the 1A1 High Pressure Feedwater Heater is required to be removed from service due to possible tube leakage.

Which **ONE** of the following completes the statements below regarding Feedwater Heaters and moisture separators?

Given the above conditions, after 1A1 High Pressure Feedwater Heater is removed from service, Unit 1's Main Generator output will be **(1)**.

By design, the moisture separator protects the Main Turbine's **(2)** pressure turbine from impingement damage.

A. (1) lower
(2) low

B. (1) lower
(2) high

C. (1) higher
(2) low

D. (1) higher
(2) high

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that when any other non-number 1, Feedwater Heater is removed from service, slightly lower generator output will occur. 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 examinee may easily confuse the location of the Moisture Separators in the Main Turbine steam flow path.

C **CORRECT**: (*See attached*) If a number 1 Feedwater Heater is removed from service, generator output will be higher. This occurs because normally extracted steam now passes through low pressure turbine.
Second part, the purpose of the moisture separators is to remove the excess moisture entrained in the steam exhausting from the high-pressure turbine prior to it going to the LP turbines. Removing the moisture protects the low-pressure turbine blading from impingement damage.

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 advantages of Feedwater Heaters and moisture separators on a typical steam cycle, as well as the impact to generator output. This question is rated as Memory due to the requirement to strictly recall procedural facts and systems training information.

Technical Reference(s): BWR Thermodynamics, Ch. 5, Thermodynamic Cycles, Rev. 4 (Attach if not previously provided)
OPL171.010, Rev. 18
1-OI-6, Rev. 51

Proposed references to be provided to applicants during examination: NONE

Learning Objective: GFES Thermal Hydraulics, Chapter 8, Obj. 24
OPL171.010, Obj. 15

Question Source:

Bank #	
Modified Bank #	GFES Thermal Hydraulics QID B129
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:

0 / 0 QID: B129 Moisture Separators/Reheaters and Feedwater Heaters	
BWR TOPIC: 293005	PWR TOPIC: 193005
KNOWLEDGE: BK1.05 [2.7/2.8]	KNOWLEDGE:
Similar questions	
Recent BWR exams	Recent PWR exams
Not used in 2001 through 2007	Not used in 2001 through 2007

What is the effect of isolating extraction steam to a high-pressure feed water heater while at 90% of rated power?

A. The core inlet subcooling remains the same while the turbine generator MWe output decreases.

B. The core inlet subcooling and the reactor power (MWt) decrease.

C. The reactor power (MWt) and the turbine generator MWe output remain the same.

D. The core inlet subcooling increases and the turbine generator MWe output increases.

0 / 0 QID: B129 Moisture Separators/Reheaters and Feedwater Heaters
D. The core inlet subcooling increases (TRUE) and the turbine generator MWe output increases. (TRUE) CORRECT – see above

The correct answer is ANSWER D. The core inlet subcooling increases and the turbine generator MWe output increases.

0 / 0 QID: B129 Moisture Separators/Reheaters and Feedwater Heaters

Explanation

What is the effect of isolating extraction steam to a high-pressure feed water heater while at 90% of rated power?

Feedwater preheating is the process in which a fraction of the steam from the turbine is removed at some intermediate point and used to preheat the condensate and feedwater prior to its being returned to the boiler. In practice, there are several stages of feedwater preheating with the steam extracted from intermediate stages of the HP and LP turbines. The heat rejected, (removed with the steam extracted to the feedwater heaters), is not lost to the cooling water in the condenser. In this case, the heat rejected is used to preheat the feedwater in the feedwater preheaters. Therefore, the total heat rejected from the cycle is reduced, and plant efficiency increases.

If feedwater heater extraction steam is secured, feedwater on the way to the boiler is not warmed by the heat of the extraction steam. A loss of feedwater heating means cooler water will enter the boiler (core inlet subcooling INCREASES). This feedwater still needs to be heated to the boiling point in the boiler. This means additional energy from the fission process must be used to raise the cooler feedwater to operating temperatures. To create the extra heat necessary to warm the cooler feedwater, reactor thermal power must increase.

This reactor thermal power increase occurs because the steam that was formerly used to preheat the feedwater is not preheating the feedwater. However this steam will now continues through the turbine performing additional work before being exhausted to the condenser. Therefore, turbine generator MWe output INCREASES.

A. The core inlet subcooling remains the same (FALSE) while the turbine generator MWe output decreases. (FALSE)

   Incorrect – less steam to heat the feedwater in the feedwater heaters means cooler feedwater and an INCREASE in core inlet subcooling. Extraction steam no longer routed to the feedwater supplies more energy to the remaining turbine stages INCREASING turbine generator MWe output with a corresponding increase in reactor power due to the positive reactivity addition of the now colder feedwater.

B. The core inlet subcooling (*decreases*) (FALSE) and the reactor power (MWt) decrease. (FALSE)

   Incorrect - less steam to heat the feedwater in the feedwater heaters means cooler feedwater and an INCREASE in core inlet subcooling. The cooler feedwater entering the reactor will add positive reactivity causing an increase in reactor power (MW_t).

C. The reactor power (MWt) (*remain the same)* (FALSE) and the turbine generator MWe output remain the same. (FALSE)

   Incorrect - A loss of feedwater heating means cooler water will enter the boiler (core inlet subcooling increases). This feedwater still needs to be heated to the boiling point in the boiler. This means additional energy from the fission process must be used to raise the cooler feedwater to operating temperatures. To create the extra heat necessary to warm the cooler feedwater, reactor thermal power must INCREASE. However the steam normally removed to provide feedwater heating will now continues through the turbine performing additional work before being exhausted to the condenser. Therefore, turbine generator MWe output INCREASES.

Excerpts from 1-OI-6:

3.2.3 Heater Isolations (continued)

L. Two signals are required to isolate extraction steam to a heater string. One is from a float switch, the other from an electric contact (LS-6-1HL) generated from the high level alarm circuit. LS-6-1HL will close on high heater level (this is a confirmatory signal and is a lower level than the float switch) OR on a Failsafe Delay Signal(FSDLAY) from the FWHLCS (this is caused by any loss of communication greater than (TBD during U1R11) seconds in the FWHLCS). Therefore, if a float switch is stuck closed, the heater will isolate when LS-6-1HL closes for either reason. See 1-45E777-32.

3.2.4 Removing Feed Water Heater

A. System parameters are allowed to stabilize after each step is performed before proceeding.

B. Limitations of Attachment 1, Maximum Turbine-Generator Load Allowed When Any Feedwater Heater is Out-of-Service, are followed when removing Feedwater heaters from service.

C. When more than one condensate and condensate booster pump is IN-SERVICE, tube side of two high or low pressure Feedwater heater strings CANNOT be isolated at same time.

D. Removing Feedwater heaters from service results in higher Reactor Power due to rising moderation of neutrons. Therefore Reactor Power should be closely monitored when Feedwater heaters are removed from service.

E. Extraction Steam is shut-off prior to isolating tube side of Feedwater heater.

F. Assuming throttle steam flow is NOT changed, the following can occur when Feedwater heater is removed from service:

1. If number one Feedwater heater is removed from service, generator output will be higher. This occurs because normally extracted steam now passes through low pressure turbine.

2. When other than number one Feedwater heater is removed from service, extraction to next higher Feedwater heater will be higher because Feedwater temperature rise across heater is greater than before. Slightly lower generator output can occur.

3. Turbine thrust bearing loading can be higher due to load imbalance caused by loss of extraction point.

4. When operating with Feedwater heater(s) out-of-service there can be loss of efficiency.

BFN Unit 1	Feedwater Heating and Misc Drains System	1-OI-6 Rev. 0051 Page 417 of 430

Attachment 1
(Page 1 of 2)

Maximum Turbine-Generator Load Allowed When Any Feedwater Heater is Out-of-Service

HEATERS OUT (Tube and Shell Side)(note 1)

	Total Feedwater Flow (Mlb/hr)(note 2)	Generator (MWe)
One HP string	11.4 Mlb/hr	1152 MWe
One LP string	12.0 Mlb/hr	1024 MWe
One HP and LP string	11.4 Mlb/hr	960 MWe

(1)Operational limit is the more restrictive of either Total FW Flow (Mlb/hr) or Generator Output (MWe)

(2 Total Feedwater flow rates are the expected flow through the Feedwater Heaters with the maximum number of tubes plugging. If higher reactor power levels are anticipated, contact Site Engineering for assistance in determining the total FW flow based on actual number of tubes plugged.

BFN Unit 1	Feedwater Heating and Misc Drains System	1-OI-6 Rev. 0051 Page 418 of 430

Attachment 1
(Page 2 of 2)

HEATERS OUT (EXTRACTION STEAM ONLY)(1)

These MWe limits are only evaluated for a maximum of 3952 MWt.(note 3)

1A1	1A2	1A3	1B1	1B2	1B3	1C1	1C2	1C3	Max Gen Output
X									No Restriction
	X								1216
		X							1216
	X	X							1152
X	X								1152
X	X				X				1152
X			X						No Restriction
X		X							1152
X	X		X						1152
X	X	X							1152
X			X			X			No Restriction
X		X	X						1152
X	X		X	X					1152
X		X			X				1152
X	X		X			X			1152
X	X	X	X						1152
X	X	X			X				1152
X	X	X				X			1152
X	X		X	X		X			1152
X	X	X	X	X					1152
X		X			X			X	1088
X	X	X	X			X			1152
X	X		X	X		X	X		1152
X	X	X	X		X				1152
X	X	X	X	X		X			1152
X		X	X		X			X	1088
X	X	X	X	X	X				1088
X	X	X	X		X	X			1152
X	X	X	X	X		X	X		1152
X	X	X	X		X			X	1152
X		X	X		X	X		X	1088
X	X	X	X	X	X	X			1088
X	X	X	X	X	X			X	1088
X	X	X	X		X	X		X	1088
X	X	X	X	X	X	X	X		1088
X	X	X	X	X	X	X		X	1088
X	X	X	X	X	X	X	X	X	1088

(3)Limitations apply to combinations indicated and equivalent combinations. For instance, restriction for operation with heaters 1A1 and 1A2 out-of-service is equally applicable to operation with heaters 1B1 and 1B2 or 1C1 and 1C2 out-of-service. To try to clarify: Any number one heater may be substituted for any other number one heater, any number two heater may be substituted for any other number two heater, and any number three heater may be substituted for any other number three heater. Contact System Engineer for further clarification.

Excerpt from OPL171.010 Lesson Plan:

OPL171.010 MAIN TURBINE REV 18

Lesson Plan Content

Outline of Instruction	Instructor Notes (Optional)
I. Moisture Separators	Figure-1
1. Purpose	ILT-15
a. The purpose of the moisture separators is to remove the excess moisture entrained in the steam exhausting from the high pressure turbine prior to it going to the LP turbines	NLO-17
b. Removing the moisture protects the low pressure turbine blading from impingement damage/moisture erosion as the steam passes through the turbines at a high velocity.	NLO-18
c. This is accomplished without any reheating by removing the low enthalpy moisture from the steam which increases the average enthalpy of the exit steam.	
2. Flow Path	
a. The steam exits the high pressure turbine through six exhaust lines.	Print 47E801-2
b. Each of the exhaust lines goes to one of the six moisture separators.	
c. The steam lines enter the moisture separator	
d. The exit steam then passes through the combined intermediate valve stop valves and intercept valves before entering the low pressure turbines.	
e. Relief valves are installed in each line leaving the moisture separators. The relief valves protect the moisture separator and cross around piping from overpressure in the event the combined intermediate valves close and the stop and control valves remain open. The relief valves discharge to their respective condensers.	ILT-4

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

Examination Outline Cross-reference:

292008 Reactor Theory – Reactor Operational Physics

K1.30 (10CFR 55.41.1)

(NORMAL REACTOR SHUTDOWN) Explain the relationship between decay heat generation and (a) power level history, (b) power production, and (c) time since reaction shut down

Level	RO	SRO
Tier #	4	------
Group #	------	------
K/A #	292008K1.30	
Importance Rating	3.5	------

Proposed Question: **# 75**

Unit 1 was operating at 100% RTP for 10 months, when a Reactor SCRAM occurred.

IMMEDIATELY following the Reactor SCRAM, decay heat is approximately **(1)**, due to **(2)** .

A. (1) 1%
(2) delayed neutrons

B. (1) 1%
(2) fission products

C. (1) 7%
(2) delayed neutrons

D. (1) 7%
(2) fission products

Proposed Answer: **D**

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 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.

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)* 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.

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): GFE Reactor Theory, Physics, Ch. 8, Rev. 4 (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 #	BFN NRC 2304 #72
Modified Bank #	
New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2023

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.

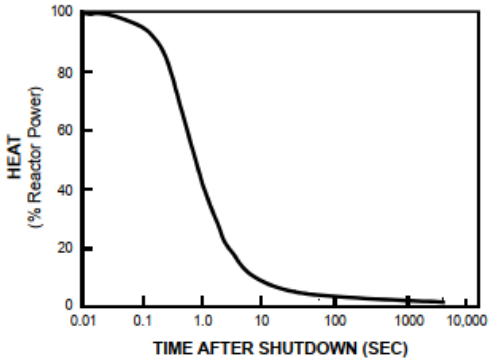


Figure 8-9 Decay Heat vs. Time after Shutdown

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

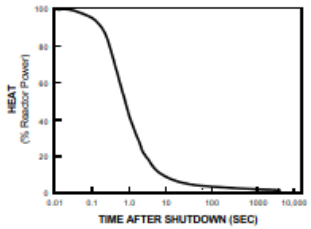
Example 8-14

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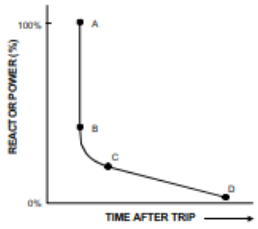
Excerpts from BWR GFES Reactor Theory, Chapter 8, Operational Physics, Instructor Guide:

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

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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

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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

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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

G2.2.23 (10CFR 55.43.2 - SRO Only)

Ability to track technical specification limiting conditions for operation

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295001G2.2.23	
Importance Rating	------	4.6

Proposed Question: **# 76**

Unit 2 is operating at 100% RTP with 2-SR-3.4.2.1, Jet Pump Mismatch and Operability, in progress when the following occurs:

- Operator reports:
  - 2-XR-68-50, Total Core Flow is 99.56 Mlb/hr
  - Calculated Recirculation Loop Jet Pump Flow mismatch is 7 Mlb/hr
- As a result, the NUSO declares performance of the SR UNSATISFACTORY

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

In accordance with Tech Spec Bases 3.4.1, Recirculation Loops Operating, the Recirculation Loop with the **(1)** flow must be considered **NOT** in operation.

In accordance with OPDP-8, Operability Determination Process and Limiting Conditions for Operation Tracking, if the INOPERABILITY will extend past the end of the **(2)**, then an eSOMS LCO Tracking Record is required.

A. (1) lower
(2) day

B. (1) lower
(2) shift

C. (1) higher
(2) day

D. (1) higher
(2) shift

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that OPDP-8 discusses operability determination as an ongoing day-to-day process, Three Required Entry Criteria (TREC) Criterion 3 impact of system leakage per day, Functional Evaluation completions within three working days, etc. Because of the various timeliness requirements specified in OPDP-8 ranging from shift to days, selection of the incorrect "day" is plausible.

B **CORRECT:** (*See attached*) In accordance with Tech Spec Bases 3.4.1, Recirculation Loops Operating, 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.

Second part, in accordance with OPDP-8, Operability Determination Process and Limiting Conditions for Operation Tracking, LCOs associated with the INOPERABLE SSC shall be entered into the Narrative Log. If the INOPERABILITY will extend past the end of the **shift**, then an LCO Tracking Log entry is also required.

C INCORRECT: First part is incorrect but plausible in Tech Spec Bases 3.4.1, Recirculation Loops Operating, discusses that Recirculation forced flow allows operation at significantly higher Power than would otherwise be possible without forced flow. Thus, it is plausible that the candidate would conclude that declaring the higher flow loop not in operation is the conservative and correct choice. 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 recall from memory Tech Spec Bases requirements necessary to analyze Recirculation loop OPERABILITY and associated ACTIONS. SRO only because of the link to 10CFR55.43(b)(2): Facility operating limitations in the Technical Specifications and their Bases. This question is rated as Memory due to the requirement to strictly recall Tech Spec Bases information and specific administrative requirements for tracking inoperable equipment.

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 3.4.1, Amend 333, 338 (Attach if not previously provided)

Tech Spec Bases 3.4.1, Amend 348

2-SR-3.4.2.1, Rev. 55

OPDP-8, Rev. 32

Proposed references to be provided to applicants during examination: NONE

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 **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 **X**

Excerpts from Tech Spec 3.4.1:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

----------NOTE----------
Single recirculation loop operation is prohibited in the MELLLA+ operating domain.

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- Not required to be performed until 24 hours after both recirculation loops are in operation. ---------- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	In accordance with the Surveillance Frequency Control Program

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 9 and 11. Single loop operation is prohibited in the MELLLA+ operating domain. This prohibition is consistent with Reference 10.
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)

Excerpt from 2-SR-3.4.2.1:

BFN Unit 2	Jet Pump Mismatch and Operability	2-SR-3.4.2.1 Rev. 0055 Page 5 of 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.

Excerpt from OPDP-8:

3.6.1 Tracking Inoperable Equipment (continued)

E. LCOs associated with the inoperable SSC shall be entered into the Narrative Log. If the inoperability will extend past the end of the shift, then an LCO Tracking Log entry is also required.

F. Obtain and document a peer check from a second SRO. Documentation of the peer check in the Narrative Logs should be logged as soon as reasonably capable and no later than the end of the current shift.

G. If an organization other than Operations is responsible for performing a required TS action (a chemistry sample for example), the Unit SRO or designee shall inform that organization of the time the equipment was declared inoperable and the associated LCO TSAS. The Unit SRO or designee shall ensure that the required actions are properly noted in the Narrative Log and LCO Tracking Log or surveillance, as appropriate.

H. When required or recurring actions are performed by other departments, Operations retains the responsibility for ensuring TS required actions and recurring actions are completed and logged.

I. When an SSC is removed from service for maintenance or other reasons, the clearance is processed in accordance with NPG-SPP-10.2, "Clearance Procedure to Safely Control Energy."

J. The following shall be completed prior to making TS equipment inoperable for planned activities:

1. As applicable, ensure the operability of redundant equipment by performing a review of the Narrative Log, eSOMS LCO Tracking, and available MCR indications.

2. Discuss importance of the LCO and any associated actions at the pre-job briefing.

3. Assign a responsible owner for the LCO and associated actions.

K. The Unit SRO evaluates the system status change and determines operability and subsequent TS LCO TSAS entry. The Unit SRO or designee will record the necessary information in the Narrative Log for all cases and in the LCO Tracking Log for clearances.

L. Recording inoperability in the LCO Tracking Log is not required when operability is controlled and tracked within Surveillance Instruction or other procedure, except when the inoperability extends beyond the end of a shift.

M. When a Periodic Surveillance Instruction is aborted, partially performed (not completed such that the due date can be reset), or not performed during the schedule work week (including SIs that have been allowed to go late), generate a LCO Tracking Log with date and time the affected equipment will become inoperable (SI regulatory due date).

N. When IST requires increased frequency of testing, generate a LCO Tracking Log with date and time the next IST performance test is required.

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

Examination Outline Cross-reference:

295016 (APE 16) Control Room Abandonment / 7

G2.2.37 (10CFR 55.43.5 - SRO Only)

Ability to determine operability or availability of safety-related equipment (SRO Only)

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295016G2.2.37	
Importance Rating	------	4.6

Proposed Question: **# 77**

Unit 3 is operating at 100% RTP, with HPCI tagged out when the following occurs:

- 0105: A fire with heavy smoke initiates in Unit 3 Control Room
- 0110: 3-AOI-100-2, Control Room Abandonment, IMMEDIATE ACTIONS complete
- 0120: Reactor Pressure Control at Panel 3-25-32, band is 800-1000 psig
- 0123: RCIC Flow Controller failed, Reactor Water Level is (-) 75 inches, lowering
- 0135: Fire Ops extinguishes the fire

Given the above conditions, the **HIGHEST** required Emergency Classification is a(an) **(2)** .

In accordance with 3-AOI-100-2, when RCIC control is transferred to Panel 3-25-32, RCIC **(2)** automatically **ISOLATE** when Reactor Pressure lowers to 50 psig.

[REFERENCE PROVIDED]

A. (1) ALERT
(2) will NOT

B. (1) ALERT
(2) will

C. (1) Site Area Emergency
(2) will NOT

D. (1) Site Area Emergency
(2) will

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that an event that results in plant control being transferred from the Control Room to the Backup Control Panel requires an EPIP Alert classification at a minimum. Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that RCIC normally automatically isolates on low Reactor Pressure, high steam line flow, high turbine exhaust pressure, and high steam space temperature.

C **CORRECT**: (*See attached*) The given conditions indicate inability to re-establish the key safety function of Reactor Water Level within 20 minutes. At time 0123, the HIGHEST Emergency Classification is Site Area Emergency (HS6). Second part, RCIC normally automatically isolates on low Reactor Pressure, high steam line flow, high turbine exhaust pressure, and high steam space temperature. However, during execution of 3-AOI-100-2, Control Room Abandonment, 3-XS-71-2 and 3 RCIC STM LINE EMER TRANS SWITCH, are placed in EMERG, removing the automatic low Reactor Pressure isolation.

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 analyze the progression of an event requiring Main Control Room abandonment and the operability or availability of relied upon safety-related equipment. Additionally, candidate must analyze conditions and determine appropriate event classification and, thus, procedures. 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 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:
(3) The progression of an event.

Technical Reference(s):	EPIP-1, Att.1, HOT INITIATING CONDTIONS-MODES 1-2-3, Rev. 67	(Attach if not previously provided)
	3-AOI-64-2C, Rev. 6; 3-AOI-100-2, Rev. 30	
	3-OI-71, Rev. 75	

Proposed references to be provided to applicants during examination: EPIP-1, Att.1, HOT INITIATING CONDTIONS-MODES 1-2-3

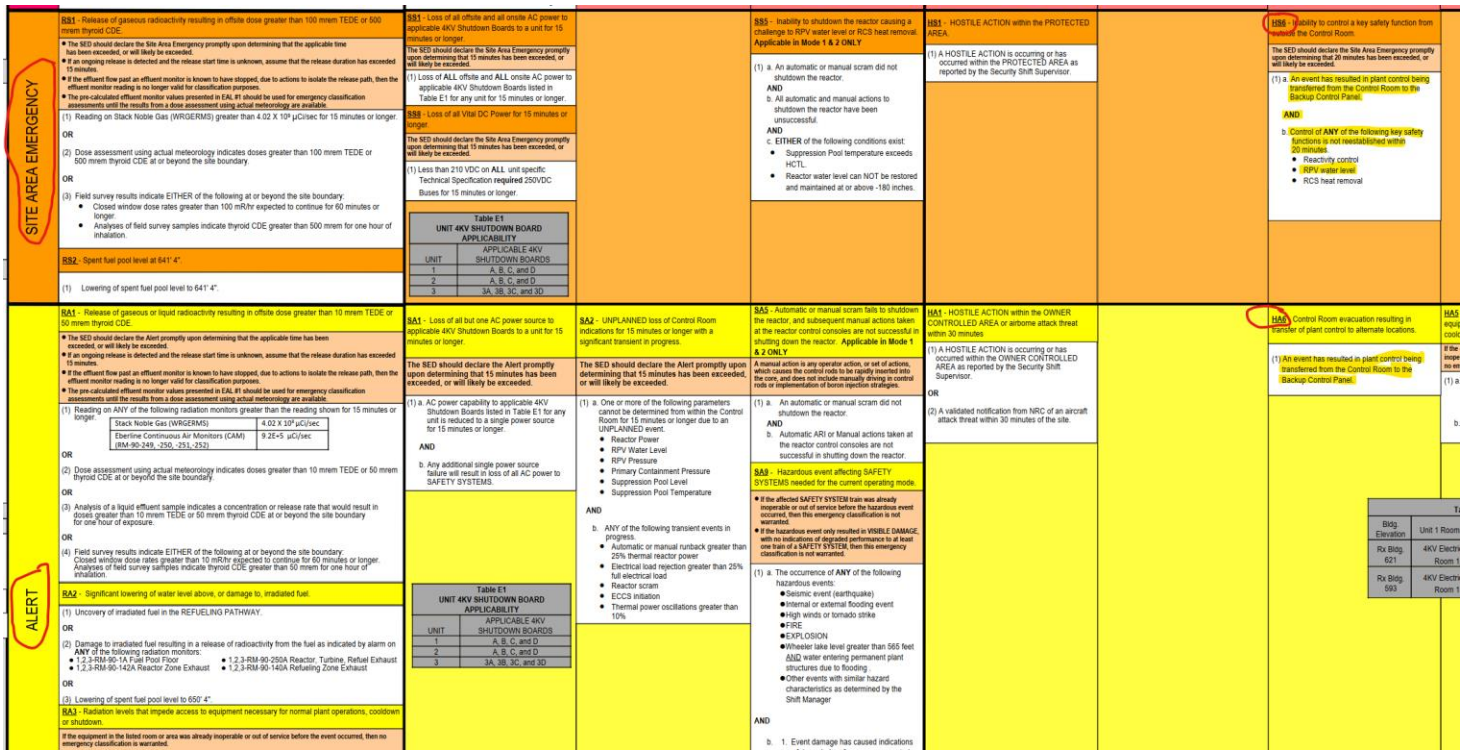
Learning Objective: OPL171.208 Obj 3

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 EPIP-1, Hot Initiating Conditions – MODES 1 – 2 - 3:



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 4.02 X 10^4 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".

SS1 - Loss of all offsite and all onsite AC power to applicable 4KV Shutdown Boards to a unit for 15 minutes or longer.

The SED should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Loss of **ALL** offsite and **ALL** onsite AC power to applicable 4KV Shutdown Boards listed in Table E1 for any unit for 15 minutes or longer.

SS8 - Loss of all Vital DC Power for 15 minutes or longer.

The SED should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Less than 210 VDC on **ALL** unit specific Technical Specification required 250VDC Buses for 15 minutes or longer.

Table E1
UNIT 4KV SHUTDOWN BOARD APPLICABILITY

UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
1	A, B, C, and D
2	A, B, C, and D
3	3A, 3B, 3C, and 3D

SS5 - Inability to shutdown the reactor causing a challenge to RPV water level or RCS heat removal. **Applicable in Mode 1 & 2 ONLY**

(1) a. An automatic or manual scram did not shutdown the reactor.
AND
b. All automatic and manual actions to shutdown the reactor have been unsuccessful.
AND
c. **EITHER** of the following conditions exist:
- Suppression Pool temperature exceeds HCTL
- Reactor water level can NOT be restored and maintained at or above -180 inches.

HS1 - HOSTILE ACTION within the PROTECTED AREA.

(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

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

ALERT

RA1 - Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

- The SED should declare the Alert 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 ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer.

Stack Noble Gas (WRGERMS)	4.02 X 10^3 µCi/sec
Eberline Continuous Air Monitors (CAM) (RM-90-249, -250, -251, -252)	9.2E+5 µCi/sec

OR

(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary.

OR

(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary for one hour of exposure.

OR

(4) Field survey results indicate EITHER of the following at or beyond the site boundary:
Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

RA2 - Significant lowering of water level above, or damage to, irradiated fuel.

(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.

OR

(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by alarm on **ANY** of the following radiation monitors:
- 1,2,3-RM-90-1A Fuel Pool Floor
- 1,2,3-RM-90-142A Reactor Zone Exhaust
- 1,2,3-RM-90-250A Reactor, Turbine, Refuel Exhaust
- 1,2,3-RM-90-140A Refueling Zone Exhaust

OR

(3) Lowering of spent fuel pool level to 650' 4".

RA3 - Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

If the equipment in the listed room or area was already inoperable or out of service before the event occurred, then no emergency classification is warranted.

SA1 - Loss of all but one AC power source to applicable 4KV Shutdown Boards to a unit for 15 minutes or longer.

The SED should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) a. AC power capability to applicable 4KV Shutdown Boards listed in Table E1 for any unit is reduced to a single power source for 15 minutes or longer.

AND

b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.

Table E1
UNIT 4KV SHUTDOWN BOARD APPLICABILITY

UNIT	APPLICABLE 4KV SHUTDOWN BOARDS
1	A, B, C, and D
2	A, B, C, and D
3	3A, 3B, 3C, and 3D

SA2 - UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.

The SED should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) a. One or more of the following parameters cannot be determined from within the Control Room for 15 minutes or longer due to an UNPLANNED event.
- Reactor Power
- RPV Water Level
- RPV Pressure
- Primary Containment Pressure
- Suppression Pool Level
- Suppression Pool Temperature

AND

b. ANY of the following transient events in progress.
- Automatic or manual runback greater than 25% thermal reactor power
- Electrical load rejection greater than 25% full electrical load
- Reactor scram
- ECCS initiation
- Thermal power oscillations greater than 10%

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.

SA9 - Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode.

- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

(1) a. The occurrence of **ANY** of the following hazardous events:
- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Wheeler lake level greater than 565 feet AND water entering permanent plant structures due to flooding
- Other events with similar hazard characteristics as determined by the Shift Manager

AND

b. 1. Event damage has caused indications

HA1 - HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.

OR

(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

HA6 - Control Room evacuation resulting in transfer of plant control to alternate locations.

(1) An event has resulted in plant control being transferred from the Control Room to the Backup Control Panel.

Bldg. Elevation	Unit 1 Room
Rx Bldg. 621	4KV Electri Room
Rx Bldg. 593	4KV Electri Room

Excerpt from 3-OI-71: Support Distractor B2, D2

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.

Excerpts from 3-AOI-100-2:

4.2 Unit 3 Subsequent Actions (continued)

NOTE

With RCIC steam supply line isolation valves 3-FCV-71-2 and 3-FCV-71-3 transfer switches in EMERG position, RCIC will **NOT** automatically isolate when Reactor Pressure drops to 50 psig.

[19] **WHEN** Reactor Pressure is less than 50 psig, **THEN**

At Panel 3-25-32, **SHUT DOWN** RCIC as follows:

[19.1] **DEPRESS** RCIC TURBINE TRIP pushbutton, 3-HS-71-9C.

[19.2] **CHECK** that the turbine comes to a complete stop using, RCIC TURBINE SPEED, 3-SI-71-42B.

BFN Unit 3	Control Room Abandonment	3-AOI-100-2 Rev. 0030 Page 49 of 92

Attachment 4
(Page 2 of 18)

Unit 3 TB AUO Panel Checklist

1.0 PART A (continued)

Switch/ Breaker Number	Component Description		Required Position	Initials
	Control Bay - 480V Reactor MOV Board 3B - El 593'			
1C	3-BKR-071-0002	RCIC STM LINE INBD ISOL VALVE BREAKER (GE-13-15)		
	3-HS-071-0002B	RCIC STM LINE INBD ISOL VALVE HAND SWITCH	OPEN	_____
	3-XS-071-0002	RCIC STM LINE EMER TRANS SWITCH	EMERG	_____

BFN Unit 3	Control Room Abandonment	3-AOI-100-2 Rev. 0030 Page 50 of 92

Attachment 4
(Page 3 of 18)

Unit 3 TB AUO Panel Checklist

1.0 PART A (continued)

Switch/ Breaker Number	Component Description		Required Position	Initials
	Control Bay - 250V DC Reactor MOV Board 3B - El 593'			
5D	3-BKR-071-0034	RCIC PUMP MIN FLOW VALVE BREAKER (GE-13-27)		
	3-HS-071-0034C	RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH	NOR	_____
	3-XS-071-0034	RCIC PUMP MIN FLOW VALVE EMER TRANS SWITCH	EMERG	_____
5B	3-BKR-071-0003	RCIC STM LINE OUTBD ISOL VALVE BREAKER (GE-13-16)		
	3-HS-071-0003C	RCIC TURB STM LINE OUTBD ISOL VALVE EMER HAND SWITCH	OPEN	_____
	3-XS-071-0003	RCIC STM LINE EMER TRANS SWITCH	EMERGENCY	_____
	NOTIFY operator at Panel 3-25-32 when Part A is complete.			_____
	STOP here until directed to perform Part B.			

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

Examination Outline Cross-reference:

295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5

G2.4.18 (10CFR 55.43.1 - SRO Only)

Knowledge of the specific bases for emergency and abnormal operating procedures.

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295028G2.4.18	
Importance Rating	------	4.0

Proposed Question: **# 78**

Which **ONE** of the following completes the statements below concerning High Drywell Temperature?

In accordance with Tech Spec Bases 3.6.1.4, Drywell Air Temperature, the highest initial Drywell Average Air Temperature (DW/T) which ensures that peak Drywell Temperature will **NOT** be reached following a Design Basis Accident (DBA) LOCA is **___(1)___**.

If all efforts to spray the Drywell have been exhausted with Drywell Temperature above **_____,** Emergency Depressurization is required in accordance with EOI-2, Primary Containment Control.

A. (1) 160 °F
(2) 336 °F

B. (1) 160 °F
(2) 350 °F

C. (1) 150 °F
(2) 336°F

D. (1) 150 °F
(2) 350 °F

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that DW/T above 160 °F is an EOI-2 entry condition. It is also listed in the DW/T leg in both steps DW/T-1 and 2. Second part is incorrect but plausible in that the Design Basis Analysis assumes initial DW/T of 150 °F to ensure that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak DW/T does not exceed the maximum calculated temperature of 336 °F.

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 Tech Spec Bases 3.6.1.4, among the inputs to the Design Basis Analysis is the initial Drywell Air Temperature (DW/T) of 150 °F. This limitation ensures that the Safety Analysis remains valid by maintaining the expected initial conditions and ensures that the peak DW/T does not exceed the maximum calculated temperature of 336 °F. Second part, Automatic Depressurization System (ADS) qualification temperature is 350 °F. If Drywell Temperature (DW/T) cannot be restored and maintained below the ADS qualification temperature, Emergency Depressurization is performed.

SRO Level Justification: Tests the candidate's ability to determine and/or interpret High Drywell Temperature as it relates to EOI-2 Drywell Temperature spray initiation bases as well as knowledge of Technical Specification Bases 3.6.1.4, Drywell Air Temperature and Design Basis Accident. Furthermore, the candidate must distinguish between the EOI-2, Primary Containment Control, entry condition and the Emergency Depressurization requirement related to High Drywell Temperature. SRO only because of link to 10CFR55.43 (1): Conditions and limitations in the facility license, and to 10CFR55.43(2): Facility operating limitations in the Technical Specifications and their 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): U2 TS Bases 3.6.1.4, Amend 254 (Attach if not previously provided)
2-EOI-2, Rev. 19
EOIPM 0-V(G), Rev. 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203, Obj. 5 (As available)

Question Source:

	Bank #	NRC BFN 2104 #81
	Modified Bank #	
	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:

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

Copy of Bank Question:

Examination Outline Cross-reference:

295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5

EA2.01 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:

- Drywell temperature

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295028EA2.01	
Importance Rating	------	4.1*

Proposed Question: **# 81**

Which **ONE** of the following completes the statements below concerning High Drywell Temperature?

In accordance with Tech Spec Bases 3.6.1.4, Drywell Air Temperature, the highest initial Drywell Average Air Temperature (DW/T) which ensures that peak Drywell Temperature will **NOT** be reached following a Design Basis Accident (DBA) LOCA is **(1)**.

In accordance with EOI-2, Primary Containment Control DW/T leg, Emergency Depressurization is required when DW/T **CANNOT** be restored and maintained below **(2)**.

A. (1) 150 °F
(2) 280 °F

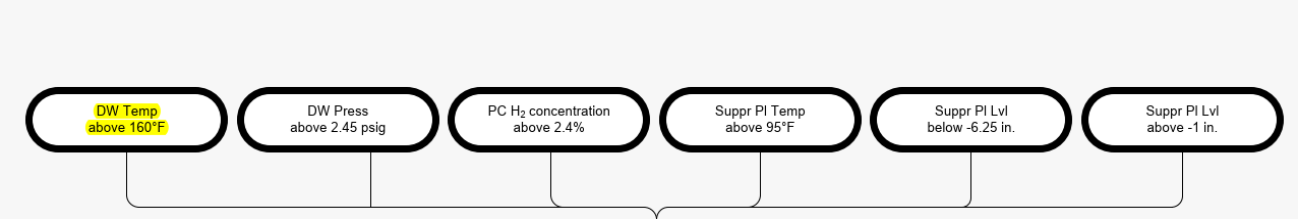
B. (1) 150 °F
(2) 350 °F

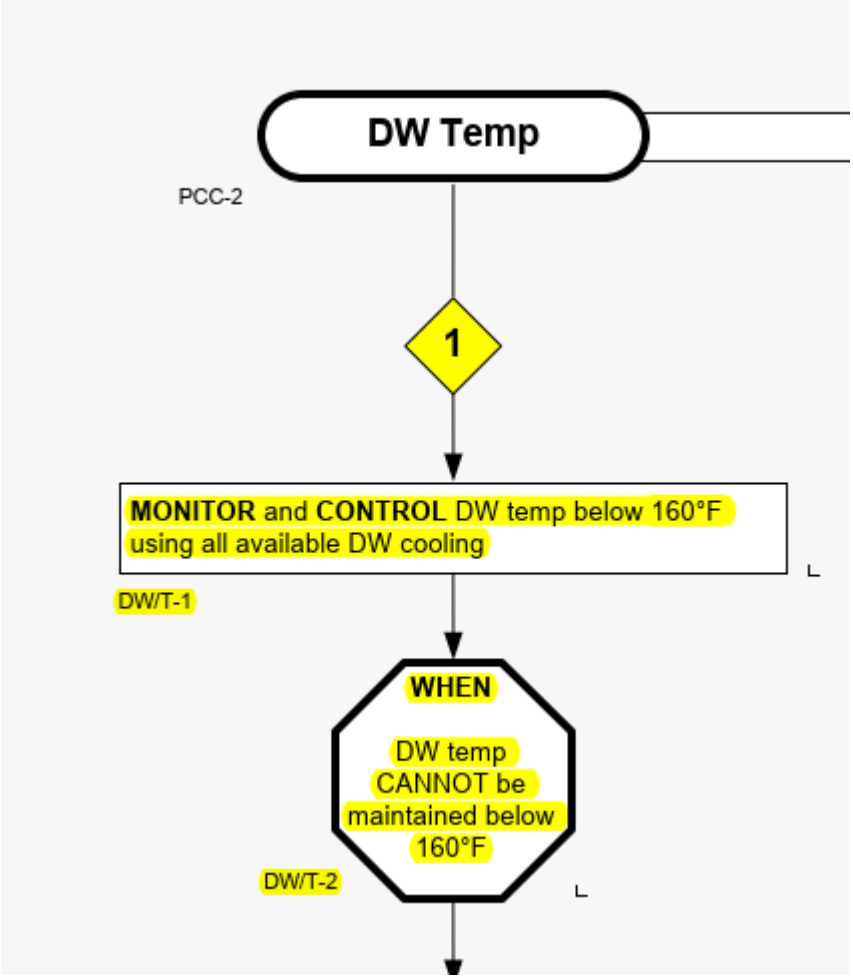
C. (1) 160 °F
(2) 280 °F

D. (1) 160 °F
(2) 350 °F

Proposed Answer: **B**

Excerpts from 2-EOI-2: Supports Distractors A1, B1





2-EOI-2 Page 1 of 1

PRIMARY CONTAINMENT CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT

Rev. 19

Excerpt from Tech Spec Bases 3.6.1.4: Also supports Distractors A2, C2

B 3.6 CONTAINMENT SYSTEMS

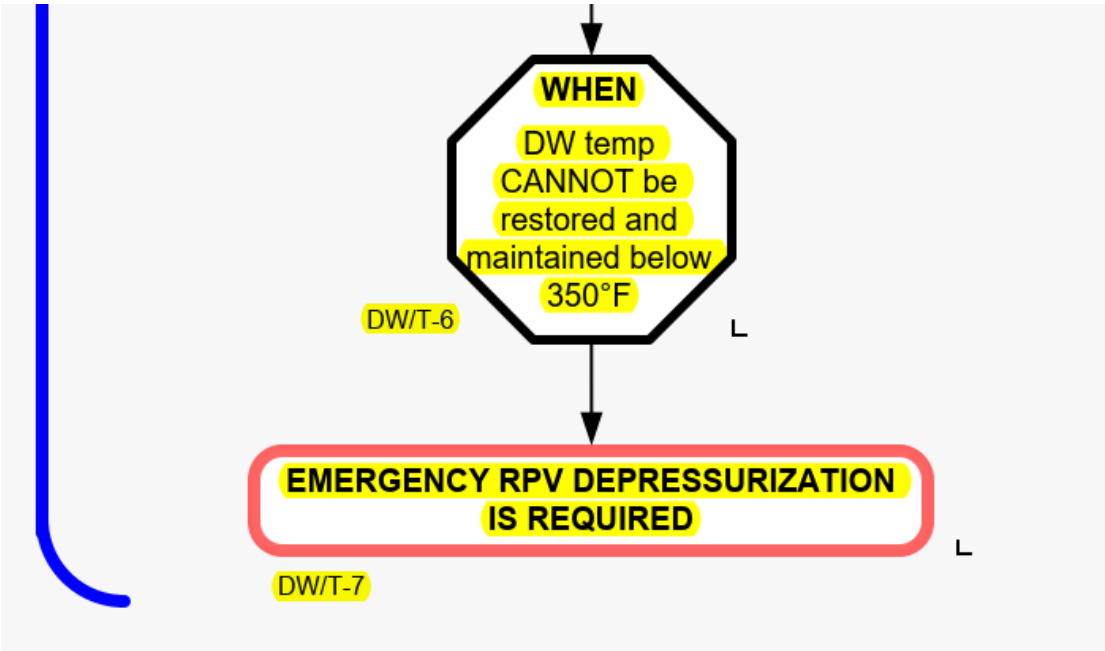
B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak drywell temperature does not exceed the maximum calculated temperature of 336°F (Ref. 2). Exceeding this temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident. Drywell air temperature satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

Excerpts from 2-EOI-2 DW/T-6 and 7:



2-EOI-2	Page 1 of 1
PRIMARY CONTAINMENT CONTROL UNIT 2 BROWNS FERRY NUCLEAR PLANT	
Rev. 19	

Excerpt from EOIPM 0-V-D related to DW/T-6 and 7:

BFN Unit 0	EOI-2, Primary Containment Control Bases	EOIPM Section 0-V(G) Rev. 0002 Page 35 of 129

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.

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

Examination Outline Cross-reference:	Level	RO	SRO
295024 (EPE 1) High Drywell Pressure / 5	Tier #	------	1
G2.4.16 (10CFR 55.43.5 - SRO Only)	Group #	------	1
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	K/A #	295024G2.4.16	
	Importance Rating	------	4.4

Proposed Question: **# 79**

Unit 2 is operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual SCRAM was inserted
- Drywell Pressure 4 psig, rising slowly
- Reactor Water Level lowered to (-) 10 inches, currently 0 inches
- **NO** Operator actions have been taken

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

Entry into **(1)** is required.

The NUSO is required to direct performance of **(2)**.

Note: 2-EOI 1, RPV Control Modes 1-3
2-EOI-2, Primary Containment Control
2-EOI Appendix-17B, RHR System Operation Drywell Sprays
2-EOI Appendix-17C, RHR System Operation Suppression Chamber Sprays

A. (1) 2-EOI-1 **AND** 2-EOI-2
(2) 2-EOI Appendix-17B **AND** 2-EOI Appendix-17C

B. (1) 2-EOI-2 **ONLY**
(2) 2-EOI Appendix-17B **AND** 2-EOI Appendix-17C

C. (1) 2-EOI-1 **AND** 2-EOI-2
(2) 2-EOI Appendix-17C **ONLY**

D. (1) 2-EOI-2 **ONLY**
(2) 2-EOI Appendix-17C **ONLY**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible in that EOI-2, Primary Containment (PC) Pressure Leg directs both Suppression Chamber Sprays and Drywell Sprays before and when reaching 12 psig, respectively. Specifically, PC/P-4, requires the NUSO to direct initiation of Suppression Chamber Sprays using Appendix-17C, RHR System Operation Suppression Chamber Sprays, before Suppression Chamber pressure rises to 12 psig. PC/P-6 directs initiation of Drywell Sprays when/if Suppression Chamber Pressure exceeds 12 psig.

B INCORRECT: First part is incorrect but plausible in that entry conditions for EOI-2 for Drywell pressure are met, but may not be determined to be met for EOI-1. Numerous entry conditions exist for EOI conditions and are easily and often confused. Second part is incorrect but plausible (*See A*).

C **CORRECT**: Entry conditions for EOI-1, RPV Control Modes 1-3, include Drywell Pressure above (+) 2.45 psig and Reactor Pressure Vessel Water Level below (+) 2 inches. Entry conditions for EOI-2, Primary Containment Control, include Drywell Pressure above (+) 2.45 psig. Based on conditions provided, entry into both EOI-1 and 2 is required.

Second part, EOI-2, Primary Containment (PC) Pressure Leg PC/P-4, requires the NUSO to direct initiation of Suppression Chamber Sprays using Appendix-17C, RHR System Operation Suppression Chamber Sprays, before Suppression Chamber pressure rises to 12 psig. Drywell Sprays are not warranted until directed at PC/P-6 when/if Suppression Chamber Pressure exceeds 12 psig.

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

SRO Level Justification: Tests the candidate's ability to determine impact of high Drywell pressure and knowledge of implementation strategy and usage hierarchy of Emergency Operating Procedures and Emergency Operating appendices. 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.

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

Technical Reference(s): 2-EOI-1, Rev. 22 (Attach if not previously provided)
2-EOI-2, Rev. 19
2-EOI Appendix-17C, Rev. 20

Proposed references to be provided to applicants during examination: None

Learning Objective: OPL171.203, Obj. 1, 2 (As available)

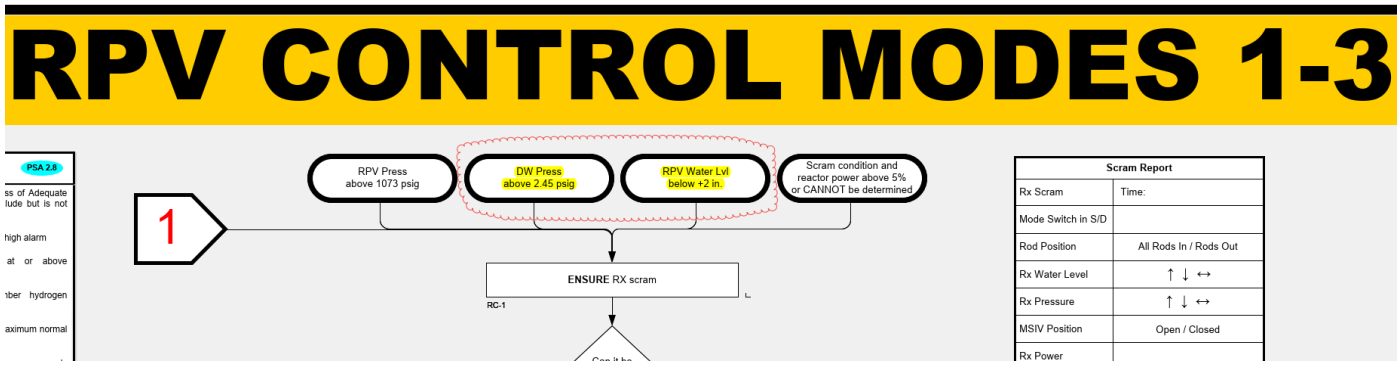
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	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	

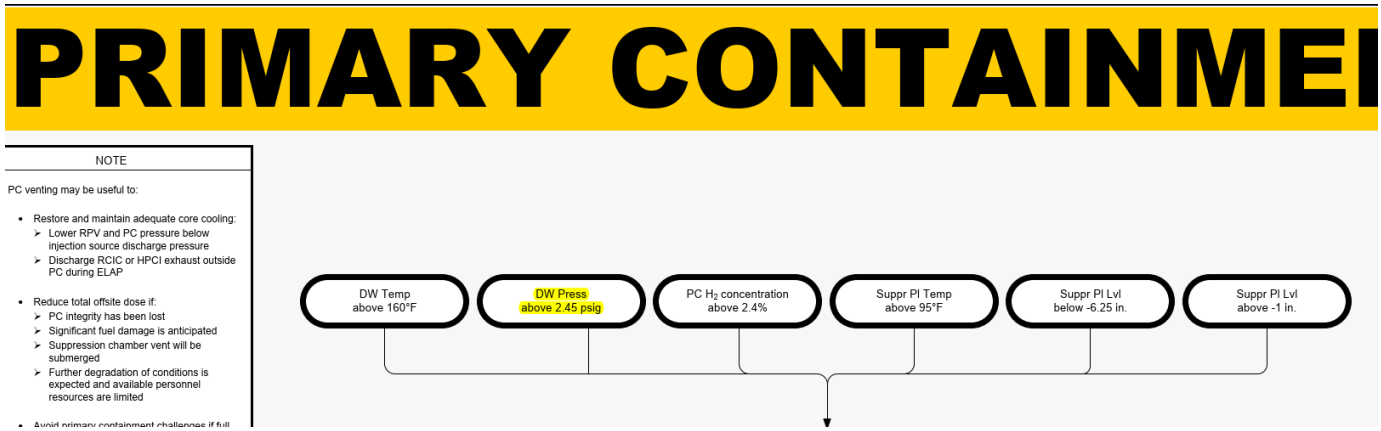
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 X

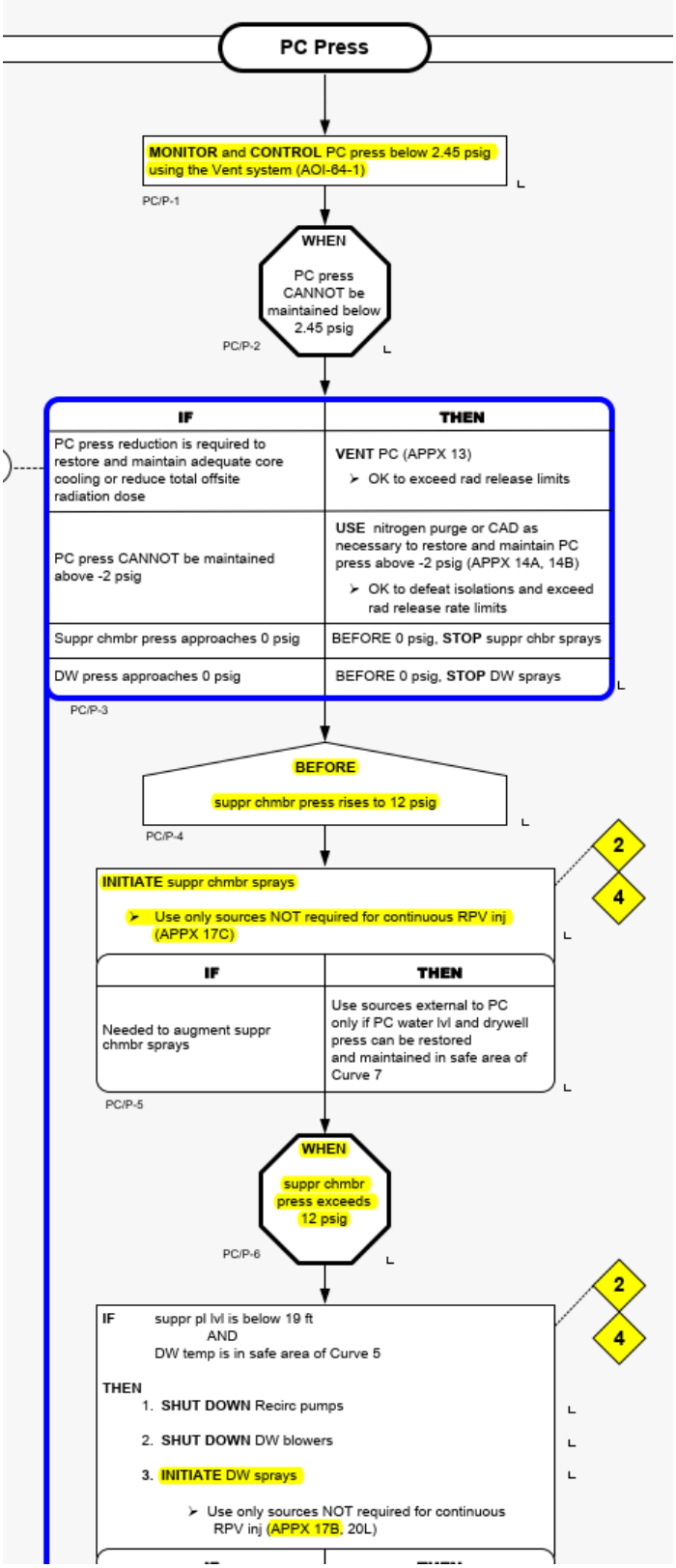
Comments:

Excerpts from EOI-1 and EOI-2:





Excerpt from 2-EOI-2:



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

Examination Outline Cross-reference:

295023 (APE 23) Refueling Accidents / 8

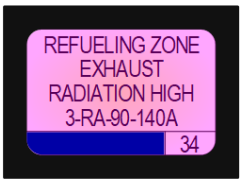
AA2.01 (10CFR 55.43.5 – SRO Only)

Ability to determine and/or interpret the following as they apply to Refueling Accidents: Radiation Levels

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295023AA2.01	
Importance Rating	------	3.9

Proposed Question: **# 80**

Unit 3 is in MODE 5 performing 0-GOI-100-3C, Fuel Movement Operations During Refueling, when the following occurs:



- REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 34) alarms
- 3-RM-90-140A **AND** 140B, Refuel Zone Channel 'A', initially spiked above 100 mR/hr, **CURRENTLY** 65 mR/hr
- 3-AOI-79-2, Inadvertent Criticality During Incore Fuel Movements, IMMEDIATE ACTIONS complete

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

In accordance with 3-EOI-3, Secondary Containment, the NUSO **(1)** direct restarting Refuel Zone Ventilation in accordance with 3-EOI Appendix-8F, Restoring Refuel Zone and Reactor Zone Ventilation Fans Following Group 6 Isolation.

In accordance with 3-AOI-79-2, the **(2)** is(are) required to be notified.

A. (1) can
(2) Reactor Engineer

B. (1) can NOT
(2) Reactor Engineer

C. (1) can
(2) Reactor Engineer and Shift Manager

D. (1) can NOT
(2) Reactor Engineer and Shift Manager

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is correct (*See C*). Second part is incorrect but plausible in that only the Reactor Engineer is required to be notified for certain AOI conditions (e.g., x-AOI-85-3, CRD System Failure). (*See attached*.)

B INCORRECT: First part is incorrect but plausible in that given the instrument spikes above 72 mr/hr initiation setpoint, Standby Gas Treatment will receive an auto-start, align to ventilation systems, and remain in service. Additionally, candidate may not recall EOI-3 Retainment Override SC-1 that allows restarting Refuel Zone Ventilation in accordance with Appendix-8F if Refuel Zone Ventilation is isolated AND Refuel Zone Ventilation Exhaust Radiation is below 72 mR/hr. Additionally, if the trip logic is incorrectly applied such as the case for downscale/inop trip initiated. One detector from the Refuel in both A and B channels results in a trip. Second part is incorrect but plausible. (*See A.*)

C **CORRECT**: (*See attached*) From the given conditions, Refuel Zone Ventilation Exhaust Radiation exceeded the Refuel Zone Ventilation Group 6 PCIS setpoint of 72 mR/hr, but has subsequently lowered to less than the setpoint. Given the conditions, EOI-3, Retainment Override SC-1 allows restarting Refuel Zone Ventilation in accordance with EOI Appendix-8F, Restoring Refuel Zone and Reactor Zone Ventilation Fans Following Group 6 Isolation. Second part, 3-AOI-79-2, Inadvertent Criticality During Incore Fuel Movements, Subsequent Actions direct notifying BOTH the Reactor Engineer and Shift Manager.

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

SRO Level Justification: Tests the candidate's ability to analyze plant conditions and interpret radiation levels as they apply to potential refueling accidents. 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 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.

Technical Reference(s): 3-AOI-85-3, Rev. 14 (Attach if not previously provided)
3-AOI-79-2, Rev. 9
3-EOI-3, Rev. 18
EOIPM Section 0-V(J), Rev. 1
3 OI-90, Rev. 66
OPL171.033, Rev. 16

Proposed references to be provided to applicants during examination: None

Learning Objective: OPL171.033, Obj. 3E, 4B (As available)
OPL171.204, Obj 3

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**

Comments:

Excerpt from 3-AOI-79-2:

BFN Unit 3	Inadvertent Criticality During Incore Fuel Movements	3-AOI-79-2 Rev. 0009 Page 6 of 8

4.2 Subsequent Actions

[1] **NOTIFY** the Shift Manager and Reactor Engineer.

[2] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOIs.

[3] **ENSURE** all control rods are inserted.

[4] **IF** criticality is still evident **AND,** at the direction of the Unit SRO, **THEN**

PERFORM the following:

[4.1] **IF** in operation, **STOP** the CRD pump.

[4.2] **IF** in operation, **ISOLATE** RWCU System as follows:

[4.2.1] **CLOSE** RWCU Inboard Suction Isolation Valve, 3-FCV-69-1.

[4.2.2] **CLOSE** RWCU OUTBD SUCT ISOLATION VALVE, 3-FCV-69-2.

[4.3] **IF** SLC is Operable, **THEN**

UNLOCK and **PLACE** 3-HS-063-0006A, SLC PUMP 3A/3B, control switch in START-A or START-B position.

[5] **NOTIFY** RADCON to conduct surveys to determine radiation levels on Refuel Floor.

NOTE

Based on recent On-Shift Analysis from the License Amendment , chemistry sampling may be delayed following plant events. In the case of a REP event, Chemistry will be required to assist with dose assessments. Dose assessments will have precedence over sampling actions in REP events

[6] **NOTIFY** Chemistry to sample AND analyze the Reactor water, if required.

[7] **REFER TO** EPIP-1 for proper notification.

[8] **NOTIFY** NRC. **REFER TO** SPP-3.5.

Excerpt from 3-EOI-3:

IF	THEN
Rx Zone ventilation exh radiation lvl is above 72 mR/hr	**ENSURE** isolation of Rx Zone and Refuel Zone ventilation AND **ENSURE** initiation of SGTS
Refuel Zone ventilation exh radiation lvl is above 72 mR/hr	**ENSURE** isolation of Refuel Zone ventilation AND **ENSURE** initiation of SGTS
Rx Zone ventilation is isolated AND Rx Zone ventilation exh radiation lvl is below 72 mR/hr	**RESTART** Rx Zone ventilation (APPX 8F) ➢ OK to defeat isolation interlocks if necessary (APPX 8E)
Refuel Zone ventilation is isolated AND Refuel Zone ventilation exh radiation lvl is below 72 mR/hr	**RESTART** Refuel Zone ventilation (APPX 8F) ➢ OK to defeat isolation interlocks if necessary (APPX 8E)
SBGT CANNOT restore and maintain Secondary Cntmt dp below 0 inches water AND Radioactivity release from Secondary Cntmt hinders operation of systems required for damage control or systems required to be operated by EOIs	**RESTART** Rx Zone ventilation (APPX 8F) ➢ OK to defeat isolation interlocks, including high radiation, if necessary (APPX 8E, APPX 8J) ➢ OK to exceed release rate limits if necessary

SC-1

Excerpt from EOIPM Section 0-V(J):

BFN Unit 0	EOI-3 Secondary Containment Control Bases	EOIPM Section 0-V(J) Rev. 0001 Page 17 of 59

1.0 EOI-3, SECONDARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SC-1 (cont'd)

Third and fourth overrides

These overrides direct the operator to restart Reactor Zone and Refuel Zone Ventilation Systems once it has been confirmed that excessive radioactivity release will not occur from system operation.

Reactor Zone and Refuel Zone Ventilation Systems are the systems normally used to maintain secondary containment temperature and differential pressure within operational limits. If isolated, it is appropriate to restart these systems and use them to restore and maintain control of secondary containment temperature and pressure once it has been confirmed that restart will not result in an excessive release of radioactivity to the environment.

Defeating high drywell pressure and low RPV water level isolation interlocks is appropriate, if needed, since application of these isolations to secondary containment ventilation is for the sole purpose of limiting radioactivity release to the environment. EOI Appendix-8E provides guidance to defeat the high drywell pressure and low RPV water level Group 6 isolation interlocks for these systems. Once assurance is provided that excessive release of radioactivity will not occur, these two isolation interlocks become dispensable. EOI Appendix-8F provides guidance to restore Reactor Zone and Refuel Zone Ventilation following a group 6 isolation.

Excerpt from 3-OI-90:

BFN Unit 3	Radiation Monitoring System	3-OI-90 Rev. 0066 Page 8 of 62

3.0 PRECAUTIONS AND LIMITATIONS

A. The following Radiation Monitoring subsystems initiate the listed automatic actions and isolations on high radiation trip signals:

1. Main Steam Line (3 times normal full-load background radiation).
   a. Mechanical Vacuum Pump trip and suction valve isolation.
2. Off-Gas Post-Treatment
   a. High or High-High - opens Adsorber Inlet Valve, 3-FCV-66-113A, and closes Adsorber Bypass Valve, 3-FCV-66-113B, if 3-XS-66-113 is in AUTO.
   b. High-High-High - closes Off-Gas System Isolation Valve, 3-FCV-66-28 (5-second time delay).
3. Refueling Zone Ventilation (72 mr/hr high radiation signal from 2 out of 2 taken once logic or downscale/inop signal from 1 out of 2 taken twice logic)
   a. Standby Gas Treatment System auto start.
   b. Refueling Zone Vent System isolation.
   c. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)
4. Reactor Zone Ventilation (72 mr/hr high radiation signal from 2 out of 2 taken once logic or downscale/inop signal from 1 out of 2 taken twice logic)
   a. Group 6 Isolation.
   b. Standby Gas Treatment System auto start.
   c. Refueling Zone Ventilation isolation.
   d. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)
5. Control Room Ventilation Monitoring (221 cpm above background high activity or two channels downscale/inop)
   a. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)

B. Abnormal or significant rises in radiation levels are required to be reported to the Unit SRO.

6.4 NUMAC Radiation Monitor Operation (continued)

NOTES

1) There are two detectors for each channel of the Reactor Zone/Refuel Zone Monitors and are indicated on each monitor as follows:

3-RM-90-140/142

Display	Description
2A	3-RE-90-142A, Reactor Zone Channel A Detector A.
2B	3-RE-90-142B, Reactor Zone Channel A Detector B.
0A	3-RE-90-140A, Refuel Zone Channel A Detector A.
0B	3-RE-90-140B, Refuel Zone Channel A Detector B.

3-RM-90-141/143

Display	Description
3A	3-RE-90-143A, Reactor Zone Channel B Detector A.
3B	3-RE-90-143B, Reactor Zone Channel B Detector B.
1A	3-RE-90-141A, Refuel Zone Channel B Detector A.
1B	3-RE-90-141B, Refuel Zone Channel B Detector B.

2) Only the "A" detector of each channel described above has input to radiation recorder 3-RR-90-144.

3) Any active trip condition will be indicated by a highlighted "TRIP" at the top of the screen. A non-highlighted "TRIP" at the top of the screen indicates that there are one or more past trip conditions that have **NOT** been acknowledged.

4) Trips on the Reactor Zone/Refuel Zone Radiation monitors will automatically reset when the alarming condition resets.

[2] **PERFORM** the following to immediately Reset Group 6 Isolation Due to Reactor Zone Radiation Monitors.

Attachment 1
(Page 2 of 3)

Radiation Monitoring System Operational Summary

Subsystem	Operation
Off-Gas Pretreatment Post-Treatment Vial Samplers	Two portable sampling units, **NOT** normally in service, can be used to draw Off-Gas samples for laboratory analysis during operation or while shutdown. Pretreatment samples can be drawn from the holdup volume inlet (or alternately from the SJAE second stage suction). Post-treatment samples can be drawn from the charcoal bed inlet and outlet. PNL 3-25-40 & 0-25-259
Reactor/Refueling Zone Exhaust Radiation Monitors 3-RE-90-140/142 3-RE-90-141/143	Each Control Room drawer contains all the monitors for that channel (both reactor and refuel zone). Each monitor has two detectors (i.e., 3-RM-90-141 Detector A and Detector B). The purpose of the second detector is to prevent spurious trips For an upscale trip to be initiated, both detectors (i.e., 3-RM-90-141 Detector A and Detector B) in a channel (A or B) must reach the trip setpoint. For a downscale/inop trip to be initiated, one detector for the associated zone in both channels must be in the downscale/inop state (i.e., 3-RM-90-140 Detector A or B and 3-RM-90-141 Detector A or B). Note that with this arrangement, one detector downscale/inop will render that channel inop for a high trip. **REFER TO** Tech Spec 3.3.6.2, for Reactor Bldg Vent Radiation Monitoring system channel(s) which are inoperable for functional testing or inoperable for calibration or maintenance. The inoperable channel is to be placed in the tripped condition. Note that placing the drawer in the inoperable state results in both radiation monitors (Reactor and Refuel Zone) being in the inoperable trip condition and is allowed by this note. High radiation in either of these systems will isolate the respective ventilation, start Standby Gas Treatment, and initiate emergency Control Room isolation and pressurization. In addition, high radiation in the reactor zone ventilation inserts a redundant refueling zone high radiation trip and initiates a Group 6 Isolation. High refueling zone radiation in any unit will cause a refueling zone isolation in all units.

Excerpt from 3-AOI-85-3 (supports distractors A2, B2):

BFN Unit 3	CRD System Failure	3-AOI-85-3 Rev. 0014 Page 10 of 12

4.2 Subsequent Actions (continued)

[3] **IF** operating CRD Pump has NOT tripped, **THEN**

PERFORM the following:(**REFER TO** 3-OI-85) (Otherwise N/A)

[3.1] **ENSURE** FULL OPEN CRD PUMPS 3A & 3B UNIT 3 SUCTION, 3-FCV-85-56.

[3.2] **IF** running CRD Pump suction pressure is approaching 18-inches HG absolute, **THEN**

ALTERNATE CRD Pumps. (Otherwise N/A)

[3.3] **IF** in-service filter differential pressure is greater than 20.5 psid, **THEN**

ALTERNATE CRD drive water filters. (Otherwise N/A)

[3.4] **IF** system flow is high or low, **THEN**

SHIFT CRD Flow Control Valves. (Otherwise N/A)

[3.5] **IF** system flow and pressure is unstable during control rod movement, **THEN**

SHIFT CRD Stabilizing Valve sets. (Otherwise N/A)

[3.6] **IF** air is suspected to be in the system, **THEN**

VENT air from Drive Water Header using DRIVE WATER HDR VENT, 3-VTV-085-0592, at elevation 565' south end of west HCU bank. (Otherwise N/A)

[4] **IF** CRD System hydraulic pressure is restored, **THEN**

RESTORE Control Rod Drive System alignment. **REFER TO** 3-OI-85. (Otherwise N/A)

[5] **NOTIFY** Reactor Engineer of CRD system failure.

Excerpt from OPL171.033:

Ventilation Radiation Monitors

A. Purpose: To record the release of radioactive material from the plant buildings to the environs and provide an alarm when preset limits are reached

B. Components and Functions — 729E814RE-7

1. Reactor zone and refueling zone monitors work independently of each other for trip actuation. REACTOR ZONE EXHAUST RADIATION HIGH (55-3A-21) and REFUELING ZONE EXHAUST RADIATION HIGH 2-RA-90-140A (55-3A-34) Alarm setpoint is 72 mr/hr. Rad-monitor auto resets when alarm is clear. — ILT/LOR OBJ 3E; NLO/NLOR OBJ 3G; TP-9
2. Trip logic for the refueling and the reactor zones is identical, and the following combinations will generate a trip:
   - Two high level trips in the same channel, (division) — Two-out-of-two, once
   - **-OR-**
   - One downscale trip in each channel (division) — One-out-of-two, twice
   - **-OR-**
   - One monitor INOP in each channel (division) — One-out-of-two, twice
   - **-OR-**
   - Loss of RPS power to either channel
3. Automatic actions
   a. Refueling area trip will start Standby Gas Treatment System and opens SBGT suction to Refuel Zone. It will cause a PCIS Group 6 isolation and start CREVs. — ILT/LOR OBJ 4B,F,G; NLO/NLOR OBJ 6
   b. Reactor Zone Trip will Isolate Control Room, Reactor Zone, and Refueling Zone ventilation, start Standby Gas Treatment System opening SBGT suction to affected unit's Reactor Zone. It will cause a PCIS Group 6 isolation and start CREVs.
   c. REACTOR ZONE EXHAUST RADIATION MONITOR DNSCL (55-3A-35) Alarms when reading ≤ 0.20 mr/hr. REFUELING ZONE EXHAUST RADIATION MONITOR DNSCL (55-3A-28) Alarms when reading ≤ 18.0 mr/hr.

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

Examination Outline Cross-reference:
295031 (EPE 8) Reactor Low Water Level / 2
EA2.04 (10CFR 55.43.5 - SRO Only)
Ability to determine and/or interpret the following as they apply to Reactor Low Water Level:

- Adequate core cooling

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295031EA2.04	
Importance Rating	------	4.7

Proposed Question: **# 81**

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

Time=0:

- A Loss of Offsite Power
- Reactor SCRAM, **ALL** Control Rods fully inserted
- **ONLY** 'A' and 'B' EDGs started and tied to their respective boards
- **NO** High Pressure injection systems are available

Subsequently:

- Reactor Water Level is (-) 162 inches, lowering 3 inches per minute
- **ALL** RHR Pumps **FAILED** to start

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

In accordance with 2-EOI-1, RPV CONTROL MODES 1-3, if all efforts to RESTORE and MAINTAIN Reactor Water Level have been exhausted, the **EARLIEST** time Emergency Depressurization is required is **(1)**.

FOLLOWING Emergency Depressurization, Reactor Water Level is (-) 200 inches, rising with Core Spray Loop I injecting at 6,400 gpm. SAMG entry **(2)** required.

A. (1) now
(2) is

B. (1) now
(2) is NOT

C. (1) six (6) minutes
(2) is

D. (1) six (6) minutes
(2) is NOT

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible in that SAMG entry is required in similar circumstances when Reactor Water Level cannot be restored and maintained above (-) 180 inches AND Spray Cooling, as defined in Table M-3, Adequate Core Cooling, cannot be established.

B **CORRECT**: (*See attached*) Provided Time=0 Reactor Water Level is (-) 162 inches with level lowering at 3 inches per minute. EOI-1, RPV Control Modes 1-3, RC/L-8 requires Emergency Depressurization (ED) to support RPV injection strategies when it is determined that Reactor Water Level cannot otherwise be restored and maintained above the top of active fuel (TAF, -162 inches). Once this conclusion is reached, ED may be initiated any time BEFORE Reactor Water Level reaches the Minimum Steam Cooling RPV Water Level (MSCRWL, -180 inches). Given that Time=0 level is at TAF with level lowering at 3 inches per minute, ED will first be required right now.

Second part, to determine if SAMG entry is required, the candidate must determine whether Reactor Water Level can be restored and maintained above (-) 180 inches AND whether Spray Cooling can be established. The post ED conditions provided are '(-) 200 inches and continuing to rise with Core Spray Loop I injecting at 6,400 gpm.' Since level is 'continuing to rise,' even though level is not currently above (-) 180 inches, it can be determined that Reactor Water Level can be restored and maintained above (-) 180 inches, thus SAMG entry is not required. Additionally, from conditions provided, Spray Cooling, as defined in Table M-3, Adequate Core Cooling, (Either CS subsystem operating with at least 6,250 gpm to the RPV AND Reactor Water Level above (-) 215 inches) has been met, thus SAMG entry is not required.

C INCORRECT: First part is incorrect but plausible in that Time=0 Reactor Water Level given is (-) 162 inches with lowering level at 3 inches per minute. At the level lowering rate, (-) 180 inches will be reached in six (6) minutes. EOI-1, RPV Control Modes 1-3, RC/L-9 requires Emergency Depressurization (ED) before Reactor Water Level reaches the Minimum Steam Cooling RPV Water Level (MSCRWL, (-) 180 inches). The candidate may incorrectly determine that ED is required WHEN Reactor Water Level reaches (-) 180 inches which, as previously stated, would occur in six (6) minutes. 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 and/or interpret Adequate Core Cooling (ACC) as it applies to low Reactor Water Level when given conditions in which ACC is questionable due to both low level and limited RPV injection capability. 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.

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

Technical Reference(s): 2-EOI-1, Rev. 22 (Attach if not previously provided)
EOIPM Section 0-V(C), Rev 3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.202 Obj. 16 (As available)
OPL171.202 Obj. 29

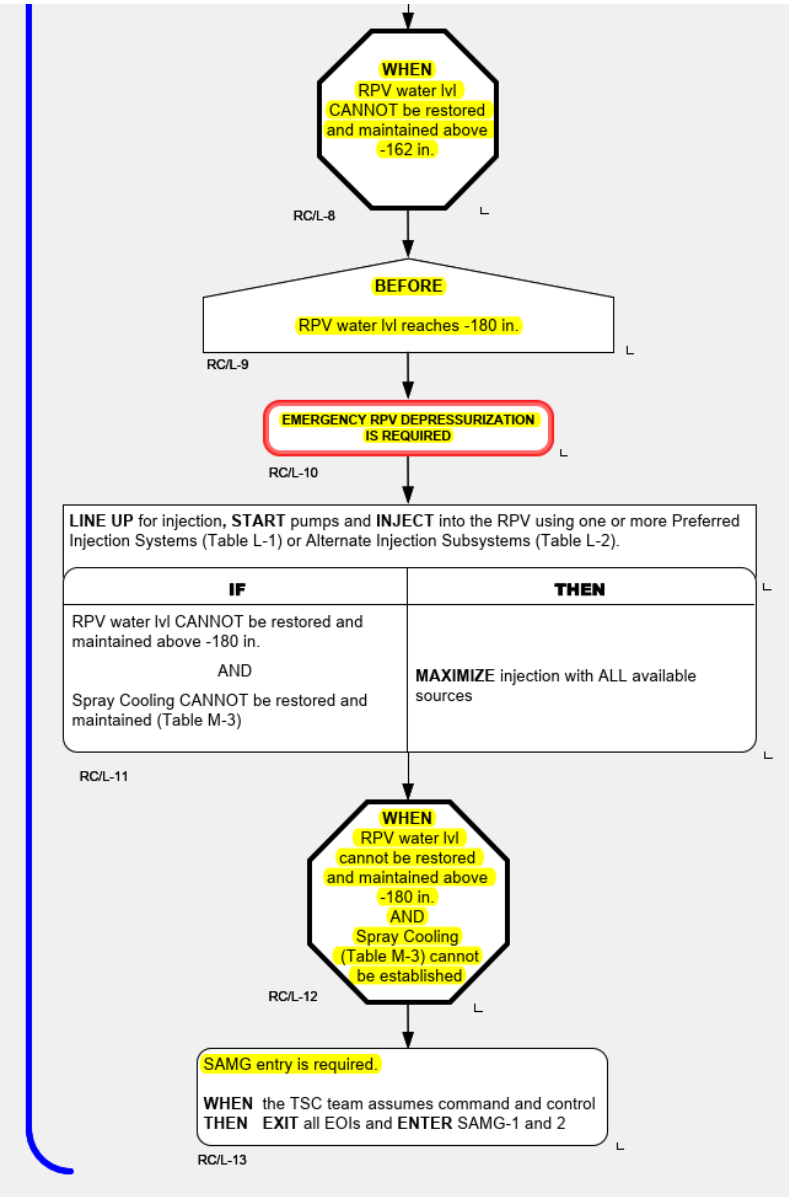
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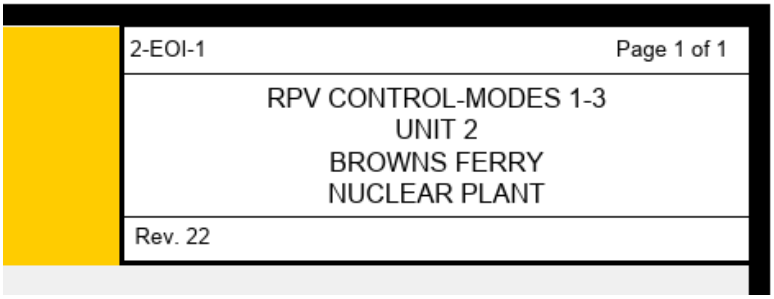
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis **X**

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

Comments:

Excerpts from 2-EOI-1: Also supports Distractors A2, C1, C2, D1





Excerpt from EOIPM Section 0-V(C):

BFN Unit 0	EOI-1, RPV Control Modes 1-3 Bases	EOIPM Section 0-V(C) Rev. 0003 Page 41 of 81

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L-8, -9, -10

Emergency RPV depressurization permits injection from low pressure systems, maximizes the total injection flow, and minimizes the flow through any primary system break.

Emergency depressurization is appropriate to support RPV injection strategies only if it is determined that RPV water level cannot otherwise be restored and maintained above the top of active fuel (TAF). Once this conclusion is reached, emergency depressurization may be initiated any time before RPV water level reaches the Minimum Steam Cooling RPV Water Level (MSCRWL). The “before” action level provides operational flexibility and requires a timing judgment. For example:

- If it is expected that available injection sources will reverse the level trend before RPV water level drops to the MSCRWL or that restoration of additional high pressure injection is imminent, the depressurization may be delayed.

- If it is believed that available injection sources are capable of restoring and maintaining RPV water level above the MSCRWL following RPV depressurization, the depressurization may be performed as soon as it is determined that RPV water level cannot be restored and maintained above the top of the active fuel.

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
EA2.04 (10CFR 55.43.5 - SRO Only)	Group #	------	1
Ability to determine and/or interpret the following as they apply to Scram condition present and reactor power above APRM downscale or unknown:	K/A #	295037EA2.04	
• Suppression pool temperature	Importance Rating	------	4.4

Proposed Question: **# 82**

An ATWS has occurred on Unit 3.

Which **ONE** of the following completes the statements below concerning Boron injection?

In accordance with EOI-1A, ATWS RPV Control, before Suppression Pool Temperature rises to **(1)**, Boron injection is required.

In accordance with the EOI Program Manual Bases, the reason Boron is injected at this temperature is **(2)**.

A. (1) 110 °F
(2) for alternate source term control

B. (1) 110 °F
(2) to shut down the Reactor prior to reaching the Heat Capacity Temperature Limit

C. (1) 120 °F
(2) for alternate source term control

D. (1) 120 °F
(2) to shut down the Reactor prior to reaching the Heat Capacity Temperature Limit

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct (*See B*). Second part is incorrect but plausible in that in accordance with the Alarm Response Procedures, Boron is injected within 2 hours of receiving 3-ARP-9-7C, Window 15, Drywell/Suppr Chamber Radiation High, for alternate source term control.

B **CORRECT**: *(See attached)* In accordance with EOI-1A, ATWS RPV Control, Step ARC/Q-8 states that BEFORE Suppression Pool Temperature rises to 110^{0} F, Boron injection is required. Second part, in accordance with EOIPM Section 0-V(D), EOI-1A, ATWS RPV Control Bases, Boron is injected prior to 110 °F Suppression Pool Temperature to avoid depressurizing the RPV with the Reactor at power prior to reaching the Heat Capacity Temperature Limit.

C INCORRECT: First part is incorrect but plausible in that according to Technical Specification 3.6.2.1, Suppression Pool Average Temperature, the Reactor must be depressurized to less than 200 psig if Suppression Pool temperature is above 120 °F. 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 interpret the impact of and bases for actions during failure to scram with regard to Suppression Pool Temperature. 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 and bases facts in relation to Boron Injection Temperature and Heat Capacity Temperature Limit curve.

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-ARP-9-7C, Rev. 41 (Attach if not previously provided)
3-EOI-1A, Rev. 6
EOIPM 0-V(D), Rev. 3
Unit 3 Tech Spec 3.6.1.2, Amend. 212

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.202, Obj. 29 (As available)

Question Source:

	Bank #	BFN NRC 2104 #80
	Modified Bank #	
	New	

(Note changes or attach parent)

Question History:

Last NRC Exam	2104

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

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:

295026 (EPE 3) Suppression Pool High Water Temperature / 5

G2.4.18 (10CFR 55.43.1 - SRO Only)

Knowledge of the specific bases for EOPs.

Level	RO	SRO
Tier #	------	1
Group #	------	1
K/A #	295026G2.4.18	
Importance Rating	------	4.0

Proposed Question: **# 80**

An ATWS has occurred on Unit 3.

Which **ONE** of the following completes the statements below concerning Boron injection?

In accordance with EOI-1A, ATWS RPV Control, before Suppression Pool Temperature rises to **(1)**, Boron injection is required.

In accordance with the EOI Program Manual Bases, the reason Boron is injected at this temperature is **(2)**.

A. (1) 110 °F
(2) for alternate source term control

B. (1) 110 °F
(2) to shut down the Reactor prior to overheating the Suppression Pool

C. (1) 120 °F
(2) for alternate source term control

D. (1) 120 °F
(2) to shut down the Reactor prior to overheating the Suppression Pool

Proposed Answer: **B**

Excerpt from 3-ARP-9-7C: Supports Distractors A2, C2

BFN Unit 3	Panel 9-7 3-XA-55-7C	3-ARP-9-7C Rev. 0041 Page 20 of 40

DRYWELL/SUPPR CHAMBER RADIATION HIGH
3-RA-90-272
15

(Page 1 of 1)

Sensor/Trip Point:

3-RM-90-272A	100 R/HR
3-RM-90-273A	160 R/HR
3-RM-90-272B	9.99 E6 R/HR (Disabled Alarm Input)
3-RM-90-273B	9.99 E6 R/HR (Disabled Alarm Input)

Sensor Location: 3-RR-90-272, Panel 3-9-54, Main Control Room.
3-RR-90-273, Panel 3-9-55, Main Control Room.

Probable Cause:
A. High radiation levels.
B. Sensor malfunction.
C. Noise spikes.

Automatic Action: None

Operator Action:
A. **CHECK** alarm on 3-RR-90-272 (Panel 3-9-54), and 3-RR-90-273 (Panel 3-9-55).
B. **CHECK** 3-RR-90-256 for rise.
C. **ATTEMPT** to isolate equipment to stop source.
D. (NRC/C)**IF** the alarm is determined to be valid, **THEN** **PERFORM** the following :
- **OPEN** UPSTREAM MSL DRAIN TO CONDENSER 3-FCV-001-0058.
- **OPEN** DOWNSTREAM MSL DRAIN TO CONDENSER 3-FCV-001-0059.
- **ENSURE** 3-PCV-001-0147 is Closed by taking STEAM SEAL REGULATOR, 3-HS-1-147 to CLOSE. (Panel 9-7) (License Amendment 223)

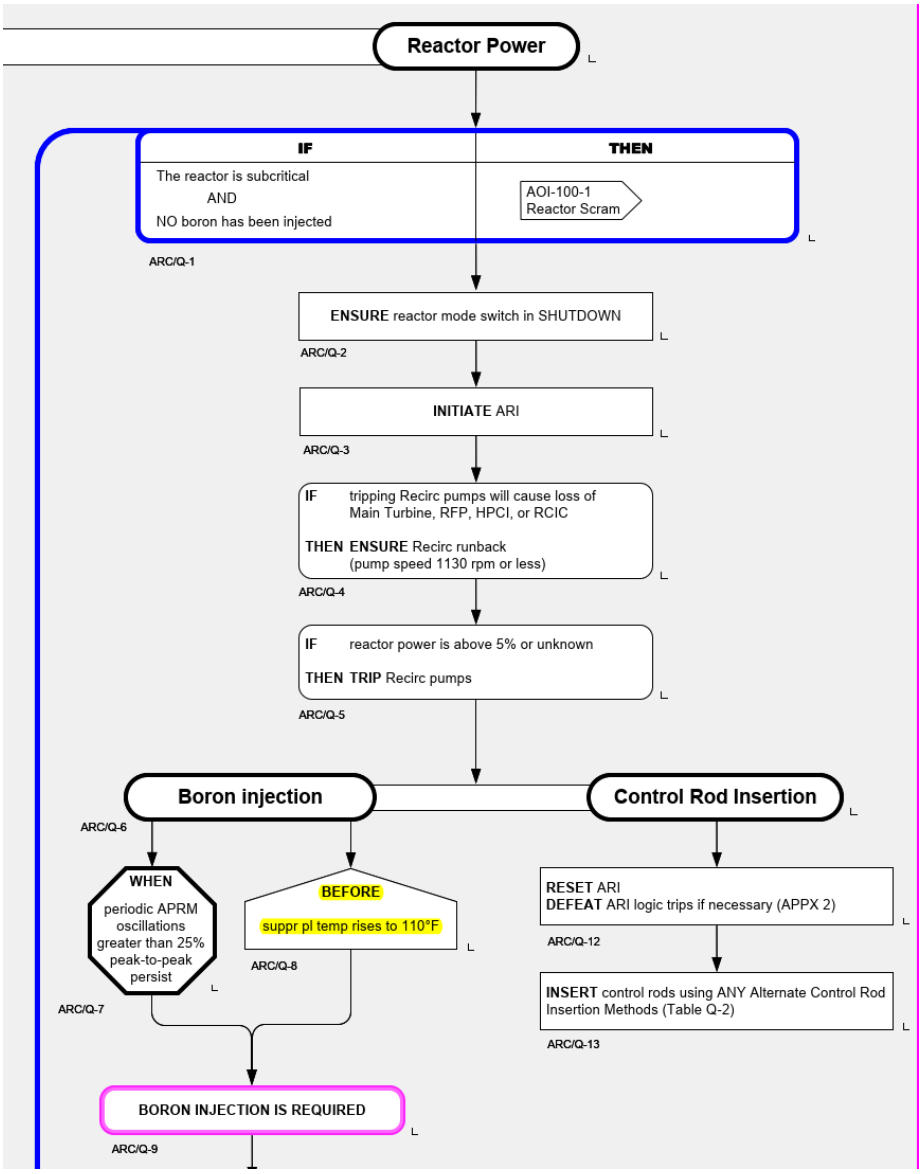
E. **IF** ALL the following conditions exist (1, 2, and 3):
1. Alarm is determined to be valid, **AND**
2. The reactor will remain subcritical without boron injection under all conditions, **AND**
3. Leakage of primary coolant into primary containment is indicated, **THEN**

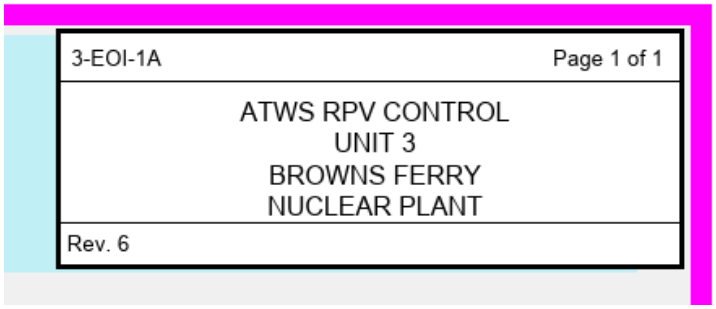
Within 2 hours of alarm, **INJECT** SLC for alternate source term control by placing SLC PUMP 3A/3B, 3-HS-63-6A in the START A or **START** B position.

F. **REFER TO** EPIPs.
G. **IF** started at Operator Action Step E, **THEN** **WHEN** SLC tank reaches 0", **STOP** the running SLC Pump
H. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References: 3-45E620-9 0-47E610-90-2 NESSD 3R-090-273A-00
Technical Specifications Section 3.3.3.1 NESSD 3R-090-272A-00

Excerpt from 3-EOI-1A:





Excerpt from EOIPM Section 0-V(D):

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/Q-8

If suppression pool temperature and RPV pressure cannot be maintained in a safe region of the Heat Capacity Temperature Limit (Curve 3), emergency RPV depressurization will be required. To avoid depressurizing the RPV with the reactor at power, it is desirable to shut down the reactor prior to reaching the Heat Capacity Temperature Limit. The Boron Injection Initiation Temperature (BIIT, **A.64**) is defined so as to achieve this goal when practicable.

The Boron Injection Initiation Temperature (BIIT) is the greater of:

- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.
- The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

The BIIT is a function of reactor power. If boron injection is initiated before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels. At higher reactor power levels, however, the suppression pool heatup rate may become so high that the Hot Shutdown Boron Weight of boron cannot be injected before suppression pool temperature reaches the Heat Capacity Temperature Limit even if boron injection is initiated early in the event.

For simplification, this step uses the suppression pool temperature at which a reactor scram is required by plant Technical Specifications to define the BIIT. Refer to EOIPM Section 0-V(B) for discussion of the BIIT.

The direction to inject boron “before” suppression pool temperature reaches the BIIT also permits injection of boron before neutron flux oscillations are observed if it is clearly beneficial to do so. Early boron injection is not *required*, however, since it may not always be appropriate or desirable. For example, if emergency RPV depressurization must be performed with the core borated, the subsequent injection of cool, unborated water to restore RPV water level can dilute and displace hot borated water in the core region. The resulting positive reactivity change will be greater than that which would occur if the injected water displaced hot *un*borated water from the core region, since boron loss would not then be a factor. Premature boron injection could thus increase the potential for a reactivity excursion.

Excerpt from TS 3.6.2.1: Supports Distractors C1, D1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Suppression pool average temperature > 120°F.	E.1	Depressurize the reactor vessel to < 200 psig.	12 hours
	AND		
	E.2	Be in MODE 4.	36 hours

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

Examination Outline Cross-reference:

295007 (APE 7) High Reactor Pressure / 3

AA2.04 (10CFR 55.43.5 – SRO Only)

Ability to determine and/or interpret the following as they apply to High Reactor Pressure

- Bypass valve capacity

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295007AA2.04	
Importance Rating	------	4.0

Proposed Question: **# 83**

Unit 2 is operating at 100% RTP when the Main Turbine tripped resulting in the following conditions:

- 2-AOI-100-1, Reactor Scram Hard Card, IMMEDIATE ACTIONS completed
- SLC **FAILED** to inject
- Reactor Water Level is being maintained (-)50 inches to (-)100 inches
- Reactor Power is 18%
- Reactor Pressure is 1030 psig, lowering

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

In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, for the **CURRENT** Reactor Power Level, a **MINIMUM** of **(1)** Main Turbine Bypass Valves are required to be OPEN to control Reactor Pressure between 800 – 1000 psig.

In accordance with 2-EOI-1A, ATWS RPV Control, the NUSO is required to direct **(2)** for Reactor Pressure Control.

Note: 2-EOI Appendix-8B, Reopening MSIVs/Bypass Valve Operation
2-EOI Appendix-11A, Alternate RPV Pressure Control Systems MSRVs

A. (1) 3
(2) 2-EOI Appendix-11A

B. (1) 3
(2) 2-EOI Appendix-8B

C. (1) 6
(2) 2-EOI Appendix-11A

D. (1) 6
(2) 2-EOI Appendix-8B

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that in accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, one Main Steam Relief Valve (MSRV) is equivalent to approximately 5 – 7% Reactor Power near rated Reactor Pressure. Additionally, 0-TI-394, Technical Support Guidelines for Use with EOIs and SAMGs, states that at 1000 psig, the MSRVs are equivalent to approximately 6.2 – 7.1% Reactor Power. With the Power level provided of 18% the candidate may mistakenly apply MSRV capacities to the number of Main Turbine Bypass Valves (TBVs) required to be open.

Second part is incorrect but plausible in that EOI-1A, ATWS RPV Control, ARC/P-4, directs the use of "Alternate RPV Pressure Control Systems (Table P-1), **if necessary** (*emphasis added*)." In the conditions given, Reactor Power is within the capacity of the TBVs and they are functioning as designed to maintain Reactor Pressure, thus the use of MSRVs via Appendix-11A is unwarranted. Additionally, direction in ARC/P-2 to manually open MSRVs until Reactor Pressure drops to the point at which all TBVs are fully open (APPX 11A) is not warranted because, again, the given Reactor Power is within the capacity of the TBVs.

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 BFN-ODM-4.20, one Main Turbine Bypass Valve (TBV) fully open is equivalent to about 2-3.5% Reactor Power near rated pressure. 0-TI-394 states that near 1000 psig, TBVs are equivalent to 2.25 to 3.5% power. With the Power level provided of 18% and choices of 3 or 6 TBVs, only 6 TBVs can be correct.

Second part, EOI-1A, ARC/P-4, directs to restore and maintain RPV Pressure between 1073 psig and 350 psig using TBVs (Appendix-8B). Alternate options are only warranted if control using TBVs is not successful.

SRO Level Justification: Tests the candidate's ability to determine and/or interpret bypass valve capacity as it applies to high Reactor 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 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): BFN-ODM-4.20, Rev. 9 (Attach if not previously provided)
0-TI-394, Rev. 15
2-EOI-1A, Rev. 7

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

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.010 Obj. 6.d (As available)

Question Source:	Bank #		
	Modified Bank #	BFN NRC 2304 #22	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2304	

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:	Level	RO	SRO
295007 (APE 7) High Reactor Pressure / 3 **AA2.02** (10CFR 55.41.10) Ability to determine and/or interpret the following as they apply to High Reactor Pressure: • Reactor Power	Tier #	1	------
	Group #	2	------
	K/A #	295007AA2.02	
	Importance Rating	4.5	------

Proposed Question: **# 22**

Unit 2 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
- The crew is implementing the RPV Pressure leg of 2-EOI-1A, ATWS RPV Control

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

In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, a **MINIMUM** of **(1)** Main Turbine Bypass Valves are required to be OPEN for the current Reactor Power Level to control Reactor Pressure between 800 – 1000 psig.

In accordance with 2-EOI-1A, Reactor Pressure is required to be **(2)** .

A. (1) 3
(2) slowly lowered to depressurize the RPV at < 100°F/hr

B. (1) 3
(2) restored and maintained < 1073 psig

C. (1) 6
(2) slowly lowered to depressurize the RPV at < 100°F/hr

D. (1) 6
(2) restored and maintained < 1073 psig

Excerpt from BFN-ODM-4.20: Also supports Distractors A1, B1

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: Also supports Distractors A1, B1

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 (9.9 Unit 2)	2,200 (2,400 unit 2)	1,100,000 (1,140,000 unit 2)
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

Excerpts from EOI-1A: Also supports Distractors A2,C2

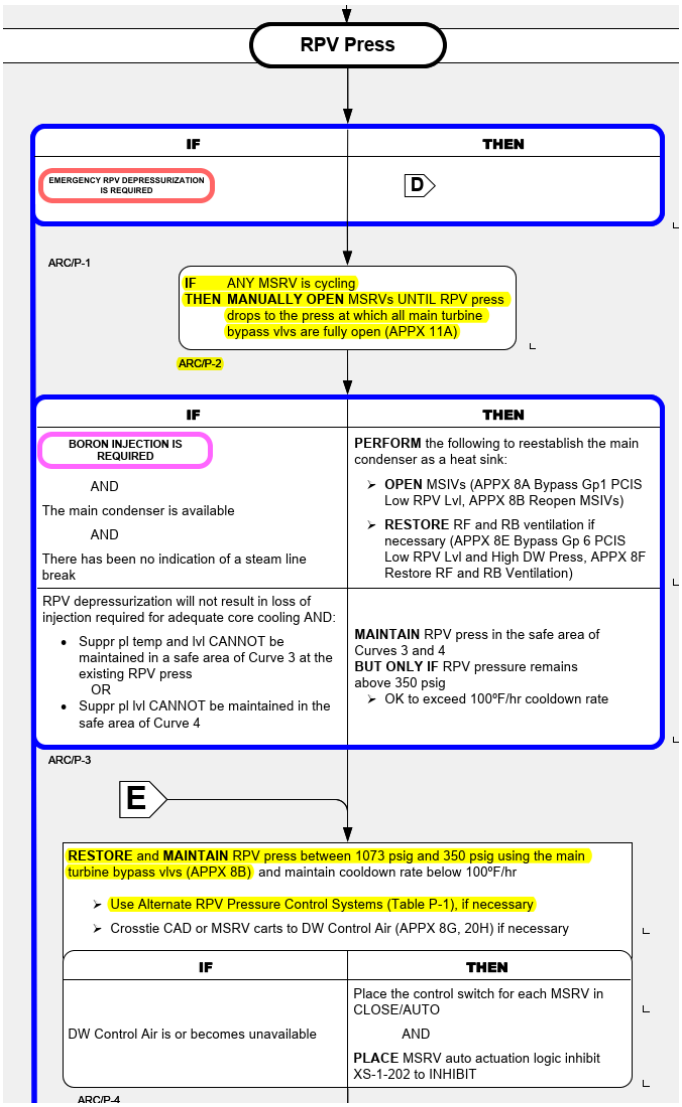


Table P-1

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:

295032 (EPE 9) High Secondary Containment Area Temperature / 5

EA2.02 (10CFR 55.43.5 – SRO Only)

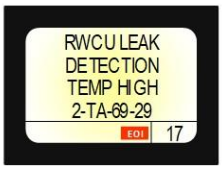
Ability to determine and/or interpret the following as they apply to High Secondary Containment Area Temperature:

- Equipment operability

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	259032EA2.02	
Importance Rating	------	3.5

Proposed Question: **# 84**

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



- RWCU LEAK DETECTION TEMPERATURE HIGH (2-9-3D, Window 17) alarms
- At 0800, 2-TE-69-29G, RWCU HX Area, is 220°F (**MAX SAFE**) **rising** at 1°F per minute
- At 0815, Reactor Water Level indications are:
  - 2-LI-3-58A, REACTOR WATER LEVEL EMERGENCY SYSTEM RANGE, LEVEL A, is (-) 4 inches, stable
  - 2-LI-208A, REACTOR WATER LEVEL NORMAL RANGE, LEVEL A, is 14 inches, stable

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

At 0815, ___**(1)**___ level instrument(s) can be used to determine or trend Reactor Water Level.

At 0840, based on provided level indications, NUSO ___**(2)**___ required to direct 2-C-4, RPV Flooding.

[REFERENCE PROVIDED]

A. (1) 2-LI-3-208A **AND** 2-LI-3-58A
   (2) is

B. (1) 2-LI-3-208A **AND** 2-LI-3-58A
   (2) is NOT

C. (1) 2-LI-3-58A **ONLY**
   (2) is

D. (1) 2-LI-3-58A **ONLY**
   (2) is NOT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that candidate may miscalculate temperature resulting from the rising trend and/or misapply 2-EOI-5, Curves and Cautions, Table 6, Secondary Containment Instrument Runs, and/or Caution 1. Second part is incorrect but plausible in that candidate may miscalculate temperature resulting from the rising trend and/or misapply 2-EOI-5, Curves and Cautions, Table 6, Secondary Containment Instrument Runs, and/or Caution 1 resulting in incorrectly determining that RPV water level cannot be determined, thus warranting entry into C-4, RPV Flooding. (*See attached*)

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*) Initial temperature is given as 220 °F, rising at 1 °F per minute. 15 minutes later temperature would be 235 °F. Referencing Table 6, RWCU HXRM column indicates that both LI-3-208A **AND** LI-3-58A are potentially impacted by elevated temperatures in the area. For the correctly determined temperature at 0815 of 235 °F, the candidate must then reference Caution 1 and apply the calculated temperature to the appropriate instrument(s) to determine impact. For LI-3-208A, with area temperature within the range of 201 °F to 250 °F (235 °F), the minimum indicated Reactor Water Level must be greater than (+) 15 inches. The level provided in the stem is (+) 14 inches, thus insufficient to be considered valid. For LI-3-58A, with area temperature within the range of 201 °F to 250 °F (235°F), the minimum indicated Reactor Water Level must be greater than (-)140 inches. The level provided in the stem is (-) 4 inches, thus sufficient to be considered valid.

Second part, initial temperature is given as 220 °F rising at 1 °F per minute. 40 minutes later, temperature would be 260 °F. Referencing Table 6, RWCU HXRM column indicates that both LI-3-208A **AND** LI-3-58A are potentially impacted by elevated temperatures in the area. For the correctly determined temperature at 0840 of 260 °F, the candidate must then reference Caution 1 and apply the calculated temperature to the appropriate instrument(s) to determine impact. For LI-3-208A, with area temperature within the range of 251 °F to 300 °F (260 °F), the minimum indicated level must be greater than (+) 20 inches. The level provided in the stem is (+) 14 inches, thus insufficient to be considered valid. For LI-3-58A, with area temperature within the range of 251°F to 300°F (260°F), the minimum indicated level must be greater than (-)130 inches. The level provided in the stem is (-)4 inches, thus sufficient to be considered valid. With LI-3-58 still a valid level indication candidate must determine that conditions to direct C-4, RPV Flooding, from retainment override(s) are not met. (*See attached*)

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

SRO Level Justification: Tests the candidate's ability to predict the impact of high Secondary Containment Temperature on equipment operability, specifically, water level instrumentation, and resulting impact(s) on procedure selection and direction. 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):	2-ARP-9-3D, Rev. 41	(Attach if not previously provided)
	2-EOI-1, Rev. 22	
	2-EOI-5, Rev. 6	
	EOIPM Section 0-V(C), Rev. 3	
	OPL171.201, Rev. 10U2	

Proposed references to be provided to applicants during examination: 2-EOI-5 (Caution 1, Table 6, Curve 8)

Learning Objective: OPL171.201 Obj. 10.j (As available)

Question Source:	Bank #		
	Modified Bank #	BFN NRC 1501 #84	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

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:

QUESTION 84 Rev 3

Unit 2 is operating at 100% power.

A leak has occurred in the RWCU Heat Exchanger Room.

- At 0800 Heat Exchanger Room maximum safe temperature (220°F) was reached.
- RWCU Heat Exchanger Room temperature is rising at 1°F per minute
- At 0815 The UO reports the following RPV water level readings:
  2-LI-3-58A reading (-) 4 inches
  2-LI-3-208A reading 16 inches

ASSUME that:
RPV water level remains constant
RWCU Heat Exchanger room temperature rate of rise remains constant.

In accordance with EOI caution 1 which ONE of the following completes the statements below?

At 0815 __ (1) __ level instrument(s) may be used to determine or trend Reactor Water level.

At 0840 based on the level indications given 2-EOI-C4, RPV flooding, __ (2) __ required to be entered.

[Reference Provided]

A. (1) 2-LI-3-208A and 2-LI-3-58A
   (2) is

B. (1) 2-LI-3-208A and 2-LI-3-58A
   (2) is not

C. (1) 2-LI-3-58A only
   (2) is

D. (1) 2-LI-3-58A only
   (2) is not

ANSWER: B

Excerpts from 2-ARP-9-3D:

BFN Unit 2	Panel 2-9-3 XA-55-3D	2-ARP-9-3D Rev. 0041 Page 21 of 42

RWCU LEAK DETECTION TEMP HIGH

2-TA-69-29

17

(Page 1 of 2)

Sensor/Trip Point:

Alarm	Alarm & Isolation
2-TE-69-29A 131°F	2-TIS-69-834A or C and B or D 185°F
2-TE-69-29B 118°F	2-TIS-69-835A or C and B or D 131°F
2-TE-69-29C 118°F	2-TIS-69-836A or C and B or D 148°F
2-TE-69-29D 135°F	2-TIS-69-837A or C and B or D 148°F
2-TE-69-29E 135°F	2-TIS-69-838A or C and B or D 139°F
2-TE-69-29F 135°F	2-TIS-69-839A or C and B or D 166°F
2-TE-69-29G 138°F	
2-TE-69-29H 138°F	

Sensor Location:
2-TE-69-29A & B - RWCU Demin Tanks 2A & 2B
2-TE-69-29C - RWCU Valve Room
2-TE-69-29D - RWCU Pump Room 2A
2-TE-69-29E - RWCU Pump Room 2B
2-TE-69-29F, G, & H - RWCU Heat Exchanger Room
2-TE-69-834A, B, C, & D - RWCU Piping in the Main Steam Tunnel
2-TE-69-835A, B, C, & D - RWCU Pipe Trench
2-TE-69-836A, B, C, & D - RWCU Pump Room 2A
2-TE-69-837A, B, C, & D - RWCU Pump Room 2B
2-TE-69-838A, B, C, & D - RWCU Heat Exchanger Room
2-TE-69-839A, B, C, & D - RWCU Heat Exchanger Room

Probable Cause:
A. RWCU System leak.
B. Ventilation off.
C. Sensor malfunction.

Automatic Action:
If this alarm is received in conjunction with RWCU ISOL 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 2-FCV-69-1, 2-FCV-69-2, and 2-FCV-69-12 close to isolate the RWCU System.

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.
B. **CHECK** RWCU Leak Detection System temperatures on Panel 2-9-22 and Panels 2-9-83, 2-9-84, 2-9-85 and 2-9-86 in the auxiliary instrument room. The 834 and 835 Temp Loops are also displayed on ICS under 'HPTURB' for Main Steam Tunnel and 'RWCU' for the Pipe Trench.

Continued on Next Page

BFN Unit 2	Panel 2-9-3 XA-55-3D	2-ARP-9-3D Rev. 0041 Page 22 of 42

RWCU LEAK DETECTION TEMP HIGH, Window 17
(Page 2 of 2)

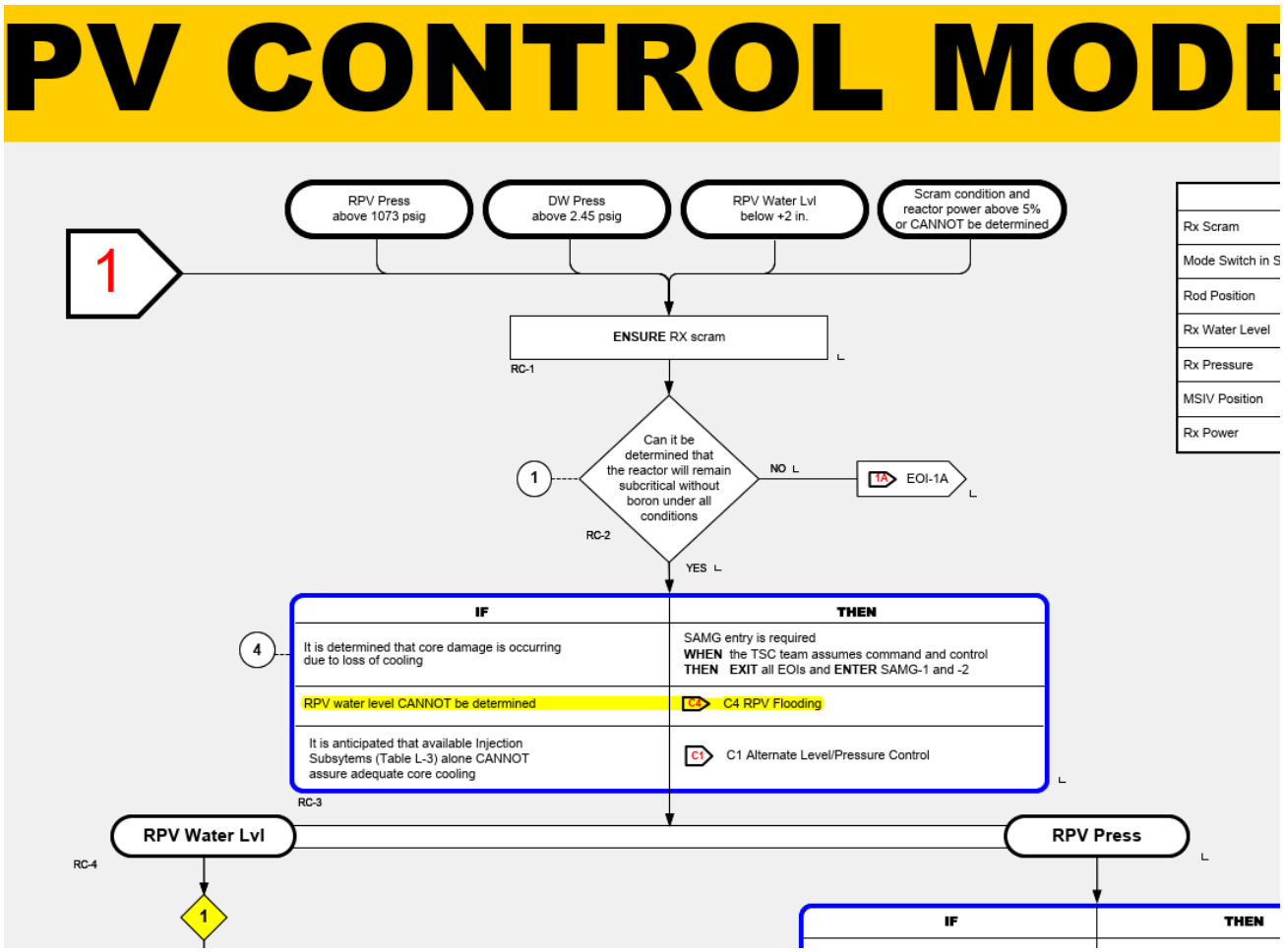
Operator Action: (Continued)

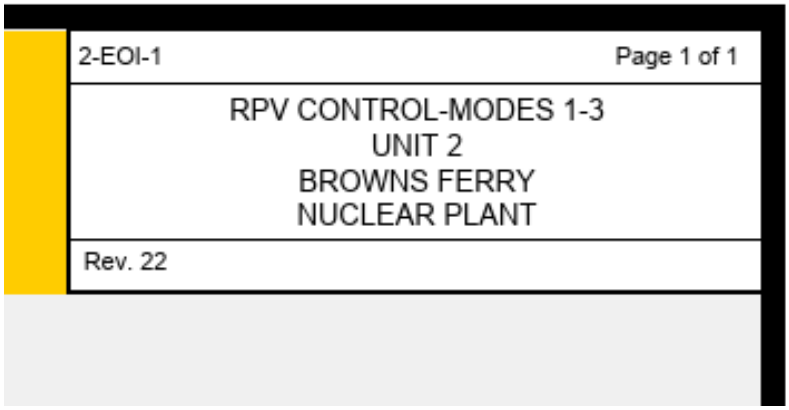
C. **IF** high temperature is indicated on TE 69-29D, E, F, G or H (2-TR-69-29, points 4-8) or TIS 69-835A, B, C, or D, **THEN** **ENTER** 2-EOI-3 Flowchart.
D. **CHECK** Rx Bldg RWCU system ARMs, 2-RI-90-13A and 14A on Panel 2-9-11, and REACTOR & REFUEL ZONE EXHAUST RADIATION, 2-RR-90-144 on Panel 2-9-2.
E. **DISPATCH** personnel to investigate the affected area.
F. **REFER TO** Tech Spec 3.3.6.1.
G. **EVALUATE** equipment associated with this alarm to determine compensatory actions required to maintain REP function. **REFER TO** NPG-SPP-18.3.5.

References:

2-45E620-2	2-730E927-7, 8, & 21	GE 2-920D351
GE 730E719-1	2-47E610-69-1	Tech Spec 3.3.6.1

Excerpts from 2-EOI-1:





Excerpts from 2-EOIPM 0-V-C:

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC-3 (Cont'd)

Second override

If RPV water level cannot be determined, the actions specified in the subsequent steps cannot be performed since RPV water level and water level trend information is required for determining which actions to take. Transferring control of RPV water level from the RPV Control flowchart to Flowchart C4, RPV Flooding is necessary to assure continued adequate core cooling under conditions where RPV water level cannot be determined.

Offscale RPV water level indications do not necessarily require entry of Flowchart C4 provided the indications are believed to be valid. Refer to the definition of "cannot be determined" in EOIPM Section 0-I(C).

Third Override

Each Injection Subsystem (Table L-3) is a high capacity motor-driven system loop independently capable of supplying high quality makeup water to the RPV. The Injection Subsystems are a subset of the Preferred Injection Systems (Table L-1), which include both those used to control RPV water level during normal plant operations at power and those categorized as emergency makeup.

The maximum RPV pressure at which Injection Subsystems can inject is relatively low. If the flow provided by Injection Subsystems will not provide adequate core cooling, available steam-driven systems and low-capacity motor-driven systems such as CRD, SLC and Alternate Injection Subsystems must be relied on until additional systems become available.

Adequate core cooling can be established by core submergence, spray cooling, or steam cooling, as given in Table M-3. Irrespective of the current RPV pressure and injection flowrate, if it is determined that one or more Injection Subsystems are capable of assuring adequate core cooling by any of these methods, transfer to Flowchart C-1 is not required; a Subsystem may be considered "available" even if not currently injecting. The RPV pressure control subsection and related contingencies provide guidance on controlling RPV pressure to support core cooling strategies and, when appropriate, depressurizing the RPV to permit injection from low pressure sources.

Excerpt from 2-EOI-5 (candidate will be provided Caution 1, Table 6, and Curve 8):

2-EOI-5 CURVES AND CAUTIONS

1

- An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp
- If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run.

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR -64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 100
		-150	N/A	101 to 150
		-145	N/A	151 to 200
		-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	Normal 0 to +60	-5	N/A	below 150
		+5	N/A	151 to 200
		+15	N/A	201 to 250
		+20	N/A	251 to 300
		+30	N/A	301 to 350
LI-3-52 LI-3-62A	Post Accident -268 to +32	on scale	N/A	N/A
LI-3-55	Shutdown Floodup 0 to +500	+10	Below 100	N/A
		+15	100 to 150	N/A
		+20	151 to 200	N/A
		+30	201 to 250	N/A
		+40	251 to 300	N/A
		+50	301 to 350	N/A
		+65	351 to 400	N/A

2 Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1, 2, 9 or 10) or with suppression pool water level below 10 ft (Vortex limit) may cause equipment damage

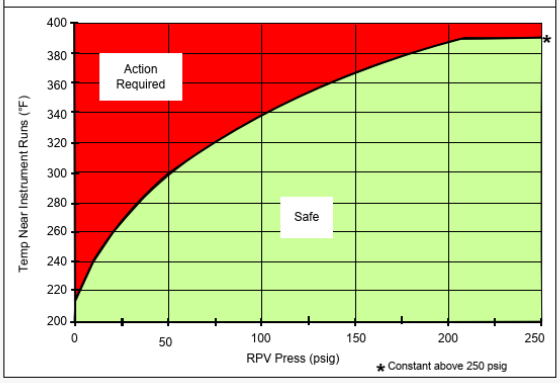
3 Elevated suppr chmbr press may trip the RCIC turbine on high exh pressure

4 Reducing PC press will reduce the available NPSH for pumps taking suction from the suppr pl

Table 6
Secondary Cntmt Instrument Runs

INSTRUMENT	SC TEMP ELEMENTS AND LOCATIONS			
	EI 621 (74-95F)	EI 593 (74-95C and D)	EI 565 (69-835A thru D)	RWCU HXRM (69-29F, G, H)
LI-3-58A	°F	°F	N/A	°F
LI-3-58B	°F	°F	N/A	N/A
LI-3-53	°F	°F	N/A	°F
LI-3-60	°F	°F	N/A	N/A
LI-3-206	°F	°F	N/A	°F
LI-3-253	°F	°F	N/A	N/A
LI-3-52	°F	°F	°F	N/A
LI-3-62A	°F	°F	°F	N/A
LI-3-208A, B	°F	°F	N/A	°F
LI-3-208C, D	°F	°F	N/A	N/A

Curve 8
RPV Saturation Temp



Excerpts from OPL171.201:

12. Can/Cannot Be Restored Above/Below — Objective 10l

The value of an identified parameter is/is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a value cannot be restored and maintained above or below a specified limit does not require immediate action simply because the current value is outside the range, but does not permit extended operation beyond the limit; the action must be taken as soon as it is apparent that the specified range cannot be attained.

F. EOI Cautions — Located on EOI-5

Cautions are provided within the EOIs to identify potentially adverse consequences of certain plant conditions or specified actions. — OF-5

A Caution is numbered sequentially and remains consistent throughout the EOIs. For example, Caution #2 in EOI-1 is the same as Caution #2 in EOI-1A. — Objective 11

The full text version of each caution reference is found on EOI-5, Curves and Cautions.

1. Caution 1

a) Caution #1 defines conditions under which RPV water level indications may be unreliable or must be considered invalid due to the effects of RPV pressure and primary and secondary containment temperatures. — See EOIPM Section 0-V(B) EOI Curves, Cautions, Tables and Variables Bases

b) The RPV water level instrument systems sense level by measuring the differential pressure (ΔP) between a variable leg water column and a reference leg water column. The reference leg is kept full of water by a condensing pot; the variable leg height depends upon RPV water level. When the actual RPV water level decreases, the variable leg height also decreases, causing the sensed ΔP to increase. The higher ΔP results in a lower indicated level. Conversely, when the actual RPV water level rises, the variable leg height increases, the ΔP decreases, and the indicated level increases.

c) The first bullet of Caution #1 addresses the effect that RPV water level instrument reference leg vertical run temperature has on the validity of indicated RPV water level trend and on the ability to determine that RPV water level is within the instrument indicating range. Except for the Shutdown Floodup instrument, drywell temperatures have little effect on indicated water level because there is little difference between the reference leg and variable leg vertical runs in the drywell. The greater difference occurs in the secondary containment for the other water level instruments.

d) The Minimum Indicated Level (MIL) for an RPV water level instrument is the highest indication resulting from a specified range of

OPL171.201, Introduction to the Emergency Operating Instructions, Rev.10U2

instrument run temperatures with actual RPV water level at the variable leg tap. The Maximum Run Temperature (MRT) is the instrument run temperature that would produce an on-scale indication with actual water level at the variable leg RPV tap. The MIL and MRT are utilized in establishing the conditions under which an RPV water level instrument may be used to determine RPV water level. Separate levels and temperatures are provided for each RPV water level instrument.

e) The second bullet of Caution #1 identifies the limiting conditions beyond which boiling of the water in the instrument legs may occur.

f) The RPV Saturation Temperature is a plot of the saturation temperature of water as a function of pressure. If the temperature of the water in an RPV water level instrument run exceeds this temperature, the water may start to boil, resulting in unreliable level indication.

g) Boiling is of concern in all instrument runs—horizontal and vertical, reference and variable, drywell and secondary containment. Boil off from the reference leg reduces the height of water in the leg. This decreases the pressure on the reference leg side of the differential pressure transmitter and increases the indicated level. Boiling in the variable leg increases the pressure on the variable leg side of the transmitter, likewise increasing the indicated level. Continued boiling would produce an increasing level trend.

h) Boiling and loss of valid level indication may not occur immediately when drywell or secondary containment area temperature exceeds Figure 8, RPV Saturation Temperature. As drywell or secondary containment area temperature increases, the temperatures in the instrument runs will lag the temperature in the surrounding area. The temperature of the water in the instrument run may therefore be lower than the indicated air temperatures.

OPL171.201, Introduction to the Emergency Operating Instructions, Rev.10U2

c) The maximum drywell pressure at which the SRVs will still open and remain open is 65 psig.

d) The elevation of the SRV solenoids is at 70 feet. This is 10 feet above the drywell instrument tap. An airspace pressure of 65 psig would result in an indicated pressure of 69 psig at an elevation of 70 feet hence the rise in the SRV limit pressure between 60 and 70 feet.

e) Above 70 feet the solenoid and pressure tap see the same backpressure from the water and this results in the SRV operability becoming limiting at containment levels above 90 feet.

8. **Curve 8**: RPV Saturation Temp

The RPV Saturation Temperature is a plot of the saturation temperature of water as a function of pressure. If the temperature of the water in an RPV water level instrument run exceeds this temperature, the water may start to boil, resulting in unreliable level indication.

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

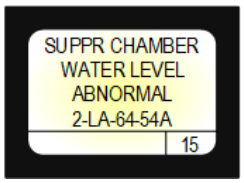
Examination Outline Cross-reference:

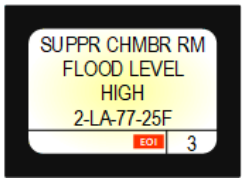
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5
G2.4.20 (10CFR 55.43.5 – SRO Only)
Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions and notes

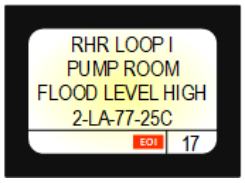
Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295036G2.4.20	
Importance Rating	------	4.3

Proposed Question: **# 85**

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, which **ONE** of the following completes the statement below in accordance with the EOIs?

At this time, HPCI ___**(1)**___ available for use and the NUSO is required to direct the crew to ___**(2)**___.

A. (1) is
(2) Emergency Depressurize in accordance with 2-EOI-3

B. (1) is
(2) Shutdown the Reactor in accordance with 2-GOI-100-12A

C. (1) is NOT
(2) Emergency Depressurize in accordance with 2-EOI-3

D. (1) is NOT
(2) Shutdown the Reactor in accordance with 2-GOI-100-12A

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B).* Second part is incorrect but plausible if the candidate incorrectly applies EOI-3 Note 3 and misdiagnoses conditions and determines that RPV depressurization, rather than Reactor shutdown, is required in accordance with EOI-3, Step SC-4. (Since a Primary System is not discharging into Secondary Containment, an Emergency Depressurization would not be required.)

B **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.

Second part, given the conditions provided, it should be determined that, in accordance with 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. (Reference Note 3) Therefore, Reactor shutdown is required in accordance with step SC-6. Note that no reference is provided to the candidate.

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 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 emergency operating notes and associated application with high Secondary Containment area water level. 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.

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-9-3B, Rev. 40 (Attach if not previously provided)
2-ARP-9-4C, Rev. 39
2-EOI-2, Rev. 19
2-EOI-3, Rev. 20

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: OPL171.204 Obj. 5, 9 (As available)

Question Source: Bank # [redacted]

Form 4.2-1	**Written Examination Question Worksheet**		
	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-9-3B:

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

Excerpts from 2-ARP-9-4C:

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0039 Page 7 of 44

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

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0039 Page 24 of 44

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

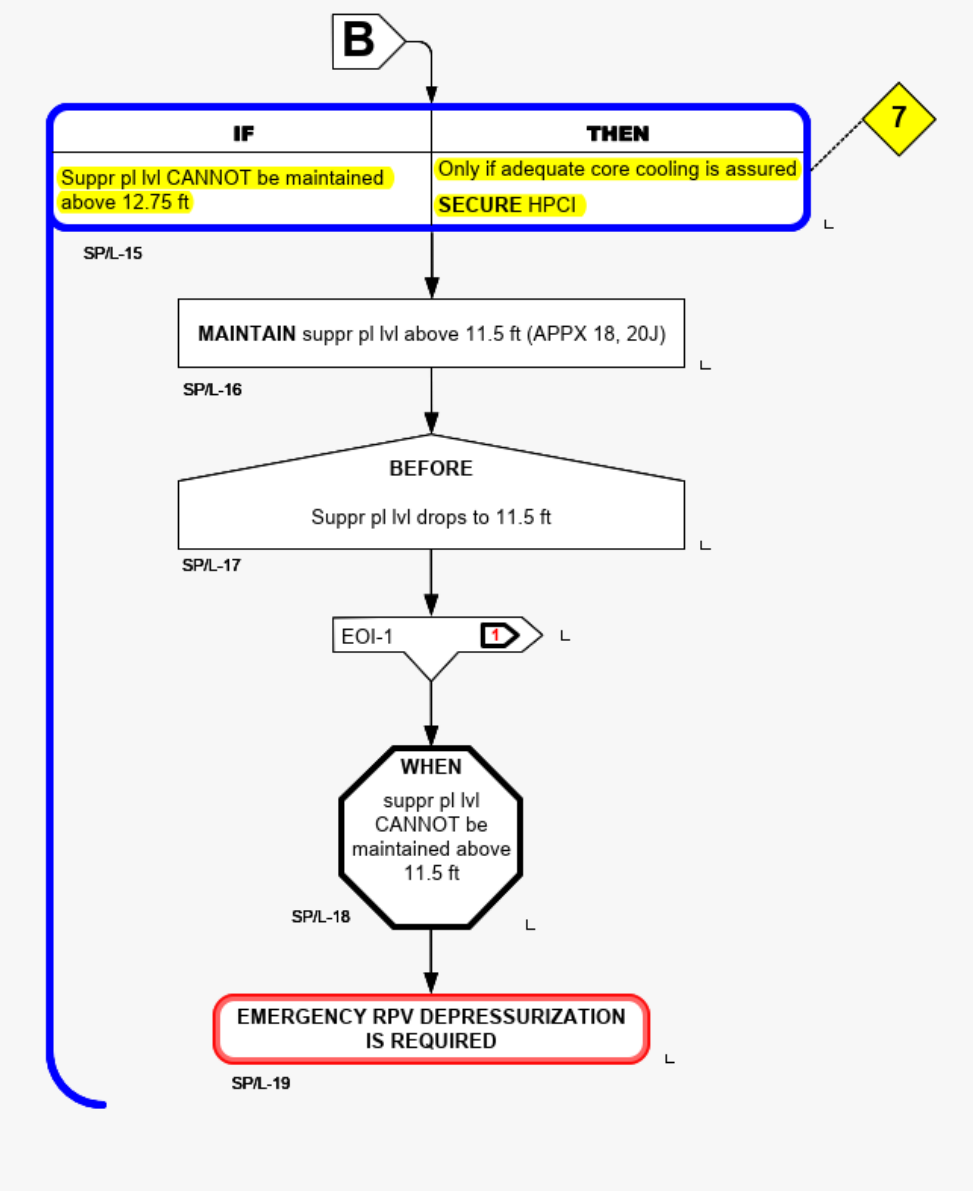
Operator Action:

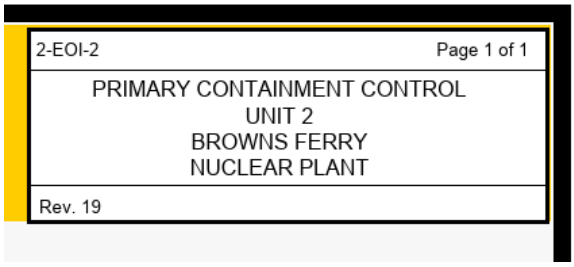
A. **DISPATCH** personnel to visually check the RHR pump room.

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.

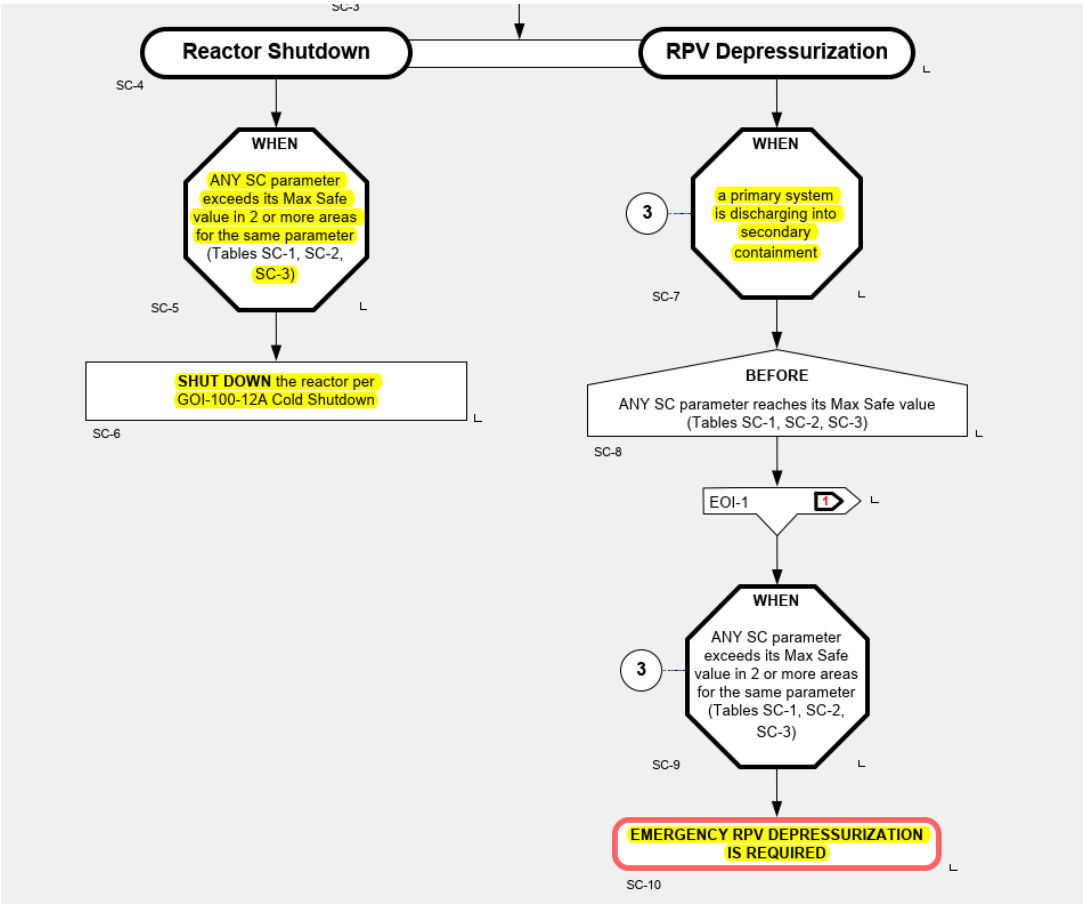
References: 0-47E610-77-1 47W600-8 47W600-100
45N620-4 FSAR Sections 13.6.2 and F.7.15

Excerpt from 2-EOI-2: Also supports Distractors C1, D1





Excerpts from 2-EOI-3: Also supports Distractors A2, C2

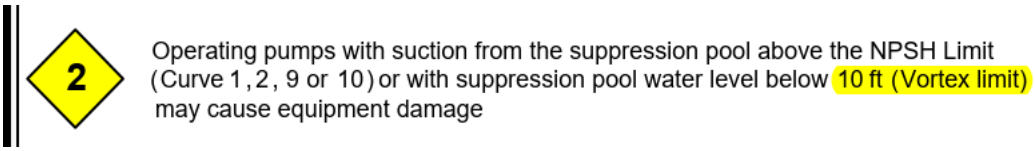


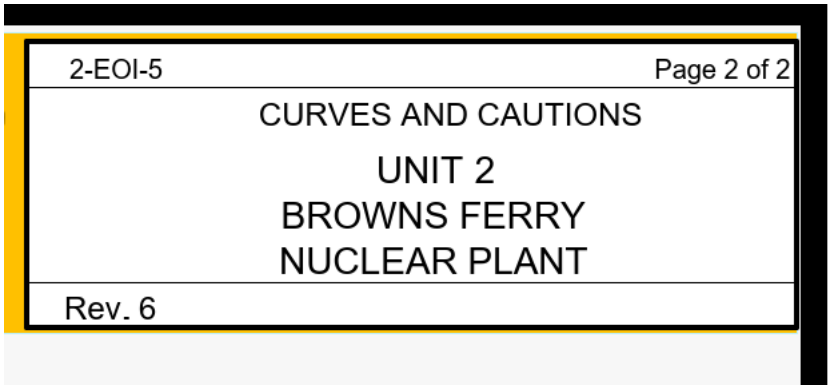
NOTE	
3	Tables SC-1 and SC-2 contain information that may be used to determine if a primary system is discharging into Secondary Cntmt (emergency depressurization will reduce discharge)

Table SC-3
Secondary Cntmt Water Level

Water Level	Max Normal Value in.	Max Safe Value in.
Floor drain sumps	66	None
Areas	2	20

Excerpt from EOI-5: Supports Distractors C1, D1





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

Examination Outline Cross-reference:

206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection

A2.05 (10CFR 55.43.5 - SRO Only)

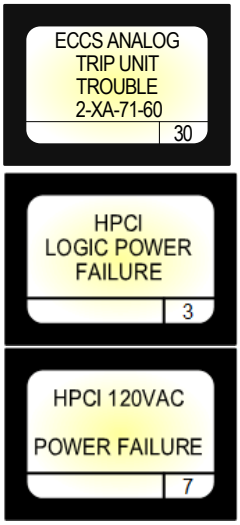
Ability to (a) predict the impacts of the following on the High Pressure Coolant Injection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- DC electrical distribution system failures

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	206000A2.05	
Importance Rating	------	3.9

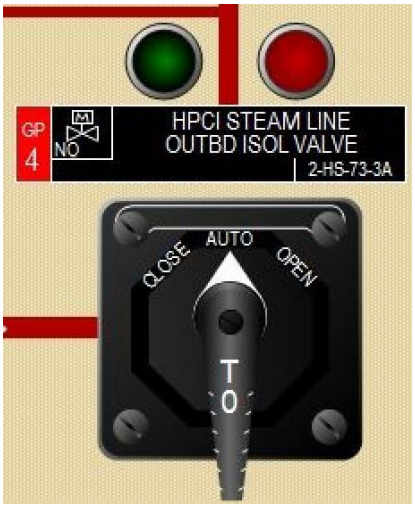
Proposed Question: **# 86**

Unit 2 is operating at 100% RTP when the following alarms and indication occurs:



- ECCS ANALOG TRIP UNIT TROUBLE (2-9-3E, Window 30)
- HPCI LOGIC POWER FAILURE (2-9-3F, Window 3)
- HPCI 120VAC Power Failure (2-9-3F, Window 7)

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



The 2A ___**(1)**___ RMOV Board is de-energized.

Unit 2 Tech Spec 3.6.1.3, PCIVs, REQUIRED ACTIONs ___**(2)**___ are required.

[REFERENCE PROVIDED]

A. (1) 480V
   (2) A.1, A.2, and B.1

B. (1) 250Vdc
   (2) A.1, A.2, and B.1

C. (1) 480V
   (2) A.1 and A.2

D. (1) 250Vdc
   (2) A.1 and A.2

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that HPCI has AC, DC, and inverted power supplies. For example, 2-FCV-73-2, HPCI Steam Line Inboard Isolation Valve, is powered from 2A 480V RMOV Board. Additionally, two alarms provided (9-3E Window 30, and 9-3F, Window 7, are received upon failure of either AC or DC power supplies.

Second part is incorrect but plausible if candidate misidentifies power supplies to HPCI steam line inboard and outboard valves and, as a result, determines that two PCIVs are INOPERABLE for a penetration flow path with two PCIVs. (*See attached*)

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

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

D **CORRECT:** (See attached) With the given alarms (HPCI Logic Power Failure, 2-9-3F, Window 3) and graphic indicating a loss of power to the hand-switch for FCV-73-3, HPCI Steam Line Outboard Isolation Valve, it can be determined that a loss of 2A 250Vdc Board has occurred.

Second part, with only a single PCIV INOPERABLE for a penetration flow path with two PCIVs, TS 3.6.1.3, Condition A, with Required Actions A.1 and A.2 are applicable.

SRO Level Justification: Tests the candidate's ability to predict the impact of DC electrical system failures on HPCI and the associated requirements to be implemented in accordance with applicable Tech Specs. 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 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.

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

Technical Reference(s): 2-ARP-9-3E, Rev. 36 (Attach if not previously provided)
2-ARP-9-3F, Rev. 43
2-45E751-1, Rev. 68
U2 Tech Spec 3.6.1.3, Amend. 334, 351

Proposed references to be provided to applicants during examination: ECCS ANALOG TRIP UNIT TROUBLE (2-9-3E, Window 30), HPCI LOGIC POWER FAILURE (2-9-3F, Window 3), 2-HS-73-3A, HPCI STEAM LINE OUTBOARD ISOLATION VALVE, indication, Unit 2 Tech Spec 3.6.1.3 (pages 3.6-9 – 3.6-13)

Learning Objective: OPL171.042, Obj. 5 (As available)

Question Source:
- Bank #
- Modified Bank # OPL171.037-012 001 #1435 (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**

Comments:

Copy of Bank Question:

1435. OPL171.037-12 001

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 power failure alarms, the following significant alarms result:

- 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)

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 to its ALTERNATE power source, the Board is considered __**(2)**__ in accordance with Tech Spec 3.8.7, "Distribution Systems – Operating" and 0-OI-57-1D, DC Electical System precautions and limitations.

A. **(1)** 2A
(2) inoperable

B. **(1)** 2B
(2) inoperable

C. **(1)** 2A
(2) operable

D. **(1)** 2B
(2) operable

Excerpts from 2-ARP-9-3E:

BFN Unit 2	Panel 9-3 2-XA-55-3E	2-ARP-9-3E Rev. 0036 Page 34 of 41

ECCS ANALOG
TRIP UNIT
TROUBLE
2-XA-71-60

30

(Page 1 of 2)

Sensor/Trip Point:

Relay 74-71-60-1X1	Relay 74-71-60-2X1
Relay 74-71-60-1X2	Relay 74-71-60-2X2
Relay 74-71-60-1AX1	Relay 74-71-60-2AX1
Relay 74-71-60-1AX2	Relay 74-71-60-2AX2
Relay 74-71-601	Relay 74-71-602

Contact ALC from Inverter

Sensor Location:

Trip UNIT:	INVERTER:
Panel 9-81 and 9-82	Shutdown Board
Unit 2 Aux. Inst.Room	Rooms C and D

Probable Cause:

A. Calibration of master trip unit(s) in progress.
B. Card out of card file.
C. Failure of a master or slave trip unit.
D. Power supply failure.
E. Inverter trouble.
F. Loss of 2A or 2B 250VDC RMOV BD.
G. Valid alarm.
H. Loss of power to cabinet fan.
I. Purging the Drywell.

Operator Action:

A. This alarm in conjunction with HPCI or RCIC logic power failure alarm may indicate a loss of the 250VDC RMOV BD.
B. **DISPATCH** personnel with key to Aux. Inst. Room, Panels 9-81 and 9-82 to check:
   - Trip unit being calibrated or calibration unit power switch in ON position.
   - Fuse 2-FU2-071-0060/1A, DDD block, and 2-FU1-71-60/1 BBA block, in Panel 9-81,.
   - Fuse 2-FU2-071-0060/2A, DDD block, and 2-FU1-71-60/2 BBA block, in Panel 9-82,.
   - Fuse 2-FU1-071-0601A (cabinet fan) in Panel 9-81.
   - Fuse 2-FU1-071-0602A (cabinet fan) in Panel 9-82.

C. **IF** fuse 2-FU1-256-1F has cleared, **THEN** **REQUEST** EM to verify capacitors and SCRs are good prior to replacing the fuse.

Continued on Next Page

ECCS ANALOG TRIP UNIT TROUBLE 2-XA-71-60, Window 30
(Page 2 of 2)

Operator Action: (Continued)

D. **DISPATCH** personnel to S/D BD RM C & D to check ECCS ATU Inverter status and the following breakers.
- 2A 250VDC RMOV BD breakers 9A1 and 11A1.
- 2B 250VDC RMOV BD breakers 1B1 and 8A.

E. **REFER TO** 2-AOI-57-11 for a loss of power to the ECCS ATU panel.
F. **NOTIFY** Instrument Maintenance for assistance in correction of conditions.
G. **REFER TO** Tech Specs 3.3.5.1, 3.5.and TRM 3.3.3.4

References: 45E670-13, -16, -19, and -22 2-45E620-1 GE 730E915
Tech Spec 3.3.5.1, 3.5, 5.4, 5.5 and Table 3.3.5.1
Tech Requirements Manual 3.3.3
FSAR Section 13.0

Excerpts from 2-ARP-9-3F:

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0043 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:

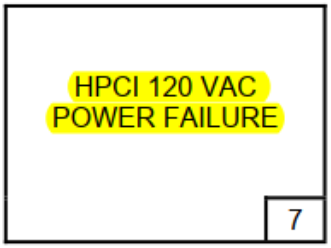
A. **DETERMINE** which logic bus has failed, **REFER TO** automatic action section.
B. **DISPATCH** personnel to check source of power failure:
   1. Logic Bus A:
      a. Power supply - 250V DC RMOV Bd 2B, Breaker 1B1.
      b. Fuse set 2-FU2-073-23A-K36 (23AF19 & 23AF20), Panel 9-32.
   2. Logic Bus B:
      a. Power supply - 250V DC RMOV Bd 2A, Breaker 11D1.
      b. Fuse sets 2-FU2-073-0039A (23AF1 & 23AF2) and 2-FU2-073-0039B (23AF19 & 23AF20), Panel 9-39.

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. 0043 Page 11 of 41



(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).

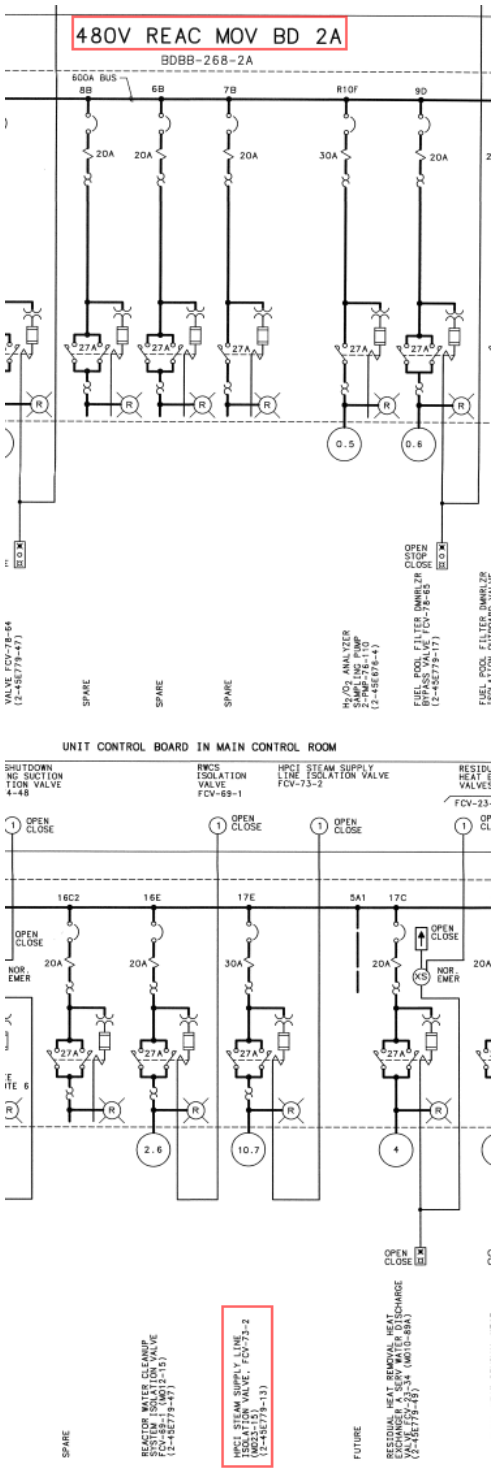
Automatic Action:
A. HPCI controller loses power. HPCI becomes inoperable.
B. If HPCI is in service, the HPCI Turbine Stop Valve, 2-FCV-73-18, closes. HPCI controller loses power. HPCI becomes inoperable.
C. 2-PI-064-67B will lose power and become inop.

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 2-45E751-1: Supports Distractors A1, C1



Excerpts from TS 3.6.1.3: Also supports Distractors A2, B2

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTES----------

1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be unisolated 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.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Only applicable to penetration flow paths with two PCIVs. ---------- One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits.	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.	4 hours except for main steam line OR In accordance with the Risk Informed Completion Time Program
	AND	8 hours for main steam line OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	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.	Once per 31 days [following isolation] for isolation devices outside primary containment AND 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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ----------NOTE---------- Only applicable to penetration flow paths with two PCIVs. ---------- One or more penetration flow paths with two PCIVs inoperable except due to MSIV leakage not within limits.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ----------NOTE---------- Only applicable to penetration flow paths with only one PCIV. ---------- One or more penetration flow paths with one PCIV inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND C.2 ----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	4 hours except for excess flow check valves (EFCVs) AND 12 hours for EFCVs Once per 31 days [following isolation]

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more penetration flow paths with MSIV leakage not within limits.	D.1 Restore leakage rate to within limit.	4 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours

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

Examination Outline Cross-reference:

209001 (SF2, SF4 LPCS) Low-Pressure Core Spray

G2.4.45 (10CFR 55.43.2 - SRO Only)

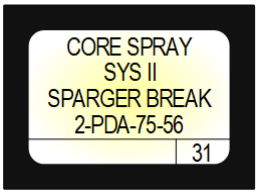
Ability to prioritize and interpret the significance of each annunciator or alarm

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	209001G2.4.45	
Importance Rating	------	4.3

Proposed Question: **# 87**

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

- CORE SPRAY SYSTEM II SPARGER BREAK (2-9-3F, Window 31) alarms
- Operator at Panel 2-25-27 reports 2-PDIS-75-56, Core Spray Sparger to Reactor Pressure Vessel D/P, is indicating 2 psid.



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

The **LOW** pressure side of 2-PDIS-75-56 senses **(1)** Core Plate pressure.

In accordance with Technical Specifications / Technical Requirements Manual Bases, Core Spray Loop II **(2)** OPERABLE.

A. (1) below
(2) is NOT

B. (1) below
(2) is

C. (1) above
(2) is NOT

D. (1) above
(2) is

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** First part is incorrect but plausible in that the Core Spray Sparger leak detection low side instrumentation is configured in conjunction with SLC injection piping as part of a “pipe within a pipe” assembly. The “inner” pipe is the SLC injection diffuser sparger and terminates just below the core plate. The “outer” pipe penetrates the vessel inside the shroud (near the core edge) and terminates in the core plate. The concentric arrangement provides input signals to various systems for both below-core-plate and above-core-plate pressures. The “inner” pipe provides the below-core plate pressure signal to the Core Spray Sparger leak detection low side. This configuration is easily and often confused.

Second part is correct (*See C*).

B INCORRECT: First part is incorrect but plausible (*See A*). Second part is incorrect but plausible in that with the plant shutdown and cooled down, the provided annunciator and PDIS indication are expected for the condition. The candidate may equate a valid psid reading for shutdown conditions with operating conditions. Doing so would lead to determining Core Spray remains OPERABLE.

C **CORRECT:** (*See attached*) The low side of the detector senses above-core plate pressure (P2) plus the pressure due to the height of water in the vessel. Under normal conditions, the high side of the detector senses core exit pressure (P3) plus pressure due to the height of water in the sensing leg. With the plant operating at rated conditions the detector reads +3.5 psid.

Second part, conditions provided are indicative of a Core Spray Sparger break. In this case, P3 would be slightly less than P2 due to the DP across the core. Therefore, the pressure differential detected would be mainly due to the height of cold water (135°F) in the high leg of piping. With the Core Spray piping broken between the reactor vessel and the shroud, piping would sense P5 instead of P3, and the high-side pressure at the detector decreases by 7 psig. Sensed low-side pressure would remain the same. The result would be a reduction in DP, causing 2-9-3F, Window 31, to alarm at 2 psid decreasing (following a 15-sec time delay). During cold shutdown conditions this alarm will normally be in. This is due to low-side pressure being greater than high side pressure (negative DP). With the indicated break in the Core Spray Sparger, that loop of Core Spray is no longer capable of performing its required function and as a result is not OPERABLE.

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

SRO Level Justification: Tests the candidate's ability to interpret the significance of an alarm associated with Core Spray not typically associated with the operating condition. Specifically, to identify and predict the impact of a broken Core Spray Sparger, and, given that conclusion, determine the Tech Spec required action.

SRO only because of the links 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 as it relates to Core Spray and Tech Specs.

Technical Reference(s):	2-ARP-9-3F, Rev. 43	(Attach if not previously provided)
	OPL171.039, Rev. 24	
	OPL171.045, Rev. 23	
	TRM Bases 3.3.3.3, Rev. 0	
	Tech Spec 3.5.1, Amend. 351	
Proposed references to be provided to applicants during examination:		CORE SPRAY SYSTEM II SPARGER BREAK (2-9-3F, Window 31)
Learning Objective:	OPL171.045, Obj. 2.d, 5.b, 6 (As available)	
Question Source:	Bank #	
	Modified Bank #	BFN NRC 2104 #32 (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:

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

Copy of Bank Question:

Examination Outline Cross-reference:

209001 (SF2, SF4 LPCS) Low-Pressure Core Spray

G2.4.31 (10CFR 55.41.10)

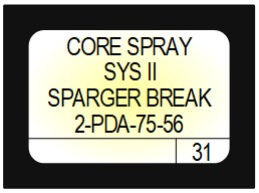
Knowledge of annunciator alarms, indications, or response procedures.

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	209001G2.4.31	
Importance Rating	4.2	------

Proposed Question: **# 32**

Which **ONE** of the following completes the statements below in accordance with the associated ARP?

- CORE SPRAY SYSTEM II SPARGER BREAK (2-9-3F, Window 31)



Core Spray Sparger break detection is used to detect a possible Core Spray System piping break inside the Reactor Vessel, **(1)** to the Core Shroud.

Due to break detection instrumentation design, the given annunciator **(2)** be sealed in during normal cold shutdown conditions on a Unit.

A. (1) external
(2) would

B. (1) external
(2) would NOT

C. (1) internal
(2) would

D. (1) internal
(2) would NOT

Proposed Answer: **A**

Excerpt from 2-ARP-9-3F: Also supports Distractors A2, C2

BFN Unit 2	Panel 9-3 2-XA-55-3F	2-ARP-9-3F Rev. 0043 Page 35 of 41

CORE SPRAY
SYS II
SPARGER BREAK
2-PDA-75-56
31

(Page 1 of 1)

Sensor/Trip Point:
2-PDIS-075-0056 Decreasing 2 psig DP (15 second time delay)

Sensor Location: Panel 25-27
El 565', R-10 S-LINE

Probable Cause:
A. Core spray piping break between reactor vessel wall and reactor shroud if alarm comes in during steady-state power operation.
B. Low core flow or unit in cold condition.
C. Sensor malfunction.

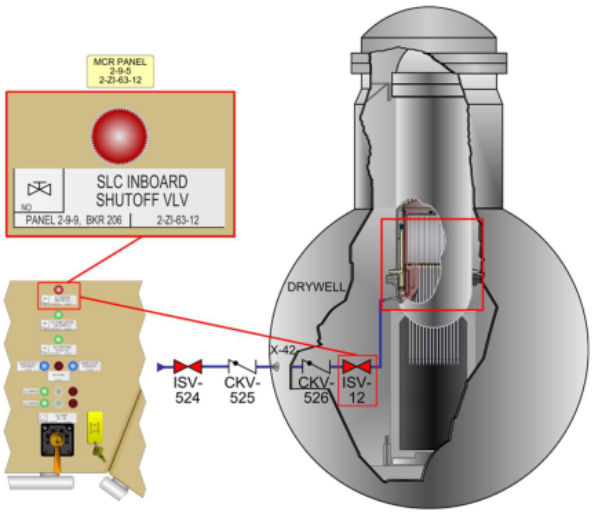
Automatic Action: None

Operator Action:
A. **DISPATCH** personnel to Panel 2-25-27 to compare 2-PDIS-75-56 with 2-PDIS-75-28. The normal reading should be approximately 3.5 psid at high power operation.
B. **IF** instrument verification is necessary, **THEN** **REQUEST** that IMs check instrument operation.
C. **IF** indications confirm a broken core spray header, **THEN**, The associated Core Spray System is inoperable **REFER TO** Tech Spec 3.5.1 and 3.5.2, TRM 3.3.3.3.
D. **IF** there are NO indications of a core spray header break, **THEN** **REFER TO** Tech Spec Table 3.3.5.1.

References: 2-45E620-1 2-47E600-59 2-47E610-75-1
GE 730E930-2 and -8 Technical Specifications 3.5.1 and 3.5.2
Technical Requirements Manual-TRM 3.3.3.3

Excerpts from OPL171.039 Lesson Plan: Also supports Distractors C1, D1

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24



2. The SLC injection piping is commonly referred to as "a pipe within a pipe". The line enters the bottom head of the reactor vessel as two concentric pipes. The "inner" pipe is the SLC injection diffuser sparger and terminates just below the core plate. The location combined with the perforations in the piping (sparger), results in good mixing and dispersion of the sodium pentaborate within the reactor. The "outer" pipe penetrates the vessel inside the shroud (near the core edge) and terminates in the core plate.

   The concentric arrangement acts to reduce thermal shock at the vessel nozzle due to the much cooler temperature of the SLC solution. In addition, this arrangement also provides input signals to various systems for both below-core-plate and above-core-plate pressures.

   The "inner" pipe provides a below-core plate pressure signal to the following:

   1. Core Differential Pressure instrument
   2. All Jet Pumps Differential Pressure instruments
   3. Jet Pump Total Developed Head Differential Pressure instrument
   4. RHR and Core Spray System's initiation and injection valve logic

NLO Obj. 2, 4e /g

NLOR Obj. 1, 3c / e, 6b / d / f

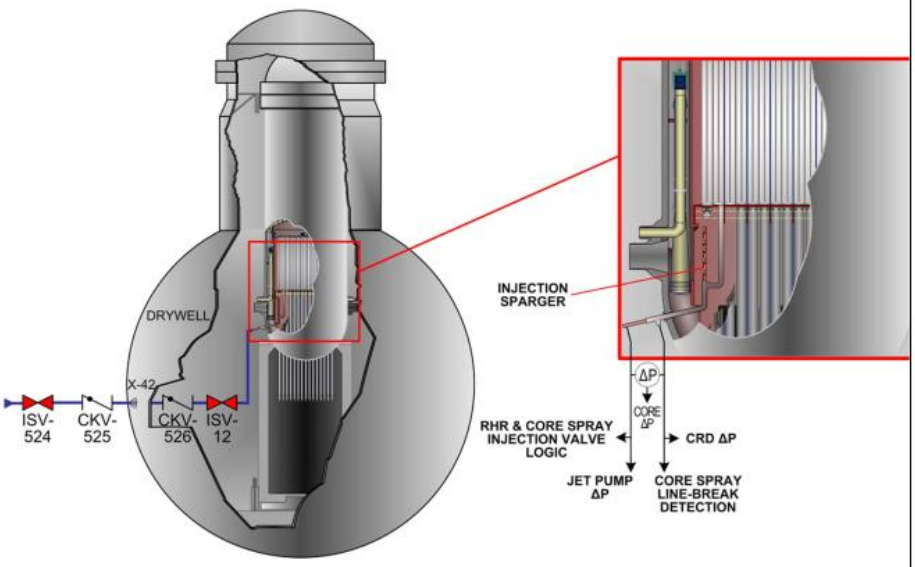
ILT Obj. 2, 4e / f, 10d, 11b / d / f

LOR Obj. 10d, 11b / d / f

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

The "outer" pipe provides an above-core plate pressure signal to the following:

1. Core Differential Pressure instrument
2. Control Rod Drive (CRD) System drive and cooling water Differential Pressure instruments
3. Core Spray System line-break detection circuitry



3.

For Suppression Pool pH Control: The flowpath is the same as for normal operation, with the exception that the boron solution is expected to leave the RPV through the pipe break and drain into the Suppression Pool inside containment.

NLO Obj. 2, 6c

NLOR Obj. 1, 7e

ILT Obj. 2, 5a, 11c, 12e

LOR Obj. 5a, 11c, 12e

OPL171.039, Standby Liquid Control (SLC) System, Rev. 24

8. **Core Spray System**

   a. As mentioned in Reactor Vessel Instrumentation above, the Core Spray System uses the SLC INJECTION SPARGER as a sensing point for line-break detection logic and pressure input to the Core Spray initiation and valve opening logic (in conjunction with a High Drywell Pressure signal of 2.45 psig). A break in the SLC injection line where it penetrates the vessel will cause the Core Spray logic to sense a Reactor Pressure of less than 450 psig. If the Drywell Pressure exceeds 2.45 psig, which it likely will due to the small-break LOCA created, the Core Spray Pumps will receive an initiation signal. The Core Spray Pumps will start and the injection valves will attempt to open, even if actual Reactor Pressure remains above 450 psig.

NLOR Obj. 6d / f
ILT Obj. 11d / f
LOR Obj. 11d / f

10. **Residual Heat Removal System**

   a. As mentioned in Reactor Vessel Instrumentation above, the RHR System uses the SLC INJECTION SPARGER as a sensing point for pressure input to the RHR initiation and valve opening logic (in conjunction with a High Drywell Pressure signal of 2.45 psig AND another Reactor Pressure input from sensors associated with Reactor Water Level Instrumentation). A break in the SLC injection line where it penetrates the vessel will cause the RHR logic to sense a Reactor Pressure of less than 450 psig. But, because RHR requires a confirmatory low Reactor Pressure signal, there would be no effect on the RHR System or the associated closure of the Reactor Recirculation Pump Discharge Valves until actual Reactor Pressure dropped to less than 450 psig.

NLOR Obj 6f
ILT Obj. 11f
LOR Obj. 11f

Excerpts from OPL171.045 Lesson Plan: Also supports Distractors C1, D1

10-K73B) when a Unit 1 LOCA is sensed provided a Unit 2 div 2 CS initiation doesn't exist.

response is normal and uninterrupted.

iv. For Unit 1, division 1 works exactly like division 2 on Unit 2. Unit 1 division 2 is the Unit 1 non-preferred division and works the same as Unit 2 division 1.

CASA & CASB send an initial trip signal to all 8 DG breakers to shed all 4kv loads on the SD boards being powered by DGs.

Eg. Bds are on DGs because a LOSP has first occurred.

o. The Diesel Generator Output breaker logic includes Unit Priority Retrip Logic.

The purpose of Unit Priority Re-trip logic is to allow the preferred pumps on the Unit with the first accident signal to continue to run but readies the second accident Unit for its preferred pump starts. This is done by selectively tripping the appropriate DG output breakers to facilitate a 4kv load shed.

p. The priority retrip will retrip the DG breakers for the 4kv boards which will have pumps starting on them. Ie. For a Unit 3 initiated CAS signal, a Unit 1 or 2 LOCA will start all 8 of that Units ECCS pumps on boards A, B, C, & D. Those 4 DG output breakers trip in that case. For a Unit 1 or 2 initiated CAS signal, the subsequent initiation from Unit 2/1 will cause diesel breaker retrip for the preferred division for that Unit only (A/B for Unit 1, or C/D for Unit 2).

6. Leak Detection

Core Spray piping penetrates the drywell, reactor vessel and shroud. If a pipe break occurred between vessel wall and the shroud, Core Spray function would be lost. Pipe break detection system monitors the integrity of the Core Spray piping and alarms in Control Room.

Obj. ILT 2.d(OF-5)
Obj LOR 1d(OF-5)
Obj NLOR 5(OF-5)
Obj. NLO 7(OF-5)

a. Pressure 1 (P1) is greater than P5 due to the jet pump driving force.

b. P1 is greater than P2 due to the pressure drop across the core plate.

c. P2 is greater than P3 due to the pressure drop across the core. (This ΔP is small.)

d. P3 is greater than P4 by 7 psi due to the pressure drop across the steam separators.

e. P4 is greater than P5 by 7" of water due to the pressure drop across the steam dryer.

f. The low side of the detector senses above-core plate pressure (P2) plus the pressure due to the height of water in the vessel. Under normal conditions the high side of the detector senses core exit pressure (P3) plus pressure due to the height of water in the sensing leg. With the plant operating at rated conditions the detector reads +3.5 psid. P3 is slightly less than P2 due to the ΔP across the core. Therefore, the pressure differential detected is mainly due to the height of cold water (135°F) in the high leg of piping. If the Core Spray piping breaks between the reactor vessel and the shroud, piping is now sensing P5 instead of P3, and the high-side pressure at the detector would decrease by 7 psig. Sensed low-side pressure will remain the same. This would cause the ΔP to decrease, causing an alarm to sound at 2 psid decreasing (following a 15-sec time delay). During cold shutdown conditions this alarm will normally be in. This is due to low-side pressure being greater than high side pressure (negative ΔP).

Actual ΔP instruments are located on Elev. 565 of Reactor Building, only alarms in control room. Instrument readings satisfy TRM requirements.

Based on Tech. Support calculations, alarm will clear at 300-330 psig, depending on drywell temp. Loop 2 typically reads higher than Loop 1. (Ref. R)

D. Relationships With Other Systems

1. Combines with other ECCS to provide adequate core cooling over the entire break spectrum.

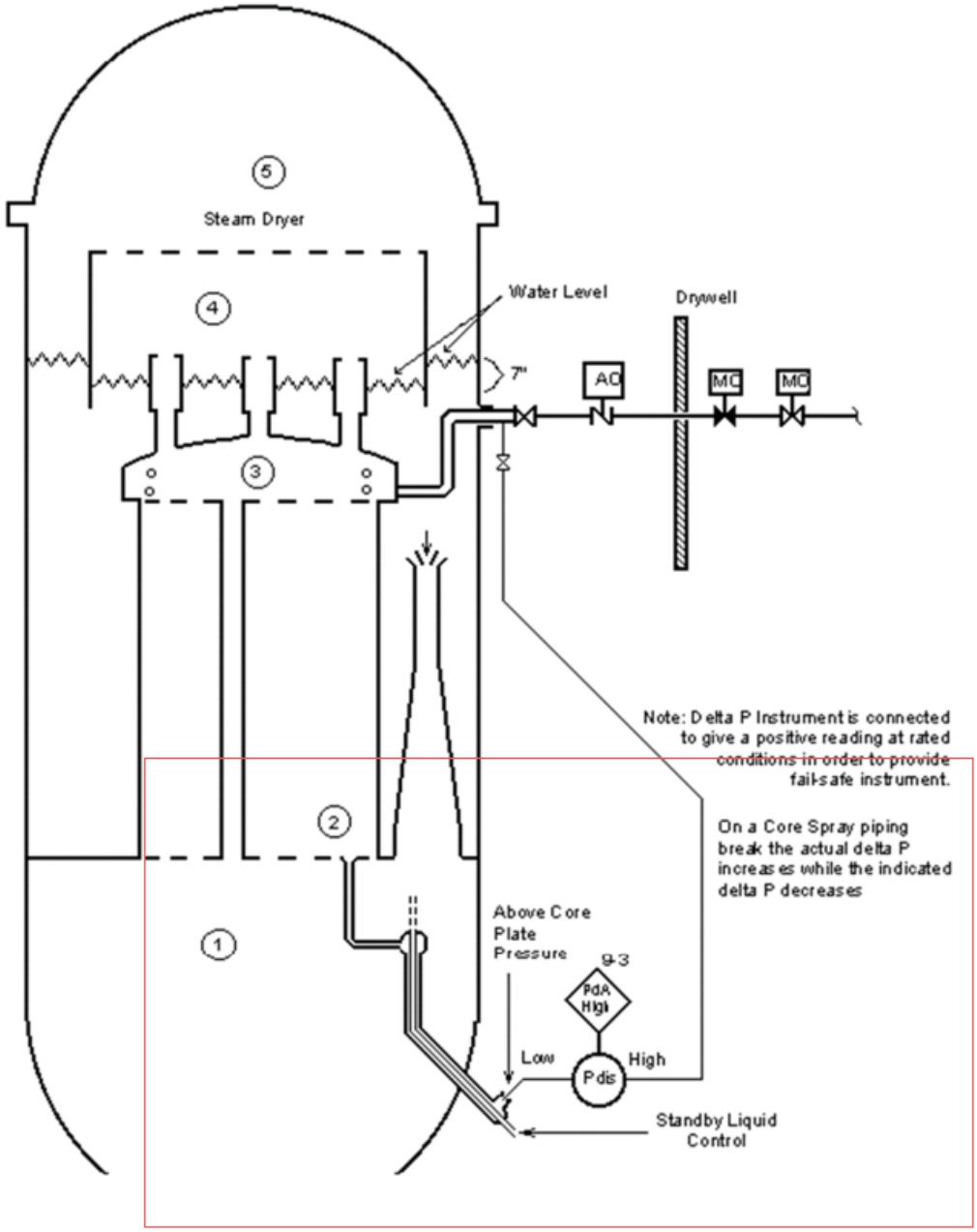
Obj. ILT 4.b,g

Obj. LOR 3.b,g

Obj. NLOR 7

OPL171.045, Core Spray System, Rev.23

CORE SPRAY PIPE BREAK DETECTION INSTRUMENTATION



Excerpts from TRM Bases 3.3.3:

TR 3.3 INSTRUMENTATION

TR 3.3.3.3 Core Spray Sparger Break Detection

BASES

BACKGROUND

Two independent loops are provided as a part of the Core Spray System. Each loop consists of two 50 percent-capacity centrifugal pumps driven by electric motors, a spray sparger in the reactor vessel above the core, piping and valves to convey water from the suppression pool to the sparger, and the associated controls and instrumentation.

The two 100-percent capacity core spray lines separately enter the reactor vessel through the two core spray nozzles that are 120 degrees apart. Each internal pipe then divides into a semicircular header with a downcomer at each end, which turns through the shroud near the top. A semicircular sparger is attached to each of the four outlets to make two practically complete circles, one above the other. Short elbow nozzles are spaced around the spargers to spray the water radially onto the tops of the fuel assemblies.

A detection system is also provided to continuously confirm the integrity of the core spray piping between the inside of the reactor vessel and the core shroud. A differential pressure switch measures the pressure difference between the bottom of the core and the inside of the core spray sparger pipe just outside the reactor vessel. If the core spray sparger piping is sound, this pressure difference will be the pressure drop across the core. If integrity is lost, this pressure drop will include the core pressure drop and the steam separator pressure drop. An increase in the normal pressure drop (decrease in the indicated differential pressure to the setpoint) initiates an alarm in the control room.

It should be noted that the instrument is in alarm during cold moderater conditions due to the instrument leg variations in densities. The alarm should clear during the increase to rated temperature and pressure. During operation at rated temperature and pressure, the alarm should remain clear.

BASES

APPLICABLE SAFETY ANALYSIS	The safety function of the Core Spray system is to maintain the fuel clad temperature to ≤2200°F during times the other ECCS systems may be incapable of maintaining vessel water inventory above the top of active fuel. The method of cooling requires that the water spray directly on top of the fuel assemblies rather than trying to maintain water above TAF. If the Core Spray sparger is broken, this will not be accomplished and will not meet its safety design function. The Core Spray sparger break detection instrumentation provides continuous monitoring of the integrity of the sparger.
LCO 3.3.3.3	The Core Spray sparger break detection alarm, within the allowable value 2 psid ±0.4 at rated temperature and pressure, provides indication of a potential break in the associated subsystem's sparger. A broken sparger will cause the subsystem to be inoperable.
APPLICABILITY	The OPERABILITY requirement is consistent with Technical Specification requirements for the times when the affected subsystem is required to be OPERABLE and the instrument is indicating meaningful readings.
ACTIONS	A.1 When a required channel is inoperable, the channel must be repaired and restored to OPERABLE status within 24 hours. B.1 When the Required Action and associated Completed Time of Condition A is not met, the subsystem must be declared inoperable immediately.

Excerpt from TS 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.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

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

Examination Outline Cross-reference:

Equipment Control

G2.2.18 (10CFR 55.43.5 – SRO Only)

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	------	3.9

Proposed Question: **# 88**

In accordance with NPG-SPP-07.2.11, Shutdown Risk Management, which **ONE** of the following completes the statements below?

Concurrence from the **(1)** is required for the release and closure of outage and system work windows that have an impact on the Shutdown Safety Functions.

RED **(2)** the outage execution risk condition that is coded to represent a **VERY HIGH** level of risk for evaluation.

A. (1) Shift Manager
(2) is

B. (1) Shift Manager
(2) is NOT

C. (1) Outage Manager
(2) is

D. (1) Outage Manager
(2) is NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** *(See attached)* In accordance with NPG-SPP-07.2.11, Shutdown Risk Management, Section 3.1, Roles and Responsibilities, specifies that the Shift Manager is responsible to concur with the release and closure of outage and system work windows that have an impact on the shutdown safety functions.

Second part, in accordance with NPG-SPP-07.2.11, 3.2.1.B.1, a very high level of risk for outage periods is RED.

B INCORRECT: First part is correct *(See A)*. Second part is incorrect but plausible in that NPG-SPP-07.2.11 specifies that risk thresholds are classified as green, yellow, orange, and red for acceptable, but increased level, high, and very high, respectively. In addition to colors associated with shutdown risk management, other colors are specified for equipment aligned for shutdown cooling, protected equipment for each unit, disabled annunciators, etc. With the multitude of colors and significance for each, a candidate may incorrectly conclude the color associated with very high risk.

C INCORRECT: First part is incorrect but plausible in that NPG-SPP-07.2.11, Section 3.1, Roles and Responsibilities, specifies that the Outage Manager is responsible for schedule development, control and review to ensure the schedule contains appropriate risk attributes. Also responsible for the fundamental functions of scheduling, coordinating, monitoring, and ensuring effective oversight of outage activities. Given these responsibilities and the nature of outage planning and cross-functional execution, a candidate may reasonably determine Outage Manager concurrence is required for the release and closure of outage and system work windows that have an impact on the Shutdown Safety Functions. Second part is correct (*See A*).

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

SRO Level Justification: Tests the candidate's knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments. SRO only because of link to 10CFR55.43(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. SRO specifically because of SRO involvement in development and execution of shutdown safety contingency plans. This question is rated as memory due to strictly recalling facts related to shutdown risk management.

Technical Reference(s): NPG-SPP-07.2.11, Rev. 16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.0239, Obj. 3 (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 NPG-SPP-07.2.11: Also supports Distractors B2, C1, D1, D2

3.1 Roles and Responsibilities (continued)

5. Are actively involved to challenge and oversee the execution of outage preparations, defense in depth reviews and key work activities with elevated risk. [R.3]

6. Pay particular attention to the actions needed to met key outage milestones related to shutdown safety, such as the production of a detailed schedule, the performance of and independent safety review, and the development and approval of contingency plans.

7. Are involved with outage scheduling logic changes to ensure the risk implications of changes that affect the shutdown safety functions are understood. [R.4]

B. Work Control Manager - In addition to the Station Senior Manager Responsibilities, the Work Control Manager is responsible for setting and reinforcing high standards and clear expectations for planning, reviewing, and executing outages.

C. Operations Manager - In addition to the Station Senior Manager Responsibilities, the Operations Manager is responsible for ensuring operations personnel are involved in all aspects for preparing and controlling outage activities and risk is properly evaluated and controlled.

D. Outage Manager - In addition to the Station Senior Manager Responsibilities, the Outage Manager is Responsible for schedule development, control and review to ensure the schedule contains appropriate risk attributes. Also responsible for the fundamental functions of scheduling, coordinating, monitoring, and ensuring effective oversight of outage activities.

E. Manager, Probabilistic Risk Assessment Applications - Owner of the shutdown risk management software. Ensures outage risk management software is maintained in accordance with NPG-SPP-12.7 Computer Software Control. Additionally Shutdown (EOOS) is maintained in accordance with NPG-SPP-09.11.3 Shutdown Equipment Out of Service Management.

F. Shift Manager - Maintains overall responsibility for control of the key shutdown safety functions. For example, activities with the potential to challenge decay heat removal, lower reactor coolant system inventory, result in a loss of electrical power, or affect reactivity such as fuel or control rod movement are overseen by the shift manager. In addition, the shift manager concurs with the release and closure of outage and system work windows that have an impact on the shutdown safety functions. [R.6]

G. Manager Fleet Outage Performance- Periodically evaluates shutdown risk management strategies through observations, benchmarking and self-assessments. Communicates new areas of risk identified by outside industry groups, regulators, or vendors to appropriate site personnel.

3.2 Instructions

3.2.1 Identify and Assess the potential risks.

A. Identify Risk

Methods are implemented to identify conditions and activities that individually and collectively have the potential to result in adverse outcomes.

1. Personnel at all levels of the organization should be engaged in identifying risk conditions and activities, and in elevating the responses to these conditions.

2. Shutdown safety is an integral part of the Outage planning process. Shutdown safety is maintained and monitored by compliance with work in accordance with the outage schedule/plan. Assessments are performed prior to and during outage execution to maintain defense - in - depth. The outage schedule is independently reviewed to assure defense in depth is maintained specific guidance for this review is listed in section 3.3 and in Attachment 1.

3. Processes such as the corrective action program/issues resolution or work management are used to screen activities for potential risk.

4. For PWRs, shutdown assessment and management methodology is typically used for Hot Shutdown (Mode 3 and Mode 4), Cold Shutdown (Mode 5), Refueling (Mode 6), and Defueled (no mode) conditions for refueling outages.

5. For BWRs, shutdown assessment and management methodology is typically used for Hot Shutdown (Mode 3), Cold Shutdown (Mode 4), Refueling (Mode 5), and Defueled (no mode) conditions for refueling outages.

B. Assess the Risk

Analyze, quantify and qualify the likelihood of adverse outcomes.

1. Risk thresholds during outage periods are defined as GREEN, YELLOW, ORANGE, and RED.

   Entry into risk thresholds definition and management approval requirements are as follows:

   a. **Green** - Represents an acceptable level of risk and requires no risk management actions.

   b. **Yellow** - Represents an acceptable but increased level of risk. Establish risk management actions per station procedures.

   c. **Orange** - Represents a high level of risk. Entry into planned or unplanned orange conditions are undesirable. Written guidance/contingency plans should be made before entering preplanned conditions of this type. The Plant Manager must authorize this evolution.

3.2.1 Identify and Assess the potential risks. (continued)

d. **Red** - Represents a very high level of risk. A planned entry into a red condition is not allowed. If a red condition is inadvertently entered, immediate actions should be taken to return the plant to higher DID conditions (i.e., Orange, Yellow, or Green). Previously written contingency plans will be utilized when attempting to restore from a Red Condition. If conditions change, or equipment failures result in red condition, prompt reversal of the evolution in progress, restoration of equipment or implementation of a contingency plan should occur.

2. Shutdown Equipment Out of Service Management (EOOS) is a computer program which qualitatively performs risk assessment. This software takes the status (examples available, unavailable, and protected) of key plant equipment and then produces an output of the relative level of safety/defense in depth of key shutdown areas (i.e., reactivity control, shutdown cooling, AC power (onsite, offsite), fuel pool cooling, inventory control, and support equipment). The models which are built to support this software include fault trees which use a building block technology to identify specific components utilized to build a system utilized in maintaining a key safety function.

3. Computer model can be utilized to determine optimum places to place specific work activities and to play "what-if" scenarios as necessary. It is primarily a tool for outage scheduling and management personnel to help plan and maintain Defense In Depth (DID) in an outage. It does not take the place of Tech Specs, procedures or the schedule in maintaining Defense-in-Depth.

4. During outage execution, a Defense-In-Depth (DID) Checklist must be filled out each shift by Operations and be provided to the Shift Outage Manager for review. Any delta(s) identified from the computer model will be reviewed by the Operations Duty manager and Outage Management to determine the correct Risk level call. This will be documented on the DID sheet and in the corrective action program/issues resolution as appropriate.

3.2.2 Minimize or Mitigate the Risk

Develop strategies, define roles and responsibilities and identify tools to prevent unnecessary risk or to mitigate residual risk.

A. Every attempt should be made to build outage plans and schedules with Key Safety Function and Unit Status of GREEN or YELLOW throughout the outage period. Planned entries into a Status of ORANGE should be minimized.

B. During the planning and execution phase of an outage, a level of equipment availability should be maintained which is greater than the minimum required supporting key safety functions, referenced in SOER 09-1, Shutdown Safety and SER 02-08, Outage Defense in Depth. The key outage safety functions are: Reactivity Control, Shutdown or Core Cooling, Inventory Makeup or Control, Electrical Power Sources (Onsite and Offsite Power), Containment Closure (Primary/Secondary Containment), Spent Fuel Pool Cooling and Fuel Pool Inventory Control.

3.4 Outage Execution

A. The Work Control Outage Group should maintain risk books and risk status boards that reflect the risk status of the outage unit based on the outage schedule and the Defense-In-Depth DID Checklist. The location of these risk books should be the Main Control Room for the outage unit and the Outage Control Center / Work Control Center. The risk status board should be located in the Outage Control Center. Additional Outage Risk Status Boards may be provided in locations to be determined by Outage Management.

B. Risk books should contain the current copy of the Safety Plan, and the latest time to boil projections, and the latest Defense In Depth (DID) sheets.

C. Risk sheets should be included in the daily outage review reports / packages as a means to communicate the current plant risk status to all departments. Each department should use the Risk Sheet as reference during pre-shift and mid-shift briefings. Risk sheets should contain current risk profile, KSF status, basis, and protected equipment. Elevated risk conditions should be emphasized. The risk sheet may be integrated into the Site STORM package.

D. Each 24 hour period, an assessment will be performed of the plant schedule using computer model. A computer model report should be generated from the updated schedule. This computer model report should be compared to information provided by Operations on the daily DID assessment, based on actual equipment availability. This comparison provides a redundant verification for determining the risk each day during the outage. The new computer model report should be compared to the original computer model report generated prior to the outage, and the Outage Safety Plan, to determine if new conditions exist. The Shift Outage Manager, with the OCC, should determine if the new conditions require further actions. If it is determined that existing contingency plans need to be modified, or new contingency plans developed, the Shift Outage Manager should oversee the development of the plan, and provide to Senior Management for approval. The Operations Manager and the Plant Manager are required to approve all contingency plan modifications. If the contingency plan modification assigns new actions to other departments, those department managers are required to approve as well.

E. The running train of decay heat removal (DHR) credited as supporting the key safety function is always protected. Work is seldom allowed on protected DHR components and equipment that directly support the decay heat removal system.

F. Shift Manager concurrence should be obtained for release and closure of outage system work windows that have impact on shutdown safety functions.

G. The Operations Outage Shift Manager should ensure proper distribution of the DID Checklist to the Main Control Room for the Outage Unit, The Outage Control Center, Operation Work Control and other locations deemed necessary.

Attachment 1
(Page 1 of 5)

Independent Safety Review Checklist

1.0 PURPOSE:

THE PURPOSE OF THIS ATTACHMENT IS TO ENSURE A SYSTEMATIC APPROACH TO SCHEDULE REVIEWS THAT ARE CONDUCTED WITH CLEAR EXPECTATIONS TO MAXIMIZE DEFENSES FOR SHUTDOWN SAFETY.

2.0 PREREQUISITES:

The Rev. C schedule has been issued IAW NPG-SPP-07.2.10 Outage Milestones.

Identified Shutdown Safety Contingency Plans have been fully Developed IAW NPG-SPP-07.2 Outage Management.

3.0 ASSEMBLE AND PREPARE THE TEAM:

NOTE

This team is established IAW NPG-SPP-07.2.10 Milestone 31. NPG-SPP-07.2.11 Shutdown Risk Management contains the requirements for each position listed.

3.1 Team Roster

Position	Name
Team Lead	
Senior Reactor Operator	
Senior Reactor Operator	
Electrical Systems	
I&C Systems	
Mechanical Systems	
Reactor Engineering	
Integrated Scheduling	
Independent Safety Evaluation	
Technical Expertise	
PRA Expertise	
External Peer	
NFPA 805 (If Applicable)	

EXAMPLE

Examination Outline Cross-reference:
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff

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 #	------	2
Group #	------	1
K/A #	223002G2.1.6	
Importance Rating	------	4.8

Proposed Question: **# 89**

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 ODM-4.20 requires that the NUSO 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 High Drywell Pressure for Reactor Water Level Control (which may indicate that a leak has occurred), candidate may conclude that given the conditions above, resetting HPCI is correct.

B **CORRECT**: (*See attached*) In accordance with BFN-ODM-4.20, the NUSO will assign the Balance of Plant Operator (BOP) to maintain Reactor Water Level Control if the Reactor Feedwater System is unavailable. Given a Group (1) One Isolation has occurred, which causes the Main Steam Isolation Valves (MSIVs) to close, thereby removing steam from the Reactor Feedwater Pump turbines. 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 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. Given that a PCIS Group 1 Isolation has occurred, MSIVs are closed therefore RFW is also isolated. 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 associated with MSIV closure resulting in loss of Feedwater. 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)
2-OI-1, Rev. 57
2-OI-3, Rev. 173

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.065 Obj. 19 (As available)

Question Source: Bank # BFN NRC 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**

Comments:

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: Also supports Distractors A2, C2

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.

Excerpt from 2-OI-1: Also supports Distractors C1, D1

BFN Unit 2	Main Steam System	2-OI-1 Rev. 0057 Page 9 of 68

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 a 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 a 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 658229-012)

Excerpt from 2-OI-3:

BFN Unit 2	Reactor Feedwater System	2-OI-3 Rev. 0173 Page 309 of 313

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 a nd RCIC turbine trip
Level 7	+39	High water level alarm (REACTOR WATER LEVEL ABNORMAL annunciation 2-9-5A 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 2-9-5A 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)

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

Examination Outline Cross-reference:

264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG

G2.4.16 (10CFR 55.43.2 - SRO Only)

Knowledge of pre- and post-maintenance operability requirements

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	264000G2.2.1	
Importance Rating	------	4.1

Proposed Question: **# 90**

0-SR-3.8.1.1(A), Diesel Generator 'A' Monthly Operability Test, is in progress to support Return to Operability (RTO).

Which **ONE** of the following completes the statements below?

In accordance with NPG-SPP-06.3, Pre-/Post-Maintenance Testing, RTO tests are required to be approved by **(1)**.

Manipulation of 0-BKR-254-000A/06, DG 'A' Logic Relay Panel, **(2)** required to be logged in the Narrative Log.

A. (1) Senior Reactor Operator
(2) is

B. (1) Senior Reactor Operator
(2) is NOT

C. (1) Maintenance Manager
(2) is

D. (1) Maintenance Manager
(2) is NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) NPG-SPP-06.3, Pre-/Post-Maintenance Testing, Section 3.2.8, Qualifications and Training, specifies that Operations reviewers of WOs who approve RTO tests are qualified as SROs. Second part, both 0-OI-82, Standby Diesel Generator System, and 0-SR-3.8.1.1(A), Diesel Generator 'A' Monthly Operability Test, explicitly require that all manipulations of the Diesel Generator Logic Breaker be logged in the Narrative Log.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that operation of emergency diesel generators require manipulation of numerous breakers, control switches, and instrumentation selectors which do not require specific log entries.

C INCORRECT: First part is incorrect but plausible in that in accordance with NPG-SPP-06.3, the Maintenance Manager maintains responsibility for the adequacy of maintenance testing following maintenance activities. 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 post-maintenance operability process requirements and associated requirements of Diesel Generator operability administrative controls. SRO only because of the links 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 procedural facts in relation to Diesel Generator operability and associated requirements.

Technical Reference(s): NPG-SPP-06.3, Rev. 4 (Attach if not previously provided)
0-SR-3.8.1.1(A), Rev. 67
0-OI-82, Rev. 183

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _____ (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 NPG-SPP-06.3:

3.2.4 Review and Approval of PMT

NOTE
Previously approved PMTs may be used for subsequent maintenance activities within the same scope provided that the PMT is correct and complete for the subsequent activity and that traceability to the original approvals is documented.

A. Technical Reviewer of Work Orders

1. Review the maintenance tests.
2. Ensure that the maintenance tests address the scope of the maintenance performed.
3. Approve the maintenance tests through approval of the WO.
4. Review the RTO testing recommendations.

B. Independent Qualified Reviewer/10 CFR 50.59 and 10 CFR 72.48 Reviewer

1. The following PMT instructions receive safety reviews in addition to the PMT reviews in this section.
   a. A IPTEs instruction receives an Independent Qualified Review (IQR), a 10 CFR 50.59 and/or 10 CFR 72.48 review as required by NPG-SPP-01.2. A IPTEs instruction also receives Plant Operations Review Committee review.
   b. A step-by-step complex PMT instruction including non-stand-alone portions of approved plant instructions incorporated in a WO receives an IQR, a 10 CFR 50.59 and/or 10 CFR 72.48 review as required by MMDP-1. Sequences of activities such as leak checks, motor current readings or rotational checks, go-no-go checks, and vibration readings are not considered complex instructions.
   c. A formal instruction prepared for the WO PMT receives an IQR, 10 CFR 50.59 and/or 10 CFR 72.48 review in accordance with NPG-SPP-01.2, Administration of Site Technical Procedures.

3.2.4 Review and Approval of PMT (continued)

C. Operations

1. Perform initial reviews of WO in accordance with the criteria specified in NPG-SPP-07.1, On Line Work Management.

2. Review/revise RTO tests as necessary to ensure TS operability and surveillance requirements are met without imposing any adverse affects on the system equipment. Contact System Engineering if assistance is needed. Changes to RTO tests are made in accordance with 3.2.4C.8 below or by revision to the work order.

3. Approve the RTO tests.

4. Ensure that the PMT satisfies all TS requirements.

5. Include status control requirements in the WO package. Specify the correct systems and component alignment required for the PMT performance and the configuration required to restore systems and components to ensure correct operating or standby mode following the completion of the PMT.

6. Ensure that the PMTs are performed at the appropriate system operating conditions or plant modes.

7. Review WO scope changes and revise RTO tests as necessary in accordance with 3.2.4C.8 below or by revision to the work order.

8. Revise RTO tests when warranted. Revisions to RTO tests may be made by two SROs without routing back through the review cycle provided that 1) the reason is documented, 2) both SROs sign the revision and 3) IQR, 10 CFR 50.59 and/or 10 CFR 72.48 review was not required previously. This type revision requires Shift Manager approval if the PMT involves a generation risk critical component as classified in MEL.

9. Waive PMT when warranted. A PMT requirement may be waived provided that the Shift Manager authorizes the waiver, the reason is documented (e.g., plant conditions prevent functional testing of a logic circuit), and affected configuration changes are verified by independent verification.

D. Engineering

1. Review WO PMTs (Maintenance Testing and RTO Testing) when requested for TS, ASME Code and Augmented Programs (e.g., ISI, R&R,IST and AIST), 10 CFR 50 Appendix J, or other complex activities to ensure that they are correct and concur with the PMT.

Supports Distractors C1, D1:

3.0 PROCESS

3.1 Roles and Responsibilities

3.1.1 PMT Review and Approval

A. Maintenance Manager

1. Implement the PMT program, including ISCs for minor or toolpouch maintenance on Run-To-Maintenance (RTM) components.
2. Maintain responsibility for the adequacy of maintenance testing following maintenance activities.
3. Monitor the overall effectiveness of the PMT program and incorporate program changes based on plant history and performance, or industry experience.
4. Maintain and update maintenance test matrixes, if applicable at the site.

B. Site Engineering

1. Review WO PMTs (Maintenance Tests and RTO Tests) when requested for TS and/or ISFSI CoC, ASME Code and Augmented Programs (e.g. ,ISI, R&R, IST, AIST), 10 CFR 50 Appendix J, or other complex activities to ensure they are correct and concur with the PMT.
2. Assist, as required, in the preparation and performance of the recommended pre- or post-maintenance testing based on design basis, vendor recommendations, standards, codes, and engineering analyses.
3. Recommend PMT program changes based on plant or equipment modifications and in-house and industry experience.

C. Scheduling Group

1. Schedule PMTs which cannot be completed immediately following completion of maintenance activities or PMT milestones to ensure timely completion and/or performance during specific plant conditions.
2. Maintain a list of outstanding PMT WOs.

Excerpts from 0-SR-3.8.1.1(A): Also supports Distractors B2, D2

BFN Unit 0	Diesel Generator A Monthly Operability Test	0-SR-3.8.1.1(A) Rev. 0067 Page 26 of 98

Date: _____

NOTES

1) Section 6.2 and 6.3 may be performed together.
2) All manipulations of the Diesel Generator Logic Breaker are required to be logged in the Narrative Log.
3) During Diesel Generator rolling activities, the Diesel Generator can still be considered available as long as the local Diesel Generator Operator remains in continuous communication with the Control Room. In this case, if a valid start signal is received while performing the Engine Roll, the local Operator will expeditiously perform Steps 6.2[10] through 6.2[14], as directed by the Control Room Operator.

6.2 Manually Rolling Diesel Generator

[1] **NOTIFY** the Unit Operator that the A Diesel Generator Logic Breaker will be opened. (N/A if Breaker is in the OFF position) _____

[2] **POSITION** the following components as specified:

COMPONENT	POSITION	
• DSL GEN A LOGIC RELAY PANEL, 0-BKR-254-000A/06	OFF	_____ 1st _____ CV
• DG A START CIRCUIT 1 CONT POWER BKR, 0-BKR-082-000A35W1	OFF	_____ 1st _____ CV
• DG A START CIRCUIT 2 CONT POWER BKR, 0-BKR-082-000A35W2	OFF	_____ 1st _____ CV
• DG A LEFT BANK AIR HDR SHUTOFF VLV, 0-SHV-086-0539A (Wall beside D/G)	UNLOCKED & CLOSED	_____ 1st _____ CV
• DG A RIGHT BANK AIR HDR SHUTOFF VLV, 0-SHV-086-0540A (Wall beside D/G)	UNLOCKED & CLOSED	_____ 1st _____ CV

[3] **REMOVE** the Right Bank Ring Gear Access Cover by unbolting it from the DG Skid. _____

[4] **ATTACH** the Manual Turning Tool to the Engine Ring Gear. _____

BFN Unit 0	Diesel Generator A Monthly Operability Test	0-SR-3.8.1.1(A) Rev. 0067 Page 44 of 98

Date: _____

NOTES

1) For the first scheduled runs during the months of June and December or if performing this procedure concurrently with 0-SR-3.8.4.2 (DG A) or any fast starts, then Section 6.7 may be marked as N/A.
2) Generator turbo charger (NO LOAD) vibration readings are required in accordance with 0-TI-230V at 900 RPM as indicated on DIESEL GENERATOR A TACHOMETER, 0-SI-082-000A.
3) The Diesel Generator will be made inoperable as a result of taking 0-BKR-254-000A/06, DSL GEN C LOGIC RELAY PANEL (logic breaker); to the "OFF" position for the purpose of a manual slow start. The Diesel Generator is considered operable when 0-BKR-254-000A/06, DSL GEN C LOGIC RELAY PANEL (logic breaker), is returned to the "ON" position. REFER TO Tech Spec 3.8.1 and 3.8.2, Operation with Inoperable Equipment.

CAUTION

If the Diesel Generator receives an Automatic Start Signal during the performance of the manual slow start, the Control Room Operator should have the logic breaker placed in the ON position as soon as possible.

6.7 Manual Slow Start of Diesel Generator

[1] On A 4KV Shutdown Board, Compartment 23,

PLACE NFPA 805 MODE SWITCH, 0-43-211-000A/23, in the ENABLE position. _____

[2] **CHECK** the following Diesel Generator Room Exhaust Fans are not Running:

- DIESEL GENERATOR RM A EXHAUST FAN A _____
- DIESEL GENERATOR RM A EXHAUST FAN B _____

[3] On VDC DSL SYS BAT BOARD A, 0-BDGG-254-0000A,

PLACE the DSL GEN A LOGIC RELAY PANEL, 0-BKR-254-000A/06 in the OFF position. _____ 1st _____ CV

Excerpts from 0-OI-82: Also supports Distractors B2, D2

5.6 Manual Slow Start at the Diesel Engine Control Cabinet

NOTE

The Diesel Generator will be made inoperable as a result of taking 0-BKR-254-000A(B)(C)(D)/06, DSL GEN A(B)(C)(D) LOGIC RELAY PANEL, (logic breaker) in the "OFF" position for the purpose of a manual slow start. The Diesel Generator is considered operable when 0-BKR-254-000A(B)(C)(D)/06, DSL GEN A LOGIC RELAY PANEL, (logic breaker) is returned to the "ON" position. **REFER TO** Tech Spec 3.8.1 and 3.8.2, Operation with Inoperable Equipment.

CAUTIONS

1) If the DG is manually slow-started prior to the stop logic being reset, the DG will shutdown 11.5 minutes after the stop buttons are pressed.
2) Modifications under DCN 69532 include a change to the DG idle start logic that results in relay sequencing. This condition leaves the start relay energized after an idle start unless reset by manual action (Idle Start Release) or a fast start initiation. Following an Idle Start, if a DG shutdown is required, the Idle Start Release pushbutton must be depressed prior to depressing the two stop pushbuttons to prevent another engine start attempt after shutdown. [PER 609502]
3) If the Diesel Generator receives an Automatic Start Signal during the performance of the manual slow start, the Control Room Operator should have the logic breaker placed in the ON position as soon as possible.

[1] **ENSURE** the following initial conditions:

[1.1] All Precautions and Limitations in Section 3.0 have been reviewed.

[1.2] Diesel Generator A(B)(C)(D) is in Standby Readiness. **REFER TO** Section 4.0.

[1.3] **RECORD** Diesel Generator A(B)(C)(D) Electrical Control Cabinet ELAPSED TIME INDICATOR reading on Attachment 2.

[1.4] **DETERMINE** the elapsed time since the diesel was last operated or rolled.

5.6 Manual Slow Start at the Diesel Engine Control Cabinet (continued)

NOTE

All manipulations of the Diesel Generator Logic Breaker are required to be logged in the Narrative Log.

[1.5] **IF** Diesel Generator A(B)(C)(D), does not need to be rolled **OR** has been operated or rolled in the last 24 hours, **THEN**

OPEN 0-BKR-254-000A(B)(C)(D)/06, DSL GEN A LOGIC RELAY PANEL (LOGIC BREAKER), **AND DO NOT PERFORM** Step 5.6[2].

[2] **PERFORM** DIESEL GENERATOR ROLLING per Section 5.7.

NOTE

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] **CHECK** the presence of EECW cooling water flow to the diesel.

[4] **DEPRESS** push-button FUEL PRIMING PUMP, 0-HS-18-49A(B)(C)(D), until fuel pressure exceeds 20 psig as indicated on PRIMING FUEL PRESS, 0-PI-18-51A(B)(C)(D).

[5] **DEPRESS** the DG-A(B)(C)(D) IDLE START, 0-HS-082-000A(B)(C)(D)/8, **AND CHECK** the following:

A. Diesel engine starts, accelerates and stabilizes at an engine speed of 440 RPM to 460 RPM.

B. Lube oil pressure rises to greater than 20 psig as indicated on the following:

- MAIN BEARINGS LUBE OIL PRESS, 0-PI-82-28A(B)(C)(D).
- DG-A TURBOCHARGER COMP BEARING LUBE OIL PRESS INDR 0-PI-082-1000A(B)(C)(D).

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

Examination Outline Cross-reference:

202002 (SF1 RSCTL) Recirculation Flow Control

G2.2.1 (10CFR 55.43.5 - SRO Only)

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	202002G2.2.1	
Importance Rating	------	4.4

Proposed Question: **# 91**

Unit 2 is in MODE 1 in Single Loop Operation with '2B' Reactor Recirculation Pump in service at 855 rpm with the following conditions:

- All appropriate LCO Actions addressed for Single Loop Operation
- Preparations in progress to place '2A' Reactor Recirculation Pump in service in accordance with 2-OI-68, Reactor Recirculation System
- 2-SR-3.4.9.3&4, Reactor Recirculation Pump Start Limitations, completed **10** minutes ago with the following data:
    - Reactor Pressure is 980 psig
    - Reactor Coolant Temperature is 544 ºF
    - Bottom Head Drain Temperature is 410 ºF
    - Recirculation Loop 'A' Temperature indicates 470 ºF
    - Recirculation Loop 'B' Temperature indicates 522 ºF

Given the above conditions, which **ONE** of the following completes the statements below in accordance with 2-OI-68 and Tech Specs?

2-SR-3.4.9.3&4 is required to be completed within **___(1)___** prior to startup of the Reactor Recirculation Pump.

Startup requirements of '2A' Reactor Recirculation Pump **___(2)___** been met.

A. (1) 4 hours
(2) have

B. (1) 15 minutes
(2) have

C. (1) 4 hours
(2) have **NOT**

D. (1) 15 minutes
(2) have **NOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: (*See attached*) First part is incorrect but plausible in that multiple Tech Spec 3.4, Reactor Coolant System, completion times and frequencies are established as 4 hours. Second part is incorrect but plausible in that in accordance with TS 3.4.9, while in MODE 2 with both Reactor Recirculation Pumps not in operation, loop to loop temperature deltas must be less than or equal to 75 °F. Misapplication of this requirement as opposed to the correct requirement of loop to loop temperature delta limit of 50 °F would lead the candidate to incorrectly conclude that pump start requirements have been satisfied. Additionally, the recognition of temperature deltas not met is made more difficult by there being no references provided to the 'below the line' Tech Spec surveillance requirements.

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

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

D **CORRECT**: (*See attached*) Tech Spec Surveillance Requirements (SR) 3.4.9.3 and 4 require that the SRs be completed within 15 minutes before starting the idle Recirculation Pump to provide adequate assurance that differential temperature limits will not be exceeded between the time of the SR and the time of the idle pump start to preclude exceeding thermal stress design limits.

Second part, startup requirements for 2A Recirc Pump have NOT been met. Specifically, while the Bottom Head coolant temperature to RPV coolant temperature is within the required 145 °F delta-T limit (544 °F – 410 °F = 134 °F; Dome to Bottom Head <145°F), the RPV coolant temperature to Recirc loop to be started (Loop 2A) is NOT within the required 50 ºF delta-T limit (544 °F – 470 °F = 74°F; Dome to Idle Loop >50°F).

SRO Level Justification: Tests the candidate's knowledge of Reactor Recirculation Pump pre-startup Tech Spec surveillance requirements and the ability to assess plant conditions and determine whether those conditions permit placing equipment into service. 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): Unit 2 Tech Spec 3.4.4, Amend. 350 (Attach if not previously provided)
Unit 2 Tech Spec 3.4.6, Amend 286/253
Unit 2 Tech Spec 3.4.9, Amend. 253
Unit 2 Tech Spec Bases 3.4.9, Rev. 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.007 Obj. 22 (As available)

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

Question Source:	Bank #		
	Modified Bank #	OPL171.007-10 002 #440	(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**

Comments:

Copy of Bank Question:

440. OPL171.007-10 002

Given the following conditions:

- Unit 2 is operating in Mode 1
- '2A' Recirculation Pump tripped due to loss of cooling water pumps
- All appropriate LCO Actions have been addressed for Single Loop Operations
- Both cooling water pumps associated with '2A' Recirculation Pump VFD have been repaired
- 2B Recirculation Pump speed indicates 855 rpm
- Preparations are in progress to restart Recirculation Pump '2A' in accordance with 2-OI-68
- 2-SR-3.4.9.3&4, Reactor Recirculation Pump Start Limitations, started at time 1212
- The following data has was recorded to restart Recirculation Pump '2A' in accordance with 2-SR-3.4.9.3&4:
  - Reactor Pressure is 980 psig
  - RPV Coolant Temp is 544º F
  - Bottom Head Drain Temperature is 410º F
  - Recirculation Loop 'A' Temperature indicates 470º F
  - Recirculation Loop 'B' Temperature indicates 522º F

Which ONE of the following completes the statement below concerning the restart of Recirculation Pump '2A' and WHY?

Recirculation Pump '2A' start may _____.

A. NOT continue because the operating Recirculation Pump speed is excessive

B. NOT continue because the differential temperature between dome and the bottom head is excessive

C. NOT continue because the differential temperature between the operating loop and the idle loop is excessive

D. continue, all pre-start requirements have been satisfied in accordance with 2-SR-3.4.9.3&4

Excerpt from Tech Spec 3.4.9:

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.3	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ---------- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	----------NOTES---------- 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. In MODE 2 with both recirculation pumps not in operation, the difference may be ≤ 75°F. ---------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

Excerpt from Tech Spec Bases 3.4.9:

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note (Note 1 for SR 3.4.9.4) that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. The Note also states the SR is only required to be met during a recirculation pump startup, since this is when the stresses occur. Therefore, ΔT limits are not required. Note 2 of SR 3.4.9.4 allows the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature to be $\leq 75^{\circ}F$ when in MODE 2 with both recirculation pumps not in operation.

(continued)

Excerpts from Tech Spec 3.4.4: Supports Distractors A1, C1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; and

c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure boundary LEAKAGE exists.	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
B. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	B.1 Reduce LEAKAGE to within limits.	4 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Unidentified LEAKAGE increase not within limit.	C.1	Reduce LEAKAGE increase to within limits.	4 hours
	OR		
	C.2	Verify source of unidentified LEAKAGE increase is not service sensitive type 304 or type 316 austenitic stainless steel.	4 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

Excerpts from Tech Spec 3.4.6: Also supports Distractors A1, C1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ 3.2 µCi/gm.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > 3.2 µCi/gm and ≤ 26.0 µCi/gm DOSE EQUIVALENT I-131.	----------NOTE---------- LCO 3.0.4.c is applicable. ---------- A.1 Determine DOSE EQUIVALENT I-131. AND A.2 Restore DOSE EQUIVALENT I-131 to within limits.	Once per 4 hours 48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Reactor coolant specific activity > 26.0 µCi/gm DOSE EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131. AND B.2.1 Isolate all main steam lines. OR B.2.2.1 Be in MODE 3. AND B.2.2.2 Be in MODE 4.	Once per 4 hours 12 hours 12 hours 36 hours

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

Examination Outline Cross-reference:

241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating

A2.07 (10CFR 55.43.5 – SRO Only)

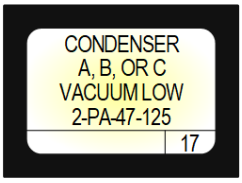
Ability to (a) predict the impacts of the following on the REACTOR/ TURBINE PRESSURE REGULATING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- *Loss of condenser vacuum*

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	241000A2.07	
Importance Rating	------	3.8

Proposed Question: **# 92**

A startup is in progress on Unit 2 with the following conditions:



- Reactor Power is 20%
- 2 CCW Pumps are in service
- CONDENSER A, B, OR C VACUUM LOW (2-9-7B, Window 17) alarms
- Vacuum is slowly degrading
- 2-AOI-47-3, Loss of Condenser Vacuum, entered
- **NO** operator actions have been taken

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

If vacuum continues to degrade, the Main Turbine will trip at **(1)** vacuum.

When the Main Turbine trips, the NUSO is required to direct the crew to continue **(2)**.

Note: 2-AOI-47-3, Loss of Condenser Vacuum

2-AOI-100-1, Reactor SCRAM

A. (1) 21.8 inches Hg
(2) 2-AOI-47-3 **ONLY**

B. (1) 21.8 inches Hg
(2) 2-AOI-47-3 **AND** perform 2-AOI-100-1

C. (1) 24.3 inches Hg
(2) 2-AOI-47-3 **ONLY**

D. (1) 24.3 inches Hg
(2) 2-AOI-47-3 **AND** perform 2-AOI-100-1

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) In accordance with 2-ARP-9-7B and 2-AOI-47-3, Loss of Condenser Vacuum, Main Turbine low vacuum trip logic is designed such that trip setpoints are established on a declining variable dependent upon Reactor Power, as well as requiring confirmatory inputs from additional condensers. The variable scale is such that trip setpoints vary from 21.8 inches Hg at 35 % Power to 21.3 inches Hg at 100% Power. Specific logic to result in a trip signal is as follows:

Condenser A voted signal greater than the turbine trip setpoint

-OR-

Condenser B voted signal greater than the turbine trip setpoint

-OR-

Condenser C voted signal greater than the turbine trip setpoint

-AND-

Any condenser at the variable alarm setpoint.

Candidate must recognize power level, the impact to the variable trip scale, and the difference between the alarm variable scale versus the trip variable scale. No reference is provided.

Second part, for the conditions provided with Reactor Power less than 26%, a SCRAM initiated from TCV Fast Closure (load reject) is auto-bypassed should a trip of the Main Turbine occur. With no SCRAM occurring, only entry into 2-AOI-47-3 is required.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible if candidate fails to recognize that, with Reactor Power at 20%, less than the 26% setpoint for TCV Fast Closure (load reject), the SCRAM is auto-bypassed should a trip of the Main Turbine occur. With no SCRAM occurring, entry into 2-AOI-100-1 is not required.

C INCORRECT: First part is incorrect but plausible if candidate improperly applies, from memory, the declining variable setpoint for Condenser low vacuum alarm rather than the declining variable setpoint for low Condenser vacuum Main Turbine trip. 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 loss of Condenser vacuum on Reactor/turbine pressure control, and based upon those impacts, use procedures to correct, control, or mitigate the consequences. Question is germane to the K/A in that a Main Turbine trip impacts and relies upon the Main Turbine Bypass System to control steam pressure when Reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. 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.

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

Technical Reference(s): 2 ARP 9 7B, Rev. 40 (Attach if not previously provided)
2-AOI-47-3, Rev. 22
2-AOI-100-1, Rev. 123
2-OI-99, Rev. 101
U2 Tech Spec Bases 3.7.5, Rev. 113

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.010, Obj. 22 (As available)

Question Source:

Bank #		
Modified Bank #	BFN NRC 1510 #83	(Note changes or attach parent)
New		

Question History: Last NRC Exam 2015

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:

QUESTION 83 Rev 0

Unit 1 is operating at 20% power with two Condenser Circulating Water Pumps in service, when the following occurs:

- CONDENSATE recorder, 1-XR-2-26 indicates (-) 24.8 inches of Hg vacuum and slowly degrading.

Note: 1-AOI-47-3, Loss of Condenser Vacuum
1-AOI-47-1, Unplanned Turbine Trip Below 30% Reactor Power
1-AOI-100-1, Reactor Scram

Which ONE of the following actions (if any) is/are required to be performed?

A. No action required at this time.

B. Enter 1-AOI-47-3 and TRIP the Main Turbine ONLY.

C. Enter 1-AOI-47-1 and VERIFY Main Turbine TRIPPED on low condenser vacuum.

D. Enter 1-AOI-47-3 and 1-AOI-100-1, SCRAM the Reactor, THEN TRIP the Main Turbine.

Answer: B

Excerpt from 2-ARP-9-7B: Also supports Distractors B2, C1, D1, D2

BFN Unit 2	Panel 9-7 2-XA-55-7B	2-ARP-9-7B Rev. 0040 Page 24 of 51

CONDENSER
A, B, OR C
VACUUM LOW
2-PA-47-125

17

Sensor/Trip Point:

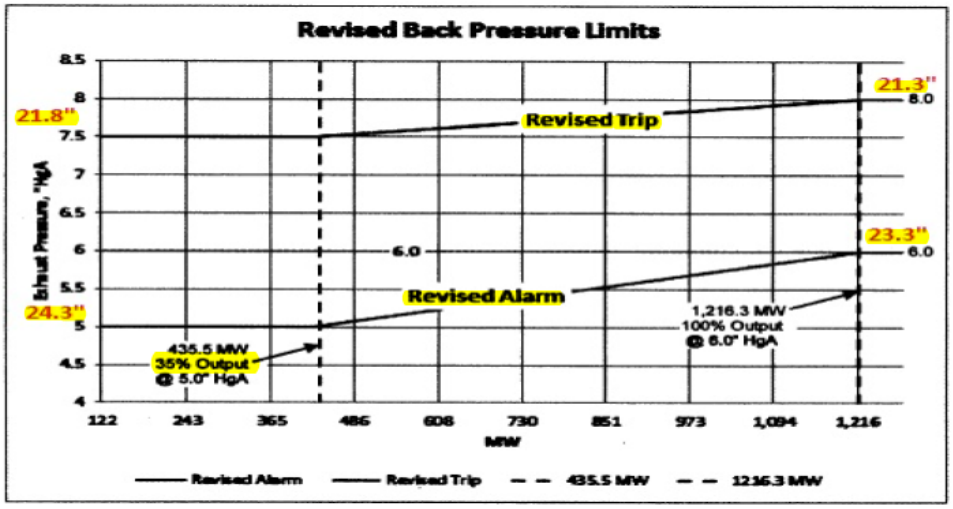
2-PT-47-72 A/B/C
2-PT-47-73 A/B/C
2-PT-47-74 A/B/C

24.3" Hg at 35% power
23.3" Hg at 100% power

EHC system

(Alarm reflashes upon receipt of other alarm inputs)

(Page 1 of 2)



Sensor Location: Under west skirt of turbine
Turbine Bldg, El 617

Probable Cause:

A. Malfunction of:
1. SJAE system.
2. CCW system.
3. Turbine sealing system.
4. Condenser vacuum breakers.
5. Sensor.
6. Cooling Tower Operation

Continued on Next Page

BFN Unit 2	Panel 9-7 2-XA-55-7B	2-ARP-9-7B Rev. 0040 Page 25 of 51

CONDENSER A, B, OR C VACUUM LOW 2-XA-47-125, Window 17
(Page 2 of 2)

Automatic Action:

Any of the following combinations will cause a turbine trip:

Condenser A, 2-PT-047-072A, 72B, 72C voted signal greater than the turbine trip setpoint

OR

Condenser B, 2-PT-047-073A, 73B, 73C voted signal greater than the turbine trip setpoint

OR

Condenser C, 2-PT-047-074A, 74B, 74C voted signal greater than the turbine trip setpoint

AND

- Any condenser at the alarm setpoint.

Operator Action:

A. **CHECK** alarm by checking vacuum lowering, MWe lowering, and exhaust hood temperature rise.
B. **IF** alarm is valid, **THEN**
REFER TO 2-AOI-47-3.

References:

2-45E620-10
2-47E610-47-3
0-45E602-6
2-45E602-16

Excerpts from 2-AOI-47-3: Also supports Distractors B2, C1, D1, D2

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a loss of condenser vacuum.

2.0 SYMPTOMS

A. Lowering in generator output.

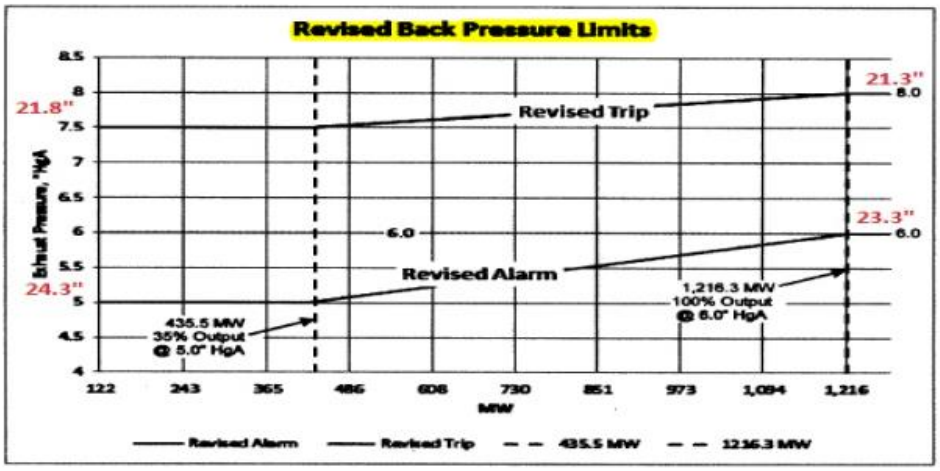
B. Annunciators in alarm could include but are NOT limited to the following:

1. CONDENSER A, B OR C VACUUM LOW 2-PA-47-125 (2-XA-55-7B, Window 17) vacuum lowering. (refer to "Revised Back Pressure Limits").
2. HOLDUP LINE INLET FLOW HIGH/LOW, 2-FA-66-111B (2-XA-55-53, Window 14) at 180 scfm.
3. OG HOLDUP LINE INLET FLOW LOW 2-FA-66-111A (2-XA-55-53, Window 4) at ≤ 8 scfm.

C. Lowering main condenser vacuum as indicated on CONDENSATE recorder, 2-XR-002-0026, Panel 9-6.

D. Rising or lowering off-gas flow as indicated on OG FLOW TO 6-HOUR HOLDUP VOLUME, 2-XR-66-103, Panel 9-53.

E. Rising or lowering off-gas flow as indicated on OG FLOW TO 6-HOUR HOLDUP VOLUME, 2-FR-66-20, Panel 9-8.



BFN Unit 2	Loss of Condenser Vacuum	2-AOI-47-3 Rev. 0022 Page 5 of 12

NOTES

1) Rising Off-Gas flow would indicate condenser inleakage if the Off-Gas System is functioning properly. Low Off-Gas flow in conjunction with low condenser vacuum could be indicative of an Off-Gas problem.

2) During operations with valid CONDENSER A, B, OR C VACUUM LOW 2-PA-47-125 alarm, and condensate temperature of 136°F or greater at the inlet of the SJAE (ICS point 2-28), reduced SJAE First Stage performance (stalling) may occur. This condition will cause reduced Off Gas flow and a loss of vacuum/turbine trip.[BFPER 02-016091-000]

3.0 AUTOMATIC ACTIONS

NOTE

To generate a turbine trip on low vacuum, one condenser section must be at the trip setpoint and another condenser must be at the alarm setpoint.. (PER 89506)

A. Any of the following will cause a turbine trip:

Condenser A, 2-PT-047-072A, 72B, 72C voted signal greater than the turbine trip setpoint

OR

Condenser B, 2-PT-047-073A, 73B, 73C voted signal greater than the turbine trip setpoint

OR

Condenser C, 2-PT-047-074A, 74B, 74C voted signal greater than the turbine trip setpoint

AND

Any condenser at the alarm setpoint.

B. RFP turbines trip and main turbine bypass valves closure occurs at -7" Hg hotwell pressure.

Excerpt from 2-AOI-100-1: Supports Distractors B2, D2

1.0 INTRODUCTION

1.1 Purpose

This instruction provides symptoms, automatic actions, and operator actions for a reactor scram.

2.0 SYMPTOMS

Any Reactor Scram

3.0 AUTOMATIC ACTIONS

A. Any withdrawn control rods fully insert.

B. Scram Discharge Volume Vent and Drain valves close.

C. Reactor Water Level Control swaps to Single Element (3 sec delay), when either RPS A or B Backup Scram channel activates.

D. Recirc Pump 75% Runback initiated.

E. Programmed Scram Response of RFPTs if initiated.

1. When Scram Response Logic initiates, the following actions occur:

   a. Red light at SCRAM RESPONSE INHIBIT/RESET switch, 2-HS-46-5 on Panel 2-9-5, illuminates.

   b. The logic polls RFP controls for availability of one RFP in AUTO. The polling sequence is RFP 2C, 2B, and 2A.

   c. Once one RFP in AUTO is found (normally RFP 2C), the polled RFPT will control reactor water level between its pressure compensated low speed clamp and an upper speed limit of 4100 rpm.

   d. The remaining pumps (normally RFP 2A and 2B) are transferred to MANUAL and RFPT speeds are set at 600 rpm. These pumps are still available for manual or automatic control to upper speed limit of 5800 rpm.

F. Hydrogen Water Chemistry System automatic shutdown on loss of feedwater flow.

G. Hydrogen Water Chemistry System automatic shutdown on Reactor Scram signal from the FWCS Panel 9-97.

Excerpt from 2-OI-99:

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0101 Page 101 of 112

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 U2 Tech Spec Bases 3.7.5:

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 21.3% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

(continued)

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

Examination Outline Cross-reference:

286000 (SF8 FPS) Fire Protection
A2.02 (10CFR 55.43.5 – SRO Only)
Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- DC electrical distribution failure

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	286000A2.02	
Importance Rating	------	2.8

Proposed Question: **# 93**

A and B Fire Pumps are currently tagged out when the following conditions occur:

- Control power to 4KV Shutdown Board 'C' is lost
- Fire Protection System Pressure is lowering

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

The Diesel Fire Pump **___(1)___** **IMMEDIATELY** automatically start.

In accordance with FPLCO 9.3.11.B, Fire Pumps and Water Distribution Mains, CONDITION 'B' **___(2)___** applicable.

REFERENCE PROVIDED

A. (1) will
(2) is NOT

B. (1) will
(2) is

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

D. (1) will NOT
(2) is

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that candidate may incorrectly determine that the Diesel Fire Pump (DFP) will IMMEDIATELY start upon an initiating signal when, in reality, the DFP has a 45 second time delay start upon receiving an initiating signal coincident with Fire Protection header pressure being less than 120 psig.
Second part is incorrect but plausible in that candidate may incorrectly determine that at least one electric Fire Pump is OPERABLE (C Fire Pump is also INOPERABLE because it is powered via the 'C' 4kV Shutdown Board. Control power to the 'C' 4kV Shutdown Board was provided as lost, thus the C Electric Fire Pump will not start and is non-functional.) Determining that C Electric Fire Pump is functional will lead the candidate to incorrectly conclude that FPLCO 9.3.11.B, CONDITION B, is applicable. Note that reference provided includes multiple redactions of pertinent FPLCO data.

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*) The Diesel Fire Pump (DFP) will NOT IMMEDIATELY start upon an initiating signal. The DFP has a 45 second time delayed start upon receiving an initiating signal coincident with Fire Protection header pressure being less than 120 psig.

Second part, the C Fire Pump is powered from the 'C' 4kV Shutdown Board, which has lost its DC sourced control power (Normal: Shutdown Battery C; Alternate: Battery Board 1). The loss of control power renders auto-start functions of the Fire Pump non-functional, requiring entry into FPLCO 9.3.11.B, CONDITION A. However, since ALL electric fire pumps are now non-functional, CONDITION B is also applicable.

SRO Level Justification: Tests the candidate's ability to predict the impacts of DC electrical distribution failure on the Fire Protection System and use procedures (FPLCOs) to mitigate the consequences. 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): 0-OI-26, Rev. 109 (Attach if not previously provided)
0-OI-57A, Rev. 177
FPLCO 9.3.11.B, Rev. 21

Proposed references to be provided to applicants during examination: NFPA 805, FPRM

Learning Objective: OPL171.049 Obj. 9d (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**

Reference Provided (redacted):

9.3.11.B **Fire Pumps and Water Distribution Mains**

FPLCO 9.3.11.B.1 The High-Pressure Fire Protection System shall be FUNCTIONAL at all times with [redacted] high-pressure fire pumps, [redacted] electric and [redacted] diesel, each with a capacity of 2,250 gpm and with their discharges aligned to the fire suppression header.

9.3.11.B.2 The High-Pressure Fire Protection System shall be FUNCTIONAL at all times with a FUNCTIONAL flow path capable of taking suction from Wheeler Reservoir and transferring the water through distribution piping with FUNCTIONAL sectionalizing control or isolation valves to the fire hydrant, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the system valve on each spray system required to be FUNCTIONAL per Sections 9.3.11.C, 9.3.11.E, and 9.3.11.F.

Note:
Separate Condition entry is allowed for each Function.

APPLICABILITY: ALL MODES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The automatic fire pump start function is NON-FUNCTIONAL	A.1.1 Ensure at least one fire pump is otherwise fully FUNCTIONAL and is available to manually start. AND A.1.2 Restore automatic fire pump start function to FUNCTIONAL status. OR A.1.3 Establish a Roving fire watch in the accessible areas of Fire Compartment 01-01, 01-05, 16-A, 16-B, 16-C, 16-D, 20-E, 21-E, and 25-1. OR A.2 Enter Condition D	[redacted] 30 days 30 days [redacted]

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Diesel fire pump is FUNCTIONAL AND No FUNCTIONAL electric fire pumps	B.1 Restore at least one electric fire pump to FUNCTIONAL status OR B.2.1 Provide an alternate backup pump or supply AND B.2.2 Restore at least one electric fire pump to FUNCTIONAL status OR B.2.3 Establish a Roving fire watch in the accessible areas of Fire Areas 3-3, 16, 21, 22, 23, and 24.	7 days 7 days 90 days 90 days
C. At least one electric fire pump is FUNCTIONAL AND the diesel fire pump NONFUNCTIONAL	C.1.1 Establish a Roving fire watch in the accessible areas of Fire Areas 1-3, 16, 25-1, and 26 (Turbine Building Elevations 565' and 586' between T1-T15, H-M lines only). AND C.1.2 Restore the diesel fire pump to FUNCTIONAL status OR C.2.1 Provide an alternate backup pump or supply AND C.2.2 Restore the diesel fire pump to FUNCTIONAL status OR C.2.3 Establish a Continuous fire watch in the accessible areas of Fire Areas 1-3, 16, 25-1, and 26 (Elevations 565 and 586 between T1-T15, H-M lines only).	24 hours 7 days 7 days 90 days 90 days

Excerpts from 0-OI-26:

5.1 Automatic Start of a Fire Pump (continued)

NOTE

Automatic start of Fire Pumps A, B, and C is locked out on the following (REFER to P&L 3.0T):

- Common Accident signal (CAS):
  High Drywell Pressure (2.45 psig) in conjunction with Low Reactor Pressure (450 psig) OR Low Low Low Reactor Water Level (-122 in.).

  AND
- Diesel Generators are supplying power to the U1/2 4160V Shutdown Boards (DGVA).

[2] The following Automatic Actions occur:

A. Control room annunciator(s) on Panel(s) 1(2,3)-9-8, or 1(2,3)-9-20, indicate the system(s) that actuated by one of the following:

- Panel 1(2,3)-9-20 - White light for activated valve indicates it received an open signal.
- Panel 1(2,3)-9-8, or 2(3)-9-20 Spray or Sprinkler annunciator indicates water pressure downstream of the valve.

CAUTION

Diesel Driven Fire Pump lube oil temperature should be maintained between 180°F and 225°F and water temperature between 165°F and 195°F during normal operation.

B. Selected Fire Pump starts.

C. Annunciator FIRE PUMP A(B)(C) RUNNING (0-XA-55-23A(B)(C), Window 41) in alarm.

D. Raw Service Water Pumps trip.

E. Raw Service Water Storage Tank Isolation Valve 0-FCV-25-32, and Raw Head Tank Isolation Valve 0-FCV-25-70, close. (Panel 1-9-20).

BFN Unit 0	High Pressure Fire Protection System	0-OI-26 Rev. 0109 Page 18 of 64

5.1 Automatic Start of a Fire Pump (continued)

NOTES

1) 15 seconds after the initiating signal, if system header pressure is less than 120 psig the second selected Fire Pump starts.
2) 30 seconds after the initiating signal if, system header pressure is less than 120 psig the third selected Fire Pump starts.
3) 45 seconds after the initiating signal if, system header pressure is less than 120 psig the Diesel Driven Fire Pump starts.

[3] **PERFORM** the following Operator Actions:

[3.1] **ENSURE** all appropriate automatic actions have occurred.

[3.2] **NOTIFY** the Intake AUO to perform Section 6.2 to flush Emergency Fire Pump Strainers.

[3.3] **LOG** in Shift Turnover Checklist for the AUO to CONTINUE flushing of Fire Pump Strainers once per shift as long as the Fire Pumps are in operation.

Excerpts from 0-OI-57A:

3.3 4kV Board Voltage and Amperage Limits

A. 4kV Shutdown bus amperage should never exceed 1200 amps due to bus rating.

B. 4kV Start Bus amperage should not exceed 3000 amps due to bus rating.

C. Print notes may require reducing auto starting loads under accident conditions by some kVA value. To determine the appropriate load reduction, use 1 hp = 1 kVA. Loads which are prevented from starting may be either 4kV load or a 480 V load which is powered from the 4kV Shutdown Board. Another method for determining kVA is to multiply board voltage, amps, square root of 3 and .001 (kVA =.001 X 1.732 X (V) X (I).) **REFER TO** Attachment 4, Board Restriction Verification Form for documentation of calculation verification.

D. (per 124528)Print notes may require reducing auto starting loads under accident conditions by some kVA value. Historically, disabling accident loads to meet the requirements of the "OPL" notes on 480V and 4160V boards is to "rack" the load breaker out of the affected board. The following are also considered acceptable methods of disabling accident (auto starting) loads:

1. Removal of the loads control power fuses.

2. Placing the loads backup control switch (43 switch) in the emergency position.

E. 4kV board voltage should remain within normal voltage limits of 3950 to 4400 volts. The degraded voltage logic will transfer the Shutdown Boards to the diesels at 3920V.

F. NRC/C] When both shutdown buses are being fed from the same USST, 0-GOI-300-1, Attachment 15.10, Abnormal Shutdown Bus Alignment Amperage Reading(s), is required to be completed every 4 hours. [NRC IR 92-15-01]

G. Whenever applicable plant Safety Related Boards are fed from the 161KV supply, their voltage must be monitored once per shift, Operations should take appropriate actions to maintain board voltage ≤ 508 volts; the Lead Electrical Engineer (Site Engineering) must be contacted if unable to maintain voltages below limits. [BFPER 03-011422-000]

Attachment 2
(Page 1 of 1)

4kV Control Power Supplies

BOARD	NORMAL CONTROL POWER SUPPLY	ALTERNATE CONTROL POWER SUPPLIES
4kV SHUTDOWN BOARD A	250V DC DISTRIBUTION PANEL SB-A	BATTERY BOARD 2
4kV SHUTDOWN BOARD B	250V DC DISTRIBUTION PANEL SB-B	BATTERY BOARD 2
4kV SHUTDOWN BOARD C	250V DC DISTRIBUTION PANEL SB-C	BATTERY BOARD 1
4kV SHUTDOWN BOARD D	250V DC DISTRIBUTION PANEL SB-D	BATTERY BOARD 3
4kV SHUTDOWN BOARD 3EA	BATTERY BOARD 1	BATTERY BOARD 2
4kV SHUTDOWN BOARD 3EB	250V DC DISTRIBUTION PANEL SB-3EB	BATTERY BOARD 3
4kV SHUTDOWN BOARD 3EC	BATTERY BOARD 3	BATTERY BOARD 1
4kV SHUTDOWN BOARD 3ED	BATTERY BOARD 2	BATTERY BOARD 3
COOLING TOWER BOARD A	250V DC DISTRIBUTION PANEL DCP-1	250V DC DISTRIBUTION PANEL DCP-2
COOLING TOWER BOARD B	250V DC DISTRIBUTION PANEL DCP-1	250V DC DISTRIBUTION PANEL DCP-2
COOLING TOWER BOARD C	250V DC DISTRIBUTION PANEL DCP-2	250V DC DISTRIBUTION PANEL DCP-1
COOLING TOWER BOARD D	250V DC DISTRIBUTION PANEL DCP-2	250V DC DISTRIBUTION PANEL DCP-1
COOLING TOWER BOARD E/F	CT 7 BATTERY BANK A	CT 7 BATTERY BANK B
RECIRC BOARD 1	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
RECIRC BOARD 2	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
RECIRC BOARD 3	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
UNIT BOARD 1A	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
UNIT BOARD 1B	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
UNIT BOARD 1C	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
UNIT BOARD 2A	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
UNIT BOARD 2B	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
UNIT BOARD 2C	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 3
UNIT BOARD 3A	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
UNIT BOARD 3B	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
UNIT BOARD 3C	250V TURBINE BLDG DIST BOARD 3	250V TURBINE BLDG DIST BOARD 1
COMMON BOARD A	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
COMMON BOARD B	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 1
BUS TIE BOARD	BATTERY BOARD 1	BATTERY BOARD 2
START BOARD 1	250V TURBINE BLDG DIST BOARD 1	250V TURBINE BLDG DIST BOARD 2
START BOARD 2	250V TURBINE BLDG DIST BOARD 2	250V TURBINE BLDG DIST BOARD 1

Excerpts from FPLCO 9.3.11.B:

9.3.11.B **Fire Pumps and Water Distribution Mains**

FPLCO 9.3.11.B.1 The High-Pressure Fire Protection System shall be FUNCTIONAL at all times with two high-pressure fire pumps, one electric and one diesel, each with a capacity of 2,250 gpm and with their discharges aligned to the fire suppression header.

9.3.11.B.2 The High-Pressure Fire Protection System shall be FUNCTIONAL at all times with a FUNCTIONAL flow path capable of taking suction from Wheeler Reservoir and transferring the water through distribution piping with FUNCTIONAL sectionalizing control or isolation valves to the fire hydrant, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the system valve on each spray system required to be FUNCTIONAL per Sections 9.3.11.C, 9.3.11.E, and 9.3.11.F.

Note:
Separate Condition entry is allowed for each Function.

APPLICABILITY: ALL MODES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The automatic fire pump start function is NON-FUNCTIONAL	A.1.1 Ensure at least one fire pump is otherwise fully FUNCTIONAL and is available to manually start. AND A.1.2 Restore automatic fire pump start function to FUNCTIONAL status. OR A.1.3 Establish a Roving fire watch in the accessible areas of Fire Compartment 01-01, 01-05, 16-A, 16-B, 16-C, 16-D, 20-E, 21-E, and 25-1. OR A.2 Enter Condition D	1 hour 30 days 30 days 1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Diesel fire pump is FUNCTIONAL AND No FUNCTIONAL electric fire pumps	B.1 Restore at least one electric fire pump to FUNCTIONAL status OR B.2.1 Provide an alternate backup pump or supply AND B.2.2 Restore at least one electric fire pump to FUNCTIONAL status OR B.2.3 Establish a Roving fire watch in the accessible areas of Fire Areas 3-3, 16, 21, 22, 23, and 24.	7 days 7 days 90 days 90 days
C. At least one electric fire pump is FUNCTIONAL AND the diesel fire pump NONFUNCTIONAL	C.1.1 Establish a Roving fire watch in the accessible areas of Fire Areas 1-3, 16, 25-1, and 26 (Turbine Building Elevations 565' and 586' between T1-T15, H-M lines only). AND C.1.2 Restore the diesel fire pump to FUNCTIONAL status OR C.2.1 Provide an alternate backup pump or supply AND C.2.2 Restore the diesel fire pump to FUNCTIONAL status OR C.2.3 Establish a Continuous fire watch in the accessible areas of Fire Areas 1-3, 16, 25-1, and 26 (Elevations 565 and 586 between T1-T15, H-M lines only).	24 hours 7 days 7 days 90 days 90 days

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

Examination Outline Cross-reference:	Level	RO	SRO
Conduct of Operations	Tier #	------	3
G2.1.4 (10CFR 55.43.2 - SRO Only)	Group #	------	------
Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no solo" operation, maintenance of active license status, 10 CFR Part 55	K/A #	G2.1.4	
	Importance Rating	------	3.8

Proposed Question: **# 94**

In accordance with OPDP-10, License Status Maintenance Reactivation and Proficiency for Non-Licensed Positions, which **ONE** of the following completes the statement below?

To **MAINTAIN** an active Senior Reactor Operator (SRO) license, an SRO shall actively perform a minimum of _____ per calendar quarter in a position credited for watch standing proficiency.

A. 40 hours **ONLY**

B. 40 hours **AND** a complete plant tour

C. five, 12-hour shifts **ONLY**

D. five, 12-hour shifts **AND** a complete plant tour

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: (*See attached*) Incorrect but plausible if candidate misapplies the OPDP-10, License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions, requirement for reactivating a license to work 40 hours performing licensed duties in tandem with an active licensed person.

B INCORRECT**:** (*See attached*) Incorrect but plausible if candidate misapplies the OPDP-10 requirement for reactivating a license to work 40 hours performing licensed duties in tandem with an active licensed person **AND** complete a complete tour of the plant of similar detail and thoroughness as AUO rounds, accompanied by an active licensed RO or SRO, as appropriate.

C **CORRECT**: (*See attached*) In accordance with OPDP-10, to maintain an active SRO license, the licensee shall actively perform the functions of an SRO for a minimum of five 12-hour shifts per calendar quarter. At least ONE complete watch (12-hour shift) per calendar quarter is required to be in a shift position credited for SRO only supervisory duties (Nuclear Unit Senior Operator or Shift Manager).

D INCORRECT:(*See attached*) Incorrect but plausible if candidate misapplies the OPDP-10 requirement to actively perform the functions of an SRO for a minimum of five 12-hour shifts per calendar quarter to maintain an active SRO license along with the requirement for reactivating a license to complete a complete tour of the plant of similar detail and thoroughness as AUO rounds, accompanied by an active licensed RO or SRO, as appropriate.

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

SRO Level Justification: Tests the candidate's knowledge of individual licensed operator responsibilities related to maintenance of active license status. SRO only because of 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 procedural facts in relation to license maintenance.

Technical Reference(s): OPDP-10, Rev. 13 (Attach if not previously provided)

Tech Spec 5.3, Amend. 322

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL 171.259 Obj. 2 (As available)

Question Source:	Bank #	BFN NRC 1510 #94	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2015	

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:

QUESTION 94 Rev 1

To maintain an active SRO license, an SRO must actively perform a minimum of _____ per calendar quarter in a position credited for watch-standing proficiency.

A. 40 hours only

B. a complete tour of the plant and 40 hours

C. 5-12 hour shifts only

D. a complete tour of the plant and 5-12 hour shifts

Answer: C

Excerpts from OPDP-10:

3.2.2 Nuclear Plant Requirements for Maintaining Active License Status (continued)

G. A Senior Reactor Operator shall actively perform the functions of a Senior Reactor Operator for a minimum of seven 8-hour or five 12-hour shifts per calendar quarter to remain active as a Senior Reactor Operator. A Senior Reactor Operator must stand at least ONE complete watch (8 or 12 hour shift) per calendar quarter in a shift position credited for SRO only supervisory duties (Unit Supervisor or Shift Manager). The remainder of the complete watches may be performed either in a credited SRO or RO position. An SRO may stand all of the required watches in credited SRO only supervisory positions and the RO portion of the license will be considered active.

H. Licensed personnel who do not meet these requirements are designated as inactive licensees. They may perform administrative duties of a licensed position, such as work control or CR Screening Committee, if they are current with training. If they are not current with training, they can perform no duties that require a license.

I. Active Licensed Maintenance for Quarterly On-Shift Time Documentation, Attachment 3, is submitted by active licensed individuals each quarter to the Operations Superintendent, or designee.

J. Non-Licensed STA Quarterly On-Shift Time Documentation, Attachment 4, is submitted by non-licensed STAs and inactive SRO licensed persons who desire to maintain proficient STA status each quarter to the Operations Superintendent.

K. The narrative logs are the legal record of watch stander assignment.

L. These forms can be submitted any time during the quarter after the required number of watches have been held. Due to the time it takes to update LMS, forms submitted at the end of one quarter may cause a license to show inactive at the beginning of the next quarter.

3.2.3 Medical Requirements

A. All licensed operators shall report any physical or psychological condition that might impair their ability to perform licensed duties, including any physical condition or corrective device that restricts the mobility of the individual or precludes the wearing of protective clothing and equipment to supervision and Health Services prior to assuming licensed duties.

B. All licensed personnel are responsible to ensure an updated NRC Form 396 is initiated and submitted to the NRC within 30 days of confirmation on a new or changed medical restriction.

C. This notification shall be made as soon as the individual has knowledge of such a situation.

1. Details of medical condition are not required to be shared with supervision, just the existence of a condition that could affect the ability to perform licensed duties.

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.

c. The 40 hours of shift functions should be completed within 30 days of completing the first shift function.

d. The 40 hours of shift functions should include at least one complete on-going shift turnover and one complete off-going shift turnover.

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:

Excerpt from Tech Spec 5.3:

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications referenced for comparable positions, as specified in TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A).

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

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

Examination Outline Cross-reference:

Conduct of Operations
G2.1.15 (10CFR 55.43.10 – SRO Only)
Knowledge administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.1.15	
Importance Rating	------	3.4

Proposed Question: **# 95**

In accordance with OPDP-1, Conduct of Operations, which **ONE** of the following completes the statements below?

Standing exemptions on the Peer Check Exemption List are approved by the **____(1)____** with **NO** allowed delegations.

In the event a plant management standing order or shift orders conflicts with plant instructions, the NUSO **____(2)____** required to direct the crew to follow the plant instructions.

A. (1) Operations Director
(2) is

B. (1) Operations Director
(2) is NOT

C. (1) Plant Manager
(2) is

D. (1) Plant Manager
(2) is NOT

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) Standing exemptions for peer checks are required to be identified on OPDP-1, Conduct of Operations, Attachment 5, Peer Check Exemption List, and approved by the Operations Director (no delegation). Second part is correct in that in the event of an order (Standing, Shift, Night, Special, etc.) conflicting with plant instructions, the crew is required to follow the instruction/procedure.

B INCORRECT: First part is part is correct (*See A*). Second part is incorrect but plausible in that multiple avenues exist to provide additional written guidance to operating crews. Examples include standing orders, shift orders, or the creation of Operational Decision-Making Issues and/or Adverse Condition Monitoring Plans when making decisions in response to degraded conditions that often fall below action thresholds defined in license documents or that are not clearly defined by existing procedures, or for emergent trip sensitivities that do not have existing processes or procedures in place. Given the nature of some of these documents, it could plausibly be concluded that additional documents such as standing orders or shift orders have precedence over plant instructions.

C First part is incorrect but plausible in that Plant Manager permission is required to startup with an outstanding outage-related Operator Workaround, Burden or Main Control Room Deficiency. Additionally, for any task where hazardous energy cannot be safely controlled by implementing requirements to safely control energy, the work is considered "working on energized equipment" and must be approved by the Plant Manager. 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 administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda, and the requirements given direction conflict with procedures. 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): OPDP-1, Rev. 58 (Attach if not previously provided)
OPDP-11, Rev. 15
NPG-SPP-10.2, Rev. 31

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _____ (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**

Excerpts from OPDP-1: Also supports Distractors C1, D1

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0058 Page 30 of 88

3.4.4 Peer Check and Exemptions

A. NPG-SPP-22.206, Verification Program, gives guidance on conducting peer checks.

B. Peer checking that interferes with oversight responsibilities will NOT be performed by the Shift Manager. NUSO peer checks should be limited in number and limited to short simple tasks located close to the horseshoe operating area, when another unit operator is unavailable.

C. The use of peer checks is governed by the following expectations:

1. Peer checks should be used for all equipment manipulations in the Control Room during steady state operations, startup, shutdown and cool down for which CV or IV is not required.

2. Peer checks are desired, but not required, for actions necessary to mitigate a transient such as Emergency or Abnormal Operating (or Procedure) immediate actions, actions to stabilize the plant during degrading conditions, or Emergency Operating Instruction or Abnormal Operating (or Procedure) directed actions.

3. Standing exemptions to the above guidance on requirements for peer checks shall be documented on Attachment 5, Peer Check Exemption List, and approved by the Operations Director (no delegation).

D. The station's list of approved peer check exemptions is maintained in the control room(s) or as a standing order.

3.4.5 Crew Composition

A. The Shift Manager maintains overall responsibility to ensure crew complement changes are evaluated for effective teamwork.

B. For Temporary, or Mixed Constituted MCR Crews, such as the result of vacation, sickness, or unexpected absence of crew members, the constituted crew shall perform the following:

1. The additional/different crewmember(s) discuss(es) their gaps from their individual operator excellence plan with the applicable Unit SRO, who then determines the appropriate mitigating strategy to employ prior to assuming control room duties. If the Unit SRO is the additional/different crewmember, this discussion will occur with the oncoming Shift Manager.

2. The Shift Manager ensures a crew brief, or shift briefing upon taking the watch, discusses the constituted crew for crew teamwork and expectations, including leadership and team effectiveness, operator fundamentals, and proficiency gaps that may exist.

3. Increase Management / Unit Oversight provided by the Shift Manager, or NUSO, Training personnel or Operations leadership as applicable.

3.11.1 General Information (continued)

B. Operator Workarounds, Burdens, Challenges and MCRDs will be documented as such in the associated work document that will correct the deficiency. The work document will be identified in the tracking method. Site Significant Focus Codes as described in NPG-SPP-07.1.4, Work Management Prioritization – On Line, will be used to identify the work document as a work around, burden, challenge or MCRD. Operator Workarounds and Burdens should be eliminated as soon as practical in accordance with site processes.

C. Operator Workarounds, Burdens, Challenges and MCRDs will be tracked by an appropriate method, including the associated work document and compensatory action being taken.

D. When deficiencies require equipment to be operated in an abnormal configuration, control in accordance with NPG-SPP-10.1, Status Control.

E. Operator Workarounds, Burdens, Challenges and MCRDs shall be evaluated to determine if an Operational Decision making Instruction (ODMI), or Adverse Condition Monitoring Plan (ACMP), is required.

F. Compensatory Actions that require plant manipulations that are outside of normal plant procedures shall be evaluated to determine if procedure revisions are needed and to evaluate 10CFR50.59 applicability.

G. Operator Workarounds, Burdens, Challenges and MCRDs shall be reviewed quarterly during the operating cycle by Operations Management for aggregate impact, considering each unit and site focus.

H. Operator Workarounds, Burdens, Challenges and MCRDs shall be periodically reviewed with plant management. Station effectiveness in minimizing the number of Operator Workarounds, Burdens and MCRDs shall be tracked through the use Key Performance Indicators or other measurements.

I. All outage-related Operator Workarounds, Burdens and MCRDs should be scheduled into the next refueling outage and considered for forced / planned maintenance outages. Plant Manager permission is required to startup with an outstanding outage-related Operator Workaround, Burden or MCRD.

J. Equipment which has been approved for abandonment by PHC shall not be included as an OWA, MCRD, or OB provided its use is not contained in any plant abnormal or emergency operating procedure. In order for this exclusion to be used, a CR log note is required documenting the date of PHC approval.

3.12 Inoperable Annunciators and Instruments

A. Instruments or annunciators that are suspected of being incorrect are promptly evaluated. If found to be in error, an alternate means shall be established to monitor the indicated/annunciated parameter.

3.12 Inoperable Annunciators and Instruments (continued)

B. Annunciators that are known to be false due to instrumentation problems, electrical problems, other problems, Main Control Room instruments that are known to be invalid or out of service are flagged and a condition report is initiated. Site Significant Focus Codes as described in NPG-SPP-07.1.4, will be used to identify the work document as an Annunciator Not Lit (AN), Annunciator Lit issue (AL), or a Disabled Alarm (DA).

C. The NUSO determines additional monitoring or compensatory measures needed until the condition is corrected with concurrence of the SM.

D. If a maintenance activity cannot promptly correct the condition that is causing an invalid alarm, evaluate having alarm cleared/disabled in accordance with OPDP-4, Annunciator Disablement.

3.13 Operations Communications Tools

Communication between plant management or staff and shift personnel can be accomplished by Standing Orders, Shift Orders, Night Orders, or Special Orders. They are informative in nature and used to disperse short-term, temporary, or immediate need-to-know information and instructions to operating shifts. They do not provide plant operating instructions or replace procedures. If an order conflicts with plant instructions, the crew is expected to follow the procedure.

3.14 Dedicated Operator Actions

A. Dedicated operator actions are designated by procedure or plant conditions and assigned to a specific individual. Dedicated actions are turned over until plant conditions no longer require a dedicated operator action.

B. Dedicated operator actions take priority over all other duties except performing immediate operator actions.

C. Any procedures or tools required shall be staged or in the operator's possession at all times.

D. Operators may be assigned collateral duties, as long as they do not prevent them from performing dedicated actions if needed.

1. Examples:

a. A problem with a reactor coolant pump results in Operations committing to trip the pump upon experiencing a reactor trip. The UO who is assigned as the dedicated operator shall first perform the immediate actions of the reactor trip procedure, then trip the pump. He cannot leave the control room without clearly turning over the dedicated action to another qualified operator.

b. An AUO is designated to vent RHR suction piping within ten minutes of a loss of RHR. The AUO may have rounds as a collateral duty but shall not enter contaminated areas (can't meet time requirement) or hold level watch during reactor coolant system drain down (conflicting dedicated actions).

Excerpt from NPG-SPP-10.2: Supports Distractors C1, D1

NPG Standard Programs and Processes	Clearance Procedure to Safely Control Energy	NPG-SPP-10.2 Rev. 0031 Page 7 of 100

1.0 PURPOSE

A. This procedure establishes standardized requirements for group tag out (hereafter referred to as clearance) to safely control hazardous energy. Implementing the requirements of this procedure demonstrates a level of safety equivalent to that obtained by the use of a lockout program. This procedure is used to ensure that before performing work on machines or equipment where the unexpected energizing, start up, or release of stored energy could occur and cause injury or property damage, the machine or equipment is isolated from its energy source and rendered non-operative. The term energy includes any system or external source of electrical, mechanical, hydraulic, pneumatic, chemical, thermal, motion, gravity, or other hazardous energy.

B. For any task where hazardous energy cannot be safely controlled by implementing the requirements of this procedure, the work is considered "working on energized equipment" and must be approved by the Plant Manager.

C. The requirements of this procedure may be used in part or waived as necessary during significant site or external events as directed by the Site Emergency Director. [C.7]

2.0 SCOPE

A. This procedure applies to work on machines and equipment under plant control where unexpected energizing, start up, or release of stored energy could occur and cause injury during servicing, maintenance, or modification activities. CAPR 99826-005 [C.4]

B. When plant status is other than "operating" or "recovery", such as deferred, cancelled, or decommissioned, TVA-TSP-18.613, Clearance Procedure to Safely Control Hazardous Energy Using Group Tagout, will apply unless the Plant Manager approves the use a different TVA approved lockout/tag out procedure. CAPR 99826-005 [C.4]

NOTE

Specific exceptions to 2.0C are described in Section 3.6. CAPR 99826-005 [C.4]

C. Servicing and/or maintenance that takes place during normal production operations is covered by this procedure if: CAPR 99826-005 [C.4]

1. An employee is required to remove or bypass a guard or other safety device; or

2. An employee is required to place any part of his or her body into an area on a piece of equipment where work is actually performed upon the material being processed (point of operation) or where an associated danger zone exists during a machine operating cycle.

D. When a contractor's clearance procedure is in use at a contractor construction site or operating facility on TVA property, TVA employees who are physically on the construction site or in the facility must receive the appropriate level of training, such as affected or authorized employee, on the contractor procedure. When TVA employees are on the site or in the facility, they must comply with the contractor procedure restrictions and prohibitions, such as danger tags and/or locks.

Excerpts from OPDP-11: Supports Distractors B2, D2

NPG Standard Department Procedure	Operational Decision-Making Issue Evaluation Process	OPDP-11 Rev. 0015 Page 7 of 46

1.0 PURPOSE

A. This Operations Departmental Procedure (OPDP) establishes requirements for the TVA Nuclear Power Group (NPG) process for evaluation of technical and operational decisions that affect plant operations, safety, reliability, and material condition. This OPDP establishes considerations to be included in the decision making process. This OPDP is based on guidance from WANO *"Principles for Effective Operational Decision -Making" and INPO "Performance Objectives and Criteria."*[C.5]

B. This OPDP establishes the process used by the plant management team when an Operational Decision Making Issue (ODMI) has been identified. An ODMI is produced for any condition that can adversely affect safe, reliable, or efficient operation which is most commonly in the form of degraded or non-conforming plant condition(s) (individual or aggregate) that could be manageable for awhile, but which might deteriorate further and force a unit into an undesired evolution, such as unexpected transient, or unplanned plant shutdown. ODMI decisions should be made with a bias towards elimination of risk. See Definitions section for more details.[C.5]

C. This OPDP establishes Adverse Condition Monitoring as the process for addressing significant plant conditions and parameters outside normal operating bands that have not yet reached plant operating procedure action or support Operational Decision Making Issue (ODMI) monitoring actions. This ensures an operational approach to adverse plant conditions from one Operations shift to another, including organizational alignment and support for issues not meeting threshold of ODMI.

D. The processes outlined in this procedure utilize cross-functional teamwork to develop and implement ODMIs and ACMPs. These processes support sustaining a high level of plant reliability through focusing on preventing and detecting equipment failures, establishing strong cross-functional collaboration/teamwork, ensuring organizational issues are addressed and using industry, vendor, and fleet operating experience.

2.0 SCOPE

A. Three typical scenarios illustrate operational and technical decisions in response to degrading plant conditions. The first scenario, basically an ODMI "Screening" involves the Shift Manager's leadership of the control room team in making immediate decisions in response to off normal conditions in accordance with plant operating procedures. The second scenario, involves the Station Management Team when making decisions in response to degraded conditions that often fall below action thresholds defined in license documents or that are not clearly defined by existing procedures. The third scenario, involves utility executives when making decisions that address long term protection of the public, the workforce and the station.

3.2.3 ODMI / ACMP Screening

A. This section provides guidance on review of a plant condition(s) that may warrant an ODMI or ACMP.

B. For any potential ODMI or ACMP, a Condition Report (CR) should be initiated to document the review / resolution of the condition. This CR is separate from the one used to repair the condition.

C. The following is a non-inclusive list of conditions that should be considered for an ODMI:

1. Unplanned entry into an LCO or actions requiring a plant shutdown in 72 hours or less with unclear resolution paths.
2. Unplanned power reductions of greater than 10% reactor power.
3. Issue(s) affecting "safety" during critical path activities during a plant shutdown condition.
4. Plant design or configuration issue(s) that may place the plant outside its design basis.
5. Selected NRC items such as Generic Letter and Bulletins.
6. Emergent trip sensitive activities that do not have existing processes or procedures in place.
7. An issue that is identified at power that could lead to a management decision to reduce power or come off line to correct.
8. Any issue, or aggregate of issues, that the Shift Manager feels the operating staff does not have sufficient monitoring ability, action steps, trigger points, or contingency guidance.

D. The following is a non-inclusive list of conditions that should be considered for an ACMP:

1. Elevated temperature on a system component above normal values that does not meet the criteria to remove it from service.
2. Elevated vibration on a pump that does not meet the criteria to remove it from service.
3. Abnormal chemistry parameters. [C.2]
4. Any issue in which one or more parameters are outside normal operating bands reducing plant operating margin.
5. Any issue requiring operator monitoring outside of normal means and/or frequency.

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

Examination Outline Cross-reference:

218000 (SF3 ADS) Automatic Depressurization

A2.02 (10CFR 55.43.5 – SRO Only)

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:

- Loss of coolant accident

Proposed Question: **# 96**

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	218000A2.02	
Importance Rating	------	4.1

Unit 2 was operating at 100% RTP, when a LOCA occurs resulting in the following:

- 2-EOI-1, RPV Control Modes 1-3, entered
- Reactor Water Level is (-) 125 inches, lowering
- Suppression Chamber Pressure is 25 psig, rising
- Core Spray and RHR Systems are **NOT** available
- Suppression Pool Level is 15 feet

Subsequently,

- Reactor Water Level is (-) 165 inches
- Available ECCS Systems are injecting
- NUSO determined Reactor Water Level **CAN** be restored and maintained above (-) 162 inches

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

The NUSO **(1)** required to direct inhibiting Automatic Depressurization System (ADS), and Emergency Depressurization (ED) is required in accordance with **(2)**.

Note: 2-EOI-1, RPV Control Modes 1-3
2-EOI-2, Primary Containment Control

A. (1) is
(2) 2-EOI-1

B. (1) is
(2) 2-EOI-2

C. (1) is NOT
(2) 2-EOI-1

D. (1) is NOT
(2) 2-EOI-2

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part is correct *(See B)*. Second part is incorrect but plausible in that Emergency Depressurization (ED) is appropriate to support RPV injection strategies ONLY if it is determined that Reactor Water Level (RWL) CANNOT otherwise be restored and maintained above the top of active fuel (TAF). Candidates may easily confuse this guidance with the requirement to ED BEFORE RWL degrades to (-)180 inches.

B **CORRECT**: (*See attached*) In accordance with EOI-1, RPV Control Modes 1-3, RC/L-6 requires inhibiting ADS if Reactor Water Level cannot be maintained greater than (-)122 inches. Defeating the ADS initiation logic ensures that Emergency Depressurization (ED) is performed under the direct control of the operating crew and only when determined to be appropriate considering actual plant conditions within the context of the EOIs.

Second part, given that Reactor Water Level is below TAF, (-)162 inches, but above (-) 180 inches, and it has been determined that Reactor Water Level CAN be restored and maintained above TAF, ED is NOT required in accordance with EOI-1. However, given the trajectory of Suppression Chamber Pressure of 25 psig, rising, and Suppression Pool Level is 15 feet, from EOI-2, PC/P-8, when Suppression Chamber Pressure CANNOT be maintained in the safe are of Curve 6, Pressure Suppression Pressure, ED is required. Action is required prior to Violating the PSP Curve 6.

C INCORRECT: First part is incorrect but plausible in that candidate may determine that allowing the system to operate automatically as designed is correct. 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 ability to predict LOCA impacts on ADS and use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations. 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 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): 2-EOI-1, Rev. 22 (Attach if not previously provided)
2-EOI-2, Rev. 19
EOIPM 0-V(C), Rev. 3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.202 Obj. 9 (As available)

Question Source:	Bank #	
	Modified Bank #	BFN NRC 1501 #89 (Note changes or attach parent)
	New	
Question History:	Last NRC Exam	2015

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	
	55.43	**X**

Copy of Bank Question:

QUESTION 89 ..... Rev 3

Unit 3 is operating at 100% power, when a LOCA occurs.

30 seconds after the LOCA the following conditions exist:

- RPV water level is (-) 125 inches and lowering
- Drywell Pressure is 15 psig and rising
- ADS BLOWDOWN AUX RELAYS ENERGIZED (3-9-3C, window 4) in alarm
- ADS BLOWDOWN TIMERS INITIATED (3-9-3C, window 11) in alarm
- ALL ECCS Systems are operating as expected

One minute later:

- ECCS systems are injecting and the Unit Supervisor has determined that Reactor water level can be restored and maintained above (-) 162 inches

Which ONE of the following is required concerning ADS and what is the required procedure for Reactor water level control?

A. Do not Inhibit ADS and execute C-1 Alternate Level Control

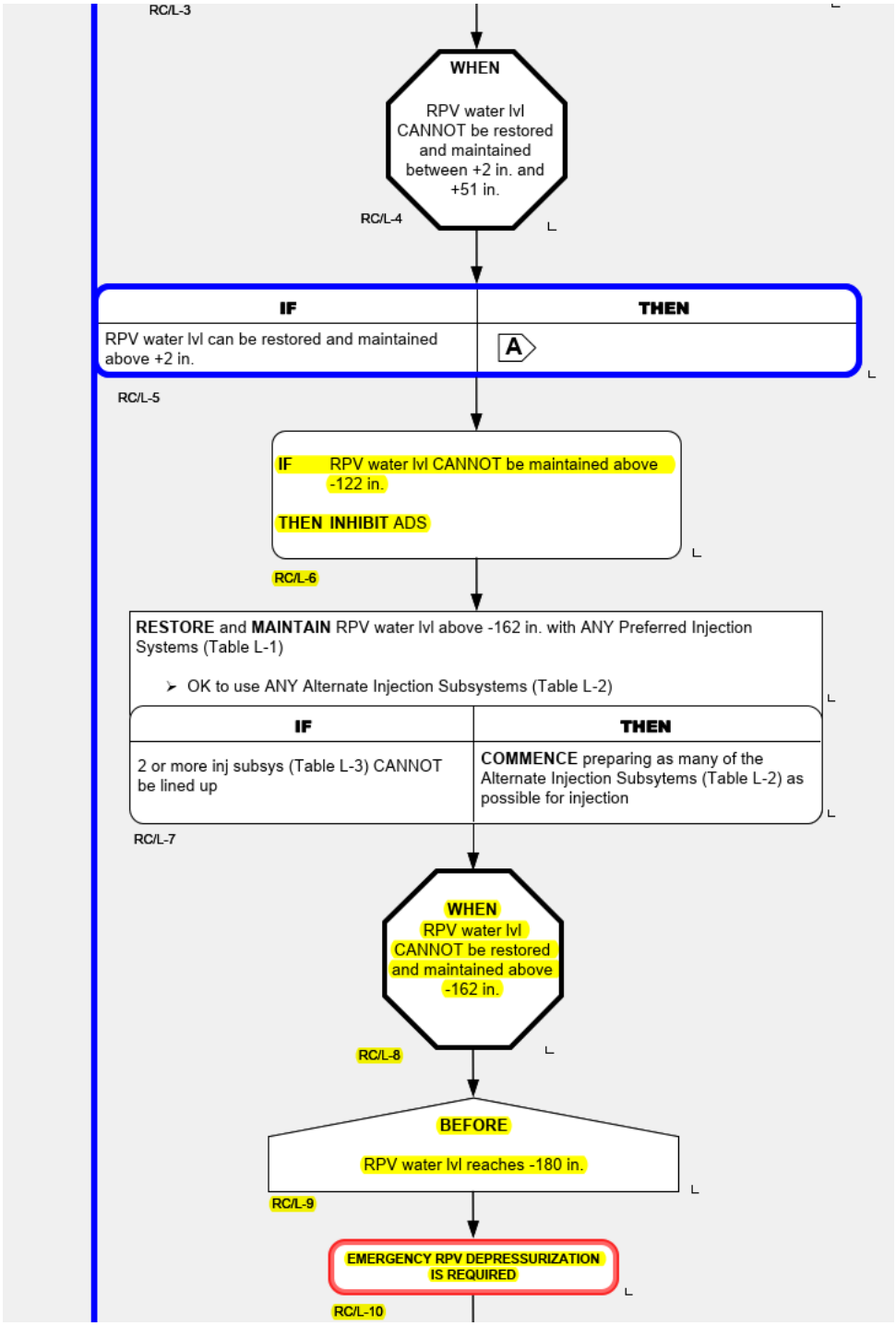
B. Do not Inhibit ADS and execute EOI-1 RC/L

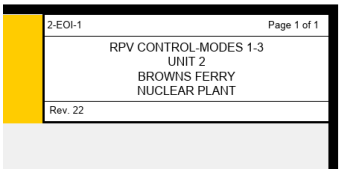
C. Inhibit ADS and execute C-1 Alternate Level Control

D. Inhibit ADS and execute EOI-1 RC/L

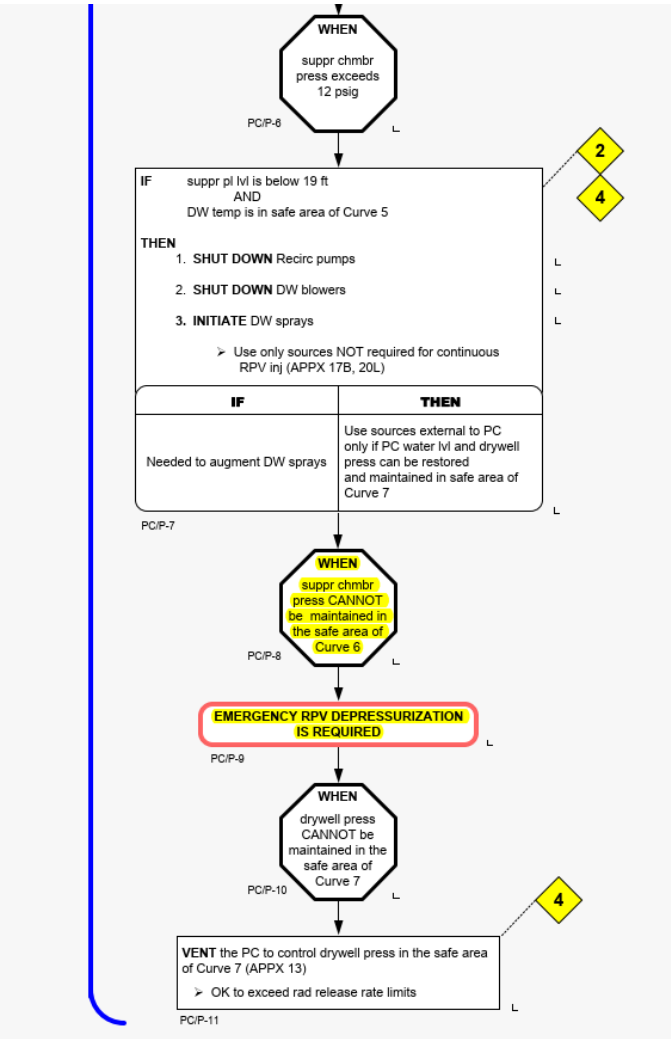
ANSWER: ... **D**

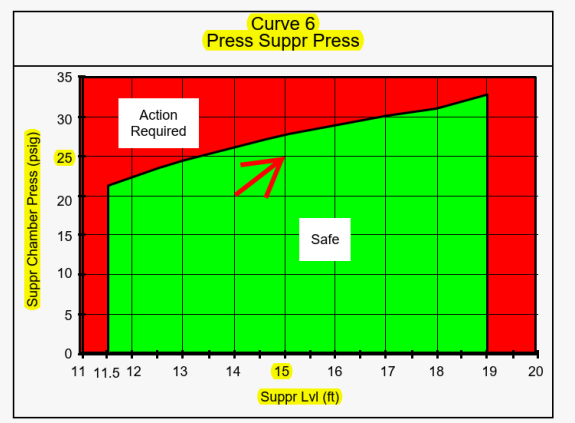
Excerpts from 2-EOI-1: Also supports Distractors A2, C2





Excerpts from EOI-2:





Excerpts from EOIPM 0-V(C): Also supports Distractors C1, D1

BFN Unit 0	EOI-1, RPV Control Modes 1-3 Bases	EOIPM Section 0-V(C) Rev. 0003 Page 34 of 81

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L-6

Permitting automatic ADS initiation may be undesirable for the following reasons:

- ADS actuation can impose a severe thermal transient on the RPV and may complicate efforts to control RPV water level.
- The conditions assumed in the design of the ADS actuation logic (e.g., no operator action for 10 minutes after event initiation) may not exist when the actions specified in this step are being performed.
- The operating crew can draw on much more information than is available to the ADS logic (e.g., equipment out of service for maintenance, operating experience with certain systems, probability of restoration of off-site power, etc.) and can better judge, based on instructions contained in the EOIs for Modes 1-3, when and how to depressurize the RPV.

Defeating the ADS initiation logic ensures that emergency depressurization is performed under the direct control of the operating crew and only when determined to be appropriate considering actual plant conditions within the context of the EOIs-Modes 1-3.

BFN Unit 0	EOI-1, RPV Control Modes 1-3 Bases	EOIPM Section 0-V(C) Rev. 0003 Page 41 of 81

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/L-8, -9, -10

Emergency RPV depressurization permits injection from low pressure systems, maximizes the total injection flow, and minimizes the flow through any primary system break.

Emergency depressurization is appropriate to support RPV injection strategies only if it is determined that RPV water level cannot otherwise be restored and maintained above the top of active fuel (TAF). Once this conclusion is reached, emergency depressurization may be initiated any time before RPV water level reaches the Minimum Steam Cooling RPV Water Level (MSCRWL). The "before" action level provides operational flexibility and requires a timing judgment. For example:

- If it is expected that available injection sources will reverse the level trend before RPV water level drops to the MSCRWL or that restoration of additional high pressure injection is imminent, the depressurization may be delayed.
- If it is believed that available injection sources are capable of restoring and maintaining RPV water level above the MSCRWL following RPV depressurization, the depressurization may be performed as soon as it is determined that RPV water level cannot be restored and maintained above the top of the active fuel.

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

Examination Outline Cross-reference:

Equipment Control

G2.2.45 (10CFR 55.43.2/5 – SRO Only)

Ability to determine or interpret technical specifications with action statements of greater than 1 hour (SRO Only)

Level	RO	SRO
Tier #	----	3
Group #	----	----
K/A #	G2.2.45	
Importance Rating	----	4.7

Proposed Question: **# 97**

Unit 2 is operating at 99% RTP when the following events occur:

- May 5 at 0300 hours, 2B SLC Pump is declared INOPERABLE due to a pump seal leak
- May 7 at 1100 hours, 2A SLC Pump is declared INOPERABLE due to a breaker fault
- May 7 at 1230 hours, 2B SLC Pump is declared OPERABLE
- Risk Informed Completion Time (RICT) is **NOT** available

Given the conditions above, which **ONE** of the following specifies the **LATEST** date and time at which 2A SLC Pump must be returned to service **WITHOUT** entering additional ACTIONS per Tech Specs?

[REFERENCE PROVIDED]

A. 0300 hours on May 12

B. 0300 hours on May 13

C. 1100 hours on May 14

D. 1100 hours on May 15

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Incorrect but plausible in that TS 3.1.7, Condition A, has a Completion Time of 7 days from time of discovery of one SLC subsystem INOPERABLE. Candidate may incorrectly apply this condition to the subsequent SLC subsystem INOPERABILITY.
(May 5 / 0300 + 7 days = May 12 / 0300)

B **CORRECT**: (*See attached*) From the given INOPERABLE times and correct application of TS 1.3, Completion Times, and TS 3.1.7, Standby Liquid Control System, the following should be concluded:
TS 3.1.7, Condition A (7 day), was entered on May 5 at 0300 for the 2B SLC Pump

TS 3.1.7, Condition A (7 day) and Condition B (8 hours) were entered on May 7 at 1100 hours for the 2A SLC Pump concurrent with the 2B SLC Pump

With the 2B SLC Pump returned to OPERABLE on May 7 at 1230 hours, TS 3.1.7, CONDITION A (7 day) for 2B SLC Pump and CONDITION B (8 hours), would be exited.

In accordance with Tech Spec 1.3, Completion Times, when a subsequent division, subsystem, component, or variable expressed in the CONDITION is discovered to be INOPERABLE or not within limits, the Completion Time(s) may be extended. To apply the Completion Time extension, two criteria must first be met. The subsequent INOPERABILITY (2A SLC Pump was for 1.5 hrs) must exist concurrent with the first INOPERABILITY (2B SLC Pump) and must remain INOPERABLE (2A SLC Pump) or not within limits after the first INOPERABILITY is resolved.

The total Completion Time allowed for completing a REQUIRED ACTION to address the subsequent INOPERABILITY shall be limited to the **more restrictive of either**:

a. The stated Completion Time, as measured from the initial entry into the CONDITION, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent INOPERABILITY.

In the scenario presented, the most restrictive extension, and therefore permitted, is the stated Completion Time, as measured from the initial entry into the CONDITION, plus an additional 24 hours:

7 days from May 5 / 0300 + 24 hours = May 13 / 0300

C INCORRECT**:** Incorrect but plausible if the candidate incorrectly applies TS 3.1.7, Condition A, full 7-day Completion Time to 2A SLC Pump INOPERABILITY concurrent with the 2B SLC Pump INOPERABILITY. (May 7 / 1100 + 7 days = May 14 / 1100)

D INCORRECT: Incorrect but plausible if the candidate incorrectly applies the 24-hour extension allowance (May 7 / 1100 + 7 days = May 14 / 1100 + 24 hours = May 15 / 1100)

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

SRO Level Justification: Tests the candidate's ability to determine/interpret Tech Specs with action statements of greater than one hour and application of 24-hour completion time extensions associated with concurrently inoperable equipment. SRO only because of the link to 10CFR55.43 (2): Facility operating limitations in the Technical Specifications and their Bases. With references provided, this question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to determine an outcome. This requires mentally using this knowledge and its meaning to determine the correct outcome.

Technical Reference(s): U2 TS 1.3, Amend. 349 (Attach if not previously provided)

U2 TS 3.1.7, Amend. 351

Proposed references to be provided to applicants during examination: U2 TS 3.1.7, Amend. 351

Learning Objective: OPL171.087 Obj. 14 (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

55.43 X

Comments:

Excerpts from Tech Spec 1.3:

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Unless otherwise specified, the Completion Time begins when a senior licensed operator on the operating shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Conditions is entered. The "otherwise specified" exceptions are varied, such as a Required Action Note or Surveillance Requirement Note that provides an alternative time to perform specific tasks, such as testing, without starting the Completion Time. While utilizing the Note, should a Condition be applicable for any reason not addressed by the Note, the Completion Time begins. Should the time allowance in the Note be exceeded, the Completion Time begins at that point. The exceptions may also be incorporated into the Completion Time. For example, LCO 3.8.1, "AC Sources – Operating," Required Action B.3, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

(continued)

1.3 Completion Times

DESCRIPTION (continued)

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.

Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition, unless otherwise specified.

However, when a subsequent division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

(continued)

Excerpt from TS 3.1.7:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

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

Examination Outline Cross-reference:

Radiation Control

G2.3.12 (10CFR 55.43.4 – SRO Only)

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.

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.3.12	
Importance Rating	------	3.7

Proposed Question: **# 98**

In accordance with 2-GOI-200-2A, Primary Containment Initial Entry, which **ONE** of the following completes the statements below?

(1) approval is required for all initial entries.

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
(2) Operations

B. (1) Shift Manager
(2) Radiation Protection

C. (1) Plant Manager
(2) Operations

D. (1) Plant Manager
(2) Radiation Protection

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT** (*See attached*) In accordance with 2-GOI-200-2A, Primary Containment Initial Entry, Shift Manager authorization is required for all initial entries. 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 Radiation 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 Plant Manager authorization is required for Drywell entry when the Reactor is at Power. 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. 11 (As available)

Question Source:	Bank #	BFN NRC 23-04 #98	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2023	

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

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

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

Copy of Bank Question: REPEAT 2304 #98

Examination Outline Cross-reference:

Radiation Control

G2.3.12 (10CFR 55.43.4 – SRO Only)

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.

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.3.12	
Importance Rating	------	3.7

Proposed Question: **# 98**

In accordance with 2-GOI-200-2A, Primary Containment Initial Entry, which **ONE** of the following completes the statements below?

___**(1)**___ approval is required for all initial entries.

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
(2) Operations

B. **(1)** Shift Manager
(2) Radiation Protection

C. **(1)** Plant Manager
(2) Operations

D. **(1)** Plant Manager
(2) Radiation Protection

Proposed Answer: **A**

Excerpts from 2-GOI-200-2A: Also supports Distractors

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
G2.4.41 (10CFR 55.43.5 – SRO Only)	Group #	------	------
Knowledge of the emergency action level thresholds and classifications (SRO Only)	K/A #	G2.4.41	
	Importance Rating	------	4.6

Proposed Question: **# 99**

At **1500**, an event occurred on Unit 2 involving fuel damage resulting in the following plant conditions:

- Stack Noble Gas (WRGERMS): 6.0 X 10^8 µCi/sec
- 2-RM-90-250, Reactor Building/Turbine Building Exhaust on Panel 2-9-2 indicates 3.3 X 10^5 µCi/sec

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

At **1515**, the **HIGHEST REQUIRED** Emergency Classification Action Level (EAL) is **___(1)___**.

Entry into 0-EOI-4, Radioactivity Release Control, **___(2)___** required.

[REFERENCE PROVIDED]

A. (1) UNUSUAL EVENT
(2) is

B. (1) UNUSUAL EVENT
(2) is NOT

C. (1) ALERT
(2) is

D. (1) ALERT
(2) is NOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is incorrect but plausible in that UNUSUAL EVENT would be selected if only the last bullet of the stem was taken into account. In accordance with EPIP-1, Attachment 1 (under Initiating Conditions (IC) RU1 (2), the given 2-RM-90-250 reading exceeds 2 times the Maximum Allowable Value for 60 minutes or longer. If the candidate misread the EPIP-1 chart, then they would confuse the release duration as exceeding 60 minutes (under RU1). However, the question is asking for the highest required EAL that exists based on 15 minutes.
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 does not accurately recall, from memory, 0-EOI-4, Radioactivity Release Control, entry conditions related to ALERT IC RA1.

C **CORRECT**: (*See Attached)* In accordance with EPIP-1, Attachment 1 (under Initiating Conditions (IC) RA1), a reading on (1) Stack Noble Gas (WRGERMS) greater than 4.02×10^8 µCi/sec for 15 minutes or longer meets the criteria for the highest REQUIRED EAL for an ALERT. As stated in stem, 15 minutes (under RA1) has elapsed since start of release. Second part, entry into 0-EOI-4 is required based on meeting the criteria for Gaseous offsite radioactivity release rate above that requiring an ALERT IC RA1 (Gaseous).

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 EOI entry conditions and immediate action steps related to High Offsite Radioactivity Release Rates regarding Emergency Action Level criteria. SRO only because of the link to 10CFR 55.43(b)(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): EPIP-1, Rev. 67 (Attach if not previously provided)
0-EOI-4, Rev. 9
0-ODCM-001, Rev. 26

Proposed references to be provided to applicants during examination: EPIP-1, Att 1, HOT INITIATING CONDTIONS-MODES 1-2-3

Learning Objective: OPL171.204 Obj. 10, 11 (As available)

Question Source: Bank # BFN NRC 1909 #88
Modified Bank # (Note changes or attach parent)
New

Question History: 2019

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:

Proposed Question: **# 88**

At **1500**, an event occurred on Unit 2 involving fuel damage.

The following plant conditions exist:

- Stack Noble Gas (WRGERMS): 6.0 X 10^{8} µCi/sec
- 2-RM-90-250, Reactor Building/Turbine Building Exhaust on Panel 1-9-2 indicates 3.3 X 10^{5} µCi/sec

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

At **1515**, the **HIGHEST REQUIRED** Emergency Classification Action Level (EAL) is **(1)**.

Entry into 0-EOI-4, Radioactivity Release Control, **(2)** required.

[REFERENCE PROVIDED]

A. **(1)** UNUSUAL EVENT
(2) is NOT

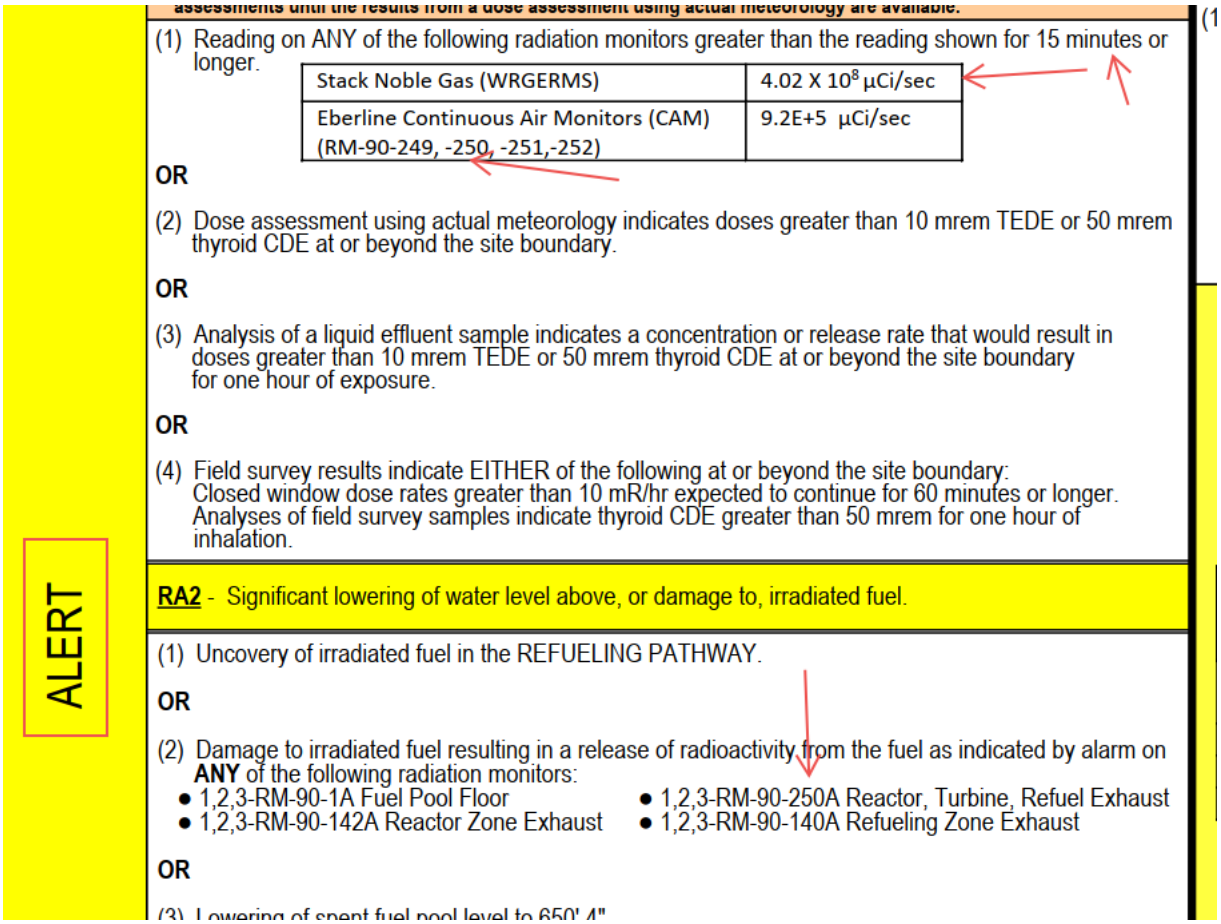
B. **(1)** UNUSUAL EVENT
(2) is

C. **(1)** ALERT
(2) is NOT

D. **(1)** ALERT
(2) is

Proposed Answer: **D**

Excerpt from EPIP-1, Attachment 1 under RADIOLOGICAL EFFLUENTS:



assessments until the results from a dose assessment using actual meteorology are available.

(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer.

Stack Noble Gas (WRGERMS)	4.02 X 10^8 µCi/sec
Eberline Continuous Air Monitors (CAM) (RM-90-249, -250, -251,-252)	9.2E+5 µCi/sec

OR

(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary.

OR

(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary for one hour of exposure.

OR

(4) Field survey results indicate EITHER of the following at or beyond the site boundary:
Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

ALERT

RA2 - Significant lowering of water level above, or damage to, irradiated fuel.

(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.

OR

(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by alarm on **ANY** of the following radiation monitors:

- 1,2,3-RM-90-1A Fuel Pool Floor
- 1,2,3-RM-90-142A Reactor Zone Exhaust
- 1,2,3-RM-90-250A Reactor, Turbine, Refuel Exhaust
- 1,2,3-RM-90-140A Refueling Zone Exhaust

OR

(3) Lowering of spent fuel pool level to 650' 4"

Excerpt from EPIP-1, Attachment 1: Supports Distractors

UNUSUAL EVENT

RU1 – Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

- **The SED should declare the Unusual Event promptly upon determining that 60 minutes 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 60 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.**

(1) Reading on Stack Noble Gas (WRGERMS) greater than 1.74 X 10^7 µCi/sec for 60 minutes or longer.

OR

(2) Reading on ANY of the following radiation monitors exceeding 2 times the Maximum Allowable Value for 60 minutes or longer.

Plant Radiation Monitoring System	2 times Maximum Allowable Value (2 x MAV)
Eberline Continuous Air Monitors (CAM) (RM-90-249, -250, -251, -252)	9.8E+4 µCi/s
GE Stack Gas Radiation Monitor (RM-90-147/148)	50,000 cps
Offgas Pretreatment Radiation Monitor (1,2,3-RM-90-157)	2.6E+05 mR/hr

OR

(3) Sample analysis for a liquid release indicates a concentration or release rate greater than 20 times the Effluent Concentration Limit for 60 minutes or longer.

RU2 – UNPLANNED loss of water level above irradiated fuel.

(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:
- Fuel Pool System Abnormal annunciator lit and SFP level less than 662'8" as indicated by 1,2,3-LI-078-0042 and 1,2,3-LI-078-0043
- Visual observation by plant personnel.

AND

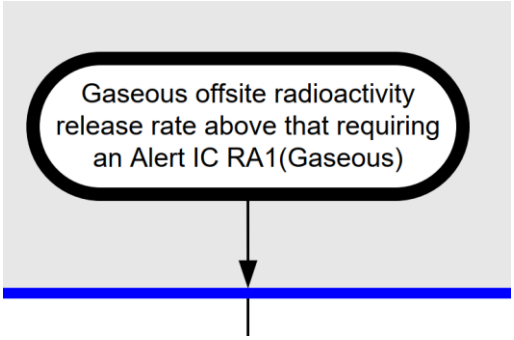
b. UNPLANNED rise in area radiation levels as indicated by the following radiation monitor.
- 1,2,3-RM-90-1A Fuel Pool Floor

Excerpt from 0-EOI-4:

0-EOI-4 RADIOACTIVITY RELEASE CONTROL MODES 1-3

RADIOACTIVITY RELEASE CONTROL
MODES 1-3
UNIT 0
BROWNS FERRY
NUCLEAR PLANT

Rev: 9



Excerpt from 0-ODCM-001:

7.0 GASEOUS EFFLUENTS

RELEASE POINTS DESCRIPTION

There are eleven monitored discharge points at BFN: a Reactor Building Exhaust for each unit, the Radwaste Building Exhaust, two sets of Turbine Deck Roof Fans for each unit, and the Stack. The Reactor and Radwaste Exhausts are located on the roof of the reactor building.

The Reactor Building Exhausts include exhaust from the refuel floor of the reactor buildings, exhaust from the Primary Containment Purge System, and exhaust from the turbine buildings. These discharge points are monitored by radiation monitors 1-,2-,3-RM-90-250.

The Radwaste Building Exhaust includes exhaust from the common radwaste building. This discharge point is monitored by radiation monitor RM-90-252.

There are nine roof fans on the roof of each unit's turbine building to provide building ventilation. These are generally used in the warmer months to control building temperature. For each unit, there are two radiation monitors. One of these monitors the exhaust through four of the fans (1-,2-RM-90-249 and 3-RM-90-251). The other monitors the exhaust through the remaining five fans (1-,2-RM-90-251 and 3-RM-90-249).

The common 600 foot plant stack receives the Condenser Offgas exhaust, the Filter Cubicle exhaust, the Steam Packing and Mechanical Vacuum exhaust, and the Standby Gas Treatment System (SBGTS) exhaust. The stack effluents are monitored by radiation monitors RM-90-147 and 148.

Figures 7.1 and 7.2 show the Offgas System, the Standby Gas Treatment System and normal building ventilation with effluent monitor locations.

Examination Outline Cross-reference:

G2.4.42 (10CFR 55.43.5 – SRO Only)

Knowledge of emergency response facilities

Level	RO	SRO
Tier #	------	3
Group #	------	------
K/A #	G2.4.42	
Importance Rating	------	3.8

Proposed Question: **# 100**

Unit 1 was operating at 100% RTP when an event occurred, resulting in the following conditions:

- A manual SCRAM inserted with an electrical ATWS conditions
- An ALERT was declared
- The on-call SED has assumed duties from the SM/SED
- TSC is staffed
- NO radiological or other hazard conditions currently exist in Reactor Building

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

The NUSO will direct the performance of 1-EOI Appendix- **(1)** at the CRD station on elevation 565 north east.

In accordance with EPIP-6, Activation and Operation of the Technical Support Center (TSC), the NUSO **(2)** required to request Operations Support Center (OSC) support through the TSC for this task.

Note: 2-EOI Appendix-1B, Venting and Repressurizing the SCRAM Pilot Air Header
2-EOI Appendix-1E, Manual Insertion of Control Rods by Venting the Over Piston Area

A. (1) 1B
(2) is

B. (1) 1B
(2) is NOT

C. (1) 1E
(2) is

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

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (*See attached*) First part is correct in that 2-EOI Appendix-1B, Venting and Repressurizing the SCRAM Pilot Air Header, is performed on the 565 foot elevation in the Northeast quadrant of the Reactor Building. Second part, in accordance with EPIP-6, Activation and Operation of the Technical Support Center (TSC), once the responsibilities of the SM/SED have been assumed by the on-call SED, command and control of the emergency response transfers to the TSC.

B INCORRECT: First part is correct (*See A*). Second part is incorrect but plausible in that control room licensed personnel typically dispatch personnel to complete outside the control room activities. Candidate may not recall that upon assumption of the SED and staffing of the TSC, command and control is transferred to the TSC and personnel are dispatched and tracked through the OSC.

C INCORRECT: First part is incorrect but plausible because EOI Appendix 1E, Manual Insertion of Control Rods by Venting the Overpiston Area, is performed on the 565 foot elevation at the East and West CRD Modules (*See attached*). Second part is correct (*See A*).

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

SRO Level Justification: Tests the candidate's knowledge of emergency response facilities such as the OSC and TSC, as well as performance location of specific EOI appendices and when required to be selected and performed. 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 and Abnormal Procedures.

Technical Reference(s): 2-EOI-1A, Rev. 7 (Attach if not previously provided)
2-EOI Appendix-1B, Rev. 5
2-EOI Appendix-1E, Rev. 7
EPIP-6, Rev. 47
BFN-ODM-4.2, Rev. 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.075, Obj. 2 (As available)

Question Source: Bank # BFN NRC 1404 #100
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2014

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 1404 #100:

QUESTION 100

An ATWS has occurred on Unit 1 with the following conditions:

- An Alert has been declared
- The On-Call SED has assumed the duties of the SED.
- There are no radiological or other hazards in the Reactor Building

Subsequently, it becomes necessary to perform 1-EOI Appendix-1B, Vent and Depressurize the Scram Pilot Air Header.

Which ONE of the following completes both statements below?

The AUO will perform 1-EOI Appendix-1B at the __(1)__.

The AUO required to perform this appendix __(2)__ requested through the TSC.

A. (1) CRD catwalk above Hydraulic Control Units
(2) is

B. (1) CRD catwalk above Hydraulic Control Units
(2) is NOT

C. (1) 565' elevation north east at the CRD station
(2) is

D. (1) 565' elevation north east at the CRD station
(2) is NOT

Answer: **C**

Excerpt from EOI-1A, ATWS RPV Control:

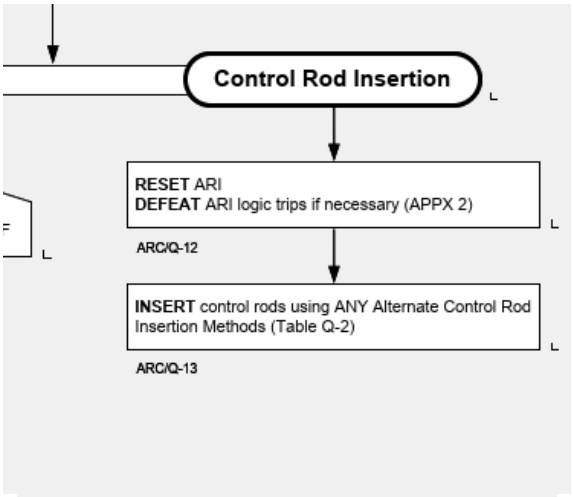


Table Q-2
Alternate Control Rod Insertion Methods

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

2-EOI-1A Page 1 of 1

ATWS RPV CONTROL
UNIT 2
BROWNS FERRY
NUCLEAR PLANT

Rev. 7

Excerpt from 2-EOI Appendix-1B:

BFN Unit 1	Venting and Repressurizing the SCRAM Pilot Air Header	1-EOI Appendix-1B Rev. 0002 Page 3 of 5

1.0 INSTRUCTIONS

LOCATION:	Unit 1 RB NE, El. 565', 1-LPNL-925-0018B
ATTACHMENTS:	Tools and Equipment

[1] **NOTIFY** Unit Operator and **CONTINUE** in this procedure. ☐

[2] **CLOSE** the following valve (El. 565, R5, N-Line):

- 1-SHV-085-0260, CONT AIR HDR SPLY SHUTOFF VALVE ☐
- 1-SHV-085-0261, CONT AIR HDR SPLY SHUTOFF VALVE ☐

[3] **INSTALL** quick disconnects on drain nipples downstream of the following drain valves (Panel 1-LPNL-925-0018B, east end):

- 1-DRIV-085-0038A, INSTR DRAIN VLV FOR PS-85-38 ☐
- 1-DRIV-085-0038B, INSTR DRAIN VLV FOR PI-85-38. ☐

[4] **OPEN** the following instrument drain valves (Panel 1-LPNL-925-0018B, east end):

- 1-DRIV-085-0038A, INSTR DRAIN VLV FOR PS-85-38 ☐
- 1-DRIV-085-0038B, INSTR DRAIN VLV FOR PI-85-38. ☐

[5] **WHEN** 1-PI-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS, indicates 0 psig, **THEN**

NOTIFY Unit Operator. ☐

Excerpt from 2-EOI Appendix-1E (supports distractors C1/D1):

BFN UNIT 1	MANUAL INSERTION OF CONTROL RODS BY VENTING THE OVER PISTON AREA	1-EOI APPENDIX-1E Rev. 1 Page 2 of 7

4.a (Continued)

NOTE

A ladder may be required to perform the following steps. Refer to Attachment 1.

b. **UNLOCK** and **CLOSE** 1-SHV-085-615, WITHDRAW RISER SOV. _____

c. **OBTAIN** necessary tools and equipment and **PERFORM** the following from the catwalk area:

1) **CONNECT** quick disconnect end of vent hose to coupling 1-TV-085-623, WITHDRAW RISER VENT BLOCK TEST VLV. _____

2) IF ..... Valve stem cap is installed for 1-VTV-085-614, WITHDRAW RISER VENT,

THEN ..... **REMOVE** valve stem cap from 1-VTV-085-614, WITHDRAW RISER VENT, using 10-in. crescent wrench. _____

CAUTION

Opening of WITHDRAW RISER VENT valve more than two turns may result in burn and contamination hazard to personnel at the valve or floor drain area.

3) **SLOWLY OPEN** 1-VTV-085-614, WITHDRAW RISER VENT, using T-shaped or L-shaped wrench NO MORE than two turns, as necessary, to cause control rod movement. _____

4) IF ..... Control rod movement is NOT observed after two turns of vent valve opening,

THEN ..... **CLOSE** 1-VTV-085-614, WITHDRAW RISER VENT, and **CONTINUE** at Step 4.c.6). _____

Excerpt from EPIP-6 (also support distractors B2/D2):

BFN Unit 0	ACTIVATION AND OPERATION OF THE TECHNICAL SUPPORT CENTER (TSC)	EPIP-6 Rev. 0047 Page 7 of 64

3.0 INSTRUCTIONS

3.1 Activation

The TSC is required to be activated at the Alert or higher emergency classification, however, activation can occur at the discretion of the Shift Manager (SM). Once an emergency classification has been declared, the SM becomes the Site Emergency Director (SED). Depending upon the emergency classification declared, steps to activate the TSC are specified in the applicable EPIP for the emergency classification. When the TSC is activated, the on-call SED will obtain a turnover from the SM/SED, ensure that minimum staffing is met for the emergency center, and assume the responsibilities of the SED from the SM/SED. Once the responsibilities of the SM/SED have been assumed by the on-call SED, command and control of the emergency response transfers to the TSC. TSC activation time is defined in the Radiological Emergency Plan.

3.2 Methods of Notification of Emergency Response Organization (ERO).

Notification of the TSC personnel can be accomplished by one or more of the following methods:

- TVA Enterprise Emergency Notification System (TEENS) is the primary method.
- Manual call-out through utilization of the call-out list.
- Plant Public Address (PA) announcement
- Activation of the Assembly and Accountability Siren

3.3 ERO Information

NPG-SPP-18.3, “Emergency Preparedness” provides the ERO with information regarding duty assignments and response to emergency call-outs.

Excerpt from BFN-ODM-4.2:

BFN Operations Directive Manual	Radiological Emergency Plan (REP) Assignments	BFN-ODM-4.2 Rev. 0007 Page 7 of 7

D. Control Room Dispatch of Operators and Area Evacuation Authority following REP Event Classification:

	May Control Room Order Area Evacuation Due To *Radiation Levels*?	May Control Room Order Direct Dispatch Of Operators...	
ANY REP Classification Has Been Declared, AND TSC Activation IS NOT Complete, Or Has NOT Been Ordered	YES	...Into ANY Area?	YES Control room staff should evaluate all available indications of radiological and other hazard conditions before dispatch.
ANY REP Classification Has Been Declared, AND TSC Activation IS Complete.	NO Unless TSC concurs that area evacuation will NOT adversely affect essential OSC team task performance.	...Into Control Bay?	YES Use operators staged in control bay per this ODM, if possible.
		...Into Electric Board Rooms?	YES Use operators staged in control bay per this ODM, if possible. If radiological or other hazard conditions exist in or near room, then request OSC team through TSC.
		...Into Diesel Generator Rooms?	YES Use operators staged in control bay per this ODM, if possible. If radiological or other hazard conditions exist in or near room, including those from *radiological releases outside secondary containment*, then request OSC team through TSC.
		...Into ANY Other Radiological Controlled Or Affected Areas (RB, TB, etc.)?	NO Request OSC team through TSC.

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-1
Scenario Source: NEW Op-Test Number: 24-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 95% Reactor Power. 2B Bus Duct Fan is tagged out of service.

Turnover: Return Reactor Power to 100%. 2A Raw Cooling Water (RCW) Pump is in AUTO. Alternate RCW Pumps, place 2A RCW Pump in service and remove 2B RCW Pump from service.

Critical Tasks:

1. When plant conditions indicate that two plant areas are above their Maximum Safe values, without delay, implement Emergency RPV Depressurization.
2. Following Emergency Depressurization, inject to the Reactor using low pressure injection systems to restore Reactor Water Level above (-) 180 inches or establish Spray Cooling with either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above (-) 215 inches to preclude SAMG entry.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	R-OATC	Raise Reactor Power with Core Flow
2.	N/A	N-BOP N-NUSO	Alternate Raw Cooling Water (RCW) Pumps
3.	XA-55-5B_21	TS-NUSO	Standby Liquid Control (SLC) Tank Level Low
4.	PMP-66-51A	C-OATC MC-OATC C-NUSO	2B Steam Packing Exhauster (SPE) Fails to Automatically Start
5.	XA-55-3A_10	C-BOP C-NUSO	Pressure Suppression Chamber (PSC) Pump Alarm
6.	SW04A	C-BOP MC-BOP C-NUSO	2A Raw Cooling Water (RCW) Pump Trip
7.	XA-55-3D_19	TS-NUSO	2-90-256, Drywell Leak Detection Radiation Monitor Sample Pump Trip
8.	FW01	C-OATC C-NUSO	2A Condensate Pump Trip
9.	HP08	M-ALL	Unisoloable High Pressure Injection Coolant (HPCI) Steam Leak
10.	PC14	C-BOP C-NUSO	'D' Outboard Main Steam Isolation Valve (MSIV) remains open after Main Steam Lines Isolation

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The Crew will Raise Reactor Power with Core Flow in accordance with 2-OI-68, Reactor Recirculation System.

2. Alternate RCW Pumps in accordance with 2-OI-24, Raw Cooling Water System. Place 2A RCW Pump in service and remove 2B RCW Pump from service.

3. SLC TANK LEVEL ABNORMAL (2-9-5B, Window 21) alarms. The ARP directs Chemistry to perform 2-SR-3.1.7.1, Conditional for Standby Liquid Control (SLC) Solution Level Check. Chemistry reports less than 4000 usable gallons available in the SLC storage tank. The NUSO will address Tech Specs

4. STEAM PACKING EXHAUSTER VACUUM LOW (2-9-7A, Window 12) alarms when 2A SPE discharge valve spuriously closes. The OATC will respond with the Alarm Response Procedure (ARP) and start the 2B SPE. (The BOP will be removed allowing the OATC to respond).

5. PSC PUMP SUCTION STRAINER DIFFERENTIAL PRESSURE HIGH (2-9-3A, Window 10) alarms. The BOP will respond with the ARP and manually restart the Pump.

6. The crew will respond to a 2A RCW Pump trip, the 2B RCW Pump will fail to start automatically and will be manually started.

7. DRYWELL LEAK DETECTION RADIATION DOWNSCALE (2-9-3D, Window 19) alarms due to the sample pump tripping. The Nuclear Unit Senior Operator (NUSO) will address Technical Specifications (Tech Specs).

8. The crew will enter 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low, in response to 2A Condensate Pump trip, inserting a Mid-Power Recirc run back.

9. An unisolable HPCI steam leak into Secondary Containment develops, the crew will attempt to isolate HPCI. The NUSO will direct a SCRAM. All Condensate pumps trip 60 seconds after the MODE SWITCH is placed in SHUTDOWN. Low pressure ECCS injection valves will not automatically open.

10. When the MODE SWITCH is placed in SHUTDOWN, a Main Steam Line ‘A’ break in the Reactor Building Steam Tunnel occurs resulting in a Main Steam Line Isolation due to high temperature in the Main Steam Vault. The ‘D’ Outboard MSIV remains open after the Main Steam Line Isolation Signal, requiring the crew to manually close the ‘D’ Outboard MSIV and transition to Panel 9-3 to maintain Reactor Pressure and Reactor Water Level.

The Scenario ends when the crew has Emergency Depressurized after Secondary Containment Temperatures reach Maximum Safe in two different areas and has restored Reactor Water Level above (-) 180 inches or established Spray Cooling.

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 Temperature 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 plant conditions indicate that two Area Temperatures exceed Maximum Safe value.

2. Following Emergency Depressurization, inject to the Reactor using low pressure injection systems to restore Reactor Water Level above (-) 180 inches or establish Spray Cooling with either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above (-) 215 inches to preclude SAMG entry.

a. Safety Significance
Maintain Adequate Core Cooling.
Prevent degradation of fission product barrier.

b. Cues
Procedural Compliance.
Reactor Water Level trend.

c. Measured by:
Observation – low pressure injection systems are used to raise Reactor Water Level.

d. Feedback:
Reactor Pressure.
Reactor Water Level.

e. Critical Task Failure Criteria:
The crew does not restore and maintain Reactor Water Level above (-) 180 inches AND Spray Cooling cannot be established, resulting in SAMG entry being required. (Table M-3 Spray Cooling is either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above -215 inches)

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 1 Page 1 of 1

Event Description: Raise Reactor Power with Core Flow

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 pre-briefed but may elect to hold 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-NRC2404-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: • 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 Pumps 2A and/or 2B speed with the RECIRC MASTER CONTROL, THEN ADJUST Recirc Pump Speed 2A & 2B using the following pushbuttons as required. • RAISE SLOW, 2-HS-96-31 • RAISE MEDIUM, 2-HS-96-32
	NRC	**End of Event 1. Proceed to Event 2, Alternate RCW Pumps.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 2 Page 1 of 2

Event Description: Alternate Raw Cooling Water (RCW) Pumps

Symptoms/Cues: Crew is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: The crew has pre-briefed but may elect to hold a re-focus brief.**
	NUSO	Directs the Balance of Plant Operator (BOP) to alternate RCW Pumps in accordance in accordance with 2-OI-24, Raw Cooling Water System, Section 6.1. Place 2A RCW Pump in service and remove 2B RCW Pump from service.
	BOP	**NOTES** 1) RCW Pumps should be operated at a minimum of once per sixty days to prevent the potential of downstream fouling when a pump is placed in service. Allowing RCW Pumps to be out of service for an extended period of time past sixty days can allow the excessive buildup of silt and clam growth on suction and discharge sides of pump. 2) P&L 3.2G When alternating RCW Pumps, if 2A RCW Pump in service, with 2B RCW Pump in standby (AUTO) and 2C RCW Pump secured. The operator should start the next sequenced pump (2B RCW pump) and place 2C RCW pump or another Unit's RCW pump in standby. [1] REVIEW all Precautions and Limitations in Section 3.0. [2] NOTIFY Chemistry prior to alternating a RCW pump. [3] For pump to be placed in continuous operation, START the RCW PUMP using the appropriate hand-switch. (REFER TO Step 3.2G for pump alteration.) • RCW PUMP 2A, 2-HS-24-7A [4] For pump to be removed from continuous operation, STOP the RCW PUMP using the appropriate hand-switch. (REFER TO Step 3.2G for pump alteration.) • RCW PUMP 2B, 2-HS-24-10A **NOTES** 1) Pump has potential to start when placed in AUTO. Review P&L 3.2E 2) Placing a running RCW pump in AUTO will bring in a motor trip out alarm if header pressure reaches 57 pounds and pump auto stops.

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 2 Page 2 of 2

Event Description: Alternate Raw Cooling Water (RCW) Pumps

Time	Position	Applicant's Actions or Behavior
		[5] For pump to be placed in automatic operation, PLACE the appropriate RCW Pump hand-switch to the mid position and PULL to AUTO. (REFER TO Step 3.2G for pump alteration.) • RCW PUMP 2B, 2-HS-24-10A [6] CHECK RCW Header Pressure are within the following: • RCW header pressure <61 psig with only 3 pumps operating in parallel. • RCW header pressure <59 psig with only 4 pumps operating in parallel. • RCW header pressure <58 psig with only 5 pumps operating in parallel. • RCW header pressure <57 psig with 6 or more pumps operating in parallel.
	Driver	**If contacted as the Unit 1 or Unit 3 Operator and requested to provide a status of RCW pumps in auto OR requested to place a RCW pump in auto, reply "Unit 1(3) has no available RCW pumps for Automatic operation".**
	NRC	**End of Event 2. Proceed to Event 3, Standby Liquid Control (SLC) Tank Level Low.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 3 Page 1 of 2

Event Description: Standby Liquid Control (SLC) Tank Level Low

Symptoms/Cues: SLC TANK LEVEL ABNORMAL (2-9-5B, Window 21) alarms

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to Fail Standby Liquid Control (SLC) Tank Level Low.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • SLC TANK LEVEL ABNORMAL (2-9-5B, Window 21)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: SLC TANK LEVEL ABNORMAL (2-9-5B, Window 21) Operator Actions: A. CHECK alarm, on Panel 2-9-5, SLC STORAGE TANK LEVEL 2-LI-63-1A by having Chemistry perform 2-SR-3.1.7.1. B. IF alarm is valid, then notify Chemistry to perform 2-SR-3.1.7.3 within 24 hours. C. REFER TO Tech Spec Sect. 3.1.7.
	NRC	**EXAMINER NOTE:** • **Time Compression is used, Chemistry completed 2-SR-3.1.7.1 should be handed to the NUSO after the crew calls for performace of 2-SR-3.1.7.1.** • **The NUSO should enter Tech Spec 3.1.7 CONDITION B. Two SLC subsystems inoperable, REQUIRED ACTION B.1 Restore one SLC subsystem to OPERABLE status, COMPLETION TIME, 8 hours, based on a Net Injectable volume of 3930 gallons determined from 2-SR-3.1.7.1**
	NUSO	

BFN Unit 2	Conditional for Standby Liquid Control (SLC) Solution Level Check	2-SR-3.1.7.1 Rev. 0012 Page 10 of 13

Attachment 1
(Page 1 of 2)

Tank Volume Conversions

Inches From Tank Bottom	Gauge Reading (%)	Gross Volume (gal)	Net Injectable Volume (gal)
106	80.2	4165.8	3890.7
106.5	80.6	4185.5	3910.4
107	81	4205.1	3930
107.5	81.4	4224.8	3949.7
108	81.8	4244.4	3969.3
108.2	82	4252.3	3977.2
108.5	82.3	4264.1	3989
109	82.7	4283.7	4008.6

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 3 Page 2 of 2

Event Description: Standby Liquid Control (SLC) Tank Level Low

Time	Position	Applicant's Actions or Behavior
	NUSO	SURVEILLANCE REQUIREMENTS SURVEILLANCE SR 3.1.7.1 Verify available volume of sodium pentaborate solution (SPB) is ≥ 4000 gallons. LCO 3.1.7 Two SLC subsystems shall be OPERABLE. APPLICABILITY: MODES 1, 2, and 3. ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours

	NRC	**End of Event 3. Proceed to Event 4, 2B Steam Packing Exhauster (SPE) Fails to Automatically Start.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 4 Page 1 of 3

Event Description: 2B Steam Packing Exhauster (SPE) Fails to Automatically Start

Symptoms/Cues: STEAM PACKING EXHAUSTER VACUUM LOW (2-9-7A, Window 12) alarms when 2A SPE discharge valve spuriously closes.

Time	Position	Applicant's Actions or Behavior
	Driver	**NOTE: Two triggers will be used for event 4, first trigger 20 and then trigger 4, each at the Chief Examiners Direction. Trigger 20 is to distract the BOP to allow the OATC to respond to trigger 4, 2B Steam Packing Exhauster (SPE) Failure to Automatically Start** • **When requested by the Chief Examiner, insert Event 20 to bring in a spurious alarm on 9-53.** • **When requested by the Chief Examiner, insert Event 4 to fail 2A SPE discharge valve to spuriously closes.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • GLYCOL STORAGE TANK TEMP HIGH (2-9-53, Window 6)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	NRC	**Based on spurious alarm, GLYCOL STORAGE TANK TEMP HIGH (2-9-53, Window 6) resetting the crew may or may not elect to dispatch personnel to the Off-Gas Bldg.**
	OATC	Acknowledges and reports the following alarm(s) to the NUSO: • STEAM PACKING EXHAUSTER VACUUM LOW (2-9-7A, Window 12)
	NUSO	Directs OATC to respond in accordance with the Alarm Response Procedure(s).
	OATC	Alarm Response Procedure: STEAM PACKING EXHAUSTER VACUUM LOW (2-9-7A, Window 12) Operator Actions: A. CHECK following: 1. Standby SPE 2B(2A), 2-HS-66-51A (2-HS-66-50A), started. (Both red and green indicating lights at handswitch of running standby blower will be illuminated; no indicating lights may be illuminated at handswitch of tripped blower.) 2. SPE 2B(2A) DISCHARGE VLV, 2-HS-66-35A(2-HS-66-34A), opens. 3. SPE 2A(2B) DISCHARGE VLV, 2-HS-66-34A(2-HS-66-35A), closes. B. IF standby blower fails to start, THEN START standby OR CHECK normal in service. REFER TO 2-OI-47C
	Driver	**If dispatched to investigate SPE breaker compartment 7E at 480V Turb MOV Bd 2C, wait one minute and report, "No abnormalities found for the 2A Steam Packing Exhauster breaker** ."

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 4 Page 2 of 3

Event Description: 2B Steam Packing Exhauster (SPE) Fails to Automatically Start

	OATC	C. IF blower is running, PERFORM the following: 1. THROTTLE in-service SPE 2B(2A) DISCHARGE VLV, 2-HS-66-35A (2-HS-66-34A), UNTIL SPE Vacuum, as indicated on SPE VACUUM, 2-PI-66-54, is between 10" and 12" Vacuum, or as appropriate for plant conditions. REFER TO 2-OI-47C. 2. ENSURE SJAE/OG CNDR CNDS FLOW, 2-FI-2-42, between $2x10^6$ lbm/hr and $3x10^6$ lbm/hr to prevent tripping SPE due to inadequate cooling. REFER TO 2-OI-2. D. N/A E. IF fans or dampers have failed, THEN CHECK electrical feeds on 480V Turb MOV Bd 2C, bkr 7E (SPE A) and 480V Turb MOV Bd 2B, bkr 7E (SPE B). F. IF standby blower was verified to have auto-started in previous steps above, AND is to be left in service, THEN PERFORM the following (Otherwise N/A): 1. IF desired to match target, THEN PLACE handswitch for running blower SPE 2B(2A), 2-HS-66-51A (2-HS-66-50A) to START. (Otherwise, N/A) 2. PLACE handswitch for tripped blower SPE BLOWER 2A(2B), 2-HS-66-50A (2-HS-66-51A) to STOP to reset dual-lit light configuration at running blower handswitch.
	NRC	**NOTE, Examinee may elect to place the Standby SPE 2B in service with the above ARP, STEAM PACKING EXHAUSTER VACUUM LOW (2-9-7A, Window 12) OR exit ARP at Step B and use the below 2-OI-47C, Seal Steam System, Section 5.3**
	OATC	2-OI-47C, Seal Steam System Section 5.3, Startup of Steam Packing Exhauster **NOTES** 1) Section 5.3 is performed at Panel 2-9-7. 2) Verify SJAE/OG CNDS FLOW, 2-FI-2-42, between $2X10^6$ lbm/hr and $3x10^6$ lbm/hr to minimize blower trips. 3) Handswitch indicating lights for the SPE's do not illuminate until one of the SPE's is started. 4) To minimize SPE motor trips when initially placing an SPE in service during preparation for establishing condenser vacuum; maintain SPE vacuum at 6-8 in H20 until the steam seal system is placed in service 5) The standby SPE will receive an auto start signal at 5 in H2O vac (PS-66-49). 6) The SPE blower handswitch, 2-HS-66-50A (51A), must be held in the start position for about 10 seconds to allow the associated discharge valve to open. 7) SPE motors may trip on overload when steam seal header pressure is below 4.5 psig. The SPE may be restarted provided motor start limitations are adhered to.

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 4 Page 3 of 3

Event Description: 2B Steam Packing Exhauster (SPE) Fails to Automatically Start

Time	Position	Applicant's Actions or Behavior
	OATC	[1] PLACE and HOLD appropriate SPE handswitch in START: • STEAM PACKING EXHR BLOWER 2B, 2-HS-66-51A. [2] ENSURE appropriate SPE Discharge Valve OPENS: • STM PKG EXHR 2B DISCHARGE VALVE, 2-FCV-66-35, using 2-HS-66-35A. [3] RELEASE SPE handswitch started in Step 5.3[1]: • STEAM PACKING EXHR BLOWER 2B, 2-HS-66-51A. [4] N/A [5] WHEN steam seals are established and the SPE is operating stable, THEN THROTTLE appropriate SPE Discharge Valve to Maintain 10 - 12 in H2O vacuum as indicated on STEAM PACKING EXH VACUUM, 2-PI-66-54: • STM PKG EXHR 2B DISCHARGE VALVE, 2-FCV-66-35.
	NRC	**End of Event 4. Proceed to Event 5, Pressure Suppression Chamber (PSC) Pump Alarm.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 5 Page 1 of 1

Event Description: Pressure Suppression Chamber (PSC) Pump Alarm

Symptoms/Cues: PSC PUMP SUCTION STRAINER DIFFERENTIAL PRESSURE HIGH (2-9-3A, Window 10) alarms. PSC pumps lock out until ARP action is performed.

Time	Position	Applicant's Actions or Behavior
	DRIVER	**When requested by the Chief Examiner, insert Event 4 to lock out the PSC pumps on high suction strainer differential pressure**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • PSC PUMP SUCTION STRAINER DIFFERENTIAL PRESSURE HIGH (2-9-3A, Window 10)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: PSC PUMP SUCTION STRAINER DIFFERENTIAL PRESSURE HIGH (2-9-3A, Window 10) Operator Actions: A. ATTEMPT to manually restart pumps. B. IF the pumps will not start, THEN DISPATCH personnel to PERFORM the following: 1. RESET the start logic by depressing local reset push-button. 2. START the pumps if possible. C. N/A D. N/A
	Driver	**If requested to report a local DP reading at 2-PDIS-075-0074, state "2-PDIS-075-0074 is about zero psid."**
	NRC	**NOTE: the above Step 'A' will fail, examinee will proceed to step 'B' and dispatch personnel to reset the start logic locally before PSC Pump(s) may be restarted.**
	NRC	**End of Event 5. Proceed to Event 6, 2A RCW Pump trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 6 Page 1 of 1

Event Description: 2A Raw Cooling Water (RCW) Pump Trip

Symptoms/Cues: White disagreement light indicates on the 2A RCW pump handswitch and distinct Motor Tripout alarm sounds

Time	Position	Applicant's Actions or Behavior
	DRIVER	**When requested by the Chief Examiner, insert Event 6 to trip 2A RCW pump.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • RCW DISCH HDR PRESS LOW HIGH (2-9-20A, Window 34) • MOTOR TRIPOUT (2-9-8C, Window 33)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: RCW DISCH HDR PRESS LOW HIGH (2-9-20A, Window 34) Automatic Action: Spare RCW Pump starts at 47 psig via 2-PS-24-7A, B when selected in AUTO Operator Actions: A. CHECK low pressure on RCW HEADER PRESSURE, 2-PI-24-18. B. On Panel 2-9-20, ENSURE standby RCW Pump RUNNING. C. NOTIFY UNIT SRO, Unit 1 and Unit 3. D. N/A E. N/A
	NRC	**NOTE: Examinee should start the 2B RCW pump**
	BOP	Alarm Response Procedure: MOTOR TRIPOUT (2-9-8C, Window 33) Operator Actions: A. CHECK Control Room for white disagreement light lit for affected equipment. B. CLEAR disagreement light. C. DISPATCH personnel to check: • Relays at associated electrical bd. • Equipment for abnormal conditions. • Safe-stop locally reset, if necessary. D. N/A E. N/A
	NRC	**End of Event 6. Proceed to Event 7, 2-90-256, Drywell Leak Detection Radiation Monitor Sample Pump Trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 7 Page 1 of 2

Event Description: 2-90-256, Drywell Leak Detection Radiation Monitor Sample Pump Trip

Symptoms/Cues: DRYWELL LEAK DETECTION RADIATION DOWNSCALE (2-9-3D, Window 19) alarms and the associated sample pump run light extinguishes.

Time	Position	Applicant's Actions or Behavior
	DRIVER	**When requested by the Chief Examiner, insert Event 7 to trip 2-90-256, Drywell Leak Detection Radiation Monitor Sample Pump.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • DRYWELL LEAK DETECTION RADIATION DOWNSCALE (2-9-3D, Window 19)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: DRYWELL LEAK DETECTION RADIATION DOWNSCALE (2-9-3D, Window 19) Operator Action: A. N/A B. DISPATCH person to examine CAM skid and CHECK that local skid mounted circuit breakers CB-1 and CB-2 located in junction box are ON. C. N/A D. IF corrective maintenance is required THEN NOTIFY Chemistry to commence its sampling procedure. E. REFER TO Tech Specs 3.4.4, 3.4.5, and TRM 3.3.10 for CAM LCO requirements and IMPLEMENT appropriate TS/TRM actions as required. F. N/A
	Driver	**If dispatched to investigate locally the 2-90-256 skid, wait one minute and report, "The 2-90-256 sample pump is not running and hot to the touch."** **If notified as chemistry to begin sampling procedure, acknowledge request.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 7 Page 2 of 2

Event Description: 2-90-256, Drywell Leak Detection Radiation Monitor Sample Pump Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**The NUSO should enter Tech Spec 3.4.5 CONDITION B. For TRM 3.3.10, there is no evidence that the 2-90-256 instrumentation is inoperable therefore it is not required for the NUSO to enter any condition for TRM 3.3.10. Although with the lack of sample flow, the NUSO may declare an info only or fail conservative and declare TRM 3.3.10 conditions A. D. and E which directs entry of TS 3.4.5 CONDITION B.**
	NUSO	LCO 3.4.5 The following RCS leakage detection instrumentation shall be OPERABLE: a. Drywell floor drain sump monitoring system; and b. One channel of either primary containment atmospheric particulate or atmospheric gaseous monitoring system. APPLICABILITY: MODES 1, 2, and 3. ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required primary containment atmospheric monitoring system inoperable.	B.1 Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	AND	
	B.2 Restore required primary containment atmospheric monitoring system to OPERABLE status.	30 days

	NRC	**End of Event 7. Proceed to Event 8, 2A Condensate Pump trip.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 8 Page 1 of 1

Event Description: 2A Condensate Pump Trip

Symptoms/Cues: White disagreement light indicates on the 2A Condensate pump handswitch. Distinct Motor Tripout alarm sounds

Time	Position	Applicant's Actions or Behavior
	DRIVER	**When requested by the Chief Examiner, insert Event 8 to trip the 2A Condensate Pump.**
	BOP	Reports trip of 2A Condensate Pump.
	Crew	Monitors for Condensate Booster Pump(s) OR Reactor Feedwater Pump(s) low suction pressure annunciators are in alarm due to loss of Condensate Pump
	NUSO	Enters 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low
	OATC	2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low Immediate Actions [1] IF Condensate Booster Pump(s) OR Reactor Feedwater Pump(s) low suction pressure annunciators are in alarm due to loss of Condensate Pump(s), THEN PERFORM the following: [1.1] DEPRESS RECIRC PUMPS MID POWER RUNBACK push-button, 2-HS-68-43. [1.2] CHECK reactor power lowers to 66% or as directed by Unit SRO. [2] N/A [3] N/A
	NRC	**NOTE: All subsequent actions of 2-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low, are N/A. However, as time permits, the BOP should review subsequent actions to ensure these steps do not apply.**
	NRC	**End of Event 8. Proceed to Event 9, Unisolable HPCI steam leak.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 1 of 14

Event Description: Unisolable HPCI steam leak

Symptoms/Cues: HPCI LEAK DETECTION TEMP HIGH (2-9-3F, Window 10) alarms

Time	Position	Applicant's Actions or Behavior
	DRIVER	**When Chief Examiner requests, insert Event 9, Unisolable HPCI steam leak.**
	NRC/ DRIVER	**NOTE: No action is required to be taken by the driver for the following malfunctions:** • **Main Steam Line 'A' break in the Reactor Building Steam Tunnel occurs when the MODE SWITCH is placed in SHUTDOWN.** • **All Condensate Pumps Trip 1 minute after the Main Steam Line Area High Temperature PCIS Isolation occurs.** • **All low pressure ECCS automatic valve opening signals will fail**
	NRC	**Approximately 4 minutes will elapse after Event 9 is triggered, before HPCI LEAK DETECTION TEMP HIGH (2-9-3F, Window 10) alarms.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • HPCI LEAK DETECTION TEMP HIGH (2-9-3F, Window 10)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: HPCI LEAK DETECTION TEMP HIGH (2-9-3F, Window 10) 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. N/A
	NRC	**NOTE:** • **BOP may elect to take action to mitigate the HPCI steam leak and attempt isolation before the ARP can be refferenced** • **NUSO may elect to SCRAM before entering 2-EOI-3, SECONDARY CONTAINMENT CONTROL due to the unisolable Primary System discharging in to Seconday Containment**
	NUSO	Directs Reactor SCRAM

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 2 of 14

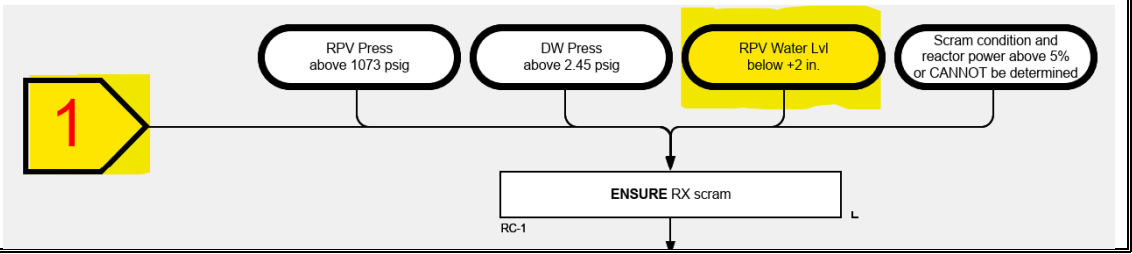
Event Description: Unisolable HPCI steam leak

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE:** • **Actions for 2-EOI-1, RPV Control Modes 1-3 are contained below the SCRAM actions of this guide.** • **Actions for 2-EOI-3, SECONDARY CONTAINMENT CONTROL are located on page 22 of this guide.** • **Approximately 4 minutes will elapse after the MODE SWITCH is placed in SHUTDOWN before reaching the MSIV High Steam Tunnel Temperature Isolation setpoint.**
	OATC	Inserts a manual Reactor SCRAM 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

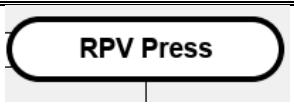
Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 3 of 14

Event Description: Unisolable HPCI steam leak

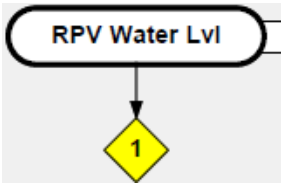
Time	Position	Applicant's Actions or Behavior
	OATC/ BOP	[1] MONITOR and CONTROL RPV Pressure to keep below 1073 psig and stable, or as directed by Unit SRO. [1.1] N/A [1.2] N/A [1.3] IF MSRVs are cycling and bypass valves are NOT available, THEN ENSURE OPEN MSRVs on Panel 2-9-3 until RPV Pressure is 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 [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" OR previously from 2-EOI-3, SECONDARY CONTAINMENT CONTROL. 

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 4 of 14

Event Description: Unisolable HPCI 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>


<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>NOT REQUIRED at this time</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-1

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
<table>
<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>
RC/P-2
</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>
<tr><td></td><td>NUSO</td><td>


ENSURE each as required:
• PCIS isolations (Groups 1, 2 and 3)
• ECCS
• RCIC
RC/L-1
</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 5 of 14

Event Description: Unisolable HPCI 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>
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Raising RPV water lvl 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>PC water lvl and drywell press CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/L-2

RESTORE and MAINTAIN RPV water lvl 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-3</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the OATC or BOP to restore or maintain Reactor Water Level in a band between +2 inches and +51 inches 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 (crosstie from unit 1/3 may be requested)
SLC (boron or test tank) - 2-EOI-Appendix-7B</td></tr>
<tr><td></td><td>Driver</td><td>If contacted as Unit 1 or 3 to crosstie the EHPM for injection, acknowledge direction given. If crew follows up, state "one hour minimum to align EHPM".</td></tr>
<tr><td></td><td>NRC</td><td>EXAMINER NOTE:
EHPM crosstie from another unit is not able to be aligned.
Due to the loss of all Condensate Pumps, Condensate and Feedwater, 2-EOI-Appendix-5A is not included within this guide.</td></tr>
</table>

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 6 of 14

Event Description: Unisolable HPCI steam leak

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.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 7 of 14

Event Description: Unisolable HPCI steam leak

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 [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.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 8 of 14

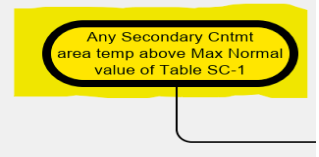
Event Description: Unisolable HPCI steam leak

Time	Position	Applicant's Actions or Behavior
	OATC	2-EOI Appendix-7B, Alternate RPV Injection System Lineup SLC System [1] N/A [2] IF RPV injection is needed immediately ONLY to prevent or mitigate fuel damage, THEN, CONTINUE at Step 1.0[10] to inject SLC Boron Tank to RPV.THEN LINE UP ALL available CRD pumps to the RPV as follows: [3] DISPATCH personnel to Unit 2 SLC pump area to line up SLC Test Tank as follows (RB NE, El 639 ft): [3.1] through [3.9] performed in the field. [4] UNLOCK and PLACE 2-HS-63-6A, SLC PUMP 2A/2B, control switch in START-A or START-B (Panel 9-5). [5] CHECK demineralized water is injecting to the RPV by observing the following: • Selected pump running, 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 Annunciator in alarm (2-XA-55-5B, Window 20). • 2-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. • System flow, as indicated by 2-IL-63-11, SLC FLOW, red light illuminated. • SLC INJECTION FLOW TO REACTOR, Annunciator in alarm (2-XA-55-5B, Window 14). [6.0] N/A [7] NOTIFY personnel in SLC area to perform the following: [7.1] and [7.2] performed in the field. [8] MONITOR and CONTROL SLC System as necessary to maintain injection. [9] N/A [10] UNLOCK and PLACE 2-HS-63-6A, SLC PUMP 2A/2B, control switch in START-A or START-B (Panel 9-5). [11.0] Refer to step [5] above for step [11] indications of SLC injection. [12] N/A [13] IF SLC tank level drops to 0%, THEN STOP SLC pumps.

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 9 of 14

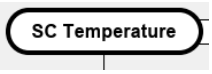
Event Description: Unisolable HPCI steam leak

Time	Position	Applicant's Actions or Behavior
	NUSO	2-EOI-3, SECONDARY CONTAINMENT CONTROL



IF	THEN
Reactor Zone Ventilation Exhaust Radiation Level is above 72 mR/hr	**NO ACTION REQUIRED**
Refuel Zone Ventilation Exhaust Radiation Level is above 72 mR/hr	**NO ACTION REQUIRED**
Reactor Zone Ventilation is isolated **AND** Reactor Zone Ventilation Exhaust Radiation level is below 72 mR/hr	**NO ACTION REQUIRED**
Refuel Zone Ventilation is isolated **AND** Refuel Zone Ventilation Exhaust Radiation Level is below 72 mR/hr	**NO ACTION REQUIRED**
SBGT CANNOT restore and maintain Secondary Cntmt dp below 0 inches water **AND** Radioactivity release from Secondary Cntmt hinders operation of systems required for damage control or systems required to be operated by EOIs	**NO ACTION REQUIRED**

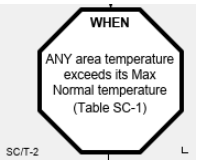
SC-1



SC-2

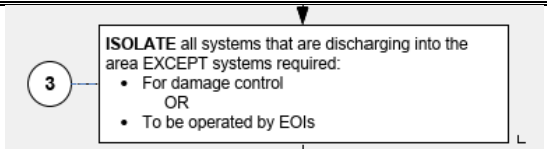
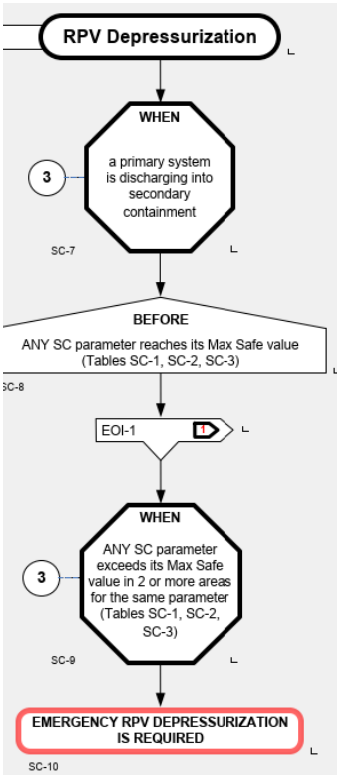
IF Rx Zone or Refuel Zone ventilation exh radiation level is below 72 mR/hr THEN operate available Rx Zone or Refuel Zone ventilation

SC/T-1



Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 10 of 14

Event Description: Unisolable HPCI steam leak

Time	Position	Applicant's Actions or Behavior
	NUSO	 SC-3 
	NUSO	Updates that Emergency Depressurization is required when two areas reach Max Safe temperatures 73-55A, HPCI Room Area (MAX SAFE 270 F), and 74-95B, RHR Sys II Pumps Area (MAX SAFE 210 F), as referenced from Table SC-1 Secondary Cntmt Area Temps
	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 plant conditions indicate that two Area Temperatures exceed Maximum Safe value.**

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 11 of 14

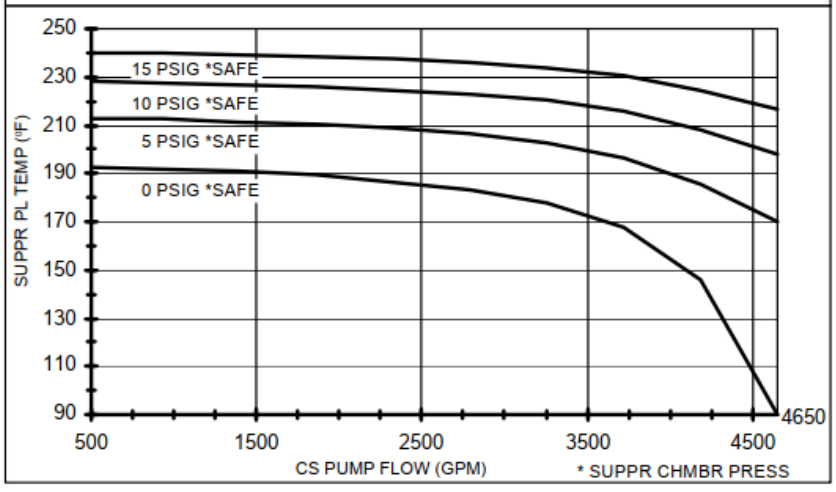
Event Description: Unisolable HPCI steam leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
2-EOI-1, RPV Control Modes 1-3
<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 Subsystems (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-6
<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>
RC/P-7
<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>
<table>
<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 5 MSRVs can be opened AND
RPV Pressure is 50 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-8
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the crew to open 6 MSRVs (ADS Valves preferred)</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the crew to inject to the Reactor using low-pressure systems:
• 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>Crew</td><td>Critical Task: Following Emergency Depressurization, inject to the Reactor using low pressure injection systems to restore Reactor Water Level above (-) 180 inches or establish Spray Cooling with either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above (-) 215 inches to preclude SAMG entry.

Critical Task Failure Criteria: The crew does not restore and maintain Reactor Water Level above (-) 180 inches AND Spray Cooling cannot be established, resulting in SAMG entry being required. (Table M-3 Spray Cooling is either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above -215 inches)</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 12 of 14

Event Description: Unisolable HPCI steam leak.

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 or OATC	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. **CURVE 1** **CS NPSH LIMITS** 

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 13 of 14

Event Description: Unisolable HPCI steam leak.

Time	Position	Applicant's Actions or Behavior
	BOP or OATC	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.

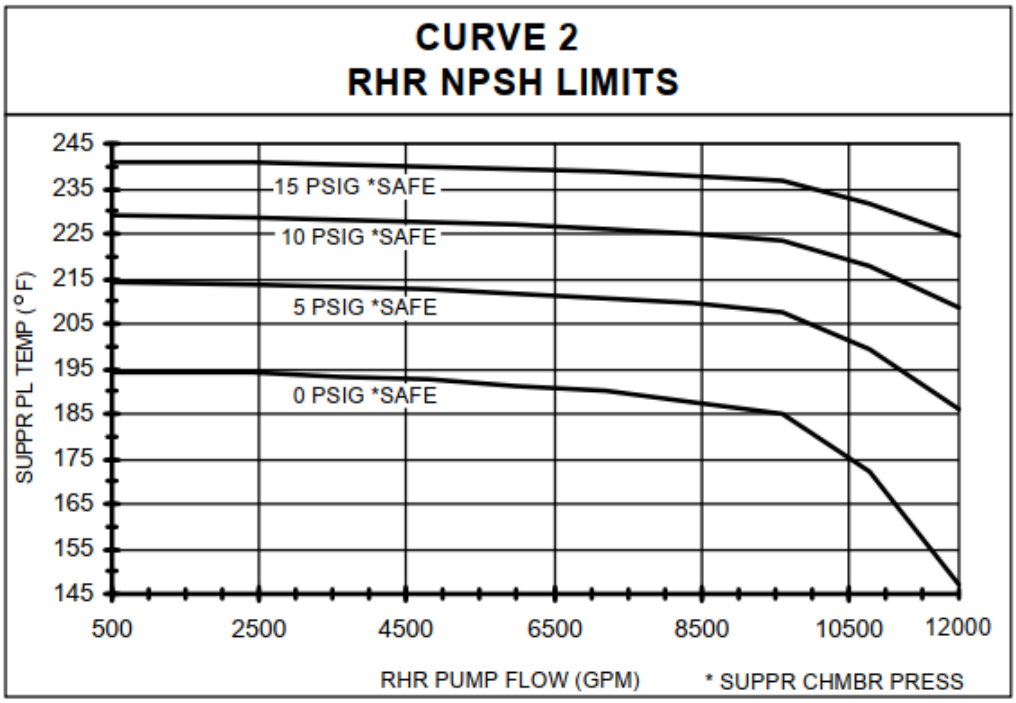
Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 9 Page 14 of 14

Event Description: Unisolable HPCI steam leak.

Time	Position	Applicant's Actions or Behavior
	BOP or OATC	[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.



Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-1 Event No.: 10 Page 1 of 1

Event Description: 'D' Outboard Main Steam Isolation Valve (MSIV) remains open after MSIV Isolation

Symptoms/Cues: Main Steam Tunnel Temperature rises on 2-TIS-1-60A, panel 2-9-3C and MAIN STEAM LINE LEAK DETECTION TEMP HIGH (2-9-3D, Window 24) alarms

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 10 is built into the scenario file; no action is required to be taken by the driver to fail open the 'D' Outboard MSIV**
	BOP	Manually closes 'D' Outboard MSIV and reports to the NUSO that 'D' Outboard MSIV failed to automatically close, manual closure is successful
	NRC	**When the crew has Emergency Depressurized after Secondary Containment Temperatures reach Maximum Safe in two different areas and has restored Reactor Water Level above (-) 180 inches or established Spray Cooling, end of Scenario.**

Scenario Setup
UNIT 2

IC	28
Exam IC	253

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-24	90	ARP-53	43	APPX-6B	12
OI-47C	34	AOI-3-1	28	APPX-7B	8
OI-68	173	AOI-100-1	123	APPX-7L	6
ARP-3A	59	T.S. 3.1.7	351		
ARP-3C	29	T.S. 3.4.5	253		
ARP-3D	41	TRM 3.3.10	0		
ARP-3F	43	EOI-1	22		
ARP-5B	33	EOI-3	20		
ARP-7A	37	APPX-5B	5		
ARP-20A	34	APPX-5C	8	2-SR-3.1.7.1	12

Simulator Setup	• Start CPERF **PRIOR** to placing the Simulator in RUN • Place protected equipment as listen on the turnover • Ensure proper lens convention on RCW pumps after IC is reset
Schedule Files(s):	ILT 2404 NRC Scenario 1 UNIT 2.sch
Event Files(s):	ILT 2404 NRC Scenario 1 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

MODE 1	DAYS ONLINE	Drywell Leakage (GPM)	Today Protected Equipment
	210		2A Bus Duct Fan (2B Bus Duct Fan OOS)
	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified **Last breaker closure** **11/23/23 05:41**	1.78	
Rx Power 95%		**Floor Drain (GPM)** 0.20	
MWe 1227		**Equipment Drain (GPM)** 1.98	

☐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

None

LCOs OF 72 HOURS OR LESS

None

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2B Bus Duct Fan is tagged out of service for maintenance.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Return Reactor Power to 100% with core flow.

Alternate RCW Pumps, place 2A RCW Pump in service and remove 2B RCW Pump from service.

OPERATOR WORK AROUNDS **OWAs – U0-0/U2-0** **Burdens - U0 – 0/U2-0** **Operator Challenges – U0 – 0/U2 - 0**

None

ODMIs/ACMPs

None

ONEAs

None

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

None

SCHEDULED ITEMS NOT COMPLETED

None

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-2
Scenario Source: NEW Op-Test Number: 24-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 100% Reactor Power. 2B Fuel Pool Cooling Pump tagged out of service. Previous shift entered Technical Specification 3.5.1 Condition 'A' and began performance of 2-SR-3.6.1.3.5, (RHR MOV I) RHR System MOV Operability Loop I, except for 2-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE.

Turnover: Control Rod 46-35 was declared OPERABLE after an HCU leak fix. Lower Reactor Power with Core Flow and move Control Rod 46-35 from position 00 to position 12 in accordance with Reactivity Control Plan (RCP). Complete 2-SR-3.6.1.3.5, (RHR MOV I) RHR System MOV Operability Loop I, Section 6.4 for 2-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE.

Critical Tasks:

1. During failure to SCRAM conditions, inject Standby Liquid Control to shut down the Reactor prior to meeting all Table M-6 conditions.

2. During failure to SCRAM conditions 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	Complete 2-SR-3.6.1.3.5
2.	N/A	R-OATC	Lower Reactor Power with Core Flow
3.	RD06R4635	C-OATC TS-NUSO	Control Rod 46-35 Difficult to Withdraw
4.	PMP-77-1A-A PMP-77-1B-A	C-OATC MC-OATC C-NUSO	Drywell Floor Drain Sump Fails to Automatically Start
5.	PC14	C-BOP TS-NUSO	Suppression Pool Leak
6.	XA-55-7B_15	C-BOP MC-BOP C-NUSO	Electro-Hydraulic Control (EHC) Pump 2A Trip
7.	SW02A SW02B	M-ALL	Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip/Electrical Anticipated Transient Without a SCRAM (ATWS)
8.	PMP-63-6A PMP-63-6B	C-BOP C-NUSO	Standby Liquid Control (SLC) Pump Trip

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. Complete 2-SR-3.6.1.3.5, (RHR MOV I) RHR System MOV Operability Loop I, for 2-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE.

2. The crew will lower Reactor Power with Core Flow in accordance with 2-OI-68, Reactor Recirculation System.

3. Control Rod 46-35 will be difficult to move from position 00. The crew will use 2-OI-85, Control Rod Drive System, Section 8.16, to move Control Rod 46-35 from position 00. Subsequently Control Rod 46-35 will become fully stuck, after attempts are made to move the Control Rod, the NUSO addresses Tech Specs.

4. DRYWELL FLOOR DRAIN SUMP LEVEL ABNORMAL (2-9-4C, Window 2) alarms. The Drywell Floor Drain Sump Pump fails to automatically start on high level and the OATC will manually start the associated Pump after referring to the Alarm Response Procedure (ARP). (The BOP will be removed allowing the OATC to respond).

5. SUPPRESSION CHAMBER WATER LEVEL ABNORMAL (2-ARP-9-3B, Window 15) alarms. The crew may begin to add water to the Suppression Pool with 2-OI-74, Residual Heat Removal System, Section 8.3. Suppression Pool Level will continue to lower until the EOI-2 entry and Tech Spec limit of (-) 6.25 inches is reached. Dispatched personnel will hear flow noise and cinch closed on a valve to stop the leak. The NUSO will address Tech Specs and enter 2-EOI-2, Primary Containment Control. Suppression pool water level will be returned to above the low-level alarm.

6. EHC HYDRAULIC FLUID HEADER PRESSURE LOW (2-ARP-9-7B, Window 1) and STANDBY EHC PUMP FAILED (2-ARP-9-7B, Window 15) alarms. The 2A EHC Pump trips and the crew will manually start the 2B EHC Pump.

7. The crew responds to a Dual RBCCW Pump Trip. Reactor SCRAM results in an Electrical ATWS and Alternate Rod Insertion (ARI) will fail. During ATWS actions the crew lowers Reactor water level to between (-)50 and (-)100 inches when the main turbine will trip. 2-EOI Appendix-1A, REMOVAL AND REPLACEMENT OF RPS SCRAM SOLENOID FUSES, will be the success path.

8. During ATWS actions, the first started SLC Pump trips, the BOP will start the second SLC Pump.

The Scenario ends when the crew inserts all Control Rods and is controlling Reactor Water Level above the Top of Active Fuel (TAF) (-) 162 inches) using any available injection system.

Critical Tasks 2

1. **Critical Task: During failure to SCRAM conditions, inject Standby Liquid Control to shut down the Reactor prior to meeting all Table M-6 conditions.**

   a. **Safety Significance**

   Standby Liquid Control is injected to lower Reactor power such that all Table M-6 conditions are not met preventing a loss of primary containment integrity.

   b. **Cues**

   Failure to inject Standby Liquid Control before **ALL** the following conditions are simultaneously met:

   - Suppression Pool Temperature is above 110 ºF
   - Reactor Power above 5%
   - RPV Level above (-)162 inches
   - MSRV open/cycling OR DW pressure above 2.4 psig

   c. **Measured by:**

   - Suppression Pool Temperature indications
   - Reactor Power on APRM displays
   - Reactor Water Level indications
   - Acoustic Monitor Display OR Drywell Pressure indications

   d. **Feedback:**

   - SLC Flow
   - Lowering Reactor Power
   - At least one Table M-6 condition not met

   e. **Critical Task Failure Criteria:**

   Crew does not inject Standby Liquid Control resulting in meeting all Table M-6 conditions.

2. During failure to SCRAM conditions 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:
ADS valves automatically open, Reactor Pressure rapidly lowers and Reactor power rises.

c. Measured by:
ADS Logic inhibited prior to an automatic initiation, Reactor power and Reactor pressure indications.

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.: 24-04 Scenario No. NRC-2 Event No.: 1 Page 1 of 2

Event Description: Complete 2-SR-3.6.1.3.5

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	**This event is to allow a normal evolution for the BOP, the NUSO is NOT required to review 2-SR-3.6.1.3.5(RHR I). Once the BOP has completed their task, notify the NUSO "Another Operator will continue 2-SR-3.6.1.3.5(RHR I)."**
	BOP	2-SR-3.6.1.3.5(RHR I) Section 6.4 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV OPERABILITY TEST **NOTE** 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV is a throttle valve and does NOT have a seal in circuit for valve operation. To ensure this valve is closed, the handswitch is held in CLOSE position for greater than 6 seconds after CLOSED is indicated. [1] RECORD date and time. [2] RECORD as found position of 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV. As found Position: OPEN □ CLOSE □ [3] ENSURE 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV is CLOSED by holding 2-HS-74-59A, RHR SYS I SUPPR POOL CLG/TEST VLV in CLOSE position for greater than 6 seconds after CLOSED is indicated. [4] OPEN 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV, as follows: [4.1] HOLD 2-HS-74-59A, RHR SYS I SUPPR POOL CLG/TEST VLV, in OPEN position until RHR SYS I SUPPR POOL CLG/TEST VLV, is OPEN. [4.2] OBSERVE 2-FCV-74-59 valve exercise for Abnormal or Erratic Operation (Reference AC 5.0B). [4.3] OBSERVE Red indicating light is ILLUMINATED and Green indicating light is EXTINGUISHED above 2-HS-74-59A. [4.4] RECORD 2-FCV-74-59 observation of valve position acceptance criteria AND observation of Abnormal or Erratic Operation Attachment 1 Table 4 Data Table 2-FCV-74-59.

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 1 Page 2 of 2

Event Description: Complete 2-SR-3.6.1.3.5

Time	Position	Applicant's Actions or Behavior
	BOP	[5] CLOSE 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV, as follows: [5.1] HOLD 2-HS-74-59A, RHR SYS I SUPPR POOL CLG/TEST VLV, in CLOSE position for greater than 6 seconds after red light EXTINGUISHES. [5.2] OBSERVE 2-FCV-74-59 valve exercise for Abnormal or Erratic Operation (Reference AC 5.0B). [5.3] OBSERVE Green indicating light is ILLUMINATED and Red indicating light is EXTINGUISHED above 2-HS-74-59A. [5.4] RECORD 2-FCV-74-59 observation of valve position acceptance criteria AND observation of Abnormal or Erratic Operation [6] IF valve position acceptance criteria results were acceptable for CLOSED and OPEN, THEN: MARK Full Cycle Exercise Acceptance Criteria Satisfied as "YES" in Attachment 1 Table 4 Data Table 2-FCV-74-59. (Otherwise, N/A) [7] IF valve position acceptance criteria was NOT acceptable for CLOSED or OPEN, THEN: [7.1] MARK Full Cycle Exercise Acceptance Criteria Satisfied as "NO" in Attachment 1 Table 4 Data Table2-FCV-74-59. (Otherwise, N/A) [7.2] PERFORM Attachment 3, Requirements for MOV Exercise Failure. (Otherwise, N/A) [8] IF Abnormal or Erratic Action, for CLOSED or OPEN, is marked YES in Attachment 1 Table 4 Data Table 2-FCV-74-59, THEN, PERFORM Attachment 2, Requirements for Abnormal or Erratic Action. (Otherwise, N/A) [9] RETURN 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VALVE, to as found position indicated in Step 6.4[2], using 2-HS-74-59A, RHR SYS I SUPPR POOL CLG/TEST VLV. [10] RECORD date and time.
	NRC	**End of Event 1. Proceed to event 2, Lower Reactor Power with Core Flow**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 2 Page 1 of 1

Event Description: Lower Reactor Power with Core Flow

Symptoms/Cues: Crew is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	NRC	**NOTE: The crew has pre-briefed but may elect to hold a re-focus brief.**
	NUSO	Directs the Operator at the Controls (OATC) to lower Reactor Power to 95% in accordance with 2-OI-68, Reactor Recirculation System and Reactivity Control Plan (RCP) RCP-NRC2404-U2S2.
	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 SLOW(MEDIUM)(FAST), 2-HS-96-17A(17B) (17C). (Otherwise, N/A). AND/OR • LOWER Recirc Pump 2B using SLOW(MEDIUM)(FAST), 2-HS-96-18A(18B) (18C). (Otherwise, 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 pushbuttons as required. • LOWER SLOW, 2-HS-96-33 • LOWER MEDIUM, 2-HS-96-34 • LOWER FAST, 2-HS-96-35
	NRC	**End of Event 2. Proceed to Event 3, Control Rod 46-35 Difficult to Withdraw**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 3 Page 1 of 6

Event Description: Control Rod 46-35 Difficult to Withdraw

Symptoms/Cues: Crew is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	Driver	**No action required. Control Rod 46-35 Difficult to Withdraw is part of the schedule file. When the crew double clutches the control rod, it will unstick and move from 00 to 02 and then become stuck again.**
	NRC	**The crew has pre-briefed Event 3 but may elect to conduct a re-focus brief. The crew may also elect to review 2-OI-85, sections 6.6.1, Initial Conditions prior to withdrawing Control Rods and 6.6.2, Actions Required during and Following Control Rod Withdrawal then continue to 6.6.3 Control Rod Notch Withdrawal** **The crew may elect to Notch Rod Withdrawal or Continuous Rod Withdrawal**
	NUSO	Directs the Operator at the Controls (OATC) to position Control Rod 46-35 in accordance with 2-OI-85, Control Rod Drive System and RCP-NRC2404-U2S2.
	OATC	2-OI-85, Control Rod Drive System Section 6.6.2, Actions Required during and Following Control Rod Withdrawal **NOTES** 1) Section 6.6.2 is applicable for all control rod withdrawals and addressed as required during or following any control rod withdrawal. 2) If rod insertion to Position 00 is required and core thermal power is ≤ 10%, entry into LCO 3.1.6 may be required. [1] IF control rod fails to withdraw, THEN: Refer to Section 8.16 for additional methods to reposition control rod. [2] N/A [3] N/A [4] N/A [5] N/A [6] N/A
	NRC	**NOTE: No action is taken initially on this page.** **The OATC will refer to the above section once the difficult control rod is found.**
	NRC	**Section 6.6.3, Control Rod Notch Withdrawal is on page 5 of this guide.** **Section 6.6.4, Continuous Rod Withdrawal is on page 6 of this guide.** **Section 8.16, Control Rod Difficult to Withdraw is on page 7 of this guide.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 3 Page 2 of 6

Event Description: Control Rod 46-35 Difficult to Withdraw

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 6.6.3, Control Rod Notch Withdrawal [1] SELECT the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40. [2] ENSURE CRD DRIVE WTR HDR DP is between 250 -270 psid on 2-PDI-85-17A by throttling CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A, as needed. [3] N/A [4] OBSERVE the following for selected control rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED. • White light on the Full Core Display ILLUMINATED • Rod Out Permit light ILLUMINATED. [5] N/A [6] PLACE CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH and RELEASE. [7] N/A [8] N/A [9] N/A
	NRC	**After Control Rod 46-35 fails to move, proceed to page 7 for Section 8.16, Control Rod Difficult to Withdraw**

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 3 Page 3 of 6

Event Description: Control Rod 46-35 Difficult to Withdraw

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 6.6.4, Continuous Rod Withdrawal **NOTES** 1) Continuous control rod withdrawal may be used when a control rod is to be withdrawn greater than three notches. 2) When in areas of high notch worth, single notch withdrawal should be used instead of continuous rod withdrawal. Information concerning high notch worth is identified by Reactor Engineering in Control Rod Coupling Integrity Check, 2-SR-3.1.3.5A. 3) When continuously withdrawing a control rod to a position other than position 48, the CRD Notch Override Switch is held in the Override position and then the CRD Control Switch is held in the Rod Out Notch position. • Both switches should be released when the control rod reaches two notches prior to its intended position. (Example: If a control rod is to be withdrawn from position 00 to position 12, the CRD Notch Override Switch and the CRD Control Switch would be used to move the control rod until reaching position 08, then both switches would be released.) • If the rod settles in a notch prior to the intended position, the CRD Control Switch should be used to withdraw the rod to the intended position. (using the above example; If the control rod settles at a notch prior to the intended position of 12, the CRD Control Switch would be used to withdraw the control rod to position 12.) [1] SELECT the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40. [2] OBSERVE the following for selected control rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED. • White light on the Full Core Display ILLUMINATED • Rod Out Permit light ILLUMINATED. [3] N/A [4] CHECK Control Rod is being withdrawn to a position greater than three notches. [5] IF withdrawing the control rod to a position other than "48", THEN: PERFORM the following: [5.1] PLACE and HOLD CRD NOTCH OVERRIDE, 2-HS-85-47, in OVERRIDE. [5.2] PLACE and HOLD CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH.
	NRC	**After Control Rod 46-35 fails to move, proceed to page 7 for Section 8.16, Control Rod Difficult to Withdraw**

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 3 Page 4 of 6

Event Description: Control Rod 46-35 Difficult to Withdraw

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 8.16, Control Rod Difficult to Withdraw [1] CHECK the control rod will not notch out. Refer to Section 6.6. [2] REVIEW all Precautions and Limitations in Section 3.0 [3] N/A **NOTE** Steps 8.16[4] through 8.16[6] should be used when the control rod is at Position 00 Step 8.16[7] should be used when the control rod is at Position 02 through 46. [4] IF the control rod problem is not believed to be air in the hydraulic system, THEN, PERFORM the following to double clutch the control rod at Position 00: [4.1] PLACE AND HOLD CRD NOTCH OVERRIDE, 2-HS-85-47, in EMERG ROD IN, for several seconds. [4.2] CHECK the control rod full in indication (double green dashes) on the Full Core Display for the associated control rod. [4.3] SIMULTANEOUSLY PLACE CRD NOTCH OVERRIDE, 2-HS-85-47, in NOTCH OVERRIDE AND CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH. [4.4] WHEN EITHER of the following occur: • Control rod begins to move, OR • It is determined the rod will not move, THEN RELEASE 2-HS-85-47 AND 2-HS-85-48. [4.5] IF the control rod successfully notches out, THEN PROCEED TO Section 6.6 and WITHDRAW the control rod to the appropriate position.
	NRC	**Initially Control Rod 46-35 is at position 00 and the crew uses step [4.0] of Section 8.16, Control Rod Difficult to Withdraw** **When returning to section 8.16, Control Rod 46-35 is beyond position 00 and the crew will use step [7] located on the following page.**

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 3 Page 5 of 6

Event Description: Control Rod 46-35 Difficult to Withdraw

Time	Position	Applicant's Actions or Behavior
		[7] IF the control rod is at or between Positions 02 and 46, THEN PERFORM the following to withdraw the control rod using elevated drive water pressure: [7.1] RAISE the CRD DRIVE WTR HDR DP, 2-PDI-85-17A, to 300 psid, using CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A. [7.2] ATTEMPT to withdraw the Control Rod using SWITCH, 2-HS-85-48. [7.3] N/A
	OATC	[7.4] IF the control rod still fails to NOTCH OUT, THEN RAISE CRD DRIVE WTR HDR DP, 2-PDI-85-17A, to 350 psid, using CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A. [7.5] ATTEMPT to withdraw the Control Rod using SWITCH, 2-HS-85-48. [7.6] LOWER 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. [7.7] IF the control rod still fails to NOTCH OUT using elevated CRD DRIVE WTR HDR DP, THENCONTACT Reactor Engineer and NOTIFY Unit SRO for further instructions
	NRC	**The NUSO should enter Tech Spec 3.1.3 CONDITION A. One withdrawn control rod stuck. REQUIRED ACTION:** **A.1 Verify stuck control rod separation criteria are met, Immediately** **AND** **A.2 Disarm the associated control rod drive (CRD) in 2 hours** **AND** **A.3 Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod in 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM** **AND** **A.4 Perform SR 3.1.1.1. in 72 hours**

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 3 Page 6 of 6

Event Description: Control Rod 46-35 Difficult to Withdraw

Time	Position	Applicant's Actions or Behavior
	NUSO	LCO 3.1.3 Each control rod shall be OPERABLE. APPLICABILITY: MODES 1 and 2. ACTIONS ----------NOTE---------- Separate Condition entry is allowed for each control rod. ----------

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	----------NOTE---------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ---------- A.1 Verify stuck control rod separation criteria are met. AND A.2 Disarm the associated control rod drive (CRD). AND	Immediately 2 hours (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod. AND A.4 Perform SR 3.1.1.1.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM 72 hours

	NRC	**End of Event 3. Proceed to Event 4, Drywell Floor Drain Sump Fails to Automatically**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 4 Page 1 of 1

Event Description: Drywell Floor Drain Sump Fails to Automatically Start

Symptoms/Cues: DRYWELL FLOOR DRAIN SUMP LEVEL ABNORMAL (2-9-4C, Window 2) alarms and the Drywell Floor Drain Sump Pump fails to automatically start on low level.

Time	Position	Applicant's Actions or Behavior
	Driver	**NOTE: Two triggers will be used for event 4, first trigger 24 and then trigger 4, each at the Chief Examiners Direction. Trigger 24 is to distract the BOP to allow the OATC to respond to event 4, Drywell Floor Drain Sump Fails to Automatically Start** • **When requested by the Chief Examiner, insert Event 24 to bring in a spurious alarm on 9-53.** • **When requested by the Chief Examiner, insert Event 4 to fail Drywell Floor Drain Sump Pump from automatically starting on low level.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • CHARCOAL BED AFTER FILTER DIFF PRESS HIGH (2-9-53, Window 18)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	NRC	**Based on spurious alarm, CHARCOAL BED AFTER FILTER DIFF PRESS HIGH (2-9-53, Window 18) resetting, the crew may or may not elect to dispatch personnel to the Off-Gas Bldg.**
	OATC	Acknowledges and reports the following alarm(s) to the NUSO: DRYWELL FLOOR DRAIN SUMP LEVEL ABNORMAL (2-9-4C, Window 2)
	NUSO	Directs OATC to respond in accordance with the Alarm Response Procedure(s).
	NRC	**Drywell Floor Drain Pumps fail to auto start once valve path is aligned via ARP The OATC should invoke OPDP-1, an automatic action is confirmed to have malfunctioned and take action to start one or both Drywell Floor Drain Pumps**
	OATC	Alarm Response Procedure: DRYWELL FLOOR DRAIN SUMP LEVEL ABNORMAL (2-9-4C, Window 2) Operator Actions: A. IF the level is high, THEN PERFORM the following: 1. OPEN DW FLOOR DRAIN OUTBD ISOL VALVE using 2-HS-77-2B 2. CHECK the sump pumps running, discharge valve open, and flow indicated on recorder DRYWELL EFFLUENT FLOW, 2-FR-77-6. 3. ENSURE sump pumps(s) stop on sump low level 4. CLOSE DW FLOOR DRAIN OUTBD ISOL VALVE using 2-HS-77-2B B. N/A C. IF level is unknown, DISPATCH personnel locally to determine sump level. D. N/A
	Driver	**When contacted as the Reactor Building AUO to determine Drywell Floor Drain Sump level, report "LIS-77-1A, Drywell Floor Drain Sump level is 36 inches".**
	NRC	**End of Event 4. Proceed to Event 5, Suppression Pool Leak.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 5 Page 1 of 4

Event Description: Suppression Pool Leak

Symptoms/Cues: SUPPR CHAMBER WATER LEVEL ABNORMAL (2-ARP-9-3B, Window 15) alarms.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to begin Suppression Pool Leak**
	NRC	**Approximately 2 and 3 minutes respectively will elapse before the below listed alarms annunciate.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • DRYWELL TO SUPPR CHAMBER. DIFF PRESS ABNORMAL (2-9-3B, Window 26) • SUPPR CHAMBER WATER LEVEL ABNORMAL (2-ARP-9-3B, Window 15)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: DRYWELL TO SUPPR CHAMBER. DIFF PRESS ABNORMAL (2-9-3B, Window 26) Operator Actions: A. CHECK alarm by checking Drywell to Suppression Chamber DP. B. N/A. C. N/A
	BOP	Alarm Response Procedure: SUPPR CHAMBER WATER LEVEL ABNORMAL (2-ARP-9-3B, Window 15) Operator Actions: 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. N/A

Form 3.3-2 Required Operator Actions

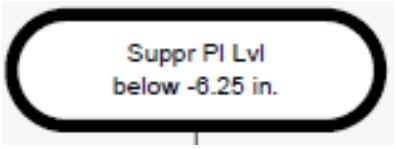
Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 5 Page 2 of 4

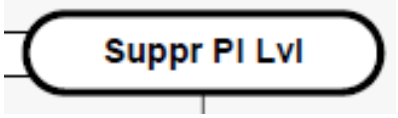
Event Description: Suppression Pool Leak

Time	Position	Applicant's Actions or Behavior
	Driver	**When dispatched to investigate the Suppression Pool leak, wait three minutes, and notify the control room, "I heard flow noise near RHR PUMP D SUCT DRAIN VLV, 2-SHV-74-525D, I cinched closed on the valve and the flow noise stopped."**
	NUSO	Directs BOP to add water to the Suppression Pool in accordance with 2-OI-74, Residual Heat Removal System Section 8.3 Adding Water to the Suppression Pool Through RCIC(HPCI) Minimum Flow Line
	BOP	2-OI-74, Residual Heat Removal System Section 8.3 Adding Water to the Suppression Pool Through RCIC(HPCI) Minimum Flow Line **NOTES** 1) This procedure should be used to add water to the suppression pool when the PSC Head Tank is in service. 2) When adjusting suppression pool level, Drywell to suppression pool differential pressure should be maintained 1.15 to 1.30 psid. 3) All operations are performed at Panel 2-9-3 unless otherwise noted. [1] ENSURE RCIC(HPCI) surveillance testing is NOT in progress. [2] NOTIFY Chemistry Section that an addition will be made to the suppression pool. [3] ENSURE OPEN RCIC CST SUCTION VLV 2-FCV-71-19, (HPCI CST SUCTION VALVE, 2-FCV-73-40). [4] N/A [5] IF RCIC OR HPCI are NOT isolated, THEN: OPEN RCIC PUMP MINIMUM FLOW VLV, 2-FCV-71-34(HPCI MAIN PUMP MINIMUM FLOW VLV, 2-FCV-73-30). [6] WHEN suppression pool level is -2 to -5.5 inches, or as desired, THEN CLOSE 2-FCV-71-34(2-FCV-73-30). [7] N/A
	NRC	**Based on leak rate, the crew may not add water with 2-OI-74, Residual Heat Removal System before entering EOI-2, Primary Containment Control and add water with EOI APPENDIX 18** **The crew may attempt Drywell/Suppression Chamber pressure equalization by using 2-OI-64, Primary Containment System Section 6.3. Only one action exists, OPEN DRYWELL DP COMP BYPASS VALVE, 2-FCV-64-141, using 2-HS-64-141 (Panel 2-9-3).**

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 5 Page 3 of 4

Event Description: Suppression Pool Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
Enters EOI-2, Primary Containment Control based on Suppression Pool level below (-) 6.25 inches.


<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>It is determined that core damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr>
</table>


<table>
<tr><td colspan="2">MONITOR and CONTROL suppression pool level (-)6.25” to (-)1” (APPX 18)</td></tr>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression pool level CANNOT be maintained below (-)1”</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppression pool level CANNOT be maintained below (-)6.25”</td><td> 
</td></tr>
</table>
SP/L-1


<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Suppression pool level CANNOT be above 12.75 feet</td><td>NO ACTION REQUIRED</td></tr>
</table>
SP/L-15
<table>
<tr><td>MAINTAIN suppression pool level above 11.5 ft (APPX 18, 20J)</td></tr>
</table>
SP/L-16
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs BOP to add water to the Suppression Pool with 2-EOI Appendix-18, Suppression Pool Water Inventory Removal and Makeup</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 5 Page 4 of 4

Event Description: Suppression Pool Leak

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI APPENDIX-18, Suppression Pool Water Inventory Removal and Makeup [1] IF Suppression Pool Water makeup is required, THEN CONTINUE in this procedure at Step 1.0[5]. [5] N/A [6] IF Directed by SRO to add water to suppression pool, THEN MAKEUP water to Suppression Pool as follows: [6.1] ENSURE OPEN 2-FCV-73-40, HPCI CST SUCTION VALVE. [6.2] OPEN 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE. [6.3] IF HPCI is NOT available for Suppression Pool makeup, THEN MAKEUP water to Suppression Pool using RCIC as follows: [6.3.1] ENSURE OPEN 2-FCV-71-19, RCIC CST SUCTION VALVE. [6.3.2] OPEN 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE. [6.4] N/A [7] WHEN Suppression Pool level reaches -5.5 in., THEN ENSURE CLOSED the following valves: • 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE • 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE
	NRC	**The NUSO should enter Tech Spec 3.6.2.2 CONDITION A. Suppression pool water level not within limits, REQUIRED ACTION A.1 Restore suppression pool water level to within limits, COMPLETION TIME, 2 hours.**
		LCO 3.6.2.2 Suppression pool water level shall be ≥ -6.25 inches with and -7.25 inches without differential pressure control and ≤ -1.0 inches. APPLICABILITY: MODES 1, 2, and 3. ACTIONS CONDITION / REQUIRED ACTION / COMPLETION TIME A. Suppression pool water level not within limits. — A.1 Restore suppression pool water level to within limits. — 2 hours
	NRC	**End of Event 5. Proceed to Event 6 Electro-Hydraulic Control Pump 2A Trip**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 6 Page 1 of 1

Event Description: Electro-Hydraulic Control (EHC) Pump 2A Trip

Symptoms/Cues: EHC HYDRAULIC FLUID HEADER PRESSURE LOW (2-ARP-9-7B, Window 1) and STANDBY EHC PUMP FAILED (2-ARP-9-7B, Window 15) alarm.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to trip Electro-Hydraulic Control (EHC) Pump 2A Trip**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • EHC HYDRAULIC FLUID HDR PRESSURE LOW (2-ARP-9-7B, Window 1) • STANDBY EHC PUMP FAILED (2-ARP-9-7B, Window 15)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: EHC HYDRAULIC FLUID HDR PRESSURE LOW (2-ARP-9-7B, Window 1) Operator Action: A. ENSURE Standby EHC PUMP 2B(2A), 2-HS-47-2A(1A) running. B. CHECK EHC HDR PRESSURE indicator, 2-PI-47-7 between 1550 and 1650 psig. C. DISPATCH personnel to inspect EHC pump unit. D. N/A
	BOP	Alarm Response Procedure: STANDBY EHC PUMP FAILED (2-ARP-9-7B, Window 15) Operator Action: A. PERFORM the following on Panel 2-9-7: 1. CHECK alarm by checking EHC HEADER PRESSURE indicator, 2-PI-47-7. 2. CHECK EHC PUMP 2B, 2-HS-47-2A and/or EHC PUMP 2A, 2-HS-47-1A 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 6. Proceed to Event 7 Dual Reactor Building Closed Cooling Water (RBCCW) Pump. Trip/Electrical Anticipated Transient Without a SCRAM (ATWS)**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 1 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Symptoms/Cues: White disagreement light indicates above the 2A (2B) RBCCW pump handswitches. Distinct Motor Tripout alarm sounds.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to trip both 2A (2B) RBCCW pumps**
	BOP	Silence and acknowledge multiple alarms. Report to the NUSO that both 2A (2B) RBCCW pumps have tripped
	NUSO	Directs entry into 2-AOI-70-1, Loss of Reactor Building Closed Cooling Water
	Driver	**If contacted as Unit 1 Operator and requested to place in service the spare RBCCW pump to Unit 2, reply "The spare RBCCW pump in unavailable".**
	BOP/ OATC	2-AOI-70-1, Loss of Reactor Building Closed Cooling Water Immediate Actions [1] IF RBCCW Pump(s) has tripped OR RBCCW SECTIONALIZING VLV, 2-FCV-70-48 is closed/closing due to low pressure, THEN PERFORM the following: • SECURE RWCU Pumps. • ENSURE RBCCW SECTIONALIZING VLV, 2-FCV-70-48 CLOSED. Subsequent Actions [1] IF Reactor is at power AND Drywell Cooling cannot be immediately restored, THEN PERFORM the following: [1.1] IF core flow is above 60%, THEN REDUCE core flow to between 50-60%. [1.2] MANUALLY SCRAM the Reactor and PLACE Mode Switch in SHUTDOWN. REFER TO 2-AOI-100-1. [1.3] SHUT DOWN both Recirc Pumps. • DEPRESS RECIRC DRIVE 2A SHUTDOWN, 2-HS-96-19. • DEPRESS RECIRC DRIVE 2B SHUTDOWN, 2-HS-96-20.
	NRC	**NOTE: OATC will reduce core flow and SCRAM. BOP performs all other actions above for 2-AOI-70-1, Loss of Reactor Building Closed Cooling Water.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 2 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	Driver	**When the OATC attempts a manual Reactor SCRAM, an electrical ATWS will occur. No action is required by the Driver**
	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.** **ARI initiation fails. Control Rods will be inserted via EOI Appendices-1-A(1-D), page 26 or EOI Appendix-1G, page 28**
	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. [2] PLACE REACTOR MODE SWITCH, 2-HS-99-5A/S1, in SHUTDOWN. [3] IF all control rods can NOT be verified fully inserted, THEN INITIATE ARI. [4] N/A [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. [5.2] ENSURE shutdown both Recirc Pumps [5.3] STOP and PREVENT injection from CONDENSATE/FEEDWATER per APP 4.
	BOP	[5.4] STOP and PREVENT injection from HPCI, RHR, and CS per APP 4.
	CREW	**Critical Task: During failure to SCRAM conditions, inject Standby Liquid Control to shut down the Reactor prior to meeting all Table M-6 conditions.** **Critical Task Failure Criteria: Crew does not inject Standby Liquid Control resulting in meeting all Table M-6 conditions.**
	BOP	[5.5] INITIATE SLC and ENSURE injection.

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 3 of 16

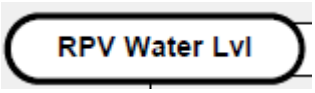
Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

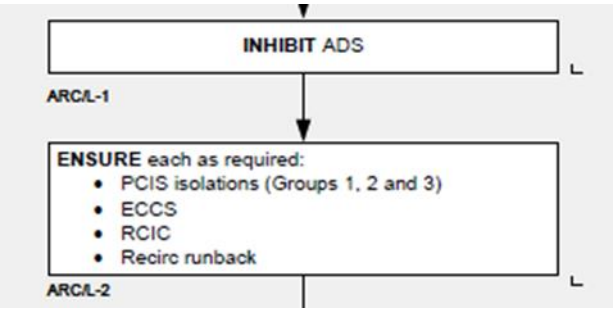
Time	Position	Applicant's Actions or Behavior
	CREW	**Critical Task: During failure to SCRAM conditions and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.** **Critical Task Failure Criteria: ADS automatic initiation with Control Rods out.**
	BOP	[5.6] INHIBIT ADS LOGIC.
	BOP	[5.7] BYPASS Grp 1 RPV Low-Low-Low Level isolation interlocks per APP 8A
	OATC	[5.8] WHEN 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 IS **OR** IS NOT Injecting. [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 IS **OR** IS NOT Injecting.
	BOP	[5.10] IF Any Main steam line is OPEN, THEN CALL for APP 8E, Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation interlocks

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 4 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NRC

DRIVER</td><td>When the crew lowers Reactor Water level below (-50) inches, a Main Turbine Trip is automatically triggered after 60 seconds.</td></tr>
<tr><td></td><td>DRIVER</td><td>Insert the Events noted below as called for when contacted as the Work Control / Outside NUSO to perform the EOI Appendices.
2-EOI APPENDIX-1A - Event 11
2-EOI APPENDIX-1B - No Event
2-EOI APPENDIX-1D - No Event
2-EOI APPENDIX-1G - No Event
2-EOI APPENDIX-8E - Event 26</td></tr>
<tr><td></td><td>NRC</td><td>EOI Appendices related to Control Rod insertion are covered starting on page 26 of 36.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-EOI-1A, ATWS RPV Control.
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV pressure is or will be lowering</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is determined that core damage is occurring due to loss Adequate Core Cooling (Table M 3)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>The reactor will 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>
</table>
ARC-2
 

ARC-3
 

</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 5 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior

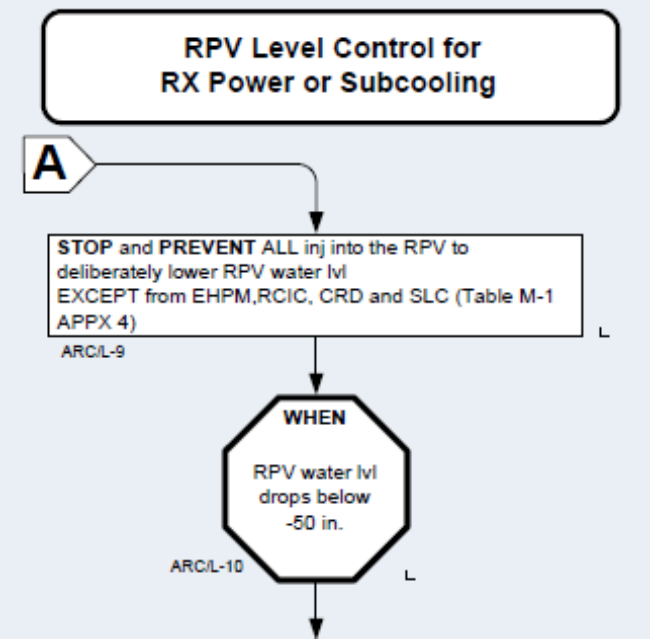
NUSO

IF	THEN
Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7	**NO ACTION REQUIRED**

ARC/L-3

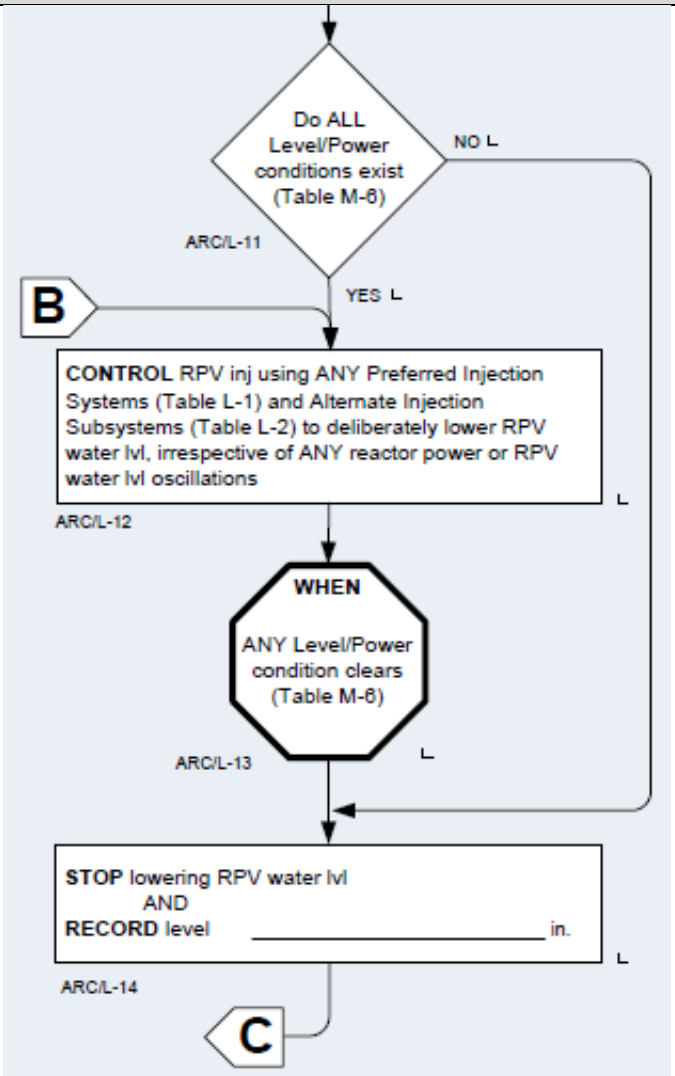
IF	THEN
Reactor Power is above 5% or unknown AND RPV Water Level is above (-) 50 in.	A
ALL Level/Power conditions exist (Table M-6)	**NO ACTION REQUIRED**
Reactor Power is above 10% or unknown AND RPV Water Level is below (-) 162 inches.AND MSRV open/cycling OR DW pressure >2.4 psig	**NO ACTION REQUIRED**

ARC/L-4



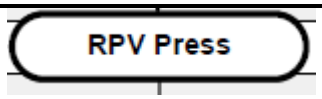
Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 6 of 15

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	Driver	
	NUSO	  RESTORE and MAINTAIN RPV water level between (-)180 inches and: Lowered level (if level was deliberately lowered in flow path A or B) OR +51 in. (if level was NOT deliberately lowered)
	NUSO	Directs the crew to maintain Reactor Water Level in any band between (-) 50 and (-) 180 inches.

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 7 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a 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>


<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>EMERGENCY RPV DEPRESSURIZATION IS REQUIRED</td><td>NO ACTION REQUIRED</td></tr>
</table>
ARC/P-1
<table>
<tr><td>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 (APPX 11A)</td></tr>
</table>
ARC/P-2
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>BORON INJECTION IS REQUIRED
AND
The Main Condenser is available.
AND
There has been no indication of a steam line break</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>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</td><td>NO ACTION REQUIRED</td></tr>
</table>
ARC/P-3
<table>
<tr><td>RESTORE and MAINTAIN RPV Pressure between 1073 psig and 350 psig using Main Turbine Bypass Valves (APPX 8B) and maintain cooldown 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</td></tr>
</table>
<table>
<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>
ARC/P-4
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs Reactor Pressure control with 2-EOI APPENDIX-8B Main Turbine Bypass Valves and/or 2-EOI APPENDIX-11A, Alternate Pressure Control System MSRVs.</td></tr>
</table>

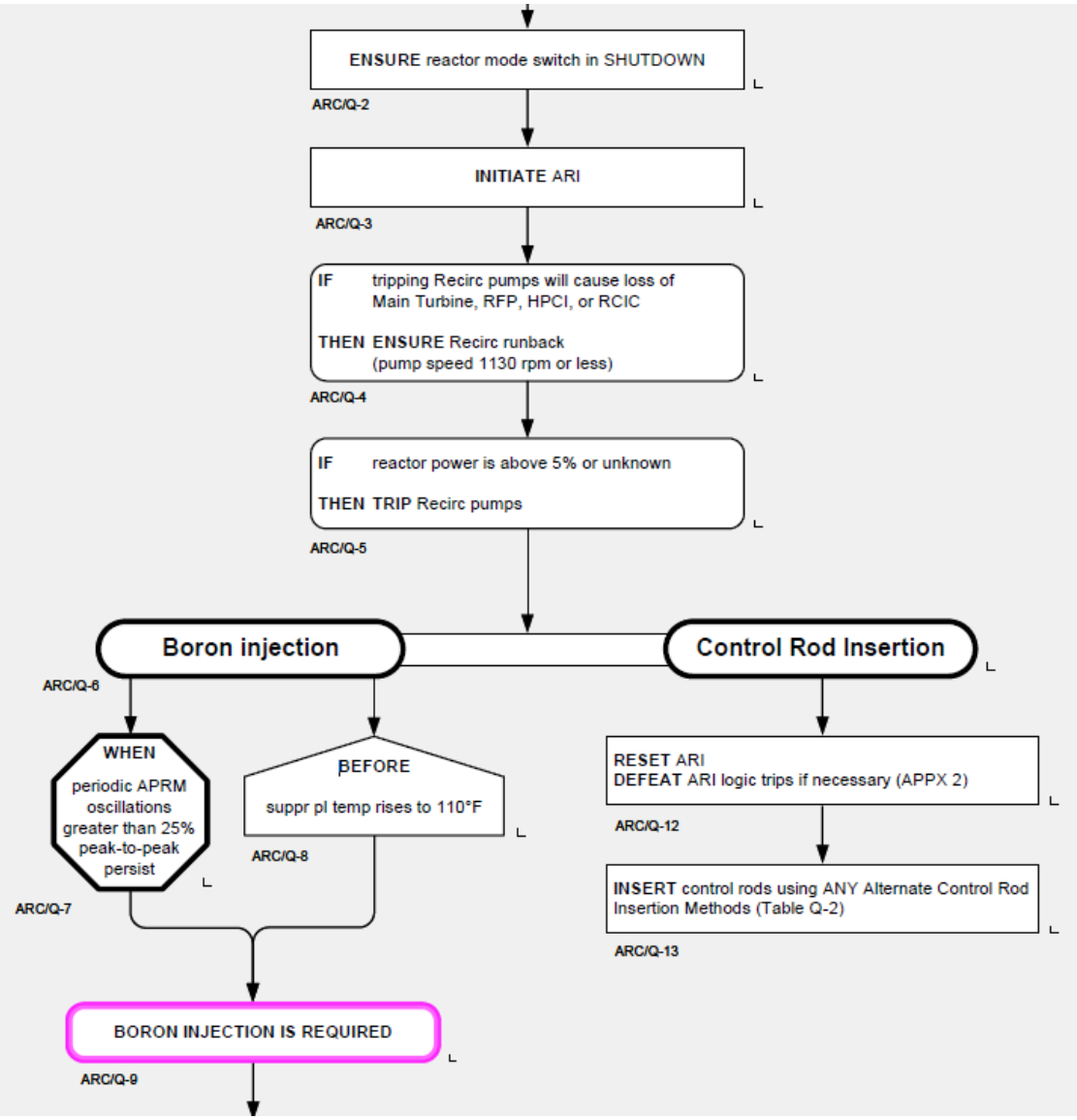
Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 8 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NRC	**Actions for 2-EOI APPENDIX-8B Main, Turbine Bypass Valves are not listed. Bypass valve operation is automatic, the crew will monitor Rector Pressure and supplement with 2-EOI APPENDIX-11A, Alternate Pressure Control System MSRVs as needed.**
	BOP	[1] N/A [2] N/A [3] OPEN MSRVs using the following sequence to control RPV pressure as directed by SRO: 1 2-PCV-1-179 MN STM LINE A RELIEF VALVE ☐ 2 2-PCV-1-180 MN STM LINE D RELIEF VALVE ☐ 3 2-PCV-1-4 MN STM LINE A RELIEF VALVE ☐ 4 2-PCV-1-31 MN STM LINE C RELIEF VALVE ☐ 5 2-PCV-1-23 MN STM LINE B RELIEF VALVE ☐ 6 2-PCV-1-42 MN STM LINE D RELIEF VALVE ☐ 7 2-PCV-1-30 MN STM LINE C RELIEF VALVE ☐ 8 2-PCV-1-19 MN STM LINE B RELIEF VALVE ☐ 9 2-PCV-1-5 MN STM LINE A RELIEF VALVE ☐ 10 2-PCV-1-41 MN STM LINE D RELIEF VALVE ☐ 11 2-PCV-1-22 MN STM LINE B RELIEF VALVE ☐ 12 2-PCV-1-18 MN STM LINE B RELIEF VALVE ☐ 13 2-PCV-1-34 MN STM LINE C RELIEF VALVE ☐ [4] N/A [5] N/A [6] N/A
	NUSO	**IF**: The Reactor is subcritical AND NO boron has been injected **THEN**: **NO ACTION REQUIRED** ARC/Q-1

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 9 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 10 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	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
		Table Q-2 **Alternate Control Rod Insertion Methods** 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	Verifies Applicable steps of ARC/Q Reator Power leg actions complete Driects, as appropriate, one or more of the following Table Q-2 Alternate Control Rod insertion Methods: • 2-EOI APPENDIX-1A, Removal and Replacement of RPS SCRAM Solenoid Fuses • 2-EOI APPENDIX-1B, Venting and Repressurizing the SCRAM Pilot Air Header • 2-EOI APPENDIX-1G, Control Rod Insertion Using Increased Cooling Water Differential Pressure • 2-EOI APPENDIX-1D, Insert Control Rods Using Reactor Manual Control System
	NRC	**If called for, no verifiable control room actions exists for 2-EOI APPENDIX-1B, Venting and Repressurizing the SCRAM Pilot Air Header**

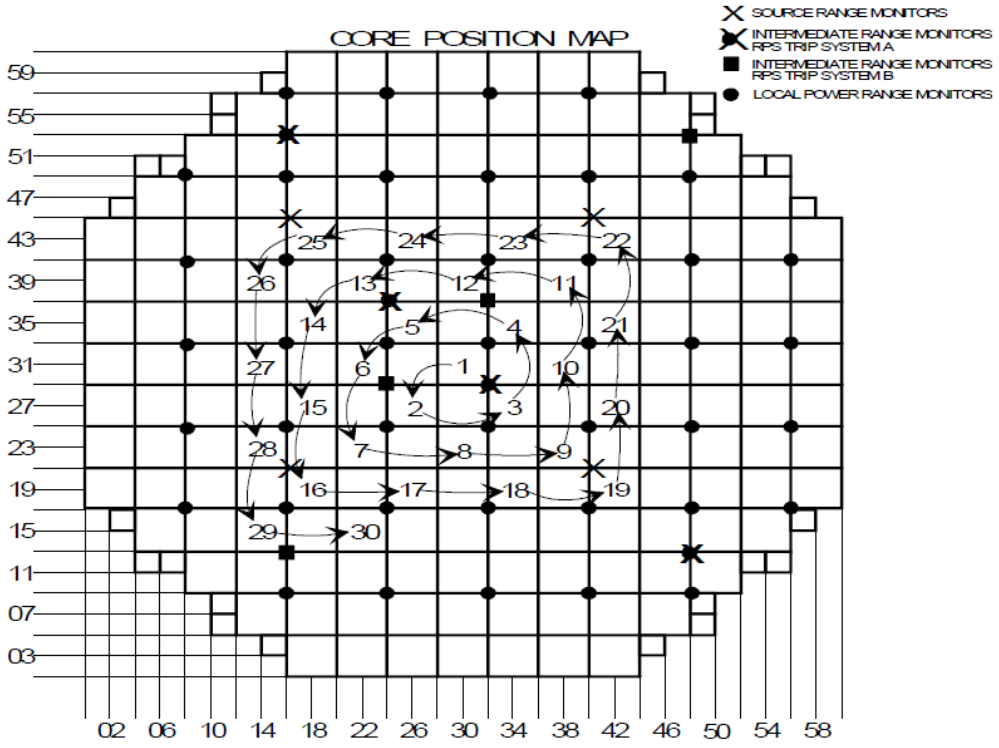
Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 11 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NRC	**Only one control room action exists for 2-EOI APPENDIX-1A, Removal and Replacement of RPS SCRAM Solenoid Fuses**
	OATC/ BOP	2-EOI APPENDIX-1A, Removal and Replacement of RPS SCRAM Solenoid Fuses [1] VERIFY CLOSED Scram Discharge Volume Vent and Drain Valves at the SCRAM DISCHARGE VOLUME VENT/DRAIN VLVS display on panel 2-9-5.
	OATC	2-EOI APPENDIX-1D, Insert Control Rods Using Reactor Manual Control System **NOTES** 1. This EOI Appendix may be executed concurrently with EOI Appendix 1A or 1B at SRO's discretion when time and manpower permit. 2. Closing 2-85-586, CHARGING WATER ISOL valve may reduce the effectiveness of EOI Appendix 1A or 1B. [1] VERIFY at least one CRD pump in service. [2] N/A [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] N/A

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 12 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	OATC	**Attachment 2** **(Page 1 of 1)** **Core Position Map**  (EXAMPLE ONLY) **NOTES** 1) This Attachment is ONLY an aid to Unit RO/SRO when the Reactor Engineer is NOT available. USE is NO mandatory and it does NOT cover ALL situations for prioritizing rod insertion sequences during ATWS. 2) Where power distribution and control rod positions are distributed throughout the full core, control rods are inserted in spiral sequence, starting at the center of the core. Each insertion should be separated by one cell from the previous insertion. 3) Where local power peaks exist and control rods are withdrawn ONLY in a local area, each control rod inserted in the affected area should be from the center of the area, separated by one cell from the previous insertion.

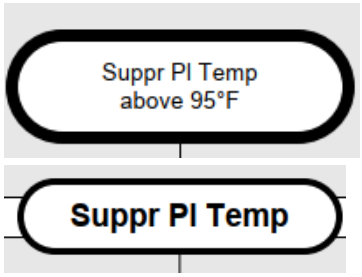
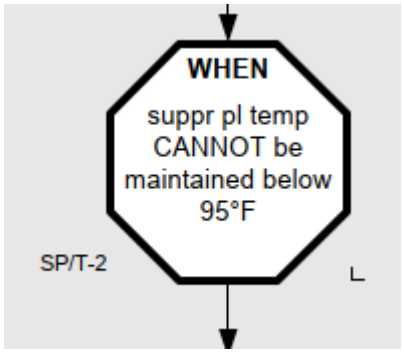
Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 13 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NRC	**The NUSO should not direct 2-EOI APPENDIX-1G *IF* EOI APPENDIX-1D is being performed**
	NUSO	2-EOI APPENDIX-1G, Control Rod Insertion Using Increased Cooling Water Differential Pressure [1] IF EOI Appendix-1D is being performed, THEN EXIT this procedure. [2] N/A [3] ENSURE at least one CRD pump running. [4] PLACE second CRD pump in service. [4.1] OPEN CRD PUMP 1B DISCH TO U2 using 2-HS-85-8A, as necessary. [5] OPEN 2-FCV -85-11, CRD SYSTEM FLOW CONTROL VALVE by performing the following: [5.1] DEPRESS CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, MANUAL operation mode transfer switch to Manual. [5.2] ADJUST CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, using Manual Control MV key until FULL OPEN position is indicated. [6] FULLY OPEN 2-PCV-85-23, CRD DRIVE WATER PRESS CONTROL VLV. [7] ENSURE CLOSED 2-FCV-85-50, CRD EXH RTN LINE SHUTOFF VALVE. [8] MONITOR control rod position and INFORM SRO of results. [9] IF during the performance of this appendix, the SCRAM is reset, THEN PERFORM either of the following: A. RETURN CRD system to normal and exit this procedure. OR B. DISPATCH personnel to FULLY OPEN the following valves as required: • 2-THV-085-0527, PUMP DISCH THROTTLING (RB NE, El 565') • 2-BYV-085-0551, PUMP TEST BYPASS (RB NE, El 565').

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 14 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	NUSO	 MONITOR and CONTROL Suppr Pool Temperature below 95°F using available Suppression Pool Cooling (APPX 17A) SP/T-1  OPERATE all available Suppression Pool Cooling using only RHR Pumps NOT required for continuous injection (APPX 17A) SP/T-3
	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.

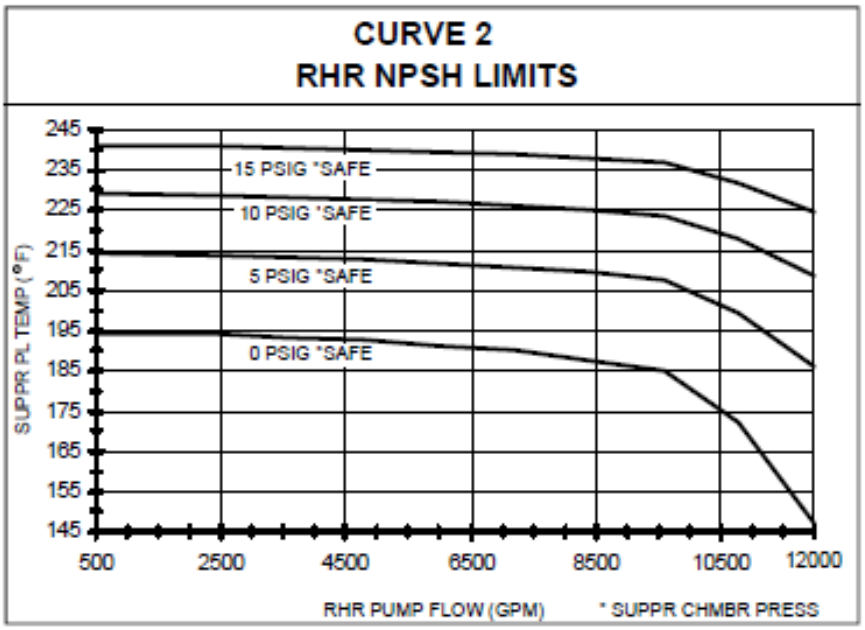
Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7 Page 15 of 16

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	BOP	[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 [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.

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 7

Event Description: Dual Reactor Building Closed Cooling Water (RBCCW) Pump Trip, Electrical Anticipated Transient Without a SCRAM (ATWS)

Time	Position	Applicant's Actions or Behavior
	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]. **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. **CURVE 2** **RHR NPSH LIMITS** 
	Driver	**When contacted as Chemistry, acknowledge any information or direction given.**
	NRC	**End of Event 7. Event 8 Standby Liquid Control (SLC) Pump Trip is automatically triggered.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-2 Event No.: 8 Page 1 of 1

Event Description: Standby Liquid Control (SLC) Pump Trip

Symptoms/Cues: White disagreement light indicates above the SLC pump handswitch. Distinct Motor Tripout alarm sounds

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 8 is built into the scenario file; no action is required to be taken by the driver to fail one SLC Pump.**
	BOP	The first selected 2A or 2B SLC pump will trip when started. Starting the second SLC pump will be the success path.
	NRC	**When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel, (-) 162 inches using available injection systems, end of Scenario.**

Scenario Setup
UNIT 2

IC	28
Exam IC	254

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-47	209	T.S. 3.1.3	253	APPX-8E	3
OI-68	173	T.S. 3.6.2.2	253	APPX-11A	5
OI-74	202	EOI-1A	7	APPX-17A	21
OI-85	162	EOI-2	19		
ARP-3B	40	APPX-1A	7		
ARP-4C	39	APPX-1B	5		
ARP-7B	40	APPX-1D	7		
ARP-53	43	APPX-1G	7		
AOI-70-1	34	APPX-2	4		
AOI-100-1	123	APPX-8B	9	2-SR-3.6.1.3.5 (RHR I)	46

Simulator Setup	• Start CPERF **PRIOR** to placing the Simulator in RUN • Place protected equipment as listen on the turnover
Schedule Files(s):	ILT 2404 NRC Scenario 2 UNIT 2.sch
Event Files(s):	ILT 2404 NRC Scenario 2 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING			Today
MODE 1	**DAYS ONLINE** **210**	**Drywell Leakage (GPM)** **1.80**	**Protected Equipment** Loop I Core Spray, Loop II RHR and Core Spray, C and D 4kV Shutdown Boards, C and D Diesel Generators, HPCI, 250V RMOV BD 2A (due to Loop I RHR inoperable)
Rx Power **100%**	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified **Last breaker closure** **11/23/23 05:41**	**Floor Drain (GPM)** **0.20**	
MWe **1308**		**Equipment Drain (GPM)** **2.0**	

☐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

Previous shift declared Loop I RHR inoperable for 2-SR-3.6.1.3.5, (RHR MOV I) RHR System MOV Operability Loop I, entered Technical Specification 3.5.1 Condition A, 6 Days remain for LCO.

Control Rod 46-35 was declared OPERABLE

LCOs OF 72 HOURS OR LESS

None

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2B Fuel Pool Cooling Pump tagged out of service for maintenance.

Spare RBCCW Pump placed in service to Unit 1.

2-SR-3.6.1.3.5, (RHR MOV I) RHR System MOV Operability Loop I, performed except for 2-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE.

Control Rod 46-35 was declared OPERABLE after an HCU leak fix.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Perform 2-SR-3.6.1.3.5, RHR MOV I for Section 6.4, 2-FCV-74-59, RHR SYSTEM I SUPPRESSION POOL COOLING/TEST VALVE. **Perform section 6.4 ONLY**

Down power 5% with core flow and move Control Rod 46-35 from position 00 to position 12 in accordance with Reactivity Control Plan (RCP).

OPERATOR WORK AROUNDS OWAs – U0-0/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 0/U2 - 0

None

ODMIs/ACMPs

None

ONEAs

None

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

None

SCHEDULED ITEMS NOT COMPLETED

None

Form 3.3-1 **Scenario Outline**

Facility: BFN-**(Run on Unit 2 ONLY)** Scenario Number: NRC-3
Scenario Source: NEW Op-Test Number: 24-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 95**%** Reactor Power. H2O2 Analyzer and 2A Reactor Feed Pump (RFP) are both out of service.

Turnover: During the previous shift, the 2A RFP spuriously tripped. Work Control is preparing a clearance and protected equipment. The Outage Control Center (OCC) is not yet manned. RPS was placed back on normal, all actions complete except the H2O2 Analyzer remains shut down, 6 days remain for associated TRM LCO 3.3.11 Condition A and TRM 3.6.2 Condition A. Place the H2O2 Analyzer in service in accordance with 2-OI-76, Containment Inerting System, Section 5.5. Return Reactor Power to 100%

Critical Tasks:

1. During failure to SCRAM conditions 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.

2. During failure to SCRAM conditions, inject Standby Liquid Control to shut down the Reactor AND spray the Suppression Chamber such that the Pressure Suppression Pressure (PSP) Curve is not challenged.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Place H2O2 Analyzer in Service
2.	N/A	R-OATC	Raise Reactor Power with Core Flow
3.	HS-75-14A	C-BOP TS-NUSO	Inadvertent 2C Core Spray Pump Start
4.	RD01A	C-OATC MC-OATC C-NUSO	2A Control Rod Drive (CRD) Pump Trip
5.	PC01C	TS-NUSO	'C' Standby Gas Fails to Start **(Unit 2 ONLY)**
6.	HS-64-3A	C-BOP MC-BOP C-NUSO	2A Reactor Zone Fans Trip
7a.	AD01J	C-OATC	Stuck Open Safety Relief Valve (SRV) 2-PCV-1-23, OATC Lowers Reactor Power
7b.	AD01J	C-BOP C-NUSO	Stuck Open Safety Relief Valve (SRV) 2-PCV-1-23, BOP Closes SRV
8.	NA	M-ALL	2-PCV-1-23 Broken Tailpipe, Hydraulic Anticipated Transient Without a SCRAM (ATWS)
9.	FCV-74-57 FCV-74-71	C-BOP C-NUSO	One Loop of Suppression Chamber Spray Fails

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will place the H2O2 Analyzer in service in accordance with 2-OI-76, Containment Inerting System, Section 5.5.

2. The Crew will Raise Reactor Power with Core Flow in accordance with 2-OI-68, Reactor Recirculation System.

3. The crew responds to an automatic start of 2C Core Spray Pump, Nuclear Unit Senior Operator (NUSO) will address Technical Specifications (Tech Specs).

4. The crew will respond to 2A CRD Pump Trip in accordance with 2-AOI-85-3, CRD System Failure.

5. Unit 1 will call and request 'C' Standby Gas to be placed in service, the fan will fail to start, NUSO will address Tech Specs. **(Unit 2 ONLY)**

6. The crew responds to a 2A Reactor Zone Fan Trip and the 2B Reactor Zone Fan fails to start automatically, BOP will start manually.

7. The crew will respond to a stuck open SRV, 2-PCV-1-23, MAIN STEAM LINE B RELIEF VALVE. The SRV initially opens and will close in response to BOP action. Subsequently the SRV fails open because of prior perturbation from valve cycling, remaining open. NUSO will direct the crew to place Suppression Pool Cooling in service due to rising Suppression Pool Temperature.
    a. The OATC inserts a Mid-Power Runback until Reactor Power lowers to below 90%.
    b. The BOP takes action to close 2-PCV-1-23 in accordance with 2-AOI-1, Stuck Open SRV.

8. An SRV tail pipe break develops on the same tail pipe as the failed SRV, triggered when the crew places Suppression Pool Cooling in service. NUSO directs a SCRAM due to rising Suppression Pool Temperature or rising Drywell Pressure, resulting in a Hydraulic ATWS.

9. The Crew will report to the NUSO the failure of Suppression Chamber Spray on whichever loop is selected first, the crew will continue to the opposite loop for success.

The Scenario ends when the crew inserts all Control Rods and is controlling Reactor Water Level above the Top of Active Fuel (TAF) (-) 162 inches) using any available injection system.

Critical Tasks 2

1. **During failure to SCRAM conditions 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:
   ADS valves automatically open, Reactor Pressure rapidly lowers and Reactor power rises.

   c. Measured by:
   ADS Logic inhibited prior to an automatic initiation, Reactor power and Reactor pressure indications.

   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

2. During failure to SCRAM conditions, inject Standby Liquid Control to shut down the Reactor AND spray the Suppression Chamber such that the Pressure Suppression Pressure (PSP) Curve is not challenged.

a. **Safety Significance**

Failure to inject SLC and spray the Suppression Chamber during an ATWS with a stuck open SRV discharging through a broken tail pipe will result in Suppression Chamber Pressure exceeding PSP, which would require Emergency Depressurization.

b. **Cues**

Reactor is SCRAMMED and ATWS actions are taken by the OATC and Reactor Power is greater than 5% or unknown.
Suppression Chamber pressure rises.

c. **Measured by**

Reactor Power on APRM displays.
Suppression Pool Level.
Suppression Chamber Pressure.

d. **Feedback**

SLC Flow.
Lowering Reactor Power.
Suppression Chamber Pressure.

e. **Critical Task Failure Criteria**

PSP curve is exceeded, and Emergency Depressurization is required.

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 1 Page 1 of 2

Event Description: Place H2O2 Analyzer in Service

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.**
	Driver	**If contacted as maintenance or Work Control and asked about the H2/O2 ANALYZER power supply breakers, reply "H2/O2 ANALYZER has been energized for 2 hours".**
	NUSO	Directs the BOP to place the H2/O2 Analyzer in Service
	BOP	2-OI-76, Containment Inerting System Section 5.5 Placing 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 H2/O2 ANALYZER ISOLATION RESET, 2-HS-76-91. [2] PLACE H2/O2 ANALYZER SUPPR CHBR/DW SELECT, 2-HS-76-110, in either SUPPR CHBR or DRYWELL. [3] IF H2/O2 ANALYZER SUPPR CHBR/DW SELECT, 2-HS-76-110 is in SUPPR CHBR, THEN CHECK the following red indicating lights illuminated: • SUPPR CHBR SMPL VLVS 2-FSV-76-55/56 OPEN, 2-IL-76-49-1 • SMPL RTN VLVS 2-FSV-76-57/58 OPEN, 2-IL-76-49-3 [4] IF H2/O2 ANALYZER SUPPR CHBR/DW SELECT, 2-HS-76-110 is in DRYWELL, THEN CHECK the following red indicating lights illuminated: • DRYWELL SMPL VLVS 2-FSV-76-49/50 OPEN, 2-IL-76-49-2 • SMPL RTN VLVS 2-FSV-76-57/58 OPEN, 2-IL-76-49-3

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 1 Page 2 of 2

Event Description: Place H2O2 Analyzer in Service

Time	Position	Applicant's Actions or Behavior
	BOP	[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 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 [6] CHECK the following red indicating light illuminates H2/O2 ANALYZER SAMPLE PUMP, 2-XI-76-110 [7] CHECK the red LOW FLOW indicating light extinguishes on H2/O2 ANALYZER, 2-MON-76-110. [8] CHECK H2/O2 ANALYZER SAMPLE PUMP, 2-XI-76-110, green indicating light extinguishes. [9] REQUEST I&C Maintenance EVALUATE requirement to PERFORM pre-operational checks of H2/O2 Analyzer, 2-LPNL-925-0340. REFER TO 2-SI-4.7.A.5.b Drywell/Suppression Chamber H2/O2 Analyzer Calibration. (N/A if recovering from PCIS Isolation or RPS Transfer)
	Driver	**If contacted as I&C Maintenance to EVALUATE requirement to PERFORM pre-operational checks of H2/O2 Analyzer, reply "The H2/O2 Analyzer pre-operational checks are not required.**
	NRC	**End of Event 1. Proceed to event 2, Raise Reactor Power with Core Flow.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 2 Page 1 of 1

Event Description: Raise Reactor Power with Core Flow

Symptoms/Cues: Crew is cued by the turnover sheet

Time	Position	Applicant's Actions or Behavior
	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-NRC2404-U2S3.
	NRC	**NOTE:** • **The crew has pre-briefed but may elect to hold a re-focus brief.** • **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 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: • 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 Pumps 2A and/or 2B speed with the RECIRC MASTER CONTROL, THEN ADJUST Recirc Pump Speed 2A & 2B using the following pushbuttons as required. • RAISE SLOW, 2-HS-96-31 • RAISE MEDIUM, 2-HS-96-32
	NRC	**End of Event 2. Proceed to Event 3, Inadvertent 2C Core Spray Pump Start**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 3 Page 1 of 2

Event Description: Inadvertent 2C Core Spray Pump Start

Symptoms/Cues: CORE SPRAY SYSTEM I PUMP C START (2-9-3C, Window 6), RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE (2-9-3C, Window 10), CORE SPRAY PUMP 2C RUNNING (2-9-23B, Window 37), alarms. Red run light illuminated above 2C Pump Handswitch.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to spuriously start 2C Core Spray Pump.**
	BOP	Reports spurious start of the 2C Core Spray Pump.
	NUSO	Directs BOP to: • Verify no auto start signal exists and to secure 2C Core Spray Pump • Notify Work Control/AUO to investigate. • May or may not dispatch personnel to rack out 2C Core Spray Pump
	BOP	Secures 2C Core Spray Pump
	Driver	**As WCC, AUO or any other plant personnel dispatched to investigate this event, acknowledge report, wait two minutes and reply "no abnormalities found at this time, I'll continue to investigate.** **If directed to rack out 2C Core Spray Pump Breaker, wait 1 minute and report "I'm at the 'B' 4kV Shutdown Board standing by to rack out 2C Core Spray Pump Breaker". When given direction, insert key 13 to rack the breaker.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 3 Page 2 of 2

Event Description: Inadvertent 2C Core Spray Pump Start

Time	Position	Applicant's Actions or Behavior			
	NRC	**The NUSO should enter Tech Spec 3.5.1 Condition A and F due to the existing LCO Tech Spec 3.5.1 Condition E for ADS valve, 2-PCV-1-19.**			
	NUSO	A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1	Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
		F. One ADS valve inoperable. AND Condition A entered.	F.1	Restore ADS valve to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
			OR		
			F.2	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
	NRC	**End of Event 3. Proceed to Event 4, 2A Control Rod Drive (CRD) Pump Trip**			

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 4 Page 1 of 1

Event Description: 2A Control Rod Drive (CRD) Pump Trip

Symptoms/Cues: White disagreement light indicates above the 2A CRD pump handswitch. Distinct Motor Tripout alarm sounds

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to trip 2A CRD Pump.**
	OATC	Reports 2A CRD Pump Trip.
	NUSO	Enters 2-AOI-85-3, CRD System Failure
	OATC	2-AOI-85-3, CRD System Failure Section 4.1 Immediate Actions [1] IF operating CRD pump has failed AND standby CRD pump is available, THEN PERFORM the following at Panel 2-9-5: (Otherwise N/A) [1.1] DEPRESS CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, MANUAL operation mode transfer switch. [1.2] ADJUST CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, using Manual Control MV key until FULL CLOSED position is indicated. [1.3] START associated standby CRD Pump using one of the following: • CRD PUMP 1B, using 2-HS-85-2A. • N/A [1.4] IF CRD Pump 1B was started, THEN OPEN CRD PUMP 1B DISCH TO U2, using 2-HS-85-8A [1.5] ADJUST CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, using Manual Control MV key to establish the following conditions: • CRD CLG WTR HDR DP, 2-PDI-85-18A, between 10 psid and 20 psid. • CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, between 40 and 65 gpm. [1.6] DEPRESS CRD SYSTEM FLOW CONTROL, 2-FIC-85-11 AUTO operation mode transfer key. [2] N/A
	NRC	**End of Event 4. Proceed to Event 5, 'C' Standby Gas Fails to Start**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 5 Page 1 of 1

Event Description: 'C' Standby Gas Fails to Start

Symptoms/Cues: When 'C' Standby Gas handswitch is taken to start, the red run light momentarily illuminates and then goes out. C' Standby Gas Inlet Damper will momentarily travel open indicating a red open light and subsequently travel back to indicate closed. Filter Bank Differential Pressure, 0-PDI-65-53 will indicate zero rather than develop and indicate a differential pressure reading.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, as Unit 1, call unit 2 NUSO and state "'C' Standby Gas needs to be started for a normal Drywell vent on unit 1".**
	BOP	0-OI-65, Standby Gas Treatment System Section 5.2 Standby Gas Treatment System Manual Initiation [1] N/A [2] N/A [3] N/A [4] START SGT FAN C as follows: [4.1] N/A [4.2] IF starting SGT FAN C from Panel 2-9-25, THEN PLACE SGTS FAN C, 0-HS-65-69A/2 in START.
	NRC	**The NUSO should enter Tech Spec 3.6.4.3 Condition A**
	OATC	A. One SGT subsystem inoperable. | A.1 Restore SGT subsystem to OPERABLE status. | 7 days
	NRC	**End of Event 5. Proceed to Event 6, 2A Reactor Zone Fans Trip**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 6 Page 1 of 3

Event Description: 2A Reactor Zone Fans Trip

Symptoms/Cues: RX BLDG VENTILATION ABNORMAL (2-9-3D, Window 3) and REACTOR ZONE DIFFERENTIAL PRESSURE LOW (2-9-3D, Window 32) alarm

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6, 2A RX Zone Fans trip.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • RX BLDG VENTILATION ABNORMAL (2-9-3D, Window 3) • REACTOR ZONE DIFFERENTIAL PRESSURE LOW (2-9-3D, Window 32)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: RX BLDG VENTILATION ABNORMAL (2-9-3D, Window 3) A. N/A B. NOTIFY Unit SROs, Unit 1 and Unit 3. C. ENSURE standby fans start. 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. IF unable to restore ventilation, THEN REFER TO 3-AOI-30B-1
	BOP	Alarm Response Procedure: REACTOR ZONE DIFFERENTIAL PRESSURE LOW (2-9-3D, Window 32) A. N/A B. REQUEST personnel to check local Reactor Building differential pressure. C. REQUEST personnel to check fans locally for any apparent problems. D. REFER TO 2-OI-30B and PLACE standby fan in service to restore normal differential pressure
	DRIVER	**NOTE: The below script applies to all 4 pages of Event 6, but is only listed here.** • **If directed to check Reactor Building differential pressure when the Reactor Zone fans are out of service, acknoledge request. Once 2B Reactor Zone fans are in service notify unit 2 MCR "Reactor Building differential pressure is (-) 0.30 inches of water."** • **If directed to check any reactor zone fans or 480V AC Rx Bldg Vent Bd 2A(2B), wait one minute then report "everything looks normal."** • **If directed to check the status of the Main Steam Vault Booster Fan, wait one minute and report, "Main Steam Vault Booster Fan is in service."**
	NRC	**The BOP may invoke OPDP-1 and start the 2B Reactor Zone Fans. Otherwise, 2-OI-30B, Sections 5.1, 6.1 and 2-AOI-30B-1 are on the following pages.** **At Chief Examiners discretion, continue to the next event rather than waiting five minutes to allow the crew to place the Reactor Zone Fans in FAST.**

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 6 Page 2 of 3

Event Description: 2A Reactor Zone Fans Trip

Time	Position	Applicant's Actions or Behavior
		2-OI-30B, Reactor Zone Ventilation System Section 5.1 Startup of Reactor Zone Supply and Exhaust Fans [1] N/A [2] N/A [3] PLACE RX ZONE FANS AND DAMPERS Switch, 2-HS-64-11A, in SLOW B. [4] CHECK dampers open and fans start as indicated by illuminated red lights above the following switches: • REACTOR ZONE SPLY OUTBD ISOL DMPR, 2-HS-64-13 • REACTOR ZONE SPLY INBD ISOL DMPR, 2-HS-64-14 • REACTOR ZONE EXH INBD ISOL DMPR, 2-HS-64-42 • REACTOR ZONE EXH OUTBD ISOL DMPR, 2-HS-64-43 • REACTOR ZONE FANS AND DAMPERS, 2-HS-64-11A [5] N/A [6] IF fast speed RX Zone Supply and Exhaust Fan operation is required, five minutes should be allowed after slow start for the discharge dampers to fully open, THEN [6.1] PLACE RX ZONE FANS AND DAMPERS switch, 2-HS-64-11A, in FAST B. [6.2] CHECK that the two green lights A remain extinguished and the two red lights B remain lit above RX ZONE FANS AND DAMPERS Switch, 2-HS-64-11A. [7] CHECK the following conditions: A. SUPPLY FANS FILTER DIFF PRESS Indicator, 2-PDI-64-22, at the filter in Reactor/Refuel Zone supply fan room, El 565' indicates less than 0.6 inches H2O. B. RX ZONE PRESS DIFFERENTIAL Indicator, 2-PDIC-64-2, on Panel 25-213 located at R10-P El 639', indicates between -0.25 inches and -0.40 inches H2O. C. N/A [8] N/A
	NRC	**At Chief Examiners discretion, continue to the next event rather than waiting five minutes to allow the crew to place the Reactor Zone Fans in FAST.**

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 6 Page 3 of 3

Event Description: 2A Reactor Zone Fans Trip

Time	Position	Applicant's Actions or Behavior
		2-OI-30B, Reactor Zone Ventilation System Section 6.1 Alternating Reactor Zone Supply and Exhaust Fans [1] N/A [2] N/A [3] N/A [4] N/A [5] PLACE REACTOR ZONE FANS AND DAMPERS Switch, 2-HS-64-11A, in OFF. [6] [II/C] CHECK dampers close and fans stop as indicated by illuminated green lights above the following switches: • REACTOR ZONE SPLY OUTBD ISOL DMPR, 2-HS-64-13 • REACTOR ZONE SPLY INBD ISOL DMPR, 2-HS-64-14 • REACTOR ZONE EXH INBD ISOL DMPR, 2-HS-64-42 • REACTOR ZONE EXH OUTBD ISOL DMPR, 2-HS-64-43 • REACTOR ZONE FANS AND DAMPERS, 2-HS-64-11A [II/B-92-072] [7] PLACE RX ZONE FANS AND DAMPERS Switch, 2-HS-64-11A, in SLOW B. [8] CHECK dampers open and fans start as indicated by illuminated red lights above the following switches: • REACTOR ZONE SPLY OUTBD ISOL DMPR, 2-HS-64-13 • REACTOR ZONE SPLY INBD ISOL DMPR, 2-HS-64-14 • REACTOR ZONE EXH INBD ISOL DMPR, 2-HS-64-42 • REACTOR ZONE EXH OUTBD ISOL DMPR, 2-HS-64-43 • REACTOR ZONE FANS AND DAMPERS, 2-HS-64-11A [9] IF fast speed RX Zone Supply and Exhaust Fan operation is required, five minutes should be allowed after slow start for the discharge dampers to fully open, THEN [9.1] PLACE RX ZONE FANS AND DAMPERS switch, 2-HS-64-11A, in FAST B. [9.2] CHECK that the two green lights A remain extinguished and the two red lights B remain lit above RX ZONE FANS AND DAMPERS Switch, 2-HS-64-11A. [10] CHECK the following conditions: A. SUPPLY FANS FILTER DIFF PRESS Indicator, 2-PDI-64-22, at the filter in Reactor/Refuel Zone supply fan room, El 565' indicates less than 0.6 inches H2O. B. RX ZONE PRESS DIFFERENTIAL Indicator, 2-PDIC-64-2, on Panel 25-213 located at R10-P El 639', indicates between -0.25 inches and -0.40 inches H2O. C. N/A [11] N/A
	NRC	**At Chief Examiners discretion, continue to the next event rather than waiting five minutes to allow the crew to place the Reactor Zone Fans in FAST.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 6 Page 3 of 3

Event Description: 2A Reactor Zone Fans Trip

Time	Position	Applicant's Actions or Behavior
	NRC	**The NUSO may elect to enter 2-AOI-30B-1, Reactor Building Ventilation Failure for RX Zone fan restoration. However, AOI entry is not required.**
		2-AOI-30B-1, Reactor Building Ventilation Failure 4.1 Immediate Action: None 4.2 Subsequent Action [1] N/A [2] N/A [3] N/A [4] IF a reactor zone ventilation fan is NOT running or a damper is closed, THEN PERFORM the following at Panel 2-9-25: (Otherwise N/A) START the alternate Reactor Zone supply and exhaust fans by placing the REACTOR ZONE FANS AND DAMPERS switch, 2-HS-64-11A, to Slow A (Slow B) position. [5] N/A [6] ENSURE STEAM VAULT EXH BOOSTER FAN running. [7] MONITOR and RECORD at 15-minute intervals MAIN STEAM LINE TUNNEL TEMP, 2-TIS-1-60A, on Panel 2-9-3. [8] NOTIFY Unit SRO if Main Steam Line Tunnel temperature on 2-TIS-1-60A reaches 150 °F [9] N/A [10] N/A [11] N/A
	NRC	**At Chief Examiners discretion, continue to the next event rather than waiting five minutes to allow the crew to place the Reactor Zone Fans in FAST.**
	NRC	**End of Event 6. Proceed to Event 7, Stuck Open Safety Relief Valve (SRV)**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 7 Page 1 of 6

Event Description: Stuck Open Safety Relief Valve Open (SRV)

Symptoms/Cues: MAIN STEAM RELIEF VALE OPEN (2-9-3C, Window 25) alarms, Acoustic Monitor indicates on panel 9-3 and lowering megawatts electrical output observed.

Time	Position	Applicant's Actions or Behavior
	DRIVER	**When requested by the Chief Examiner, insert Event 7 to stick open SRV, 2-PCV-1-23, MAIN STEAM LINE B RELIEF VALVE.**
	NRC	**SRV, 2-PCV-1-23 will close for one minute when MSRV AUTO ACTUATION LOGIC INHIBIT, 2-XS-1-202 is placed in INHIBIT. When the SRV subsequently sticks open the SRV will remain stuck open for the remainder of the scenario.** **NUSO may or may declare info only tracking for Tech Spec 3.4.3. The safety function of 12 SRV remain operable with 2-PCV-1-23 stuck open.**
	BOP	Reports stuck open SRV.
	CREW	Confirms stuck open SRV when indicated megawatts electrical output lowers.
	NUSO	Enters 2-AOI-1-1, Relief Valve Stuck Open
	OATC	2-AOI-1-1, Relief Valve Stuck Open Section 4.1 Immediate Actions [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 DSCHRGE 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.
	BOP	2-AOI-1-1, Relief Valve Stuck Open Section 4.1 Immediate Actions [3] WHILE OBSERVING the indications for the affected SRV on the Acoustic Monitor; CYCLE the affected relief valve control switch as required up to three times: • CLOSE to OPEN to CLOSE positions [4] N/A 4.2 Subsequent Actions Section 4.2.1 N/A Section 4.2.2 Attempt to close valve from Panel 9-3: [1] PLACE the SRV TAILPIPE FLOW MON POWER SWITCH in the OFF position. [2] PLACE the SRV TAILPIPE FLOW MON POWER SWITCH in the ON position. [3] N/A

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 7 Page 2 of 6

Event Description: Stuck Open SRV

Time	Position	Applicant's Actions or Behavior
	BOP	[4] PLACE MSRV AUTO ACTUATION LOGIC INHIBIT, 2-XS-1-202 in INHIBIT: [5] IF relief valve closes, THEN OPEN breaker or PULL fuses as necessary using Attachment 1 (Unit 2 SRV Solenoid Power Breaker/Fuse Table). Reports to the NUSO closure of SRV, 2-PCV-1-23.
	DRIVER	**When contacted to OPEN breaker or PULL fuses for SRV, 2-PCV-1-23 as necessary using Attachment 1 or section 4.2.3 step [10], wait five minutes, insert key 17, then report "2C 250V RMOV, compartment 1B1 breaker is open for SRV, 2-1-23".**
	CREW	BOP reports subsequent opening of SRV, 2-PCV-1-23. Crew confirms stuck open SRV by observation of lowering megawatts electric. NUSO directs reperforming actions of 2-AOI-1-1, Relief Valve Stuck Open
	NRC	**Based on crew communications and response times performing 2-AOI-1-1, Relief Valve Stuck Open, the NUSO may direct placing Suppression Pool Cooling in service with 2-OI-74 Residual Heat Removal System or 2-EOI Appendix-17A after entry into 2-EOI-2 PRIMARY CONTAINMENT CONTROL.** • **2-EOI-2 PRIMARY CONTAINMENT CONTROL, begins on page 15** • **Steps for 2-EOI Appendix-17A, RHR System Operation Suppression Pool Cooling begin on page 16.** • **Steps for 2-OI-74 Residual Heat Removal System Section 8.5 Initiation of Loop I(II) Suppression Pool Cooling follow below**
	BOP	2-OI-74 Residual Heat Removal System Section 8.5 Initiation of Loop I(II) Suppression Pool Cooling [1] N/A [2] N/A [3] NOTIFY other units of placing Loop I(II) of RHR in suppression pool cooling, the subsequent start of common equipment (i.e., RHRSW pumps) and associated alarms are to be expected. [4] NOTIFY Radiation Protection for impending action to initiate Suppression Pool Cooling. RECORD name and time of Radiation Protection representative notified in NOMS narrative log. [5] IF possible, THEN BEFORE placing RHRSW in service, NOTIFY Chemistry that RHRSW sampling is to be initiated (RHRSW sampling requirements). [6] ENSURE at least one RHRSW Pump is operating on each EECW Header.

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 7 Page 3 of 6

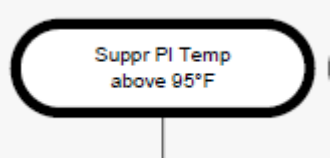
Event Description: Stuck Open SRV

Time	Position	Applicant's Actions or Behavior
	NRC	**Steps for placing Suppression Pool Cooling in service for Loop I and II have been combined below.**
	BOP	[7] PLACE RHR Pump and HX A(C)(B)(D) in service as follows: [7.1] START an RHRSW Pump to supply RHR HX A(C)(B)(D). [7.2] ESTABLISH RHRSW flow by performing one the following: [7.2.1] N/A [7.2.2] THROTTLE OPEN RHR HX 2A(2C)(2B)(2D) RHRSW OUTLET VLV, 2-FCV-23-34(40)(46)(52), to maintain 4000 and 4500 gpm RHRSW flow on 2-FI-23-36(42)(48)(54), RHR HTX 2A(2C)(2B)(2D)(total flow from all units per each header). [7.3] ENSURE CLOSED RHR SYS I(II) LPCI INBD INJ VALVE, 2-FCV-74-53(67). [7.4] IF NO RHR PUMP (2A OR 2C) (2B OR 2D) is operating, THEN ENSURE CLOSED RHR SYS I(II) SUPPR POOL CLG/TEST VALVE, 2-FCV-74-59(73). [7.5] ENSURE CLOSED RHR SY I(II) SUP CHBR SPRAY VLV, 2-FCV-74-58(72). [7.6] ENSURE CLOSED RHR SYS I(II) DW SPRAY OUTBD VLV, 2-FCV-74-60(74). [7.7] ENSURE OPEN RHR SYS I(II) SUP CHBR/POOL ISOL VLV, 2-FCV-74-57(71). [7.8] N/A [7.9] START RHR PUMP A(C)(B)(D) using 2-HS-74-5A(16A)(28A)(39A). [7.10] THROTTLE RHR SYS I(II) SUP POOL CLG/TEST VLV, 2-FCV-74-59(73), to maintain RHR flow indicated on RHR SYS I(II) CTMT SPRAY FLOW, 2-FI-74-56(70): ONE RHR PUMP IN ONE LOOP 7,000 to 10,000 gpm & Blue light illuminated. TWO RHR PUMPS IN ONE LOOP <13,000 gpm & Blue light illuminated. [8] CHECK pump motor breaker charging spring charged for motors operated in this section, amber breaker spring charged light on and closing spring indicates charged.
	NRC	**While placing Suppression Pool Cooling in service, when either SUP POOL CLG/TEST VLV, 2-FCV-74-59(73) is opened, the Broken Tailpipe on 2-PCV-1-23 begins to ramp.**

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 7 Page 4 of 6

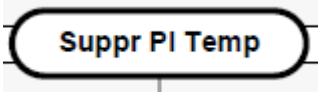
Event Description: Stuck Open SRV

Time	Position	Applicant's Actions or Behavior
	NUSO	Enters 2-EOI-2 PRIMARY CONTAINMENT CONTROL, directs placing all available suppression pool cooling in service at 95 °F Suppression Pool Temperature.



IF	THEN
It is determined that core damage is occurring due to loss of cooling	**NO ACTION REQUIRED**

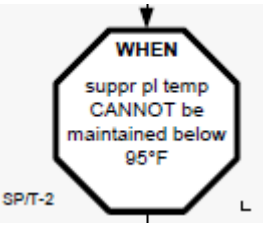
PCC 1



PCC 2

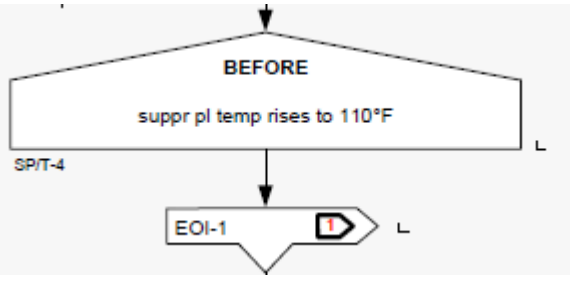
MONITOR and CONTROL suppression pool temp below 95 °F using available suppression pool cooling (APPX 17A)

SP/T 1



OPERATE all available suppression pool cooling using only RHR pumps NOT required for continuous injection (APPX 17A)

SP/T 3



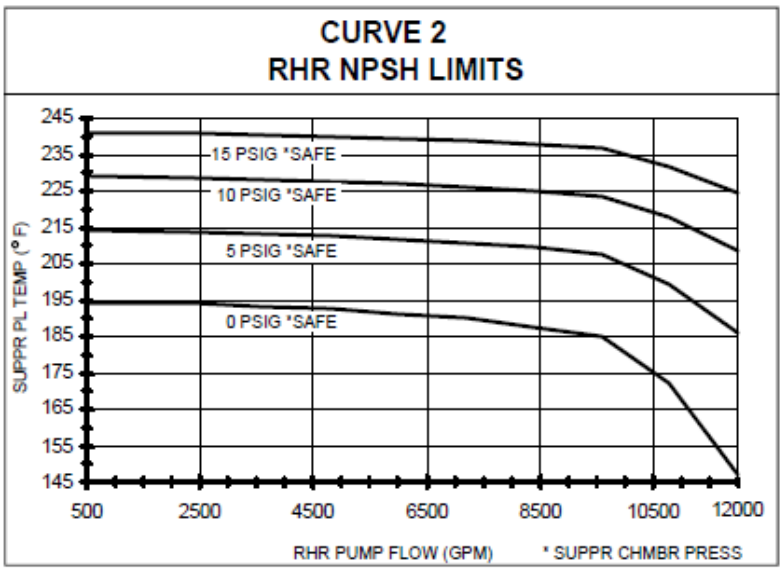
Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 7 Page 5 of 6

Event Description: Stuck Open SRV

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI Appendix-17A RHR System Operation Suppression Pool Cooling [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 THEN BYPASS LPCI injection valve open interlock AS NECESSARY: PLACE 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS. PLACE 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in BYPASS. [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 in service RHRSW outlet valves to obtain 1700 to 4500 gpm: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV [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. [2.4] 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] IF LPCI Initiation signal exists, THEN MOMENTARILY PLACE 2-XS-74-121(129), RHR SYS I (II) CTMT SPRAY/CLG VLV SELECT, in SELECT. [2.6] 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. [2.7] OPEN 2-FCV-74-57(71), RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV. [2.8] ENSURE desired RHR pump(s) for Suppression Pool Cooling are operating.

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 7 Page 6 of 6

Event Description: Stuck Open SRV

Time	Position	Applicant's Actions or Behavior
	BOP	[2.9] THROTTLE OPEN 2-FCV-74-59(73), RHR SYS I(II) SUPPR POOL CLG/TEST VLV, 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. • At or below 13,000 gpm for two pump operation. [2.10] ENSURE CLOSED 2-FCV-74-7(30), RHR SYS I(II) MIN FLOW VALVE. [2.11] MONITOR RHR Pump NPSH using Attachment 1. [2.12] NOTIFY Chemistry that RHRSW is aligned to in service RHR HXs. [2.13] IF further 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]. Attachment 1, NPSH Monitoring  • Suppression pool level below 10.0 ft • System flow rate decreasing with constant valve position • System flow rate/discharge psi less than expected for system conditions • Pump discharge psi lower than expected or fluctuating excessively • Pump motor amps lower than expected or fluctuating excessively • Pump suction pressure low (local indication)
	NRC	**When the crew places Suppression Pool Cooling in service event 8 is automatically triggered.**
	NRC	**End of Event 7. Proceed to Event 8, 2-PCV-1-23 Broken Tailpipe, Anticipated Transient Without a SCRAM (ATWS).**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 1 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic Anticipated Transient Without a SCRAM (ATWS)

Symptoms/Cues: DRYWELL TO SUPPR CHAMBER. DIFF PRESS ABNORMAL (2-9-3B, Window 26), Suppression Chamber Pressure and Temperature rises above Drywell Pressure and Temperature

Time	Position	Applicant's Actions or Behavior
	DRIVER	**Event 8 is automatically triggered when the crew places Suppression Pool Cooling in service. When the OATC attempts a manual Reactor SCRAM, a hydraulic ATWS will occur.**
	NRC	**The NUSO may elect to SCRAM when Suppression Pool Cooling is in service concurrent with the stuck open SRV and Suppression Pool Temperature unable to be maintained less than or equal to 95°F or prior to the SCRAM setpoint 2.45 pounds Drywell Pressure.**
	NUSO	Directs 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. [2] PLACE REACTOR MODE SWITCH, 2-HS-99-5A/S1, in SHUTDOWN. [3] IF all control rods can NOT be verified fully inserted, THEN INITIATE ARI. [4] N/A [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. [5.2] ENSURE shutdown both Recirc Pumps [5.3] STOP and PREVENT injection from CONDENSATE/FEEDWATER per APP 4
	BOP	[5.4] STOP and PREVENT injection from HPCI, RHR, and CS per APP 4.
	NRC	**Part 2 of the below Critical Task 1 is found on page 30 of this guide.**
	CREW	**Critical Task 1 Part 1: During failure to SCRAM conditions, inject Standby Liquid Control to shut down the Reactor AND spray the Suppression Chamber such that the Pressure Suppression Pressure (PSP) Curve is not challenged.** **Critical Task Failure Criteria: PSP curve is exceeded, requiring Emergency Depressurization.**
	BOP	[5.5] INITIATE SLC and ENSURE injection.

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 2 of 16

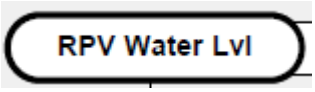
Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

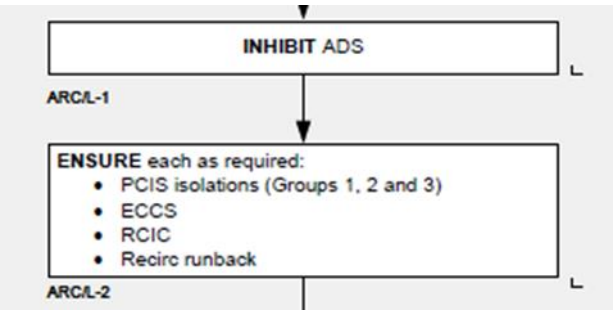
Time	Position	Applicant's Actions or Behavior
	CREW	**Critical Task 2: During failure to SCRAM conditions and the Reactor not shutdown, to prevent an uncontrolled Reactor depressurization and subsequent power excursion, inhibit ADS or control Reactor Water Level such that no automatic ADS actuation occurs.** **Critical Task Failure Criteria: ADS automatic initiation with Control Rods out.**
	BOP	[5.6] INHIBIT ADS LOGIC.
	BOP	[5.7] BYPASS Grp 1 RPV Low-Low-Low Level isolation interlocks per APP 8A
	OATC	[5.8] WHEN 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 IS **OR** IS NOT Injecting. [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 IS OR IS NOT Injecting.
	BOP	[5.10] IF Any Main steam line is OPEN, THEN CALL for APP 8E, Bypassing Group 6 Low RPV Level and High Drywell Pressure Isolation interlocks

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 3 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>DRIVER</td><td>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 28
Wait one minute and reply "2-EOI APPENDIX-1F field actions complete."
• 2-EOI APPENDIX-2 – Event 20
Wait one minute and reply "2-EOI APPENDIX-2 complete."
• 2-EOI APPENDIX-8E – Event 23
Wait one minute and reply "2-EOI APPENDIX-8E is complete."
The above communications may be bundled provided appendices are simultaneously called for.

If contacted to close 2-FCV-85-586, CRD CHARGING WTR VLV, wait 1 minute and report "standing by to close 2-FCV-85-586", when directed insert Event 24, report "2-FCV-85-586 is closed".
If contacted to open 2-FCV-85-586, wait 1 minute, insert Event 25, report "2-FCV-85-586 is open"</td></tr>
<tr><td></td><td>NRC</td><td>EOI Appendices for Control Rod insertion begin on page 26.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-EOI-1A, ATWS RPV Control.
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV pressure is or will be lowering</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>It is determined that core damage is occurring due to loss Adequate Core Cooling (Table M 3)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>The reactor will 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>
</table>
ARC-2
 

ARC-3
 
</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 4 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior

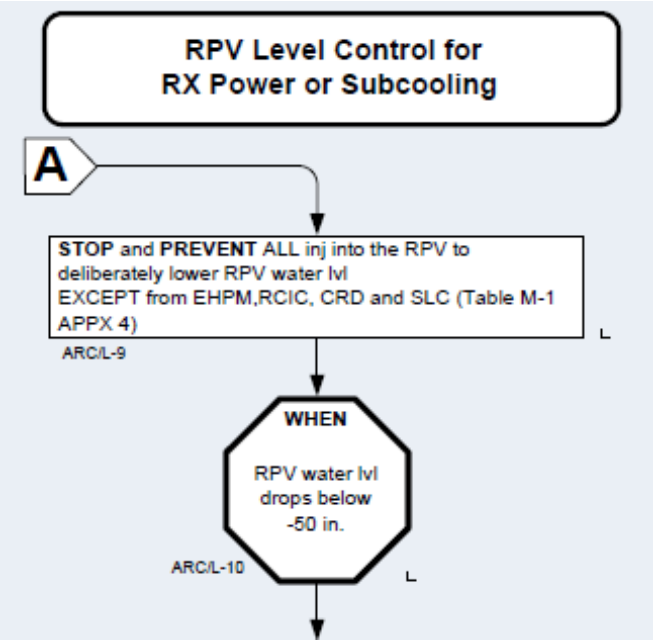
NUSO

IF	THEN
Primary Containment Water Level and Drywell Pressure CANNOT be maintained in the safe area of Curve 7	**NO ACTION REQUIRED**

ARC/L-3

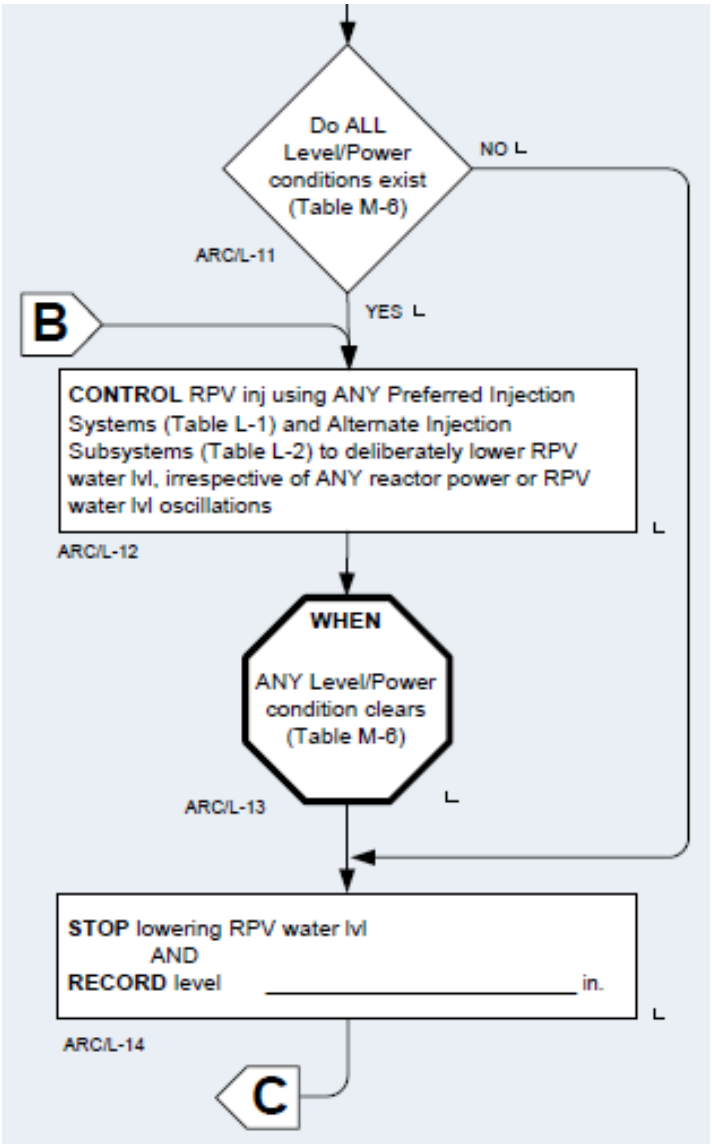
IF	THEN
Reactor Power is above 5% or unknown AND RPV Water Level is above (-) 50 in.	A
ALL Level/Power conditions exist (Table M-6)	**NO ACTION REQUIRED**
Reactor Power is above 10% or unknown AND RPV Water Level is below (-) 162 inches.AND MSRV open/cycling OR DW pressure >2.4 psig	**NO ACTION REQUIRED**

ARC/L-4



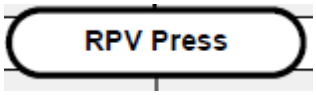
Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 5 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior
	NUSO	  RESTORE and MAINTAIN RPV water level between (-)180 inches and: Lowered level (if level was deliberately lowered in flow path A or B) OR +51 in. (if level was NOT deliberately lowered)
	NUSO	Directs the crew to maintain Reactor Water Level in any band between (-) 50 and (-) 180 inches.

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 6 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>EMERGENCY RPV DEPRESSURIZATION IS REQUIRED</td><td>NO ACTION REQUIRED</td></tr>
</table>
ARC/P-1
<table>
<tr><td>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 (APPX 11A)</td></tr>
</table>
ARC/P-2
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>BORON INJECTION IS REQUIRED
AND
The Main Condenser is available.
AND
There has been no indication of a steam line break</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>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</td><td>NO ACTION REQUIRED</td></tr>
</table>
ARC/P-3
<table>
<tr><td>RESTORE and MAINTAIN RPV Pressure between 1073 psig and 350 psig using Main Turbine Bypass Valves (APPX 8B) and maintain cooldown 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</td></tr>
</table>
<table>
<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>
ARC/P-4
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs Reactor Pressure control with 2-EOI APPENDIX-8B Main Turbine Bypass</td></tr>
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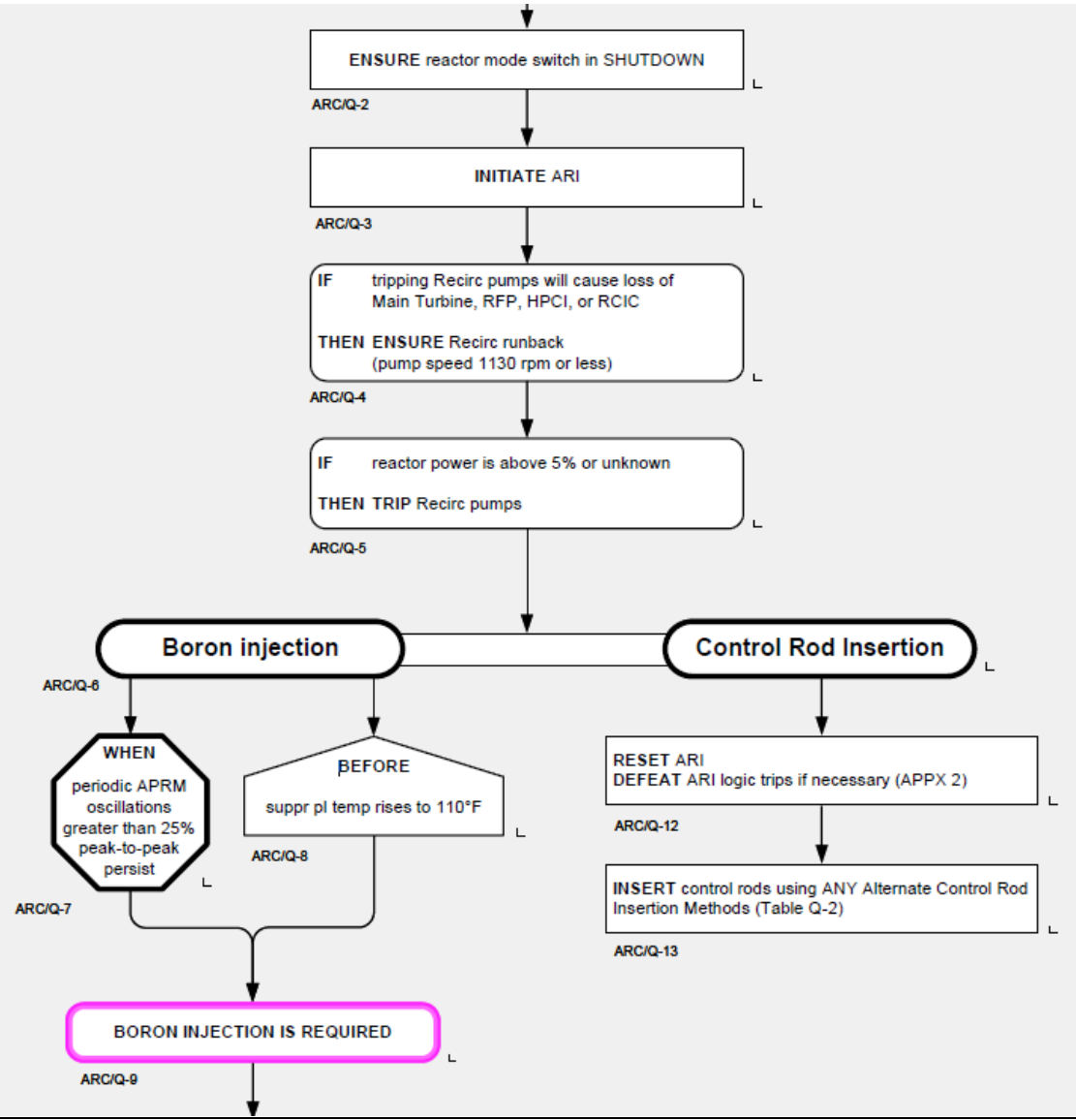
Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 7 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior
	NRC	**Actions for 2-EOI APPENDIX-8B Main, Turbine Bypass Valves are not listed. Bypass valve operation is automatic, the crew will monitor Rector Pressure.**
	NUSO	

IF	THEN
The Reactor is subcritical AND NO boron has been injected	**NO ACTION REQUIRED**

ARC/Q-1



Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 8 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

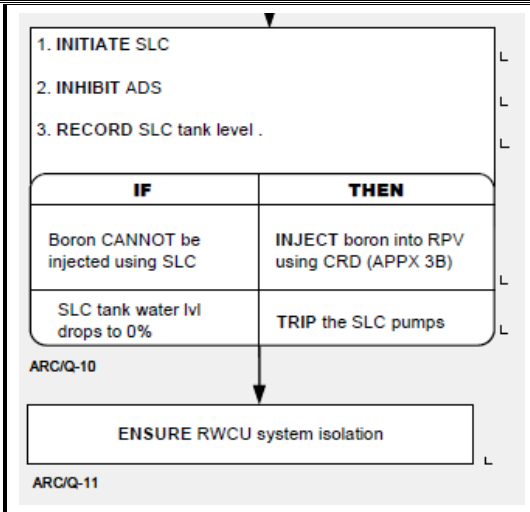
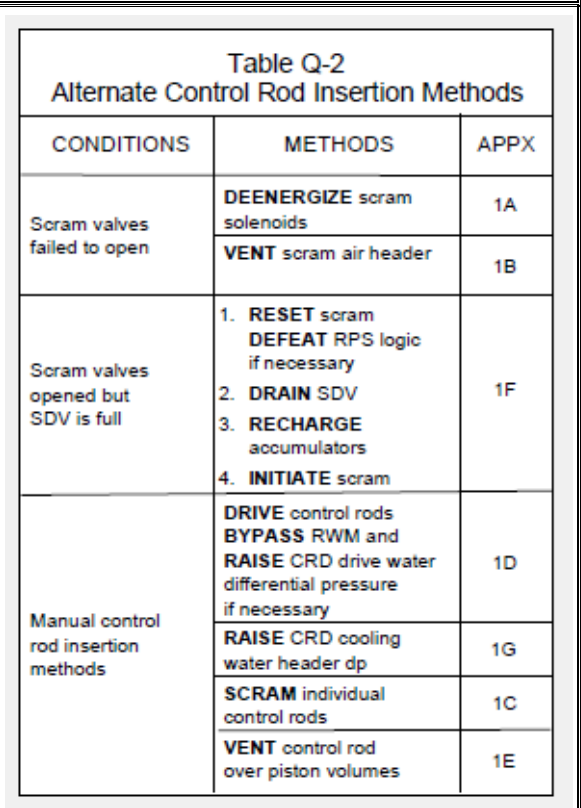
Time	Position	Applicant's Actions or Behavior
	NUSO	 
	NUSO	Verifies Applicable steps of ARC/Q Reator Power leg actions complete Driects, as appropriate, one or more of the following Table Q-2 Alternate Control Rod insertion Methods: • 2-EOI APPENDIX-1F, Manual Scram • 2-EOI APPENDIX-1D, Insert Control Rods Using Reactor Manual Control System • 2-EOI APPENDIX-2, DEFEATING ARI LOGIC TRIPS

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

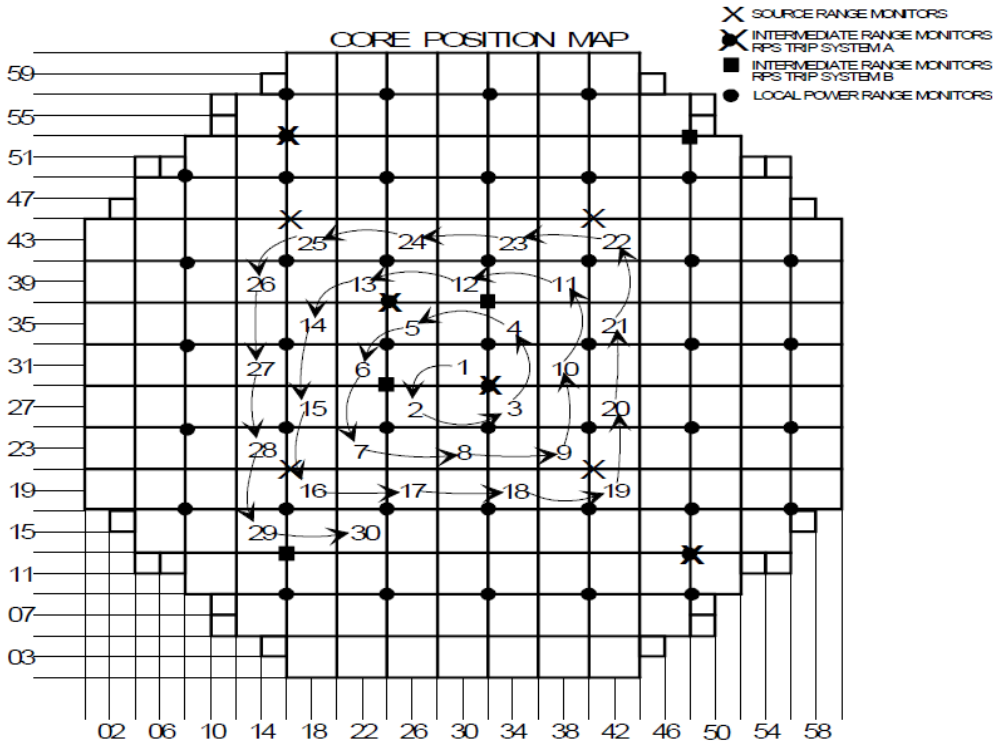
Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 9 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior
	OATC	2-EOI APPENDIX-1D, Insert Control Rods Using Reactor Manual Control System **NOTES** 1. This EOI Appendix may be executed concurrently with EOI Appendix 1A or 1B at SRO's discretion when time and manpower permit. 2. Closing 2-85-586, CHARGING WATER ISOL valve may reduce the effectiveness of EOI Appendix 1A or 1B. [1] VERIFY at least one CRD pump in service. [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).

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 10 of 16

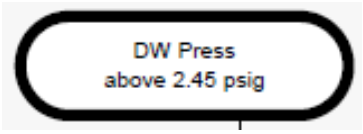
Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior
	OATC	**Attachment 2** **(Page 1 of 1)** **Core Position Map**  (EXAMPLE ONLY) **NOTES** 1) This Attachment is ONLY an aid to Unit RO/SRO when the Reactor Engineer is NOT available. USE is NO mandatory and it does NOT cover ALL situations for prioritizing rod insertion sequences during ATWS. 2) Where power distribution and control rod positions are distributed throughout the full core, control rods are inserted in spiral sequence, starting at the center of the core. Each insertion should be separated by one cell from the previous insertion. 3) Where local power peaks exist and control rods are withdrawn ONLY in a local area, each control rod inserted in the affected area should be from the center of the area, separated by one cell from the previous insertion.

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 11 of 16

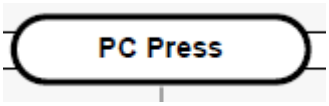
Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior
	NUSO	Reenters 2-EOI-2 PRIMARY CONTAINMENT CONTROL

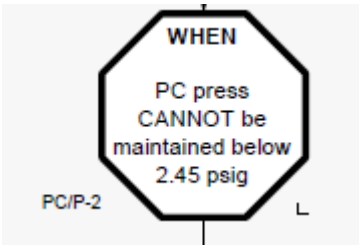


IF	THEN
It is determined that core damage is occurring due to loss of cooling	**NO ACTION REQUIRED**

PCC-1

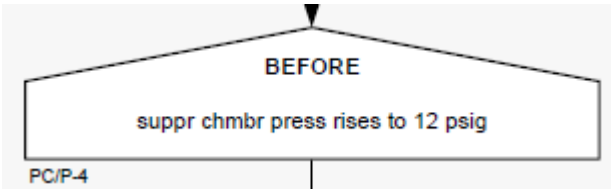


PCC-2



IF	THEN
PC press reduction is required to restore and maintain adequate core cooling or reduce total offsite radiation dose	**NO ACTION REQUIRED**
PC press CANNOT be maintained above (-)2 psig	**NO ACTION REQUIRED**
Suppr chmbr press approaches 0 psig	**NO ACTION REQUIRED**
DW press approaches 0 psig	**NO ACTION REQUIRED**

PC/P-3



INITIATE suppression chamber sprays
- Use only sources NOT required for continuous RPV injection (APPX 17)

IF	THEN
Needed to augment suppression chamber sprays	**NO ACTION REQUIRED**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 12 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs 2-EOI APPENDIX-17C, RHR System Operation Suppression Chamber Sprays
	BOP	2-EOI APPENDIX-17C, RHR System Operation 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 THEN BYPASS LPCI injection valve auto interlock as necessary: PLACE 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS. PLACE 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL 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. [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 SYS I (II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. [5.3] MOMENTARILY PLACE 2-XS-74-121(129), RHR SYS I (II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT. [5.4] IF 2-FCV-74-53 (67), RHR SYS I (II) LPCI INBD INJ VALVE, is OPEN, THEN ENSURE CLOSED 2-FCV-74-52 (66), RHR SYS I (II) LPCI OUTBD INJECT VALVE. [5.5] ENSURE OPERATING the desired RHR System I(II) pump(s) for Suppression Chamber Spray [5.6] ENSURE OPEN 2-FCV-74-57(71), RHR SYS I (II) SUPPR CHBR/POOL ISOL VLV [5.7] OPEN 2-FCV-74-58 (72), RHR SYS I (II) SUPPR CHBR SPRAY VALVE

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 13 of 16

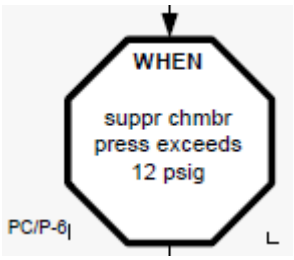
Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

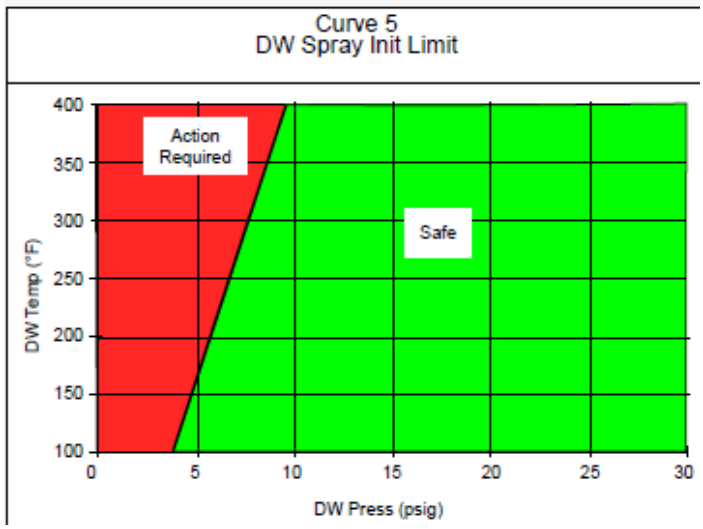
Time	Position	Applicant's Actions or Behavior
	NRC	**Part 1 of the below Critical Task 1 is found on page 18 of this guide.**
	CREW	**Critical Task 1 Part 2: During failure to SCRAM conditions, inject Standby Liquid Control to shut down the Reactor AND spray the Suppression Chamber such that the Pressure Suppression Pressure (PSP) Curve is not challenged.** **Critical Task Failure Criteria: PSP curve is exceeded, requiring Emergency Depressurization.**
	BOP	[5.7] OPEN 2-FCV-74-58 (72), RHR SYS I (II) SUPPR CHBR SPRAY VALVE
	BOP	[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 SYS I (II) MIN FLOW VALVE. [5.10] RAISE System flow with second RHR Sys I (II) pump if needed. [5.11] MONITOR RHR Pump NPSH using Attachment 2. [5.12] ENSURE RHRSW pump supplying desired RHR Heat Exchanger(s). [5.13] THROTTLE in-service RHRSW outlet valves to obtain 1700 to 4500 gpm: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV [5.14] NOTIFY Chemistry that RHRSW is in-service RHR Heat Exchangers. [6] WHEN EITHER of the following exists: Before Suppression Pool pressure drops below 0 psig **OR** Directed by SRO to stop Suppression Chamber Sprays, THEN STOP Suppr Chamber Sprays as follows: [6.1] CLOSE 2-FCV-74-58(72), RHR SYS I (II) SUPPR CHBR SPRAY VALVE.

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 14 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior



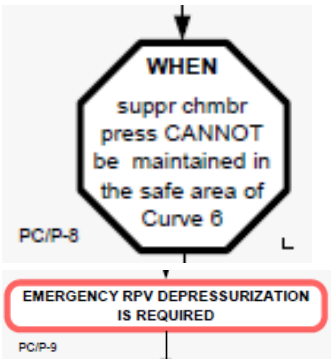


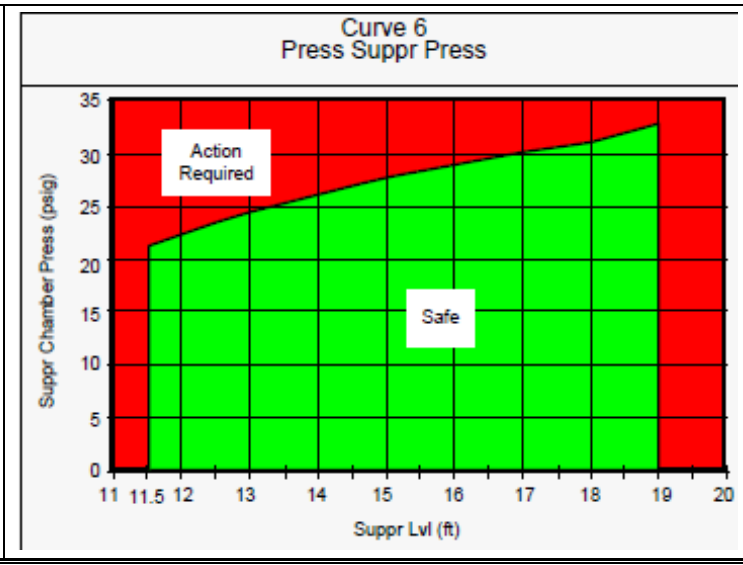
NUSO

IF suppression pool level is below 19 ft AND
DW temp is in safe area of Curve 5
THEN
1. SHUT DOWN Recirc pumps
2. SHUT DOWN DW blowers
3. INITIATE DW sprays
   - ➢ Use only sources NOT required for continuous RPV injection (APPX 17B)

IF	THEN
Needed to augment Drywell Sprays	**NO ACTION REQUIRED**

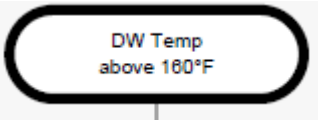
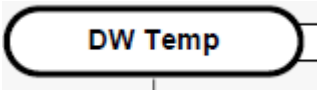
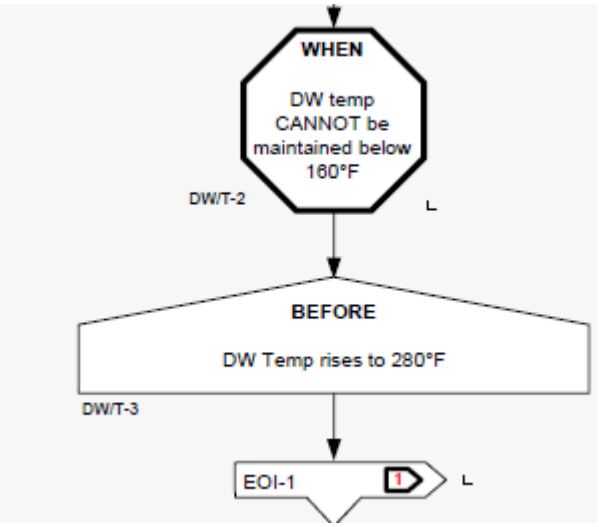
PC/P-7





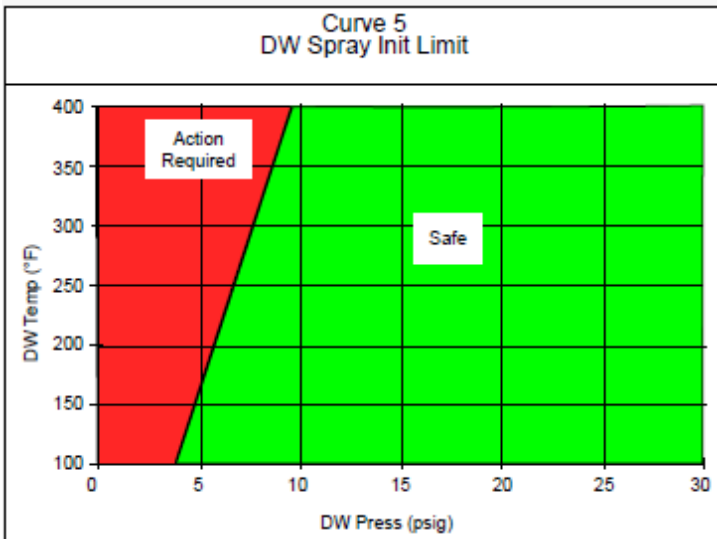
Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 15 of 16

Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Directs Drywell Sprays from either the Primary Containment Pressure Leg after Suppression Chamber Pressure reaches 12 psig or the Drywell Temperature Leg before Drywell Temperature rises to 280°F.</td></tr>
<tr><td></td><td>NUSO</td><td>Reenters 2-EOI-2 PRIMARY CONTAINMENT CONTROL
 
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>It is determined that core damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr></table>PCC-1
 
PCC-2
 
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>If Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr></table>DW/T-4</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 8 Page 16 of 16

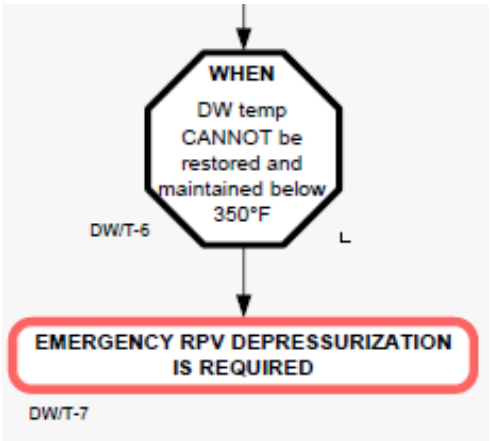
Event Description: 2-PCV-1-23 Broken Tailpipe, Hydraulic ATWS

Time	Position	Applicant's Actions or Behavior
	NUSO	

IF suppression pool level is below 19 ft AND
DW temp is in safe area of Curve 5
THEN
1. SHUT DOWN Recirc pumps
2. SHUT DOWN DW blowers
3. INITIATE DW sprays
   - ➢ Use only sources NOT required for continuous RPV injection (APPX 17B)

IF	THEN
Needed to augment Drywell Sprays	**NO ACTION REQUIRED**

DW/T-5



Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-3 Event No.: 9 Page 1 of 1

Event Description: One Loop of Suppression Chamber Spray Fails

Symptoms/Cues: When the operator attempts to open the first selected 2-FCV-74-58 or 2-FCV-74-72, RHR SYS I (II) SUPPR CHBR SPRAY VALVE light indication above handswitch is lost and Suppression Chamber Spray flow cannot be achieved.

Time	Position	Applicant's Actions or Behavior
	DRIVER	**Event 9 is built into the scenario file; no action is required to be taken by the driver to fail one loop of Suppression Chamber Spray.**
	BOP	The first selected 2-FCV-74-58 or 2-FCV-74-72 RHR SYS I (II) SUPPR CHBR SPRAY VALVE will fail to open. Aligning the second loop of Suppression Chamber Sprays will be the success path.
	NRC	**When the crew has inserted all Control Rods and has control of Reactor Water Level above the Top of Active Fuel, (-) 162 inches using available injection systems, end of Scenario.**

Scenario Setup
UNIT 2

IC	28
Exam IC	255

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-30B	30	AOI-100-1	123	APPX-17B	20
OI-65	57	T.S. 3.5.1	351	APPX-17C	20
OI-68	173	T.S. 3.6.4.3	334		
OI-74	202	EOI-1A	7		
OI-76	79	EOI-2	19		
ARP-3C	29	APPX-1D	7		
ARP-3D	41	APPX-1F	6		
ARP-23B	35	APPX-2	19		
AOI-1-1	30	APPX-8B	9		
AOI-85-3	28	APPX-8E	3		

Simulator Setup	• Start CPERF **PRIOR** to placing the Simulator in RUN • Place protected equipment as listen on the turnover
Schedule Files(s):	ILT 2404 NRC Scenario 3 UNIT 2.sch
Event Files(s):	ILT 2404 NRC Scenario 3 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ONLINE	Drywell Leakage (GPM)	Protected Equipment
			ADS MSRVs, 2-PCV-1-5, 2-PCV-1-22, 2-PCV-1-30, 2-PCV-1-31, 2-PCV-1-34, 250V RMOV BDS 2A, 2B and 2C
	210		
	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified **Last breaker closure** **11/23/23 05:41**	1.78	
Rx Power **95%**		**Floor Drain (GPM)** **0.20**	
MWe **1236**		**Equipment Drain (GPM)** **1.98**	

☐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

13 days remain for Tech Spec 3.5.1 Condition E. One ADS valve inoperable due to Main Steam Line B Relief Valve, 2-PCV-1-19 out of service. Required Action to Restore ADS valve to OPERABLE Status in 14 days. The safety function of, 2-PCV-1-19 remains OPERABLE.

6 days remain for TRM 3.3.11 Condition A and TRM 3.6.2 Condition A due to H2O2 Analyzer shut down.

LCOs OF 72 HOURS OR LESS

None

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

During the previous shift, the 2A RFP spuriously tripped. Work Control is preparing a clearance and protected equipment. The Outage Control Center (OCC) is not yet manned.

Maintenance complete and untagged the H2O2 Analyzer.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Place the H2O2 Analyzer in service for a 24-hour confidence run, in accordance with 2-OI-76, Containment Inerting System, Section 5.5.

OPERATOR WORK AROUNDS **OWAs – U0-0/U2-0** **Burdens - U0 – 0/U2-0** **Operator Challenges – U0 – 0/U2 - 0**

None

ODMIs/ACMPs

None

ONEAs

None

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

None

SCHEDULED ITEMS NOT COMPLETED

None

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-4
Scenario Source: NEW Op-Test Number: 24-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 100% Reactor Power. 2B Reactor Water Cleanup (RWCU) Pump is tagged out of service.

Turnover: Lower Reactor Power with Core Flow. Perform Automatic Start Test of Reactor Feed Pump Turbine (RFPT) 2C Oil Pumps in accordance with 2-OI-3, Reactor Feedwater System. The prior shift completed 2A and 2B RFPT oil pumps testing.

Critical Tasks:
1. When a primary system is discharging into secondary containment, isolate the system prior to exceeding Maximum Safe Operating Limit in two areas.

2. With a Primary System discharging into Primary Containment, direct Drywell Sprays before Drywell Temperature exceeds 280 ºF to prevent reaching component qualification and structural design limits.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	R-OATC	Lower Reactor Power with Core Flow
2.	XA-55-4B_17	N-BOP N-NUSO	RFPT Oil Pump Automatic Start Test
3.	XA-55-4B_17	C-BOP MC-BOP TS-NUSO	RWCU 2-FCV-69-1, INBOARD ISOLATION VALVE, Fails to Automatically Isolate
4.	RD02A	C-OATC MC-OATC C-NUSO	2-FCV-85-11A, CRD FLOW CONTROL VALVE, Fails Closed
5.	XA-55-7B_3	C-BOP C-NUSO	Electro-Hydraulic Control (EHC) Pump Discharge Filter Differential Pressure High
6a.	TH10A TH11A	C-BOP TS-NUSO	2A Recirc Pump Seal Leak/Pump Trip and Urgent Load Reduction
6b.	N/A	C-OATC	2A Recirc Pump Seal Leak/Pump Trip and Urgent Load Reduction
7.	TH11A	M-ALL	Unisolable 2A Recirc Pump Second Seal Leak
8.	RC09 RC10	C-BOP MC-BOP C-NUSO	RCIC Fails to Isolate on High Temperature

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will lower Reactor Power with Core Flow in accordance with 2-OI-68, Reactor Recirculation System.

2. The crew will perform the Reactor Feed Pump oil pump auto start test in accordance with 2-OI-3, Reactor Feedwater System, Section 8.15.

3. The crew will respond to a spurious RWCU isolation. 2-FCV-69-1, INBOARD ISOLATION VALVE, fails to automatically isolate. The BOP will isolate 2-FCV-69-1. Nuclear Unit Senior Operator (NUSO) will address Technical Specifications (Tech Specs).

4. The crew will respond to CONTROL ROD DRIVE (CRD) DRIVE WATER FILTER DIFFERENTIAL PRESSURE HIGH (2-9-5A, Window 3) when 2-FCV-85-11A, CRD FLOW CONTROL VALVE, fails closed requiring 2-FCV-85-11B to be placed in-service in accordance with 2-OI-85, Control Rod Drive System, Section 6.3.

5. The crew will respond to EHC PUMP DISCHARGE FILTER PRESSURE HIGH, (2-9-7B, Window 3) and alternate EHC pumps.

6. The crew responds to a seal leak on 2A Recirc Pump. 2A Recirc Pump coincidentally trips due to a keyed radio when inspecting the VFD cabinet. The crew enters 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease.
    a. The BOP performs 2-AOI-68-1 actions, including closing 2A Recirc Pump Discharge Valve
    b. The OATC will drive Control Rods in accordance with the 2-AOI-68-1 and Recirc Pump Trip portion of a standing Reactivity Control Plan (RCP)

7. The crew responds to rising Drywell Pressure when a second seal fails on 2A Recirc Pump. 2A Recirc Pump Suction Valve is stuck open causing an unisolable seal leak.

8. RCIC steam line breaks, the crew responds to RCIC failing to isolate on high temperature, both the inboard and outboard isolation valves fail automatically, manual isolation is successful. Damaged fuel reveals itself after the SCRAM and high radiation will occur in Secondary Containment if RCIC is not isolated.

The Scenario ends when the crew has sprayed the Drywell and is controlling Reactor Water Level above the Top of Active Fuel (TAF) (-) 162 inches using any available injection system.

Critical Tasks 2

1. **When a primary system is discharging into secondary containment, isolate the system prior to exceeding Maximum Safe Operating Limit in two areas.**

   a. **Safety Significance**

   EOI-3 directs action be taken when a Maximum Normal operating value is exceeded. Failing to do so can significantly change the mitigation strategy as an unnecessary release will result, unduly endangering plant personnel and the public.

   b. **Cues**

   Annunciators: Reactor Building High Radiation, High Area Temp Steam Leak, are in alarm.

   c. **Measured by**

   Isolate RCIC to avoid exceeding two areas above Maximum Safe.

   d. **Feedback**

   One or more isolation valves for the affected system are closed resulting in Primary System no longer discharging into the Secondary Containment and improving conditions in the affected areas.

   e. **Critical Task Failure Criteria:**

   The crew fails to isolate a system leaking from Primary to Secondary Containment when a system is isolable.

2. **With a Primary System discharging into Primary Containment, direct Drywell Sprays before Drywell Temperature exceeds 280 ºF 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 280 ºF Drywell Temperature.

d. **Feedback**
Lowering Drywell Temperature.
RHR System Flow to Containment Sprays.

e. **Critical Task Failure Criteria:**
Drywell Sprays are not directed before Drywell Temperature reaches 280 ºF.

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 1 Page 1 of 1

Event Description: Lower Reactor Power with Core Flow

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 pre-briefed but may elect to hold a re-focus brief.**
	NUSO	Directs the Operator at the Controls (OATC) to lower Reactor Power to 95% in accordance with 2-OI-68, Reactor Recirculation System and Reactivity Control Plan (RCP) RCP-NRC2404-U2S4.
	NRC	**The crew may elect to reduce Reactor Power using any method below.**
		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 SLOW(MEDIUM)(FAST), 2-HS-96-17A(17B) (17C). (Otherwise, N/A). AND/OR • LOWER Recirc Pump 2B using SLOW(MEDIUM)(FAST), 2-HS-96-18A(18B) (18C). (Otherwise, 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 pushbuttons as required. • LOWER SLOW, 2-HS-96-33 • LOWER MEDIUM, 2-HS-96-34 • LOWER FAST, 2-HS-96-35
	NRC	**End of Event 1. Proceed to event 2, RFPT Oil Pump Automatic Start Test.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 2 Page 1 of 1

Event Description: RFPT Oil Pump Automatic Start Test

Symptoms/Cues: Crew is cued by the turnover sheet.

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the BOP to perform Automatic Start Test of Reactor Feed Pump Turbine (RFPT) 2C Oil Pumps
		2-OI-3 Reactor Feedwater System Section 8.15 Automatic Start Test of RFPT 2C Oil Pumps [1] OBTAIN Unit SRO approval to perform this test. [2] ENSURE position of the following switches: RFPT 2C 2C1 MAIN OIL PMP, 2-HS-3-154A in Normal after START or STOP. RFPT 2C 2C2 MAIN OIL PMP, 2-HS-3-258A in Normal after START or STOP. RFPT 2C EBOP 2C3, 2-HS-3-153A, in Auto after START or STOP. [3] TEST EBOP 2C3 as follows: [3.1] DEPRESS and HOLD 2C3 EBOP TEST pushbutton, 2-HS-3-156A. [3.1.1] CHECK the following: Red (running) light and amber (auto start) light at pushbutton illuminated. RFPT OIL PUMP AUTO START, 2-XA-55-6B Window 29 in alarm. [3.2] RELEASE 2C3 EBOP TEST pushbutton, 2-HS-3-156A. [3.3] PLACE RFPT 2C EBOP 2C3, 2-HS-3-153A, in START (return to AUTO). [3.3.1] CHECK the following: Amber (auto start) light not lit,2C3 EBOP TEST pushbutton, 2-HS-3-156A. RFPT OIL PUMP AUTO START, 2-XA-55-6B, Window 29, will reset. [3.4] PLACE RFPT 2C EBOP 2C3 switch, 2-HS-3-153A, in STOP. CHECK Red light at 2C3 EBOP TEST pushbutton extinguished. [4] PERFORM applicable action: [4.1] IF both RFPT 2C 2C1 and 2C2 Main Oil Pumps on, THEN GO TO Step 8.15[6]. [6] NOTIFY Unit SRO the Automatic Start Test of RFPT 2C Oil Pumps complete.
	NRC	**End of Event 2. Proceed to event 3, RWCU 2-FCV-69-1, INBOARD ISOLATION VALVE, Fails to Automatically Isolate.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 3 Page 1 of 4

Event Description: RWCU 2-FCV-69-1, INBOARD ISOLATION VALVE, Fails to Automatically Isolate

Symptoms/Cues: White disagreement light indicates above the 2A RWCU pump handswitch. Distinct Motor Tripout alarm sounds. RWCU NON-REGEN HX DISCH TEMP HIGH (2-9-4B, Window 17) alarms.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to isolate RWCU.**
	BOP	Manually closes 2-FCV-69-1, INBOARD ISOLATION VALVE. Reports that the RWCU system received an isolation signal from high outlet temperature on the Non-Regenerative HX and 2-FCV-69-1, INBOARD ISOLATION VALVE failed to close automatically, manual isolation worked.
	NRC	**The NUSO may or may not elect to enter 2-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation given that RWCU is fully isolated after manual isolation of 2-FCV-69-1, INBOARD ISOLATION VALVE.**
	Driver	**If contacted as Work Control to request a clearance, acknowledge request.** **If requested to verify RWCU area temperatures in the Aux Instrument Room, wait 1 minute and reprot, "All temps normal and in band."**
	BOP	2-AOI-64-2A, Group 3 Reactor Water Cleanup Isolation Section 4.1 Immediate Actions [1] PERFORM the following: • ENSURE CLOSED RWCU INBD SUCTION ISO VLV, 2-FCV-69-1. • ENSURE CLOSED RWCU OUTBD SUCTION ISO VLV, 2-FCV-69-2. • ENSURE CLOSED RWCU RETURN ISO VLV, 2-FCV-69-12. • ENSURE TRIPPED Reactor Water Cleanup Recirc Pumps 2A and 2B. Section 4.2 Subsequent Actions [1] N/A [2] N/A [3] N/A [4] PERFORM necessary Heat Balance adjustments. REFER TO 2-OI-69. [5] N/A [6] N/A [7] NOTIFY Chemistry that RWCU is OOS and to perform the following evaluations. • The need to begin sampling Reactor Water. • The need to remove the Durability Monitor from service. [8] IF the isolation cannot be reset, THEN PERFORM the following: [8.1] ISOLATE CRD seal water by closing 2-69-592(A) and 2-69-614(B) (RX BLDG). [8.2] N/A [9] EVALUATE Technical Requirements Manual 3.4.1, Coolant Chemistry [10] N/A [11] N/A

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 3 Page 2 of 4

Event Description: RWCU 2-FCV-69-1, INBOARD ISOLATION VALVE, Fails to Automatically Isolate

Time	Position	Applicant's Actions or Behavior
	NRC	**At Chief Examiners discretion, once satisfied with Crew actions and NUSO TS call, proceed to the next event rather than waiting for NHB substitutions.**
	BOP	2-OI-69, Reactor Water Cleanup System Section 7.1, ICS Temperature Point Substitution for Heat Balance [1] IF removing Reactor Water Cleanup System from service when operating at power, THEN PERFORM RWCU ICS Temperature Point Substitution for Heat Balance adjustments. **NOTE** The following values are to be substituted for RWCU Inlet and Outlet temperatures so RWCU parameters provide conservative input to the Integrated Computer System (ICS) thermal power calculation. • 525 degrees F for 69-6A, RWCU LOOP INLET TEMP. • 420 degrees F for 69-6D, RWCU LOOP OUTLET TEMP. [1.1] TYPE SV in the Command block at the top of the ICS display and DEPRESS Enter key to cause the SUBSTITUTE VALUE screen to be displayed. [1.2] ENTER the following information in the entry blocks of the SUBSTITUTE VALUE screen to substitute the temperature for point 69-6A: [1.2.1] In the Point ID field, TYPE 69-6A. [1.2.2] In the New Value field, TYPE 525 [1.2.3] In the Modified By field, TYPE your initials. [1.2.4] In the Reason field of the INSERT VALUE screen, TYPE short description of reason (like "system shutdown"). [1.2.5] After all above entries made, then SELECT Execute to insert the substitution. [1.3] ENTER the following information in the entry blocks of the SUBSTITUTE VALUE screen to substitute the temperature for point 69-6D: [1.3.1] In the Point ID field, TYPE 69-6D. [1.3.2] In the New Value field, TYPE 420 [1.3.3] In the Modified By field, TYPE your initials. [1.3.4] In the Reason field, TYPE short description of reason. [1.3.5] After all above entries made, then SELECT Execute to insert the substitution. [1.4] DEPRESS the Esc key to exit the SUBSTITUTE VALUE screen.

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 3 Page 3 of 4

Event Description: RWCU 2-FCV-69-1, INBOARD ISOLATION VALVE, Fails to Automatically Isolate

Time	Position	Applicant's Actions or Behavior
	NRC	**The NUSO should:** • **Enter Tech Spec 3.6.1.3 Condition A** • **Reference TRM 3.4.1, ensuring notification to Chemistry that Technical Surveillance Requirement 3.4.1.1 requires 4-hour sampling**
	NUSO	2-FCV-69-1, INBOARD ISOLATION VALVE CONDITION / REQUIRED ACTION / COMPLETION TIME A. ----------NOTE---------- Only applicable to penetration flow paths with two PCIVs. ---------- One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits. 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. 4 hours except for main steam line OR In accordance with the Risk Informed Completion Time Program AND 8 hours for main steam line OR In accordance with the Risk Informed Completion Time Program AND (continued)
	NRC	**Tech Spec 3.6.1.3 Condition A and TSR 3.4.1.1 continued below.**

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 3 Page 4 of 4

Event Description: RWCU 2-FCV-69-1, INBOARD ISOLATION VALVE, Fails to Automatically Isolate

Time	Position	Applicant's Actions or Behavior
	NUSO	A. (continued) 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. — Once per 31 days [following isolation] for isolation devices outside primary containment AND 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
		TR 3.4.1 TECHNICAL SURVEILLANCE REQUIREMENTS SURVEILLANCE: TSR 3.4.1.1 ----------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
	NRC	**End of Event 3. Proceed to event 4, 2-FCV-85-11A, CRD FLOW CONTROL VALVE, Fails Closed.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 4 Page 1 of 2

Event Description: 2-FCV-85-11A, CRD FLOW CONTROL VALVE, Fails Closed

Symptoms/Cues: CONTROL ROD DRIVE (CRD) CHG WTR HDR PRESSURE HIGH (2-9-5A, Window 10) alarms and 2-FCV-85-11A indicates full closed

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to fail closed 2-FCV-85-11A, CRD FLOW CONTROL VALVE**
	OATC	Acknowledges and reports the following alarm(s) to the NUSO: • CRD CHG WTR HDR PRESSURE HIGH (2-9-5A, Window 10)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	OATC	Alarm Response Procedure: CRD CHG WTR HDR PRESSURE HIGH (2-9-5A, Window 10 A. CHECK psi high on CRD ACCUM CHG WTR HDR 2-PI-85-13A, Panel 2-9-5. B. CHECK 2-FCV-85-11A (B) in service. C. IF in-service controller has failed, THEN REFER TO 2-OI-85. D. N/A
	NRC	**The crew may also elect to enter 2-AOI-85-3, CRD System Failure to assist in diagnosis and response.**
	OATC	2-AOI-85-3, CRD System Failure 4.1 Immediate Actions N/A 4.2 Subsequent Actions [1] N/A [2] N/A [3] IF operating CRD Pump has NOT tripped, THEN PERFORM the following: [3.1] ENSURE FULL OPEN CRD PUMPS 2A & 1B UNIT 2 SUCTION, 2-FCV-85-65. [3.2] N/A [3.3] N/A [3.4] IF system flow is high or low, THEN SHIFT CRD Flow Control Valves [3.6] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 4 Page 2 of 2

Event Description: 2-FCV-85-11A, CRD FLOW CONTROL VALVE, Fails Closed

Time	Position	Applicant's Actions or Behavior
	OATC	2-OI-85, Control Rod Drive System Section 6.3 Shifting CRD Flow Control Valves [1] N/A [2] N/A [3] PERFORM the following for Flow Control Valve being brought into service from Reactor Bldg. El 565': [3.1] OPEN FCV-85-11B INLET SOV, 2-SHV-085-0561. [3.2] OPEN FCV-85-11B OUTLET SOV, 2-SHV-085-0562. [3.3] ENSURE OPEN PCV BYPASS SOV TO FCV-85-11A(B), 2-85-317(318). [3.4] CHECK OPEN PCV 85-11 SOV, 2-85-247. [3.5] CHECK OPEN HDR ISOL TO FCV-85-11A & B, 2-85-313. [3.6] CHECK FCV-85-11A three-way iso vlv handle in Horizontal for 2-85-251. [3.7] CHECK FCV-85-11B, three-way iso vlv handle in Horizontal for 2-85-252.
	Driver	**NOTE: Four separate "field actions/communications" are required to support Shifting CRD Flow Control Valves. Only part 2 requires Driver to insert a key.** **When contacted for each action of 2-OI-85 Section 6.3, perform the following:** **Part 1 is step [3], call back one minute later reporting "All of Step [3] is complete for placing 2-FCV-85-11B in service".** **Part 2 is step [6], insert key 14 and immediately call back reporting "SELECTOR Switch, 2-XS-85-11 in VALVE B".** **Part 3 is step [8], immediately call back reporting "Step [8] is complete for placing 2-FCV-85-11B in service".** **Part 4 is step [11], immediately call back reporting "2-FI-85-22, is 6 gpm".**
	OATC	[4] DEPRESS CRD SYS FLOW CON, 2-FIC-85-11, MAN operation mode xfr key. [5] REDUCE CRD SYS FLOW using 2-FIC-85-11, to approximately 40 gpm with Manual Control MV key on 2-FIC-85-11. [6] N/A for OATC this is performed in the field, 2-XS-85-11, selected to VALVE B. [7] CHECK selected in-service valve opening and out-of-service valve closing. [8] PERFORM the following for Flow Control Valve being removed from service: [8.1] CLOSE FCV-85-11A INLET SOV, 2-SHV-085-0563. [8.2] CLOSE FCV-85-11A OUTLET SOV, 2-SHV-085-0564. [9] ADJUST CRD SYS FLOW CON, 2-FIC-85-11, Man Con MV key to 40 to 65 gpm. [10] DEPRESS CRD SYS FLOW CON, 2-FIC-85-11 AUTO operation mode xfr key. [11] CHECK CRD STABILIZING FLOW, 2-FI-85-22, is app 6 gpm (locally). [12] CHECK CRD DR WATER HDR FLOW, 2-FI-85-15A, is approximately 0 gpm.
	NRC	**End of Event 4. Proceed to event 5, Electro-Hydraulic Control (EHC) Pump Discharge Filter Differential Pressure High**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 5 Page 1 of 2

Event Description: Electro-Hydraulic Control (EHC) Pump Discharge Filter Differential Pressure High

Symptoms/Cues: EHC PUMP DISCHARGE FILTER PRESSURE HIGH, (2-9-7B, Window 3)

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to develop EHC Pump Discharge Filter Differential Pressure High.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • EHC PUMP DISCHARGE FILTER PRESSURE HIGH, (2-9-7B, Window 3)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: EHC PUMP DISCHARGE FILTER PRESSURE HIGH, (2-9-7B, Window 3) A. CHECK EHC HEADER PRESSURE indicator, 2-PI-47-7. B. DISPATCH personnel to EHC pump unit to check DP by performing the following: 1. PLACE 2-HS-47-82 in the TEST position. Illuminated light indicates bad filter. 2. CHECK 2-PDI-047-0240, EHC PUMP A DISC FILTER DP C. IF alarm is valid, PLACE standby pump in service. REFER TO 2-OI-47A. D. IF filter is dirty, THEN INITIATE CR to mechanical maintenance to change filter.
	Driver	**If asked for EHC pump DP, wait 1 minute and report "EHC PUMP A DISC FILTER DP, 2-PDI-047-0240 is 110 pounds". (This note only applies if 2A EHC pump is in service)** **If asked to PLACE 2-HS-47-82 in the TEST position wait 1 minute and report "When 2-HS-47-82 is in the TEST position, light is illuminated".**
	BOP	2-OI-47A, EHC System Section 6.3 EHC Pump Alternation [1] N/A [2] N/A [3] START the standby EHC PUMP 2B using EHC PUMP 2B, 2-HS-47-2A. [4] Locally check EHC pump 2B, 2-PI-47-2, between 1550 and 1750 psig, [5] N/A
	Driver	**If asked for a pre or post start inspection on 2B EHC pump, report "all personnel are clear, pre-start checks are SAT" or "post start checks are SAT".** **If asked for 2B EHC pump discharge pressure, report "2-PI-47-2 FILTER DP, is 1650 psig".**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 5 Page 2 of 2

Event Description: Electro-Hydraulic Control (EHC) Pump Discharge Filter Differential Pressure High

Time	Position	Applicant's Actions or Behavior
	Driver	**If asked to check EHC Pump 2B suction filter and EHC AUX Pump suction filter, report "suction filter is clean" or "AUX Pump suction filter is clean".** **If asked to ensure manual flow control valve, 2-47-588, is set at 2.5 turns clockwise., report "2-47-588, is set at 2.5 turns clockwise".**
	BOP	Section 6.3 EHC Pump Alternation Continued [6] STOP EHC PUMP 2A using EHC HYD FLUID PUMP 2A, 2-HS-47-1A. [7] CHECK EHC HEADER PRESSURE, 2-PI-47-7, between 1550 and 1650 psig. [8] N/A [9] N/A [10] CHECK the started EHC PUMP 2B MTR amps, 2-EI-47-2, 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] [NER/C] ENSURE MANUAL FLOW CONTROL VALVE, 2-47-588, is set at 2.5 turns clockwise. [INPO SOER 84-006]
	NRC	**End of Event 5. Proceed to event 6, 2A Recirc Pump Seal Leak/Pump Trip and Urgent Load Reduction**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 6 Page 1 of 3

Event Description: 2A Recirc Pump Seal Leak/Pump Trip and Urgent Load Reduction

Symptoms/Cues: RECIRC PUMP A NO. 1 SEAL LEAKAGE ABN, (2-9-4A, Window 25)

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to begin 2A Recirc Pump Seal Leak and subsequent 2A Recirc Pump trip.**
	NRC	**2A Recirc Pump trips two minutes after Event 6 begins**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • RECIRC PUMP A NO. 1 SEAL LEAKAGE ABN, (2-9-4A, Window 25)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: RECIRC PUMP A NO. 1 SEAL LEAKAGE ABN, (2-9-4A, Window 25) A. DETERMINE initiating cause by comparing No. 1 and 2 seal cavity pressure indicators on Panel 2-9-4 or ICS. • N/A. • N/A • Failure of No. 1 seal - No. 2 seal psi is > than 50% of the pressure of No. 1. • N/A B. N/A C. N/A **NOTES** 1) Possible indications of dual seal failure include: • Window 18 on this panel alarming in conjunction with this window. • Rising drywell pressure and/or temperature. • Increased leakage into the drywell sump. • Increased vibration of the recirc pump. 2) Single recirculation loop operation (SLO) is prohibited in the MELLLA+ operating domain. REFER TO Tech Spec 3.4.1. D. N/A
	Crew	Reports 2A Recirc Pump trip.
	NUSO	Enters 2-AOI-68-1, Recirc Pump Trip/Core Flow Decrease Directs OATC to INSERT control rods to less than 67.0% loadline per 0-TI-464, Reactivity Control Plan and 2-OI-85.
	BOP	2-AOI-68-1 Recirc Pump Trip/Core Flow Decrease 4.1 Immediate Actions N/A 4.2 Subsequent Actions [1] N/A [2] IF a single Recirc Pump tripped, THEN CLOSE tripped Recirc Pump 2A discharge valve 2-FCV-068-0003. [2.1] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 6

Event Description: 2A Recirc Pump Seal Leak/Pump Trip and Urgent Load Reduction

Time	Position	Applicant's Actions or Behavior
	OATC	[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 per 0-TI-464, Reactivity Control Plan and 2-OI-85.
	BOP	[2.3] N/A [2.4] WHEN the Recirc Pump discharge valve has been closed for at least five minutes, THEN (MARK N/A if Recirc Pump will be isolated for maintenance.) OPEN Recirc Pump 2A(2B) discharge valve 2-FCV-068-0003 as necessary to maintain Recirc Loop in thermal equilibrium. [2.5] N/A [2.6] N/A [2.7] N/A [2.8] NOTIFY Reactor Engineer to PERFORM the following: [2.8.1] 2-SR-3.4.1(SLO), Reactor Recirculation System Single Loop Operation [2.8.2] 0-TI-248, Core Flow Determination in Single Loop Operation Remaining Steps [3] through [10] N/A

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 6

Event Description: 2A Recirc Pump Seal Leak/Pump Trip and Urgent Load Reduction

Time	Position	Applicant's Actions or Behavior
	NRC	**The NUSO should enter Tech Spec 3.4.1 Condition A and B**
	NUSO	3.4.1 Recirculation Loops Operating ACTIONS CONDITION / REQUIRED ACTION / COMPLETION TIME A. Requirements of the LCO not met. — A.1 Satisfy the requirements of the LCO. — 24 hours B. Operating in the MELLLA+ operating domain with a single recirculation loop in operation. — B.1 Initiate action to exit the MELLLA+ operating domain. — Immediately
	Driver	**End of Event 6. Proceed to event 7, Unisolable 2A Recirc Pump Second Seal Leak**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 1 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

Symptoms/Cues: RECIRC PUMP A NO. 2 SEAL LEAKAGE HIGH, (2-9-4A, Window 18)

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to develop a second seal leak on 2A Recirc Pump. When the crew places the MODE switch in SHUTDOWN, RCIC steam line breaks concurrent with damaged fuel**
	NRC	**During Event 7 the crew will SCRAM, when the MODE switch is in SHUTDOWN Event 8 will be automatically triggered. Event 8 actions are listed on page 24 of this guide.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • RECIRC PUMP A NO. 2 SEAL LEAKAGE HIGH, (2-9-4A, Window 18)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	NRC	**RECIRC PUMP 2A SUCTION VALVE, 2-HS-68-1 will fail to close.** **BOP may attempt to isolate 2A Recirc pump with direction from RECIRC PUMP A NO. 1 SEAL LEAKAGE ABN, (2-9-4A, Window 25) or RECIRC PUMP A NO. 2 SEAL LEAKAGE HIGH, (2-9-4A, Window 18)**
	BOP	Alarm Response Procedure: RECIRC PUMP A NO. 2 SEAL LEAKAGE HIGH, (2-9-4A, Window 18) A. N/A B. N/A **NOTES** 1) Possible indications of dual seal failure include: • Window 18 on this panel alarming in conjunction with this window. • Rising drywell pressure and/or temperature. • Increased leakage into the drywell sump. • Increased vibration of the recirc pump. 2) Single recirculation loop operation (SLO) is prohibited in the MELLLA+ operating domain. REFER TO Tech Spec 3.4.1. C. IF dual seal failure is indicated, THEN PERFORM the following: 1. N/A 2. N/A 3. N/A. 4. CLOSE RECIRC PUMP 2A SUCTION VALVE, 2-HS-68-1. 5. CLOSE RECIRC PUMP 2A DISCHARGE VALVE, 2-HS-68-3A. 6. N/A 7. N/A (Unisolable)

Form 3.3-2 Required Operator Actions

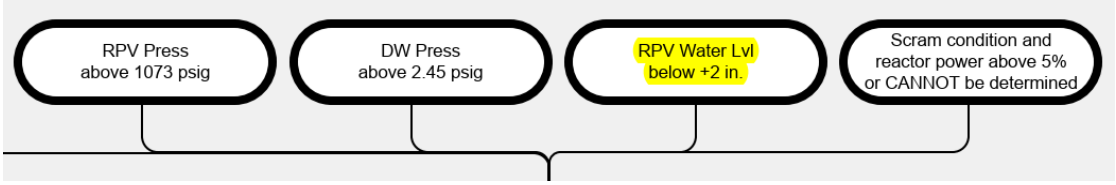
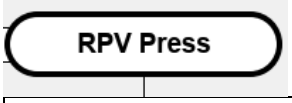
Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 2 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs Reactor SCRAM based on rising Drywell Pressure
	OATC	Inserts a manual Reactor SCRAM 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: • 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. [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.

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 3 of 10

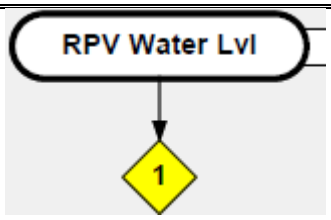
Event Description: Unisolable 2A Recirc Pump Second Seal Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>OATC / BOP</td><td>[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, AND CONTINUE as required.</td></tr>
<tr><td></td><td>NUSO</td><td>Enters 2-EOI-1, RPV Control Modes 1-3 following the Reactor SCRAM based on Reactor Water Level below +2".
 
 
<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>NOT REQUIRED at this time</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-1

<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>
RC/P-2</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>
<tr><td></td><td>NRC</td><td>Depending on NUSO's prioritization of unfolding event, an RPV cool down may be directed below 100ºF/hr now or later when the plant is further stabilized.</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 4 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


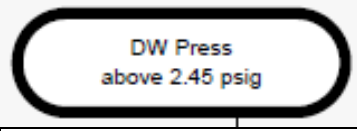
ENSURE each as required:
• PCIS isolations (Groups 1, 2 and 3)
• ECCS
• RCIC
RC/L-1

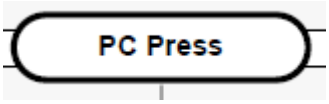
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Raising RPV water lvl above +51in. will facilitate use of shutdown cooling, steam-driven injection systems, or Alternate Injection Subsystems (Table L-2)</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>PC water lvl and drywell press CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/L-2

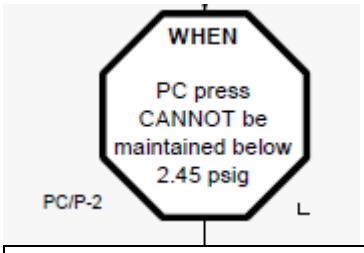
RESTORE and MAINTAIN RPV water lvl between +2 in. and +51 in. with ANY Preferred Injection Systems (Table L-1)
➢OK to use ANY Alternate Injection Subsystems (Table L-2)
RC/L-3
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the OATC or BOP to restore or maintain Reactor Water Level in a band between +2 inches and +51 inches using:
• Condensate and Feedwater - 2-EOI-Appendix-5A</td></tr>
</table>

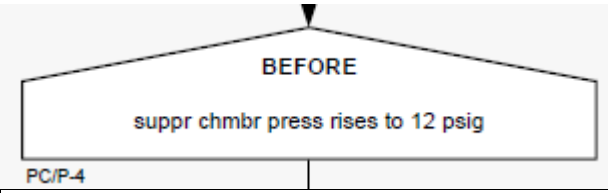
Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 5 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
Enters 2-EOI-2 PRIMARY CONTAINMENT CONTROL


<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>It is determined that core damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr>
</table>
PCC-1


PCC-2


<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>PC press 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>PC press CANNOT be maintained above (-)2 psig</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>Suppr chmbr press approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
<tr><td>DW press approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
PC/P-3


INITIATE suppression chamber sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17)
<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>Directs 2-EOI APPENDIX-17C, RHR System Operation Suppression Chamber Sprays</td></tr>
</table>

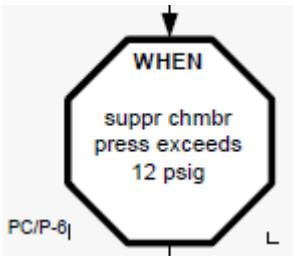
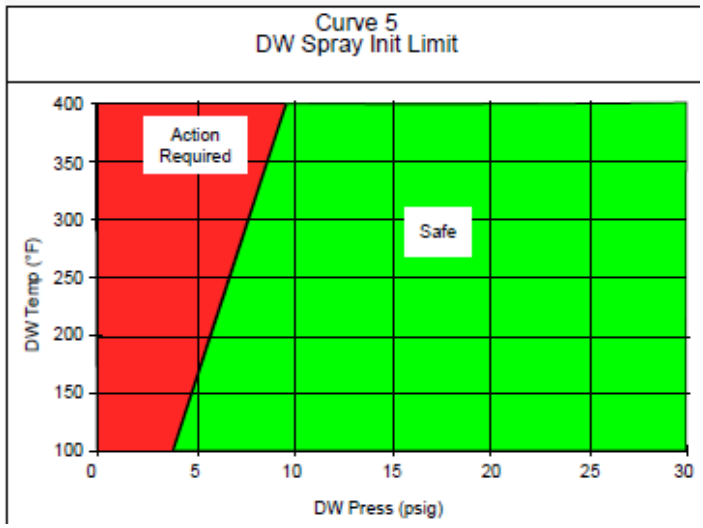
Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 6 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

Time	Position	Applicant's Actions or Behavior
	BOP	2-EOI APPENDIX-17C, RHR System Operation 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 THEN BYPASS LPCI injection valve auto interlock as necessary: PLACE 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS. PLACE 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL 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. [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 SYS I (II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. [5.3] MOMENTARILY PLACE 2-XS-74-121(129), RHR SYS I (II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT. [5.4] IF 2-FCV-74-53 (67), RHR SYS I (II) LPCI INBD INJ VALVE, is OPEN, THEN ENSURE CLOSED 2-FCV-74-52 (66), RHR SYS I (II) LPCI OUTBD INJECT VALVE. [5.5] ENSURE OPERATING the desired RHR System I(II) pump(s) for Suppression Chamber Spray [5.6] ENSURE OPEN 2-FCV-74-57(71), RHR SYS I (II) SUPPR CHBR/POOL ISOL VLV [5.7] OPEN 2-FCV-74-58 (72), RHR SYS I (II) SUPPR CHBR SPRAY VALVE

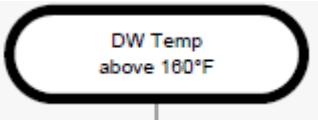
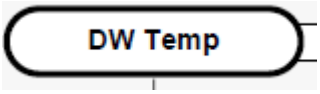
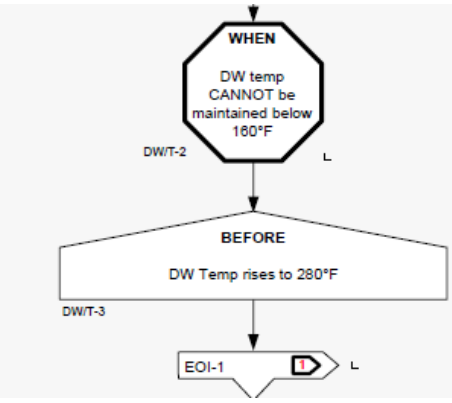
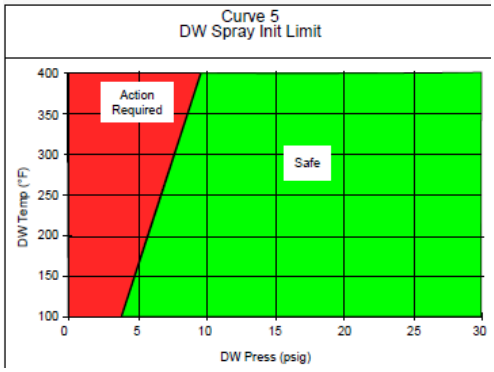
Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 7 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

Time	Position	Applicant's Actions or Behavior
		[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 SYS I (II) MIN FLOW VALVE. [5.10] RAISE System flow with second RHR Sys I (II) pump if needed. [5.11] MONITOR RHR Pump NPSH using Attachment 2. [5.12] ENSURE RHRSW pump supplying desired RHR Heat Exchanger(s). [5.13] THROTTLE in-service RHRSW outlet valves to obtain 1700 to 4500 gpm: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV [5.14] NOTIFY Chemistry that RHRSW is in-service RHR Heat Exchangers. [6] WHEN EITHER of the following exists: Before Suppression Pool pressure drops below 0 psig OR Directed by SRO to stop Suppression Chamber Sprays, THEN STOP Suppr Chamber Sprays as follows: [6.1] CLOSE 2-FCV-74-58(72), RHR SYS I (II) SUPPR CHBR SPRAY VALVE.
		  IF suppression pool level is below 19 ft AND DW temp is in safe area of Curve 5 THEN 1. SHUT DOWN Recirc pumps 2. SHUT DOWN DW blowers 3. INITIATE DW sprays ➢ Use only sources NOT required for continuous RPV injection (APPX 17B) **IF** / **THEN** Needed to augment Drywell Sprays — **NO ACTION REQUIRED** PC/P-7

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 8 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>Directs Drywell Sprays from either the Primary Containment Pressure Leg after Suppression Chamber Pressure reaches 12 psig or the Drywell Temperature Leg before Drywell Temperature rises to 280°F.</td></tr>
<tr><td></td><td>NUSO</td><td>2-EOI-2 PRIMARY CONTAINMENT CONTROL
 
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>It is determined that core damage is occurring due to loss of cooling</td><td>NO ACTION REQUIRED</td></tr></table>PCC-1
 
PCC-2</td></tr>
<tr><td></td><td>NUSO</td><td> 
  </td></tr>
<tr><td></td><td>NUSO</td><td><table><tr><th>IF</th><th>THEN</th></tr><tr><td>If Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr></table>DW/T-4
IF suppression pool level is below 19 ft AND
DW temp is in safe area of Curve 5
THEN
1. SHUT DOWN Recirc pumps
2. SHUT DOWN DW blowers
3. INITIATE DW sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B)
<table><tr><th>IF</th><th>THEN</th></tr><tr><td>Needed to augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr></table>DW/T-5</td></tr>
</table>

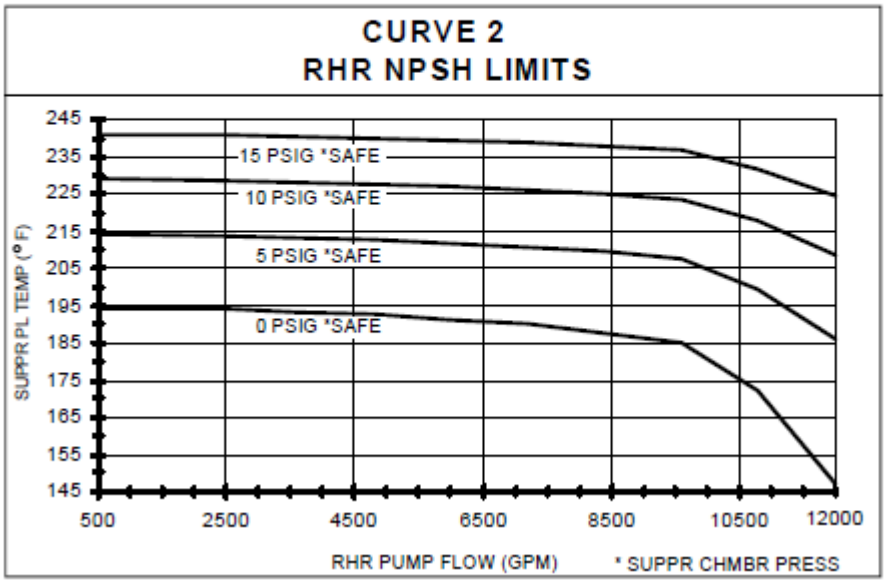
Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 9 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

Time	Position	Applicant's Actions or Behavior
	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 THEN BYPASS LPCI injection valve auto open signal as necessary: • PLACE 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYP SEL IN BYP. • PLACE 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYP SEL IN BYP. [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. • 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 OVRD, in MANUAL OVERRIDE [6.3] MOMENTARILY PLACE 2-XS-74-121 (129), RHR SYS I (II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT. [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 INJ VLV. [6.5] ENSURE OPERATING the desired Sys I (II) RHR pump(s) for Drywell Spray.
	CREW	**Critical Task: With a Primary System discharging into Primary Containment, direct Drywell Sprays before Drywell Temperature exceeds 280 ºF to prevent reaching component qualification and structural design limits.** **Critical Task Failure Criteria: Drywell Sprays are not directed before Drywell Temperature reaches 280 ºF.**
	BOP	[6.6] OPEN the following valves: • 2-FCV-74-60(74), RHR SYS I(II) DW SPRAY OUTBD VLV. • 2-FCV-74-61(75), RHR SYS I(II) DW SPRAY INBD VLV.
	NRC	**The Critical Task is met after Drywell Spray is directed, and a Drywell Spray flow path is aligned via either Loop I or Loop II of RHR.**
	BOP	[6.7] ENSURE CLOSED 2-FCV-074-0007 (0030), RHR SYS I (II) MIN FLOW VALVE. [6.8] IF more Drywell Spray flow is needed, PLACE a second 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 in-service RHRSW outlet vlvs to 1700 to 4500 gpm RHRSW flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 7 Page 10 of 10

Event Description: Unisolable 2A Recirc Pump Second Seal Leak

Time	Position	Applicant's Actions or Behavior
	BOP	[6.12] NOTIFY Chemistry that RHRSW is in-service RHR Heat Exchangers. [7] WHEN EITHER of the following exists: • BEFORE drywell pressure drops below 0 psig • OR directed to stop Drywell Sprays, THEN STOP Drywell Sprays as follows: [7.1] ENSURE CLOSED the following valves: • N/A • 2-FCV-74-60(74), RHR SYS I(II) DW SPRAY OUTBD VLV • 2-FCV-74-61(75), RHR SYS I(II) DW SPRAY INBD VLV [7.2] ENSURE OPEN 2-FCV-74-7(30), RHR SYSTEM I (II) MIN FLOW VALVE. **Attachment 2 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.  **NO remaining steps of 2-EOI APPENDIX-17B, RHR System Operation Drywell Sprays apply for this scenario.**
	NRC	**End of Event 7. Event 8, RCIC Fails to Isolate on High Temperature, is automatically triggered.**

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-4 Event No.: 8 Page 1 of 1

Event Description: RCIC Fails to Isolate on High Temperature

Symptoms/Cues: RCIC STEAM LINE LEAK DETECTION TEMP HIGH (2-9-3D, Window 10), RX BLDG AREA RADIATION HIGH (2-9-3A, Window 22) alarms.

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 8 is built into the scenario file; no action is required to be taken by the driver for RCIC to fail to isolate on high temperature**.
	BOP	Reports the following: • RCIC STEAM LINE LEAK DETECTION TEMP HIGH (2-9-3D, Window 10) • RX BLDG AREA RADIATION HIGH (2-9-3A, Window 22)
	CREW	**Critical Task: When a system fails to isolate, manually isolate the system prior to exceeding Maximum Safe Operating Limit in two areas.** **Critical Task Failure Criteria: The crew fails to isolate a system leaking from Primary to Secondary Containment when a system is isolable.**
	BOP	Closes one of the following valves: • RCIC STEAM LINE INBD ISOLATION VLV, 2-FCV-71-2, • RCIC STEAM LINE OUTBD ISOLATION VLV, 2-FCV-71-3
	NRC	**When the crew has sprayed the Drywell and is controlling Reactor Water Level above the Top of Active Fuel (TAF) (-) 162 inches using any available injection system, end of Scenario**

Scenario Setup
UNIT 2

IC	28
Exam IC	256

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-3	173	AOI-64-2A	27	APPX-17C	20
OI-47A	53	AOI-68-1	39		
OI-68	173	AOI-85-3	28		
OI-85	162	AOI-100-1	123		
ARP-3A	59	TRM 3.4.1	21		
ARP-3D	41	T.S. 3.6.1.3	334		
ARP-4A	52	EOI-1	22		
ARP-4B	52	EOI-2	19		
ARP-5A	65	APPX-5A	9		
ARP-7B	40	APPX-17B	20		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	ILT 2404 NRC Scenario 4 UNIT 2.sch
Event Files(s):	ILT 2404 NRC Scenario 4 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 1	DAYS ONLINE	Drywell Leakage (GPM)	Protected Equipment
	210	1.78	None
Rx Power 100%	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified **Last breaker closure 11/23/23 05:41**	Floor Drain (GPM) 0.20	
MWe 1309		Equipment Drain (GPM) 1.98	

☐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

None

LCOs OF 72 HOURS OR LESS

None

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

2B Reactor Water Cleanup (RWCU) Pump is tagged out of service for maintenance.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Down power 5% with core flow in accordance with Reactivity Control Plan (RCP).

Perform Automatic Start Test of Reactor Feed Pump Turbine (RFPT) 2C oil pumps in accordance with 2-OI-3, Reactor Feedwater System, Section 8.15. The prior shift completed 2A and 2B RFPT oil pumps testing.

OPERATOR WORK AROUNDS OWAs – U0-0/U2-0 Burdens - U0 – 0/U2-0 Operator Challenges – U0 – 0/U2 - 0

None

ODMIs/ACMPs

None

ONEAs

None

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

None

SCHEDULED ITEMS NOT COMPLETED

None

Form 3.3-1 **Scenario Outline**

Facility: BFN Scenario Number: NRC-5
Scenario Source: NEW Op-Test Number: 24-04

Examiners: _____ Operators: NUSO: _____
_____ OATC: _____
_____ BOP: _____

Initial Conditions: 2% Reactor Power. The Emergency High Pressure Make-up (EHPM) Pump is tagged out of service.

Turnover: Perform 2-OI-47, Turbine-Generator System, Section 6.10.1, Alternate Bus Duct Cooling Fans, placing 2B Bus Duct Fan in service. Continue Reactor start-up, raise Reactor Power with Control Rods.

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. Following Emergency Depressurization, inject to the Reactor using low pressure injection systems to restore Reactor Water Level above (-) 180 inches or establish Spray Cooling with either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above (-) 215 inches to preclude SAMG entry.

Event Number	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-NUSO	Alternate Bus Duct Cooling Fans
2.	N/A	R-OATC	Raise Reactor Power using Control Rods
3.	RD22	I-OATC MC-OATC I-NUSO	2-FIC-85-11, CRD SYSTEM FLOW INDICATING CONTROLLER, Signal Fails Low
4.	XS-92-7/42B XS-92-7/42H	C-OATC TS-NUSO	Intermediate Range Monitor (IRM) 'H' Fails Upscale **AND** IRM 'B' Fails Downscale
5.	XA-55-4C_13	C-BOP C-NUSO	Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low
6.	XA-55-6B_10	TS-NUSO	2-RM-90-259A, CONTROL ROOM RADIATION MONITOR, Fails Downscale
7.	TH18A	C-BOP MC-BOP C-NUSO	2A1 Recirc Drive Cooling Water Pump Trips, Standby Fails to Automatically Start
8.	FW18	M-All	Feedwater 'A' Line Break
9.	RC04	C-BOP MC-BOP C-NUSO	Reactor Core Isolation Cooling (RCIC) Flow Controller Automatic Mode Fails

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC) Manual Control

Events

1. The crew will alternate Bus Duct Cooling Fans in accordance with 2-OI-47, Turbine-Generator System, Section 6.10.1.

2. The crew will raise Reactor Power using Control Rods in accordance with 2-OI-85, Control Rod Drive System.

3. The crew will respond to 2-FIC-85-11, CRD SYSTEM FLOW INDICATING CONTROLLER, failing low, the OATC will take manual control of the Control Rod Drive (CRD) flow controller.

4. IRM HIGH (2-9-5A, Window 26) alarms and half SCRAM occurs from IRM 'H' failing upscale. The crew will respond, bypassing IRM 'H' and resetting the half SCRAM. IRM DOWNSCALE (2-9-5A, Window 19) alarms and a Control Rod Block occurs from IRM 'B' failing downscale. No operator action is required. The Nuclear Unit Senior Operator (NUSO) addresses Technical Specifications (Tech Specs).

5. RBCCW SURGE TANK LEVEL LOW (2-9-4C, Window 13) alarms, the crew will respond in accordance with the Alarm Response Procedure and add water.

6. CONTROL ROOM ISOLATION RAD MONITOR DOWNSCALE OR INOP (2-9-6B, Window 10) alarms and the NUSO addresses Tech Specs for 2-RM-90-259A, CONTROL ROOM RADIATION MONITOR, downscale.

7. RECIRC DRIVE 2A COOLANT FLOW LOW (2-9-4A, Window 12) alarms. The crew will respond to a 2A1 Recirc Drive Cooling Water Pump trip, the 2A2 Standby Pump fails to automatically start and must be started manually.

8. The crew will respond to a Feedwater Line 'A' break requiring a SCRAM. The Reactor Bottom Head leaks when the MODE Switch is placed in SHUTDOWN causing a loss of Reactor Water inventory.

9. When RCIC is used for Reactor Vessel injection, the automatic flow controller will fail and the crew will take manual control of RCIC.

The Scenario ends when the crew has Emergency Depressurized and has restored Reactor Water Level above (-) 180 inches or established Spray Cooling.

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.
   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**
   Reactor Pressure trend.
   MSRV open status indicators.

   e. **Critical Task Failure Criteria:**
   Emergency Depressurization is not directed before Reactor Water Level reaches (-) 180 inches.

2. Following Emergency Depressurization, inject to the Reactor using low pressure injection systems to restore Reactor Water Level above (-) 180 inches or establish Spray Cooling with either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above (-) 215 inches to preclude SAMG entry.

a. Safety Significance
Maintain Adequate Core Cooling.
Prevent degradation of fission product barrier.

b. Cues
Procedural Compliance.
Reactor Water Level trend.

c. Measured by:
Observation – low pressure injection systems are used to raise Reactor Water Level.

d. Feedback:
Reactor Pressure.
Reactor Water Level.

e. Critical Task Failure Criteria:
The crew does not restore and maintain Reactor Water Level above (-) 180 inches AND Spray Cooling cannot be established, resulting in SAMG entry being required. (Table M-3 Spray Cooling is either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above -215 inches)

Form 3.3-2 Required Operator Actions

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 1 Page 1 of 1

Event Description: Alternate Bus Duct Cooling Fans

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 pre-briefed but may elect to hold a re-focus brief.**
	NUSO	Directs the Balance of Plant Operator (BOP) to alternate Bus Duct Cooling Fans in accordance with 2-OI-47, Turbine-Generator System, Section 6.10.1
	BOP	2-OI-47, Turbine-Generator System Section 6.10.1, Alternate Bus Duct Cooling Fans **CAUTIONS** 1) Starting Generator Bus Duct Heat Exchanger Fan while rotating greater than 100 rpm in the reverse direction creates potential to damage the fan. U-2 GEN BUS DUCT HTX FAN 2A(2B) BACKDRAFT DMPR, 2-DMP-262-0051(0052), should preclude reverse rotation of the associated fan at speeds greater than 100 rpm. It is acceptable to try restarting the fan one time immediately if the breaker trips on the first attempt. 2) Be aware of any abnormal vibrations or audible sounds in dual fan operation with inlet vane dampers full open ref. EWR20MEB262128 Rev. 0. [2] Starting Bus Duct Cooling Fan B/Stopping Fan A [2.4] MOMENTARILY PLACE GEN BUS DUCT HX FAN B, 2-HS-262-0002A, in START on Panel 2-9-8 • CHECK for proper fan operation. [2.5] MOMENTARILY PLACE GEN BUS DUCT HX FAN A, 2-HS-262-0001A, in STOP on Panel 2-9-8.
	NRC	**EXAMINER NOTE: Steps [2.1] [2.2] [2.3] are peformed locally at the Bus Duct Fans. The BOP will perfom steps [2.4] [2.5].**
	Driver	**When called to perform steps [2.1] [2.2] [2.3], wait one minute and report, "2-OI-47, section 6.10.1, Alternating Operating Bus Duct Cooling Fans, steps [2.1] [2.2] [2.3] are all complete."**
	NRC	**End of Event 1. Proceed to Event 2, Raise Reactor Power using Control Rods.**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 2 Page 1 of 1

Event Description: Raise Reactor Power using Control Rods

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. The crew may also elect to review 2-OI-85, sections 6.6.1, Initial Conditions prior to withdrawing Control Rods and 6.6.2, Actions Required during and Following Control Rod Withdrawal then continue to 6.6.3 Control Rod Notch Withdrawal**
	NUSO	Directs the Operator at the Controls (OATC) to raise Reactor Power using Control Rods in accordance with 2-OI-85, Control Rod Drive System the Reactivity Control Plan (RCP). RCP-NRC2404-U2S5.
	OATC	2-OI-85, Control Rod Drive System 6.6.3 Control Rod Notch Withdrawal [1] SELECT the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40. [2] ENSURE CRD DRIVE WTR HDR DP is between 250 -270 psid on 2-PDI-85-17A by throttling CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A, as needed. [3] IF selected control rod has been identified as a "fast notching." control rod by engineering, THEN THROTTLE OPEN 2-PCV-085-0023 CRD, CRD DRIVE WATER PRESS CONTROL VLV, to as low as 180 psid using 2-HS-85-23A. (Otherwise N/A) [4] OBSERVE the following for selected control rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED. • White light on the Full Core Display ILLUMINATED • Rod Out Permit light ILLUMINATED. [5] ENSURE ROD WORTH MINIMIZER operable and LATCHED in to correct ROD GROUP when Rod Worth Minimizer is enforcing. [6] PLACE CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH and RELEASE. [7] OBSERVE control rod settles into desired position AND ROD SETTLE light extinguishes.
	NRC	**End of Event 2, proceed to Event 3, 2-FIC-85-11, CRD SYSTEM FLOW INDICATING CONTROLLER, Signal Fails Low**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 3 Page 1 of 1

Event Description: 2-FIC-85-11, CRD SYSTEM FLOW INDICATING CONTROLLER, Signal Fails Low

Symptoms/Cues: Event is automatically triggered, 2-FIC-85-11, Signal Fails Low and 2-FCV-85-11A responds by fully opening

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 3 to fail low the signal to 2-FIC-85-11.**
	OATC	Acknowledges and reports the following alarm(s) to the NUSO: CRD DRIVE WTR FILTER DIFF PRESS HIGH, 2-PDA-85-10 (2-9-5A, Window 03)
	NUSO	Directs OATC to respond in accordance with the Alarm Response Procedure(s).
	OATC	Alarm Response Procedure: CRD DRIVE WTR FILTER DIFF PRESS HIGH, 2-PDA-85-10 (2-9-5A, Window 03) Operator Actions: A. IF alarm resets immediately AND 2-PDI-85-17A & 18A indicate hard downscale, THEN CHECK CLOSED Lighting Cabinet 202, Breaker 3. B. CHECK alarm as follows: 1. CHECK signal from CRD SYSTEM FLOW CONTROL, 2-FIC-85-11, to the flow regulating valve 2-FCV-85-11A or -11B abnormally high. 2. DISPATCH PERSONNEL to 2-LPNL-925-0018A, to check local DP gauge, 2-PDIS-085-0010. C. N/A D. N/A
	Driver	• **If contacted to CHECK CLOSED Lighting Cabinet 202, Breaker 3, wait two minutes and report "Lighting Cabinet 202, Breaker 3 was found in the closed position and nothing abnormal is noted".** • **If contacted to check local DP gauge, 2-PDIS-085-0010, wait two minutes and report "2-PDIS-085-0010 indicates 22 PSID"**
	NRC	**The crew may attempt to switch Flow Control Valves in accordance with 2-AOI-85-3, CRD System Failure. However, if the Flow Transmitter is failed low, neither set of Flow Control Valves will operate in automatic. 2-FIC-85-11, CRD SYSTEM FLOW CONTROL, must be placed in MANUAL.**
	OATC	Determines that the CRD Flow Transmitter has failed low, causing 2-FCV-85-11A, CRD LINE A FLOW CONTROL VALVE to OPEN. 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.
	NRC	**End of Event 3, proceed to Event 4 Intermediate Range Monitor (IRM) 'H' Fails Upscale and IRM 'B' Fails Downscale.**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 4 Page 1 of 6

Event Description: Intermediate Range Monitor (IRM) 'H' Fails Upscale and IRM 'B' downscale.

Symptoms/Cues: IRM 'H' fails upscale causing a half SCRAM and a Control Rod Block. IRM 'B' fails downscale causing Control Rod Block. The two simultaneous failures cause numerous alarms, listed below in the guide.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 4 to Fail IRM 'H' Upscale and IRM 'B' downscale.**
	NRC	**EXAMINER NOTE: Due to the numerous alarms the BOP may assist in addressing alarms. The OATC is expected to take actions on panel 2-9-5.**
	OATC	Acknowledges and reports the following alarm(s) to the NUSO: IRM 'H' upscale alarms. • NEUTRON MONITORING SYS HALF SCRAM (2-9-4A, Window 16) • CONTROL ROD WITHDRAWAL BLOCK (2-9-5A, Window 7) • IRM HIGH (2-9-5A, Window 26) • IRM CH B, D, F, H HI-HI/INOP (2-9-5A, Window 34) • REACTOR CHANNEL B AUTO SCRAM, (2-9-5B, Window 02) IRM 'B' downscale alarms. • CONTROL ROD WITHDRAWAL BLOCK (2-9-5A, Window 7) • IRM DOWNSCALE (2-9-5A, Window 19)
	NUSO	Directs OATC and BOP to respond to Alarm Response Procedure(s).
	OATC	Alarm Response Procedure: NEUTRON MONITORING SYS HALF SCRAM (2-9-4A, Window 16) Operator Actions: A. CHECK vertical section of Panel 2-9-5 for APRM HIGH or Inop (red light). B. IF the cause is a sensor malfunction, THEN one IRM may be bypassed in each RPS channel, and one APRM channel may be bypassed. REFER TO 2-OI-92 cautions. C. IF in the startup mode (Mode 2), THEN CHECK the range switch position. D. MONITOR the APRM/IRM recorders on Panel 2-9-5. E. N/A F. N/A G. N/A H. IF initiating condition is corrected, THEN PERFORM the following: 1. RESET Half Scram with UNIT SRO permission. 2. REFER to 2-OI-99. I. N/A

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 4 Page 2 of 6

Event Description: IRM 'H' Fails Upscale and IRM 'B' downscale.

Time	Position	Applicant's Actions or Behavior
	OATC	Alarm Response Procedure: CONTROL ROD WITHDRAWAL BLOCK (2-9-5A, 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. IF NO corresponding alarm exists, THEN 1. N/A. 2. N/A 3. N/A 4. WHEN REACTOR MODE SWITCH is in START-UP position, THEN CHECK IRM detectors NOT FULL IN. E. REFER TO Tech Spec Table 3.3.2.1-1, TRM Table 3.3.4-1.
	OATC	Alarm Response Procedure: IRM HIGH (2-9-5A, Window 26) Operator Actions: A. ACKNOWLEDGE alarm and IRM response by multiple indications. B. CHECK control rod block. C. IF IRM response is NOT as expected, THEN BYPASS initiating channel. REFER TO 2-OI-92A. D. N/A E. N/A F. NOTIFY Reactor Engineer. G. REFER TO Tech Spec Table 3.3.1.1-1, TRM Tables 3.3.4-1 and 3.3.5-1

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 4 Page 3 of 6

Event Description: IRM 'H' Fails Upscale and IRM 'B' downscale.

Time	Position	Applicant's Actions or Behavior
	OATC	Alarm Response Procedure: IRM CH B, D, F, H HI-HI/INOP (2-9-5A, Window 34) Operator Actions: A. STOP any reactivity changes. B. CHECK alarm by multiple indications. C. PLACE range switch in a position to clear alarm or BYPASS initiating channel to reset Half-Scram. REFER TO 2-OI-92A. D. With SRO permission, RESET Half Scram. REFER TO 2-OI-99. E. N/A F. N/A G. NOTIFY Instrument Maintenance that functional tests of any monitor indicating an inop condition, including a downscale reading, are required before the instrument can be considered operable. [NRC IE item 86-40] H. NOTIFY Reactor Engineer. I. REFER TO Tech Spec Table 3.3.1.1-1, TRM Tables 3.3.4-1 and 3.3.5-1. J. N/A
	OATC	Alarm Response Procedure: REACTOR CHANNEL B AUTO SCRAM (2-9-5B, Window 02) Operator Actions: A. CHECK channel B relays dropped out by checking scram solenoid and backup scram valve lights extinguished. B. N/A C. N/A D. N/A E. With SRO permission, RESET half-scram. REFER TO 2-OI-99. F. N/A G. N/A I. REFER TO Tech Spec Table 3.3.1.1-1, TRM Table 3.3.4-1 and 3.3.5-1. J. N/A

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 4 Page 4 of 6

Event Description: IRM 'H' Fails Upscale and IRM 'B' downscale.

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs the OATC to bypass IRM 'H' in accordance with 2-OI-92A, Intermediate Range Monitors. Directs the OATC to RESET half-scram. REFER TO 2-OI-99.
	OATC	2-OI-92A, Intermediate Range Monitors Section 6.1, Bypassing an IRM Channel **CAUTION** [QA/QC] NPG-SPP 10.4 requires approval of the Plant Manager or his designee prior to any planned operation with IRMs bypassed unless bypassing is specifically allowed within approved procedures. [ISE-NPS-92-RO1] **NOTES** 1) It is not necessary for a bypassed IRM channel to have its detector inserted into the core. 2) Only one IRM in each trip system can be bypassed at a time. 3) All operations are performed on Panel 2-9-5 unless specifically stated otherwise. [1] REVIEW all precautions and limitations in Section 3.0. [2] PLACE the appropriate IRM Bypass selector switch to the BYPASS position: • IRM BYPASS, 2-HS-92-7A/S4A • IRM BYPASS, 2-HS-92-7A/S4B [3] CHECK that the Bypassed light is illuminated
	OATC	Alarm Response Procedure: IRM DOWNSCALE (2-9-5A, Window 19) Operator Actions: A. ACKNOWLEDGE alarm and IRM response by multiple indications. B. N/A C. N/A D. N/A E. NOTIFY Reactor Engineer. F. REFER TO Tech Spec Tables 3.3.1.1-1, TRM Tables 3.3.4-1 and 3.3.5-1.

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 4 Page 5 of 6

Event Description: IRM 'H' Fails Upscale and IRM 'B' downscale.

Time	Position	Applicant's Actions or Behavior
	NRC	**The following note applies to all ARP references listed in event 4.** • **Tech Spec Table 3.3.2.1-1 DOES NOT APPLY.** • **TRM Table 3.3.4-1 NOT REQUIRED but NUSO may declare INFO ONLY.** • **TRM Table 3.3.5-1 NOT REQUIRED but NUSO may declare INFO ONLY.** **(For TRM 3.3.4-1 and 3.3.5-1, 6 operable IRM channel remain, therefor no TRM LCO applies)** **(For Tech Spec 3.3.2.1, this is for Rod Block Monitor and Rod Worth Minimizer, NOT IRMs)** **The NUSO should enter TS 3.3.1.1 CONDITION A**
	Driver	**If contacted as the Reactor Engineer, Instrument Mechanic, or Work Control, acknowledge any information or direction given.**
	NUSO	RPS Instrumentation 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. ----------

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 4 Page 6 of 6

Event Description: IRM 'H' Fails Upscale and IRM 'B' downscale.

Time	Position	Applicant's Actions or Behavior
	NUSO	

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours OR In accordance with the Risk Informed Completion Time Program
	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 OR In accordance with the Risk Informed Completion Time Program

	NRC	**End of Event 4. Request that the Driver insert Event 5, Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 5 Page 1 of 1

Event Description: Reactor Building Closed Cooling Water (RBCCW) Surge Tank Low

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. RBCCW SURGE TANK LEVEL LOW (2-9-4C, Window 13) alarms.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 5 to cause a low-level alarm from the RBCCW Surge Tank.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • RBCCW SURGE TANK LEVEL LOW (2-9-4C, Window 13)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: RBCCW SURGE TANK LEVEL LOW (2-9-4C, Window 13) Operator Actions: A. ADD water to the RBCCW Surge Tank for approximately one minute or until low level alarm resets using the following: • RBCCW SYS SURGE TANK FILL VLV, 2-FCV-70-1 (Panel 2-9-4) OR • FCV-70-1 BYPASS VLV, 2-HCV-2-1369 (locally). B. N/A. C. N/A D. N/A
	NRC	**The examinee is expected to add water to the RBCCW Surge Tank with 2-FCV-70-1 on (Panel 2-9-4).**
	NRC	**End of Event 5, proceed to Event 6, 2-RM-90-259A, CONTROL ROOM RADIATION MONITOR, Fails Downscale.**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 6 Page 1 of 3

Event Description: 2-RM-90-259A, CONTROL ROOM RADIATION MONITOR, Fails Downscale

Symptoms/Cues: Event is initiated by the simulator booth when requested by the Chief Examiner. CONTROL ROOM ISOLATION RAD MONITOR DOWNSCALE OR INOP 0-RA-90-259B/2 (2-9-6B, Window 10) alarms.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 6 to cause 2-RM-90-259A, CONTROL ROOM RADIATION MONITOR downscale.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • CONTROL ROOM ISOLATION RAD MONITOR DOWNSCALE OR INOP 0-RA-90-259B/2 (2-9-6B, Window 10)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	BOP	Alarm Response Procedure: CONTROL ROOM ISOLATION RAD MONITOR DOWNSCALE OR INOP 0-RA-90-259B/2 (2-9-6B, Window 10) Operator Actions: A. CHECK alarm by monitoring the following: 1. 1-RI-90-8A on Panel 1-9-11. 2. STACK GAS / CONT RM RADIATION, 0-RR-90-147, Panel 1-9-2. • CREVs Rad MON, 0-RM-90-259A • CREVs Rad MON, 0-RM-90-259B B. NOTIFY Unit SRO. (Unit SRO will determine when to declare the cam inoperative). C. IF equipment is inoperable, THEN NOTIFY instrument department. D. CHECK Tech. Spec. Section 3.3.7.1 for inoperable monitor limitations. E. OBSERVE LCO for any inoperable radiation monitor. F. NOTIFY Rad Pro to PLACE an alarming portable CAM in the control room when required channel INOP. G. N/A
	NRC	**The examinee may elect to dispatch personnel locally to 0-RM-90-259A(B) AND/OR contact Unit 1 for readings, any single method is acceptable.**
	Driver	• **If contacted as Radiation Protection, Instrument Mechanics or Work Control, acknowledge any information or direction given.** • **If contacted to check 0-RM-90-259A(B) as Unit 1 or an AUO, then wait two minutes and report "0-RM-90-259A is indicating downscale and 0-RM-90-259B is reading normal and steady in band"**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 6 Page 2 of 3

Event Description: 2-RM-90-259A, CONTROL ROOM RADIATION MONITOR, Fails Downscale

Time	Position	Applicant's Actions or Behavior
	NRC	**EXAMINER NOTE: The NUSO should enter TS 3.3.7.1 CONDITION D, REQUIRED ACTION D.1, COMPLETION TIME Once per 24 Hours.**
	NUSO	CREV System Instrumentation 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.

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 3.3.7.1.5 SR 3.3.7.1.6	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 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 3.3.7.1.5 SR 3.3.7.1.6	≤ 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 3.3.7.1.5 SR 3.3.7.1.6	≤ 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

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 6 Page 3 of 3

Event Description: 2-RM-90-259A, CONTROL ROOM RADIATION MONITOR, Fails Downscale

Time	Position	Applicant's Actions or Behavior
	NUSO	See table below
	NRC	**End of Event 6, proceed to Event 7, 2A1 Recirc Drive Cooling Water Pump Trips, Standby Fails to Automatically Start.**

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
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

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 7 Page 1 of 1

Event Description: 2A1 Recirc Drive Cooling Water Pump Trips, Standby Fails to Automatically Start

Symptoms/Cues: RECIRC DRIVE 2A COOLANT FLOW LOW (2-9-4A, Window 12) alarms.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 7 to trip 2A1 Recirc Drive Cooling Water Pump.**
	BOP	Acknowledges and reports the following alarm(s) to the NUSO: • RECIRC DRIVE 2A COOLANT FLOW LOW (2-9-4A, Window 12) • RECIRC DRIVE 2A PROCESS ALARM (2-9-4A, Window 28)
	NUSO	Directs BOP to respond in accordance with the Alarm Response Procedure(s).
	NRC	**NUSO may direct or BOP may invoke OPDP-1 and take manual action to start 2A2 Recirc Drive Cooling Water Pump before pulling ARPs.**
	BOP	Alarm Response Procedure: RECIRC DRIVE 2A COOLANT FLOW LOW (2-9-4A, Window 12) Operator Actions: A. ENSURE RECIRC DRIVE cooling water pump running. B. DISPATCH personnel to the RECIRC DRIVE to check the operation of the Recirc Drive cooling water system.
	BOP	Alarm Response Procedure: RECIRC DRIVE 2A PROCESS ALARM (2-9-4A, Window 28) Operator Actions: A. N/A B. REFER TO ICS screen "VFDAAL" and determine cause of alarm C. N/A D. N/A E. N/A F. For all other alarms, or any problems encountered NOTIFY system engineering. G. N/A
	Driver	**If contacted as the Reactor Engineer, Work Control, or an AUO acknowledge any information or direction given.**
	NRC	**End of Event 7, proceed to Event 8, Feedwater Line 'A' break AND Reactor Bottom Head leaks when the MODE Switch is placed in SHUTDOWN causing a loss of Reactor Water inventory.**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 1 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Symptoms/Cues: Condensate and Feedwater flow initially rising yet flow to the vessel lowers. Condensate Booster Pump and Reactor Feedwater pump NPSH alarms.

Time	Position	Applicant's Actions or Behavior
	Driver	**When requested by the Chief Examiner, insert Event 8, break Feedwater Line 'A'. Reactor Bottom Head leaks when the MODE Switch is placed in SHUTDOWN causing a loss of Reactor Water inventory.**
	NRC	**When the Reactor MODE SWITCH is placed in SHUTDOWN, the following Events will occur:** • **RCIC Automatic Flow controller, FIC-71-36-A will fail to work in automatic yet be available to the crew when placed in manual (See Event 9 on page 35)** • **A 100% Reactor Bottom Head leak develops and a Recirc Suction Pipe leak ramps to 1% over 5 minutes to assist with the loss of Reactor Water inventory.** • **The HPCI Auxiliary Oil Pump is failed rendering HPCI unavailable and will not be returned to the crew (this failure is part of the schedule file, and not triggered by the MODE SWITCH in SHUTDOWN).**
	NUSO	When the Reactor Feedwater leak is apparent to the crew, directs the OATC to insert a manual Reactor SCRAM.
	OATC	Inserts a manual Reactor SCRAM 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: • 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

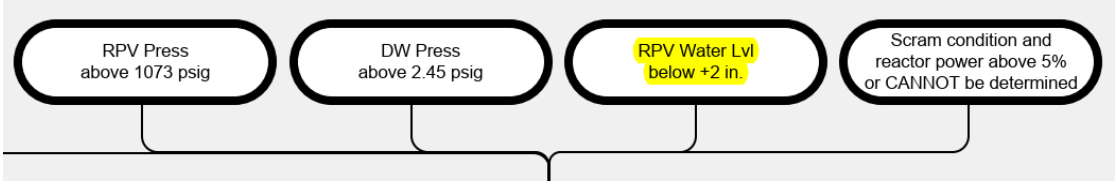
Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 2 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory

Time	Position	Applicant's Actions or Behavior
	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.
	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.

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 3 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	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.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 4 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


<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>NOT REQUIRED at this time</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-1

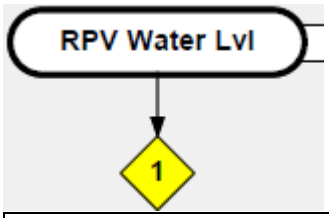
STABILIZE RPV Pressure below 1073 psig using the Main Turbine Bypass Valves (APPX 8B)

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

<table>
<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>
RC/P-2
</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>
<tr><td></td><td>NRC</td><td>Depending on NUSO's prioritization of unfolding event, an RPV cool down may be directed below 100ºF/hr now or later when the plant is further stabilized.</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 5 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


ENSURE each as required:
• PCIS isolations (Groups 1, 2 and 3)
• ECCS
• RCIC
RC/L-1

<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Raising RPV water lvl above +51in. 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>PC water lvl and drywell press CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/L-2

RESTORE and MAINTAIN RPV water lvl 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-3
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the OATC or BOP to restore or maintain Reactor Water Level in a band between +2 inches and +51 inches using any of the following EOI Appendices:
• CRD - 2-EOI-Appendix-5B
• RCIC - 2-EOI-Appendix-5C
• EHPM Pump - 2-EOI-Appendix-7L (crosstie from unit 1/3 may be requested)
• SLC (boron or test tank) - 2-EOI-Appendix-7B</td></tr>
<tr><td></td><td>Driver</td><td>If contacted as Unit 1 or 3 to crosstie the EHPM for injection, acknowledge direction given. If crew follows up, state "one hour minimum to align EHPM".</td></tr>
<tr><td></td><td>NRC</td><td>EXAMINER NOTE:
• Reactor Water Level approaching the Top of Active Fuel (TAF) can be monitored by typing "PAL" on any ICS PC screen.
• EHPM crosstie from another unit will not be able to be aligned.</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 6 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

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

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 7 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

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 [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')

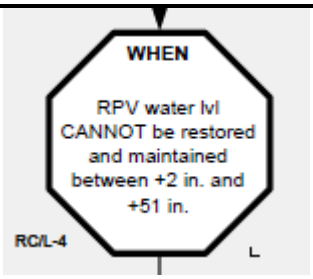
Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 8 of 20

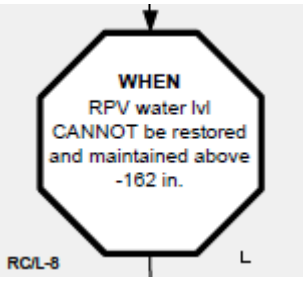
Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	OATC	2-EOI Appendix-7B, Alternate RPV Injection System Lineup SLC System [1] N/A [2] IF RPV injection is needed immediately ONLY to prevent or mitigate fuel damage, THEN, CONTINUE at Step 1.0[10] to inject SLC Boron Tank to RPV.THEN LINE UP ALL available CRD pumps to the RPV as follows: [3] DISPATCH personnel to Unit 2 SLC pump area to line up SLC Test Tank as follows (RB NE, El 639 ft): [3.1] through [3.9] performed in the field. [4] UNLOCK and PLACE 2-HS-63-6A, SLC PUMP 2A/2B, control switch in START-A or START-B (Panel 9-5). [5] CHECK demineralized water is injecting to the RPV by observing the following: • Selected pump running, 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 Annunciator in alarm (2-XA-55-5B, Window 20). • 2-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. • System flow, as indicated by 2-IL-63-11, SLC FLOW, red light illuminated. • SLC INJECTION FLOW TO REACTOR, Annunciator in alarm (2-XA-55-5B, Window 14). [6.0] N/A [7] NOTIFY personnel in SLC area to perform the following: [7.1] and [7.2] performed in the field. [8] MONITOR and CONTROL SLC System as necessary to maintain injection. [9] N/A [10] UNLOCK and PLACE 2-HS-63-6A, SLC PUMP 2A/2B, control switch in START-A or START-B (Panel 9-5). [11.0] Refer to step [5] above for step [11] indications of SLC injection. [12] N/A [13] IF SLC tank level drops to 0%, THEN STOP SLC pumps.

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 9 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

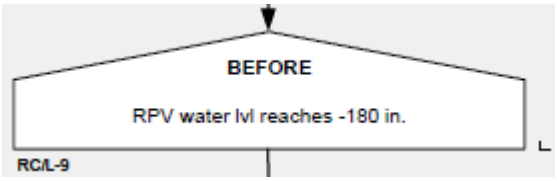
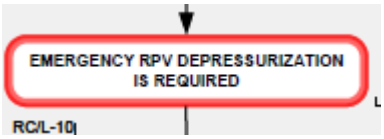
<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>


RC/L-4
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>RPV water lvl can be restored and maintained above +2 in.</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/L-5
<table>
<tr><td>IF RPV water lvl CANNOT be maintained above -122 in.
THEN INHIBIT ADS</td></tr>
</table>
RC/L-6
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the OATC or BOP to Inhibit ADS</td></tr>
<tr><td></td><td>BOP</td><td>Inserts key and inhibits both divisions of ADS</td></tr>
<tr><td></td><td>NUSO</td><td>
RESTORE and MAINTAIN RPV water lvl above -162 in. with ANY Preferred Injection Systems (Table L-1)
➢OK to use ANY Alternate Injection Subsystems (Table L-2)
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>PC water lvl and drywell press CANNOT be maintained in the safe area of Curve 7</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/L-7


RC/L-8
</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 10 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	NUSO	 RC/L-9  RC/L-10
	NRC	**The NUSO may determine that Emergency Depressurization is required any time before Reactor Water Level reaches the Minimum Steam Cooling RPV Water Level (MSCRWL) of (-)180 inches if it is believed that Reactor Water Level cannot be restored and maintained above the Top of Active Fuel (TAF) of (-) 162 inches AND it is believed that low-pressure injection sources will restore Reactor Water Level above TAF.**
	NUSO	Determines and makes a crew update that Emergency Depressurization is required.
	Crew	**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 before Reactor Water Level reaches (-) 180 inches.**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 11 of 20

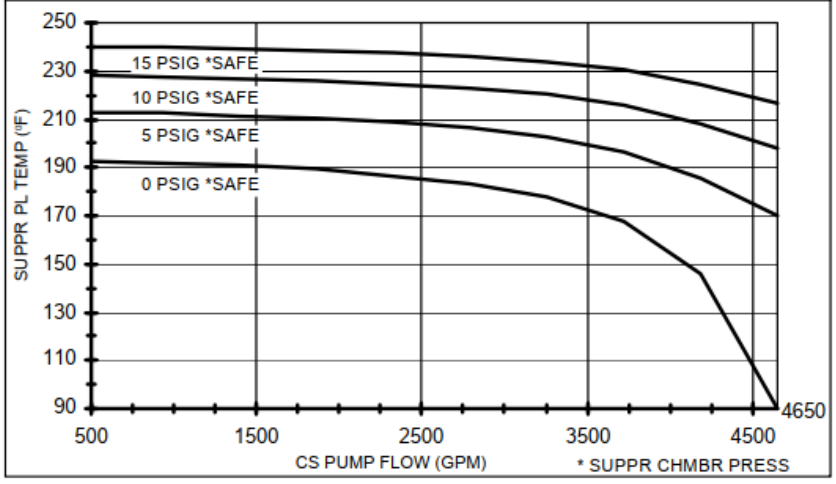
Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td>
<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 Subsystems (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-6
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)
RC/P-7
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
<table>
<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 5 MSRVs can be opened AND
RPV Pressure is 50 psi or more above Suppression Chamber Pressure</td><td>NO ACTION REQUIRED</td></tr>
</table>
RC/P-8
</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the crew to open 6 MSRVs (ADS Valves preferred)</td></tr>
<tr><td></td><td>NUSO</td><td>Directs the crew to inject to the Reactor using low-pressure systems:
• 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>Crew</td><td>Critical Task: Following Emergency Depressurization, inject to the Reactor using low pressure injection systems to restore Reactor Water Level above (-) 180 inches or establish Spray Cooling with either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above (-) 215 inches to preclude SAMG entry.

Critical Task Failure Criteria: The crew does not restore and maintain Reactor Water Level above (-) 180 inches AND Spray Cooling cannot be established, resulting in SAMG entry being required. (Table M-3 Spray Cooling is either Core Spray subsystem operating with at least 6,250 gpm to the RPV AND RPV water level above -215 inches)</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 12 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

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 or OATC	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. **CURVE 1** **CS NPSH LIMITS** 

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 13 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	BOP or OATC	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.

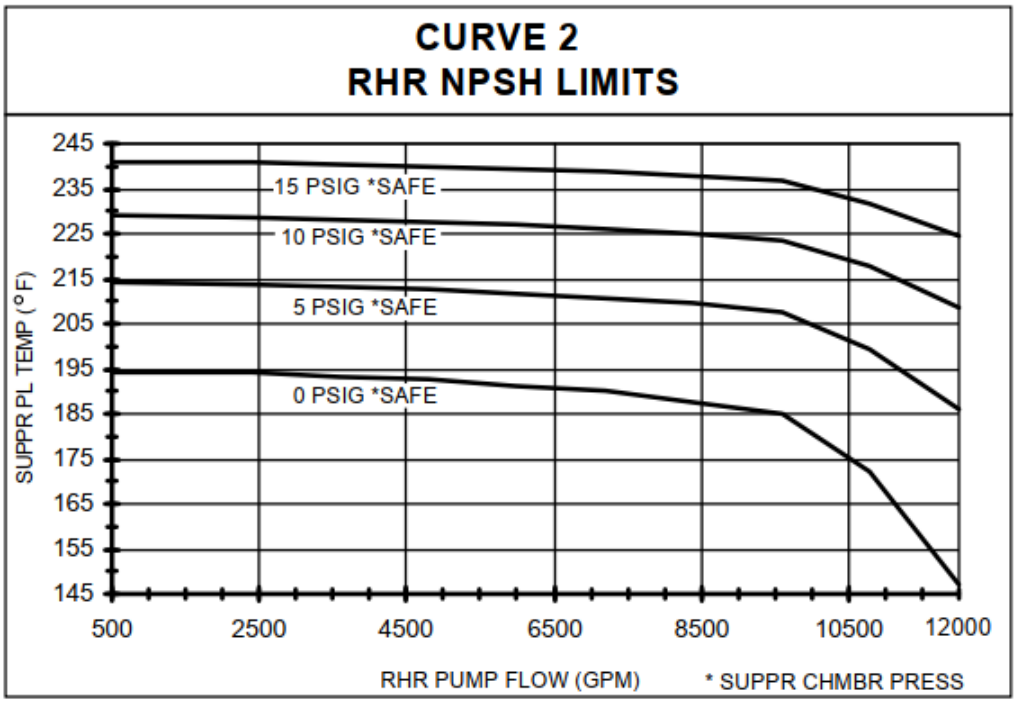
Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 14 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
		[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

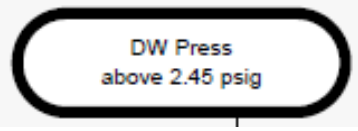
Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used.



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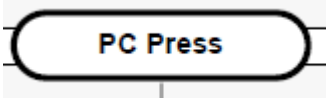
Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	NUSO	Enters 2-EOI-2 PRIMARY CONTAINMENT CONTROL

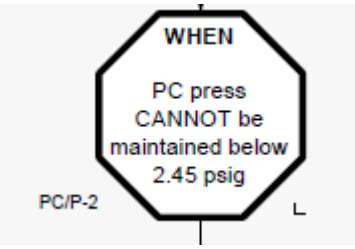


IF	THEN
It is determined that core damage is occurring due to loss of cooling	**NO ACTION REQUIRED**

PCC-1

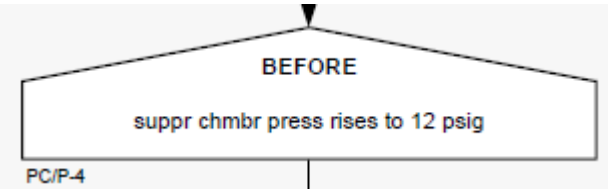


PCC-2



IF	THEN
PC press reduction is required to restore and maintain adequate core cooling or reduce total offsite radiation dose	**NO ACTION REQUIRED**
PC press CANNOT be maintained above (-)2 psig	**NO ACTION REQUIRED**
Suppr chmbr press approaches 0 psig	**NO ACTION REQUIRED**
DW press approaches 0 psig	**NO ACTION REQUIRED**

PC/P-3



INITIATE suppression chamber sprays
- Use only sources NOT required for continuous RPV injection (APPX 17)

IF	THEN
Needed to augment suppression chamber sprays	**NO ACTION REQUIRED**

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 16 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	NUSO	Directs 2-EOI APPENDIX-17C, RHR System Operation Suppression Chamber Sprays
	BOP	2-EOI APPENDIX-17C, RHR System Operation 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 THEN BYPASS LPCI injection valve auto interlock as necessary: PLACE 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS. PLACE 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL 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. [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 SYS I (II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. [5.3] MOMENTARILY PLACE 2-XS-74-121(129), RHR SYS I (II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT. [5.4] IF 2-FCV-74-53 (67), RHR SYS I (II) LPCI INBD INJ VALVE, is OPEN, THEN ENSURE CLOSED 2-FCV-74-52 (66), RHR SYS I (II) LPCI OUTBD INJECT VALVE. [5.5] ENSURE OPERATING the desired RHR System I(II) pump(s) for Suppression Chamber Spray [5.6] ENSURE OPEN 2-FCV-74-57(71), RHR SYS I (II) SUPPR CHBR/POOL ISOL VLV [5.7] OPEN 2-FCV-74-58 (72), RHR SYS I (II) SUPPR CHBR SPRAY VALVE

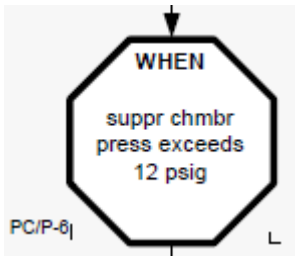
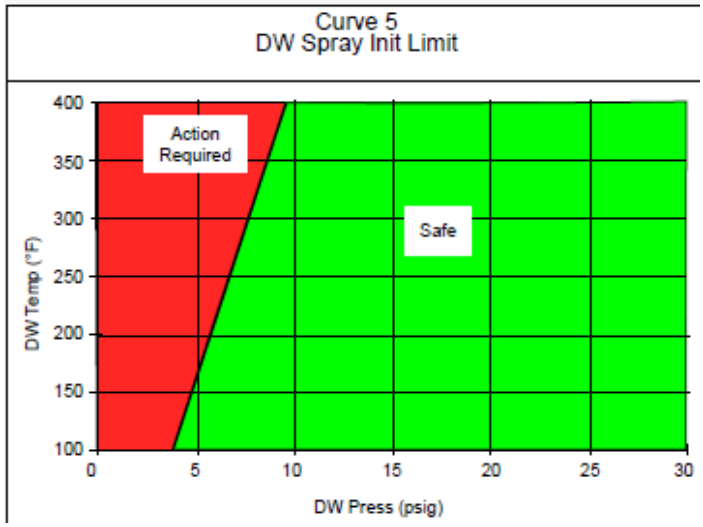
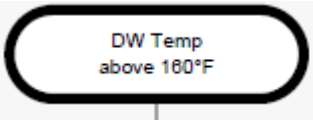
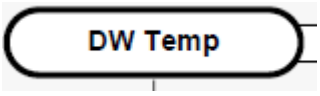
Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 17 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	BOP	[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 SYS I (II) MIN FLOW VALVE. [5.10] RAISE System flow with second RHR Sys I (II) pump if needed. [5.11] MONITOR RHR Pump NPSH using Attachment 2. [5.12] ENSURE RHRSW pump supplying desired RHR Heat Exchanger(s). [5.13] THROTTLE in-service RHRSW outlet valves to obtain 1700 to 4500 gpm: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV [5.14] NOTIFY Chemistry that RHRSW is in-service RHR Heat Exchangers. [6] WHEN EITHER of the following exists: Before Suppression Pool pressure drops below 0 psig OR Directed by SRO to stop Suppression Chamber Sprays, THEN STOP Suppr Chamber Sprays as follows: [6.1] CLOSE 2-FCV-74-58(72), RHR SYS I (II) SUPPR CHBR SPRAY VALVE.

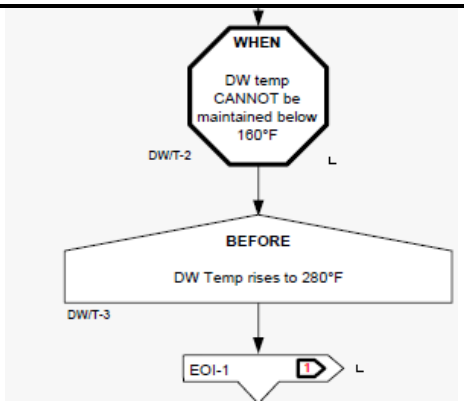
Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 18 of 20

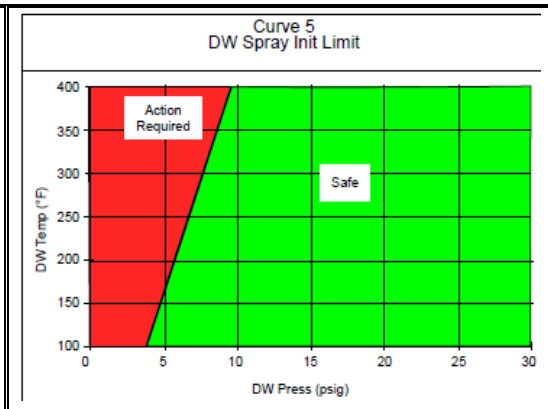
Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	NUSO	  IF suppression pool level is below 19 ft AND DW temp is in safe area of Curve 5 THEN 1. SHUT DOWN Recirc pumps 2. SHUT DOWN DW blowers 3. INITIATE DW sprays ➢ Use only sources NOT required for continuous RPV injection (APPX 17B) **IF** / **THEN** Needed to augment Drywell Sprays — **NO ACTION REQUIRED** PC/P-7
	NUSO	Directs Drywell Sprays from either the Primary Containment Pressure Leg after Suppression Chamber Pressure reaches 12 psig or the Drywell Temperature Leg before Drywell Temperature rises to 280°F.
	NUSO	 **IF** / **THEN** It is determined that core damage is occurring due to loss of cooling — **NO ACTION REQUIRED** PCC-1  PCC-2

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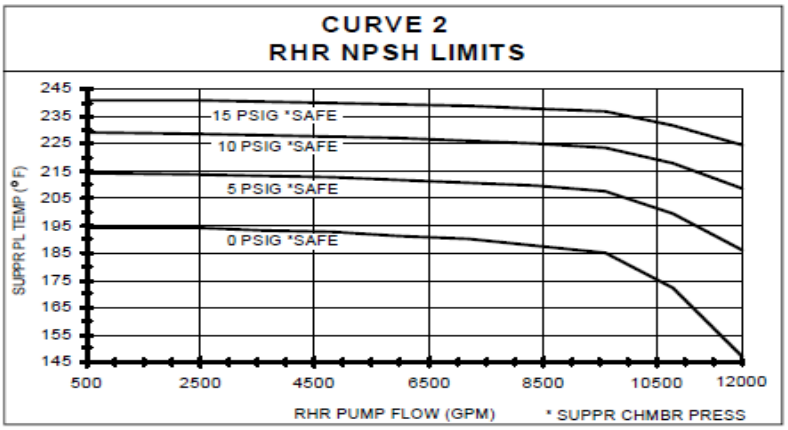
Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

<table>
<tr><th>Time</th><th>Position</th><th>Applicant's Actions or Behavior</th></tr>
<tr><td></td><td>NUSO</td><td> 


</td></tr>
<tr><td></td><td>NUSO</td><td>
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>If Drywell Pressure approaches 0 psig</td><td>NO ACTION REQUIRED</td></tr>
</table>
DW/T-4
<table>
<tr><td>IF suppression pool level is below 19 ft AND
DW temp is in safe area of Curve 5
THEN
1. SHUT DOWN Recirc pumps
2. SHUT DOWN DW blowers
3. INITIATE DW sprays
➢ Use only sources NOT required for continuous RPV injection (APPX 17B)</td></tr>
</table>
<table>
<tr><th>IF</th><th>THEN</th></tr>
<tr><td>Needed to augment Drywell Sprays</td><td>NO ACTION REQUIRED</td></tr>
</table>
DW/T-5</td></tr>
<tr><td></td><td>BOP</td><td>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 THEN BYPASS LPCI injection valve auto open signal as necessary:
• PLACE 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYP SEL IN BYP.
• PLACE 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYP SEL IN BYP.
[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.
• 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 OVRD, in MANUAL OVERRIDE
[6.3] MOMENTARILY PLACE 2-XS-74-121 (129), RHR SYS I (II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT.
[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 INJ VLV.
[6.5] ENSURE OPERATING the desired Sys I (II) RHR pump(s) for Drywell Spray.</td></tr>
</table>

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 8 Page 20 of 20

Event Description: Feedwater Line 'A' break, Reactor Bottom Head leak, loss of Reactor Water inventory.

Time	Position	Applicant's Actions or Behavior
	BOP	[6.6] OPEN the following valves: • 2-FCV-74-60(74), RHR SYS I(II) DW SPRAY OUTBD VLV. • 2-FCV-74-61(75), RHR SYS I(II) DW SPRAY INBD VLV. [6.7] ENSURE CLOSED 2-FCV-074-0007 (0030), RHR SYS I (II) MIN FLOW VALVE. [6.8] IF more Drywell Spray flow is needed, PLACE a second 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 in-service RHRSW outlet vlvs to 1700 to 4500 gpm RHRSW flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV [6.12] NOTIFY Chemistry that RHRSW is in-service RHR Heat Exchangers. [7] WHEN EITHER of the following exists: • BEFORE drywell pressure drops below 0 psig • OR directed to stop Drywell Sprays, THEN STOP Drywell Sprays as follows: [7.1] ENSURE CLOSED the following valves: • N/A • 2-FCV-74-60(74), RHR SYS I(II) DW SPRAY OUTBD VLV • 2-FCV-74-61(75), RHR SYS I(II) DW SPRAY INBD VLV [7.2] ENSURE OPEN 2-FCV-74-7(30), RHR SYSTEM I (II) MIN FLOW VALVE. **Attachment 2 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. 
		End of Event 8. Event 9, Reactor Core Isolation Cooling (RCIC) Flow Controller Automatic Mode Fails, is automatically triggered.

Op Test No.: 24-04 Scenario No. NRC-5 Event No.: 9 Page 1 of 1

Event Description: Reactor Core Isolation Cooling (RCIC) Flow Controller Automatic Mode Fails

Symptoms/Cues: When RCIC starts, it will initially appear to operate correctly however RCIC will not operate in automatic.

Time	Position	Applicant's Actions or Behavior
	Driver	**Event 9 is automatically entered when the Reactor MODE SWITCH is placed in SHUTDOWN. No action is required by the Driver to cause RCIC Flow Controller automatic mode to fail.**
	NRC	• **When the Reactor MODE SWITCH is placed in SHUTDOWN, the RCIC Automatic Flow controller, FIC-71-36-A will fail to work in automatic. The crew must place the RCIC controller in Manual to achieve injection to the Reactor.** • **Scripting for 2-EOI-Appendix-5C, Injection System Lineup RCIC actions is located on page 20 of 31 in this guide.**
	BOP	Places RCIC in manual for injection and reports that the RCIC flow controller automatic mode has failed.
	Driver	**If contacted as the Work Control/Outside NUSO or an AUO to investigate, acknowledge the direction.**
	NRC	**When the crew has Emergency Depressurized the Reactor and restored Reactor Water Level above (-) 180 inches or established Spray Cooling, end of Scenario.**

Scenario Setup
UNIT 2

IC	38
Exam IC	277

Procedure	Revision	Procedure	Revision	Procedure	Revision
OI-47	209	AOI-100-1	123	APPX-6B	12
OI-85	162	TRM 3.3.5	67	APPX-6C	13
OI-92A	29	T.S. 3.3.1.1	351	APPX-6D	8
OI-99	101	T.S. 3.3.7.1	253	APPX-6E	8
ARP-4A	52	OPDP-1	57	APPX-7L	6
ARP-4C	39	EOI-1	22		
ARP-5A	65	EOI-2	19		
ARP-5B	33	APPX-5B	5		
ARP-6B	22	APPX-5C	8		
AOI-85-3	28	APPX-6A	7		

Simulator Setup	Start CPERF **PRIOR** to placing the Simulator in RUN
Schedule Files(s):	ILT 2404 NRC Scenario 5 UNIT 2.sch
Event Files(s):	ILT 2404 NRC Scenario 5 UNIT 2.evt

UNIT 2 SHIFT TURNOVER MEETING

Today

MODE 2	DAYS ONLINE	Drywell Leakage (GPM)	Protected Equipment
	0	1.78	None
Rx Power 2.0%	PRA (Phoenix) -Green 500Kv GRID - Qualified 161Kv Grid -Qualified	Floor Drain (GPM) 0.20	
MWe 0		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

Declared U1 EHPM System Non-Functional and entered FPLCO A.2.2.1 Condition A

LCOs OF 72 HOURS OR LESS

None

SIGNIFICANT ITEMS DURING PREVIOUS SHIFT/RADIOLOGICAL CHANGES

The Emergency High Pressure Make-up Pump (EHPM) tagged out of service to change Pump Motor Bearing Oil.

MAJOR EQUIPMENT CHANGES PLANNED FOR THIS SHIFT

Alternate Bus Duct Cooling Fans, placing 2B Bus Duct Fan in service.

Continue Reactor start-up. Page 301 of 2-SR-3.1.3.5(A) begin at Control Rod 22-43

OPERATOR WORK AROUNDS **OWAs – U0-0/U2-0** **Burdens - U0 – 0/U2-0** **Operator Challenges – U0 – 0/U2 - 0**

None

ODMIs/ACMPs

None

ONEAs

None

FIRE RISK SIGNIFICANT ITEMS OOS/FPLCO Actions Due

EHPM OOS, FPLCO A.2.2.1, REQUIRED ACTION A.3 is due in 30 days

SCHEDULED ITEMS NOT COMPLETED

None

Job Performance Measure (JPM)

SITE: BFN		JPM TITLE:	Inject to the Reactor per 2-EOI Appendix-5D, Injection System Lineup HPCI
JPM NUMBER:	727A-U2	REVISION:	1

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-32 / Lineup Injection Systems – HPCI in accordance with EOI Appendix-5D		
K/A RATINGS:	RO: 4.3		
K/A No. &STATEMENT:	206000 SF2 High Pressure Coolant Injection System A4.02: Ability to manually operate and/or monitor in the Control Room: Flow Controller		
RELATED PRA INFORMATION:	SSCs in the Top 90% for CDF		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 727A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Places 2-HS-73-47A, HPCI AUXILIARY OIL PUMP, in START
- Opens 2-FCV-73-44, HPCI PUMP INJECTION VALVE
- Opens 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE

Alternate Path:

- Trips HPCI and verifies 2-FCV-73-18, HPCI TURBINE STOP VALVE, closes

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI Appendix-5D

VALIDATION TIME: 9 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	08/10/19	All	Initial issue
1	12/12/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-EOI Appendix-5D	10
OPDP- 1	57

SIMULATOR SETUP

IC	N/A
Exam IC	259

Console Operator Instructions	• **Reset to IC 259** • **Hang danger tags on 2-HS-71-2 and 2-HS-71-3** • **Open and run schedule file F:\2404 jpms\JPMA 727A\JPM A.sch Ensure Event File F:\2404 jpms\JPMA 727A\JPM A.evt is opened by the schedule file.** • **Place the simulator in RUN to ensure stable conditions** • **Verify the HPCI Controller is in automatic and set to 5300 GPM**

Malfunctions	Description	Event	Severity	Delay	Initial set
HP02	HPCI AUTOMATIC TURBINE TRIP SYSTEM FAILURE (RLY-23A-K14)	N/A	N/A	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
HS-73-40A	HPCI CST SUCTION VALVE	1	CLOSE	10	N/A
HS-73-27A	HPCI SUPPRESSION POOL OUTBD SUCT	0	CLOSE	N/A	Y
HS-73-26A	HPCI SUPPRESSION POOL INBD SUCT	0	CLOSE	N/A	Y

Schedule Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Event Files	Description	Event	Severity	Delay	Initial set
ZAOSI7351 > 3500	T-HPCI, speed 3500 rpm	1	N/A	N/A	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. A Manual Reactor SCRAM was inserted, with the following plant conditions:

- The Reactor Core Isolation Cooling (RCIC) System is inoperable
- Main Steam Isolation Valves (MSIVs) are closed
- Another Operator has been tasked with Reactor Pressure Control

INITIATING CUES:

The Unit Supervisor directs you to restore Reactor Water Level to +2 to +51 inches, using the High Pressure Coolant Injection (HPCI) System as directed by 2-EOI Appendix-5D, Injection System Lineup – HPCI.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **IF** Suppression Pool Level drops below 12.75 ft. during HPCI operation, **THEN TRIP** HPCI and **CONTROL** injection using other options. Expected Action(s): Checks Suppression Pool Water Level and notes that it is currently >12.75 feet.	_____ SAT _____UNSAT _____N/A
Step 2: [2] **IF** Suppression Pool Level CANNOT be maintained below 5.25 in, **THEN EXECUTE** 2-EOI-Appendix-16E concurrently with this procedure to bypass HPCI High Suppression Pool Water Level Suction Transfer Interlock. Expected Action(s): Checks Suppression Pool Water Level and notes that it currently less than 5.25 inches.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **IF** BOTH of the following exist: • High temperature exists in the HPCI area, **AND** • SRO directs bypass of HPCI High Temperature Isolation Interlocks, **THEN PERFORM** the following: [3.1] **EXECUTE** 2-EOI-Appendix-16L concurrently with this procedure. [3.2] **RESET** auto isolation logic using HPCI AUTO-ISOL LOGIC A (B) RESET pushbuttons. Expected Action(s): Determines that 2-EOI-Appendix-16L is not required and the Auto Isolation Logic is reset.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **VERIFY** at least one Standby Gas Train (SGT) in operation. Expected Action(s): Verifies that at least one SGTS is in operation.	_____ SAT _____UNSAT _____N/A
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. Step 5: [5] **VERIFY** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, 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 Expected Action(s): Verifies the HPCI Flow Controller is set as desired.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
NOTE HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open. Step 6: [6] **IF** high Reactor Water Level trip logic is actuated, **THEN** [6.1] **DEPRESS** HPCI TURBINE TRIP RX LVL HIGH RESET pushbutton. [6.2] **CHECK** HPCI TURBINE TRIP LVL HIGH amber light has extinguished. Expected Action(s): Verifies that the high Reactor Water Level trip is not actuated.	_____ SAT _____UNSAT _____N/A
Step 7: [7] **PLACE** 2-HS-73-47A, HPCI AUXILIARY OIL PUMP, in START. Expected Action(s): Places 2-HS-73-47A, HPCI AUXILIARY OIL PUMP, in START.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 8: [8] **PLACE** 2-HS-73-10A, HPCI STEAM PACKING EXHAUSTER handswitch in START. Expected Action(s): Places 2-HS-73-10A, HPCI STEAM PACKING EXHAUSTER handswitch in START.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 9: [9] **OPEN** 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE Expected Action(s): Opens 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE.	_____ SAT _____UNSAT _____N/A
Step 10: [10] **OPEN** 2-FCV-73-44, HPCI PUMP INJECTION VALVE. Expected Action(s): Opens 2-FCV-73-44, HPCI PUMP INJECTION VALVE.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 11: [11] **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE, to start HPCI Turbine. Expected Action(s): Opens 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VALVE.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 12: [12] **CHECK** proper HPCI operation by observing the following: A. HPCI Turbine 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. Expected Action(s): Verifies that the HPCI Turbine accelerates and injects to the Reactor.	_____ SAT _____UNSAT _____N/A

EXAMINER NOTE: Begin Alternate Path – 10 seconds after the HPCI Turbine accelerates to 3500 RPM, 2-HS-73-40, HPCI CST SUCTION VALVE, will close. Examinee will likely complete all of the following page before Alternate Path actions on page 11 of this guide.

STEP / STANDARD	SAT / UNSAT
CAUTION HPCI PUMP MIN FLOW VALVE, 2-FCV-073-0030, automatically opens when system flow is at or below 900 GPM (lowering) only if a system initiation signal is present. Manually opening the min flow valve may be required for pump min flow protection. Step 13: [13] **ADJUST** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, as necessary to control injection. Expected Action(s): Adjusts the Flow Controller to control injection.	_____ SAT _____UNSAT _____N/A
Step 14: [14] **VERIFY** HPCI Auxiliary Oil Pump stops and the Shaft Driven Oil Pump operates properly. Expected Action(s): Verifies that the HPCI Auxiliary Oil Pumps stops and that the Shaft Driven Oil Pump is operating properly.	_____ SAT _____UNSAT _____N/A
Step 15: [15] **WHEN** HPCI Auxiliary Oil Pump stops, **THEN PLACE** HPCI AUXILIARY OIL PUMP handswitch in AUTO. Expected Action(s): Places the HPCI Auxiliary Oil Pump handswitch in AUTO.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 16: HPCI PUMP SUCTION PRESS LOW, 2-9-3F, Window 13 Operator Action: A. **ENSURE** automatic action has occurred. Automatic Action: A. Turbine Trip 1. 2-FCV-73-18, HPCI TURBINE STOP VALVE, closes 2. 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes Expected Action(s): **EXAMINER NOTE: Verifying that 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes is NOT a critical step.** Determines that HPCI Turbine Trip Automatic Action did not occur. Trips HPCI by holding 2-HS-73-18A, HPCI TURBINE TRIP: **AND** When HPCI indicate zero RPM places 2-HS-73-47A, HPCI AUX OIL PUMP in PULL TO LOCK **OR** Closes 2-HS-73-16A HPCI TURBINE STEAM SUPPLY VALVE, OR 2-HS-73-2A HPCI STEAM LINE INBD VALVE, OR 2-HS-73-3A HPCI STEAM LINE OUTBD VALVE	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: When informed that HPCI did not trip as required on a low suction condition, acknowledge the report and 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 an Operator on Unit 2. A Manual Reactor SCRAM was inserted, with the following plant conditions:

- The Reactor Core Isolation Cooling (RCIC) System is inoperable
- Main Steam Isolation Valves (MSIVs) are closed
- Another Operator has been tasked with Reactor Pressure Control

INITIATING CUES:

The Unit Supervisor directs you to restore Reactor Water Level to +2 to +51 inches, using the High Pressure Coolant Injection (HPCI) System as directed by 2-EOI Appendix-5D, Injection System Lineup – HPCI.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Reduce Reactor Pressure per EOI Appendix-11F, Alternate RPV Pressure Control Systems-RFPT on Minimum Flow	
JPM NUMBER:		48A-U2	REVISION:	8

TASK APPLICABILITY:	☐SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-58 / Lineup Alternate RPV Pressure Control Systems-RFPT on Minimum Flow in Accordance with EOI Appendix 11F		
K/A RATINGS:	RO: 4.4 SRO: 4.1		
K/A No. &STATEMENT:	241000 SF3 RTPRS Reactor/Turbine Pressure Regulating System A2.02; Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High Reactor Pressure		
RELATED PRA INFORMATION:	CDF Key Operator Action= 27.4% Risk Reduction		
SAFETY FUNCTION:	3		

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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 48A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Closes 2-FCV-3-19(12)(5), RFPT 2A(2B)(2C) Discharge Valve
- Depresses 2-HS-3-124A(150A)(175A), RFPT 2A(2B)(2C) RFPT TRIP RESET
- Places 2-HS-46-112A(138A)(163A), RFPT 2A(2B)(2C) START/LOCAL ENABLE, in START
- Raises RFPT Speed using 2-HS-46-8A(9A)(10A) RFPT 2A(2B)(2C) SPEED CONT RAISE/LOWER. Maintains 2A(2B)(2C) RFPT Discharge Pressure <1250 PSIG as indicated on 2-PI-3-16A(9A)(2A), RFPT DISCHARGE PRESSURE

Alternate Path:

- Determines that 2-FCV-3-19(12)(5), RFPT 2A(2B)(2C) DISCHARGE VALVE, will not close

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI Appendix -11F

VALIDATION TIME: 10 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
7	08/09/17	All	JPM converted to new format
8	12/12/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-EOI Appendix-11F	5

SIMULATOR SETUP

IC	N/A
Exam IC	260

Console Operator Instructions	• **Reset to IC 260** • **Open and run schedule file F:\2404 jpms\JPM B 48A\JPM B – 48A.sch. Ensure Event File F:\2404 jpms\JPM B 48A\JPM B – 48A.evt is opened by the schedule file.** • **Place the simulator in RUN to ensure stable conditions**

Schedule File F:\2404 jpms\JPM B 48A\JPM B – 48A.sch

Actions	Description	Event	Severity	Delay	Initial set
Schedule F:\2404\jpms\JPM B 48A\RFPT A.sch	Run RFPT A Schedule File	1	N/A	N/A	N/A
Schedule F:\2404\jpms\JPM B 48A\RFPT B.sch	Run RFPT B Schedule File	2	N/A	N/A	N/A
Schedule F:\2404\jpms\JPM B 48A\RFPT C.sch	Run RFPT C Schedule File	3	N/A	N/A	N/A
Event **F:\2404 jpms\JPM B 48A\JPM B – 48A.evt**	Run Event File	N/A	N/A	N/A	N/A
Malfunctions	**Description**	**Event**	**Severity**	**Delay**	**Initial set**
FCV-03-19	MOTOR_OPERATED_ VALVE RFP A DISCH ISOL VLV	N/A	FAIL_POWER_ NOW	N/A	N/A
FCV-03-12	MOTOR_OPERATED_ VALVE RFP B DISCH ISOL VLV	N/A	FAIL_POWER_ NOW	N/A	N/A
FCV-03-5	MOTOR_OPERATED_ VALVE RFP C DISCH ISOL VLV	N/A	FAIL_POWER_ NOW	N/A	N/A
Overrides	**Description**	**Event**	**Severity**	**Delay**	**Initial set**
ZLOHS0319A_1	HS-3-19A GREEN RFP 2A DISCHARGE VALVE	N/A	OFF	N/A	OFF
ZLOHS0319A_2	HS-3-19A RED* RFP 2A DISCHARGE VALVE	N/A	ON	N/A	ON
ZLOHS0312A_1	HS-3-12A GREEN RFP	N/A	OFF	N/A	OFF

	2B DISCHARGE VALVE				
ZLOHS0312A_2	HS-3-12A RED* RFP 2B DISCHARGE VALVE	N/A	ON	N/A	ON
ZLOHS035A_1	HS-3-5A-GREEN RFP C DISCHARGE VALVE	N/A	OFF	N/A	OFF
ZLOHS035A_2	HS-3-5A-RED* RFP C DISCHARGE VALVE	N/A	ON	N/A	ON

Schedule file RFPT A.sch

Overrides	Description	Event	Severity	Delay	Initial set
HS-3-19A	RFP 2A DISCHARGE VALVE		OPEN		
Actions	**Description**	**Event**	**Severity**	**Delay**	**Initial set**
Delete override ZLOHS0312A_1	HS-3-12A GREEN RFP 2B DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS0312A_2	HS-3-12A RED* RFP 2B DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS035A_1	HS-3-5A-GREEN RFP C DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS035A_2	HS-3-5A-RED* RFP C DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete malfunction FCV-03-12	MOTOR_OPERATED_ VALVE RFP B DISCH ISOL VLV	N/A	N/A	N/A	N/A
Delete malfunction FCV-03-5	MOTOR_OPERATED_ VALVE RFP C DISCH ISOL VLV	N/A	N/A	N/A	N/A
Create Event 2 - desc Clear B After RFPT A	Delete Events for remaining RFPTs	N/A	N/A	N/A	N/A
Create Event 3 - desc Clear C After RFPT A	Delete Events for remaining RFPTs	N/A	N/A	N/A	N/A

Schedule file RFPT B.sch

Overrides	Description	Event	Severity	Delay	Initial set
HS-3-12A	RFP 2B DISCHARGE VALVE		OPEN		
Actions	**Description**	**Event**	**Severity**	**Delay**	**Initial set**
Delete override ZLOHS0319A_1	HS-3-19A GREEN RFP 2A DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS0319A_2	HS-3-19A RED* RFP 2A DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS035A_1	HS-3-5A-GREEN RFP C DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS035A_2	HS-3-5A-RED* RFP C DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete malfunction FCV-03-19	MOTOR_OPERATED_ VALVE RFP A DISCH ISOL VLV	N/A	N/A	N/A	N/A
Delete malfunction FCV-03-5	MOTOR_OPERATED_ VALVE RFP C DISCH ISOL VLV	N/A	N/A	N/A	N/A
Create Event 1 - desc Clear A After RFPT B	Delete Events for remaining RFPTs	N/A	N/A	N/A	N/A
Create Event 3 - desc Clear C After RFPT B	Delete Events for remaining RFPTs	N/A	N/A	N/A	N/A

Schedule file RFPT C.sch

Overrides	Description	Event	Severity	Delay	Initial set
HS-3-19A	RFP 2A DISCHARGE VALVE	4	OPEN		
Actions	**Description**	**Event**	**Severity**	**Delay**	**Initial set**
Delete override ZLOHS0319A_1	HS-3-19A GREEN RFP 2A DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS0319A_2	HS-3-19A RED* RFP 2A DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS0312A_1	HS-3-12A GREEN RFP 2B DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete override ZLOHS0312A_2	HS-3-12A RED* RFP 2B DISCHARGE VALVE	N/A	N/A	N/A	N/A
Delete malfunction FCV-03-19	MOTOR_OPERATED_ VALVE RFP A DISCH ISOL VLV	N/A	N/A	N/A	N/A
Delete malfunction FCV-03-12	MOTOR_OPERATED_ VALVE RFP B DISCH ISOL VLV	N/A	N/A	N/A	N/A
Create Event 1 - desc Clear A After RFPT A	Delete Events for remaining RFPTs	N/A	N/A	N/A	N/A
Create Event 2 - desc Clear B After RFPT A	Delete Events for remaining RFPTs	N/A	N/A	N/A	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. The Reactor has SCRAMMED and the Turbine Bypass Valves are not responding properly for Reactor Pressure control due to a Control Unit Failure.
2-EOI-1, RPV Control, has been entered and followed to RC/P-4.

INITIATING CUES:

The Unit Supervisor directs you to place a Reactor Feedwater Pump (RFPT) in Alternate Pressure Control as directed by 2-EOI Appendix-11F, Alternate RPV Pressure Control Systems RFPT on Minimum Flow and establish a Reactor Cooldown, not to exceed 100ºF/hr.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **IF** BOTH of the following exist: • Emergency RPV Depressurization is required, **AND** • Group 1 Isolation Signal exists, **THEN EXIT** this procedure and **ENTER** EOI Appendix 11H. Expected Action(s): Verifies that a Group 1 Isolation signal does not exist by observing that the Containment Isolation Status System Amber Initiation Lights (PCIS Logic A and PCIS Logic B) on Panel 2-9-4 for the Main Steam Isolation Valves (MSIVs) are extinguished.	_____ SAT _____UNSAT _____N/A
Step 2: [2] **VERIFY** MSIVs open. Expected Action(s): Verifies that MSIVs are open.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **VERIFY** Hotwell Pressure at or below (-) 7 in. Hg. Expected Action(s): Verifies Main Condenser Vacuum at or below (-) 7 inches of Mercury using 2-XR-2-2, HOTWELL TEMPERATURE AND PRESSURE, on Panel 9-6, or on ICS by using computer point D383, D384, or D385.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **PLACE** RFPTs in service as follows: [4.1] **VERIFY** the following: [4.1.1] At least one Condensate Pump running. [4.1.2] At least one Condensate Booster Pump running. [4.1.3] Condensate System aligned to supply suction to RFPs. Expected Action(s): Verifies that at least one Condensate Pump and one Condensate Booster Pump are running; verifies that the Condensate System is aligned to supply suction to the 2A(2B)(2C) RFPT by checking open the Low Pressure Heater Inlet and Outlet Valves and RFPT 2A(2B)(2C) Suction Valve.	_____ SAT _____UNSAT _____N/A
Step 5: [4.2] **VERIFY** Main Oil Pump running for each RFPT to be started. Expected Action(s): Verifies 2A(2B)(2C) RFPT MAIN OIL PUMP(S) is/are running by checking the RED light(s) above the associated Main Oil Pump hand switches.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Begin Alternate Path. The first RFPT chosen will have a failed open Discharge Valve.	
Step 6: [4.3] **VERIFY CLOSED** 2-FCV-3-19(12)(5), RFPT 2A(2B)(2C) DISCHARGE VALVE. Expected Action(s): Determines that 2-FCV-3-19(12)(5), RFPT 2A(2B)(2C) DISCHARGE VALVE, will not close.	**Critical Step** _____ SAT _____UNSAT _____N/A
NOTE: The candidate may choose any RFPT to place in Reactor Pressure Control. The first RFPT chosen will have a failed open Discharge Valve.	
Driver – verify that the malfunctions on the remaining RFPT Discharge valves clear.	

STEP / STANDARD	SAT / UNSAT
Step 7: Attempts to use another RFPT: [4] **PLACE** RFPTs in service as follows: [4.1] **VERIFY** the following: [4.1.1] At least one Condensate Pump running. [4.1.2] At least one Condensate Booster Pump running. [4.1.3] Condensate System aligned to supply suction to RFPs. Expected Action(s): Verifies that at least one Condensate Pump and one Condensate Booster Pump are running; verifies that the Condensate System is aligned to supply suction to 2A(2B)(2C) RFPT by checking open the Low Pressure Heater Inlet and Outlet Valves and RFPT 2A(2B)(2C) Suction Valve.	_____ SAT _____UNSAT _____N/A
Step 8: [4.2] **VERIFY** Main Oil Pump running for each RFPT to be started. Expected Action(s): Verifies 2A(2B)(2C) RFPT MAIN OIL PUMP(S) is/are running by checking the RED light(s) above the associated Main Oil Pump hand switches.	_____SAT _____UNSAT _____N/A
Step 9: [4.3] **VERIFY CLOSED** 2-FCV-3-19(12)(5), RFPT 2A(2B)(2C) Discharge Valve. Expected Action(s): Closes 2-FCV-3-19(12)(5), RFPT 2A(2B)(2C) Discharge Valve.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 10: [4.4] **DEPRESS** 2-HS-46-8A(9A)(10A) RFPT 2A(2B)(2C) SPEED CONT RAISE/LOWER, and **VERIFY** amber light is illuminated. Expected Action(s): Depresses 2-HS-46-8A(9A)(10A) RFPT 2A(2B)(2C) SPEED CONTROL RAISE/LOWER and verifies the amber light is illuminated.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 11: [4.5] **DEPRESS** 2-HS-3-124A(150A)(175A), RFPT 2A(2B)(2C) TRIP RESET. Expected Action(s): Depresses 2-HS-3-124A(150A)(175A), RFPT 2A(2B)(2C) RFPT TRIP RESET.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 12: [4.6] **PLACE** 2-HS-46-112A(138A)(163A), RFPT 2A(2B)(2C) START/LOCAL ENABLE, in START. Expected Action(s): Places 2-HS-46-112A(138A)(163A), RFPT 2A(2B)(2C) START/LOCAL ENABLE, in START.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 13: [4.7] **CHECK** RFPT 2A(2B)(2C) speed increases to approximately 600 RPM. Expected Action(s): Verifies that 2A(2B)(2C) RFPT accelerates to approximately 600 RPM.	_____SAT _____UNSAT _____N/A
Step 14: [4.8] **VERIFY OPEN** 2-FCV-3-13, RFPT 2A(2B)(2C) MIN FLOW VALVE. Expected Action(s): Verifies open 2-FCV-3-20(13)(6), RFPT 2A(2B)(2C) MIN FLOW VALVE.	_____SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 15: **CAUTION** RFP discharge pressure is limited to below 1250 PSIG to avoid system damage. [4.9] **PLACE** 2-HS-46-8A(9A)(10A) RFPT 2A(2B)(2C) SPEED CONT RAISE/LOWER in RAISE to raise RFPT speed, maintaining discharge pressure less than 1250 psig. Expected Action(s): Raises RFPT Speed using 2-HS-46-8A(9A)(10A) RFPT 2A(2B)(2C) SPEED CONT RAISE/LOWER. Maintains 2A(2B)(2C) RFPT Discharge Pressure <1250 PSIG as indicated on 2-PI-3-16A(9A)(2A), RFPT DISCHARGE PRESSURE.	**Critical Step** _____ SAT _____UNSAT _____N/A
Note: If Reactor Pressure does not respond quickly enough in the candidate's judgement, another RFPT may be placed in Reactor Pressure Control if desired. The Examiner may stop the JPM at any point when satisfied with the candidate's actions.	
CUE: Another Operator will be tasked with Reactor Pressure Control.	

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. The Reactor has SCRAMMED and the Turbine Bypass Valves are not responding properly for Reactor Pressure control due to a Control Unit Failure.
2-EOI-1, RPV Control, has been entered and followed to RC/P-4.

INITIATING CUES:

The Unit Supervisor directs you to place a Reactor Feedwater Pump (RFPT) in Alternate Pressure Control as directed by 2-EOI Appendix-11F, Alternate RPV Pressure Control Systems RFPT on Minimum Flow and establish a Reactor Cooldown, not to exceed 100ºF/hr

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Raise Main Turbine Load
JPM NUMBER:	756A-U2	REVISION:	0

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-047-NO-04 Roll the Main Turbine, and Synchronize and Load the Main Generator		
K/A RATINGS:	RO: 3.1		
K/A No. &STATEMENT:	245000 SF4 MTGEN Main Turbine Generator and Auxiliary Systems K1.16; Knowledge of the physical connections and/or cause and effect relationships between the Main Turbine Generator and Auxiliary Systems and the following systems: Lube oil system		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 756A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- DEPRESSES Load Set RAISE, 2-HS-47-75C, pushbutton to raise load on the Main Turbine

Alternate Path:

- MANUALLY TRIPS the Main Turbine

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-OI-47, 2-ARP-9-7A

VALIDATION TIME: 9 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	12/14/23	All	Initial issue

Procedure Revisions

Procedure	Revision
2-OI-47	208
2-ARP-9-7A	37

SIMULATOR SETUP

IC	N/A
Exam IC	261

Console Operator Instructions	• **Reset to IC 261** • **Open and run schedule file F:\2404 jpms\JPMC 756A\JPM C.sch. Ensure Event File F:\2404 jpms\JPMC 756A\JPM C.evt is opened by the schedule file.** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
TU05	MTOT LEVEL LOW/LEAK	1	80	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Schedule Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Event Files	Description	Event	Severity	Delay	Initial set
ZAOZI0162 < 50	WHEN BPV 2 IS LESS THAN 50 PERCENT OPEN	1	N/A	N/A	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. Generator Synchronization and Loading is in progress in accordance with 2-OI-47, Turbine-Generator System, section 5.5:

- Another Operator has completed section 5.5 steps [1] through step [19]
- Generator output is between 60 Mwe and 80 MWe

INITIATING CUES:

The Unit Supervisor directs you to load the Main Turbine continuing at step [20] of 2-OI-47, Turbine-Generator System, Section 5.5 Generator Synchronization and Loading.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [20] **INTERMITTENTLY DEPRESS** Load Set RAISE, 2-HS-47-75C, pushbutton **UNTIL** ALL turbine bypass valves are closed. Expected Action(s): INTERMITTENTLY DEPRESSES Load Set RAISE, 2-HS-47-75C, pushbutton.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Begin Alternate Path – after Turbine Bypass Valve 2 closes more than 50% a leak develops draining the Main Turbine Oil Tank (MTOT).	
Step 2: A. CHECK TURB BRG OIL TANK LEVEL indicator 2-LI-47-13 on Panel 2-9-7. B. DISPATCH personnel to check main turbine oil tank level 2-LI-47-126 and float switch. C. IF centrifuge is in service, THEN ENSURE system lineup. REFER TO 0-OI-20. D. CHECK main turbine oil system for leaks. E. REFILL MTOT. REFER TO 0-OI-20. Expected Action(s): Checks MTOT level indicator 2-LI-47-13 on Panel 2-9-7. Dispatches personnel to check main turbine oil tank level 2-LI-47-126 and float switch.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: 2-9-7A, MAIN TURB OIL TANK LEVEL LOW, Window 13 will alarm first. The Examinee may not dispatch personnel based on the rate of MTOT leak and then prioritize 2-9-7A, MAIN TURB OIL TANK LEVEL LOW-LOW, Window 18, once in alarm.	

STEP / STANDARD	SAT / UNSAT
Step 3: D. IF the alarm is valid and a complete loss of Main Turbine Lube Oil is expected, THEN PERFORM the following (Otherwise N/A): D. 1. **INITIATE** a Manual SCRAM, 2-AOI-100-1. Expected Action(s): Determines that alarm 2-9-7A, MAIN TURB OIL TANK LEVEL LOW-LOW is valid. Initiates a Manual SCRAM	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: A second Operator will be monitoring panel 9-5. When the Examinee determines that a reactor SCRAM is required, the second Operator may insert the SCRAM and Examinee may continue with BOP actions.	
Step 4: D. 2. **MANUALLY TRIP** the Main Turbine. Expected Action(s): MANUALLY TRIPS the Main Turbine.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: When the Reactor has been SCRAMMED and the Main Turbine has been tripped, or at Examiners discretion, inform the Examinee "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 an Operator on Unit 2. Generator Synchronization and Loading is in progress in accordance with 2-OI-47, Turbine-Generator System, section 5.5:

- Another Operator has completed step [1] through step [19]
- Generator output is between 60 Mwe and 80 MWe

INITIATING CUES:

The Unit Supervisor directs you to load the Main Turbine continuing at step [20] of 2-OI-47, Turbine-Generator System, Section 5.5 Generator Synchronization and Loading.

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Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Traversing Incore Probe (TIP) Isolation Failure (U2 ONLY)	
JPM NUMBER:	281-U2	REVISION:	1	

TASK APPLICABILITY:	☐SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-064-AB-06/ TIP Isolation Failure		
K/A RATINGS:	RO: 3.1		
K/A No. &STATEMENT:	223002 SF5 PCIS Primary Containment Isolation System/Nuclear Steam Supply Shutoff K1.13; Knowledge of the physical connections and/or cause and effect relationships between the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and the following systems: Traversing Incore Probe System		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 281-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Places MANUAL switch in REV. Determines that TIP B does not retract, and associated ball valve cannot be closed.:
- Obtains key PA 235
- Inserts key into keylock 2-XS-94-510 SHEAR VALVE INDEX. Turns keylock 2-XS-94-510 SHEAR VALVE INDEX B to FIRE position then to MONITOR position.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-AOI-64-2E

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	2/10/2013	All	Initial issue
1	12/14/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-AOI-64-2E	17

SIMULATOR SETUP

IC	N/A
Exam IC	263

Console Operator Instructions	• **Reset to IC 263** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Schedule Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Event Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	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. High Drywell pressure caused a Reactor Scram, with the following plant conditions:

- Traversing Incore Probe (TIP) operations were in progress on Unit 2
- The Unit Supervisor has entered EOI-1 and 2.

INITIATING CUES:

The Unit Supervisor directs you verify a Group 8 Isolation in accordance with 2-AOI-64-2E, Traversing Incore Probe Isolation.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] IF any EOI entry condition is met, **THEN ENTER** appropriate EOI(s). (Otherwise N/A) Expected action(s): Determines that no other EOI entry conditions exist and N/As this step.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: If Examinee begins to evaluate the plant for other EOI entry conditions then inform the candidate "No other EOI entry conditions exist.".	
Step 2: [2] **MONITOR** TIP detector to ensure it stops at its shield chamber limit and the ball valve closes. [2.1] **IF** detector does **NOT** stop, **THEN PLACE** Mode Switch in OFF. (Otherwise N/A) Expected action(s): Determines that TIP B remains in the core and N/As this step.	_____ SAT _____UNSAT _____N/A
Step 3: [3] **ENSURE** automatic actions. (N/A for guide tube leak) Expected action(s): **EXAMINER NOTE: Examinee may elect to verify TIP ball valves associated with A, C, D and E are closed.** Determines that a guide tube leak occurred and N/As this step.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4] **IF** a guide tube leak is detected or detector is stuck, **THEN PERFORM** the following for any TIP channel with an open ball valve. (Otherwise N/A) [4.1] **ENSURE** MODE switch in MAN. [4.2] **PLACE** MANUAL switch in REV. [4.3] WHEN detector reaches IN-SHIELD position, **THEN PLACE** MANUAL switch to OFF. (N/A if detector does not reach IN-SHIELD) [4.4] **PLACE** MAN VALVE CONTROL Switch to CLOSED. (N/A if detector does not reach IN-SHIELD) Expected action(s): **EXAMINER NOTE: ONLY** [4.2] PLACE MANUAL switch in REV is considered the **Critical Step.** Places MANUAL switch in REV. Determines that TIP B does not retract, and associated ball valve cannot be closed.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [4.5] **IF** TIP detector does not retract and associated ball valve close, **THEN PERFORM** the following to activate the TIP shear valves: (Otherwise N/A) [4.5.1] **OBTAIN** Shift Manager permission and key PA 235. Expected action(s): Obtains key PA 235	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: When examine proceeds to obtain key PA 235 then provide key PA 235 to the candidate.	
Step 6: [4.5.2] **INSERT** key into keylock of TIP panel for TIP that has not fully withdrawn. 2-XS-94-510 SHEAR VALVE INDEX B [4.5.3] **TURN** key to FIRE position then to MONITOR position. 2-XS-94-510 SHEAR VALVE INDEX B Expected action(s): Inserts key into keylock 2-XS-94-510 SHEAR VALVE INDEX. Turns keylock 2-XS-94-510 SHEAR VALVE INDEX B to FIRE position then to MONITOR position.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [4.5.4] **WITHDRAW** key and Return to Shift Manager Expected action(s): Withdraw key and Return to Shift Manager.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: When Examinee withdraws key, inform the candidate "Your task is complete.". **EXAMINER NOTE: Examiner may elect to end the JPM when satisfied that Critical Step 6 is met.**	

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. High Drywell pressure caused a Reactor Scram, with the following plant conditions:

- Traversing Incore Probe (TIP) operations were in progress on Unit 2
- The Unit Supervisor has entered EOI-1 and 2.

INITIATING CUES:

The Unit Supervisor directs you verify a Group 8 Isolation in accordance with 2-AOI-64-2E, Traversing Incore Probe Isolation.
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Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Transfer 1B Start Bus to Alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System (U2 ONLY)	
JPM NUMBER:		757-U2	REVISION:	0

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-57A-NO-08/ Perform a Control Room Transfer of 4kV Start Bus 1A (2A) and 1B (2B) Power Supplies		
K/A RATINGS:	RO: 3.7		
K/A No. &STATEMENT:	262001 SF6 AC Electrical Distribution A4.01: Ability to manually operate and/or monitor in the control room: Breakers and disconnects		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 757-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Places 0-43-204-1B 4kV START BUS 1B MAN/AUTO SELECT to MAN.
- Places and holds 0-HS-204-1/2A, 4kV START BUS 1B ALT FDR BKR 1414 to CLOSE **AND** places 0-HS-204-1/6A, 4kV START BUS 1B NORM FDR BKR 1516 to TRIP

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-OI-57A

VALIDATION TIME: 10 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	12/12/23	All	Initial issue

Procedure Revisions

Procedure	Revision
0-OI-57A	177

SIMULATOR SETUP

IC	N/A
Exam IC	264

Console Operator Instructions	• **Reset to IC 264** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Schedule Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Event Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	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. Preparations are underway to unload CSST B for scheduled maintenance.

- All Precautions and Limitations have been reviewed and are being addressed by another Operator.
- AUO reports from the field that the Start Bus area is clear of all personnel and all pre-start checks are complete.

INITIATING CUES:

The Unit Supervisor directs you transfer 4kV Start Bus 1B to alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System, Section 8.21.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] **REVIEW** all Precautions and Limitations listed in P&L Section 3.0. Expected action(s): Marks step as complete, Precautions and Limitations listed in P&L Section 3.0 are addressed by another Operator as stated in the Initial Conditions	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: If Examinee begins to review Precautions and Limitations then inform the candidate "Another Operator is addressing Precautions and Limitations, continue with your task".	
Step 2: [2] PLACE 0-43-204-1B(2B) 4kV START BUS 1B(2B) MAN/AUTO SELECT to MAN. Expected action(s): Places 0-43-204-1B 4kV START BUS 1B MAN/AUTO SELECT to MAN.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 3: [3] To transfer 4kV START BUS 1B(2B) to the ALT FDR BKR 1414(1418), PERFORM the following: [3.1] CHECK CSST A VOLTAGE is between 3950 and 4400 Volts on 0-EI-245-A, CSST A START BD 1 VOLTS. Expected action(s): Determines that CSST A VOLTAGE is between 3950 and 4400 Volts on 0-EI-245-A, CSST A START BD 1 VOLTS.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [3.2] PLACE and HOLD 0-HS-204-1/2A(0-HS-204-2/1A), 4kV START BUS 1B(2B) ALT FDR BKR 1414(1418) to CLOSE. [3.3] PLACE 0-HS-204-1/6A(0-HS-204-2/5A), 4kV START BUS 1B(2B) NORM FDR BKR 1516(1512) to TRIP Expected action(s): Places and holds 0-HS-204-1/2A, 4kV START BUS 1B ALT FDR BKR 1414 to CLOSE AND places 0-HS-204-1/6A, 4kV START BUS 1B NORM FDR BKR 1516 to TRIP	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: • **To meet this critical step, the alternate breaker handswitch must be held in close while the normal breaker handswitch taken to trip, resulting in a fast transfer of Start Bus 1B.** • **Once step 4 is complete to the examiner may elect to inform the examinee that "another operator will complete this task".**	
Step 5: [3.4] CHECK CLOSED 4kV START BUS 1B(2B) ALT FDR BKR 1414(1418). [3.5] CHECK OPEN 4kV START BUS 1B(2B) NORM FDR BKR 1516(1512) [3.6] RELEASE breakers 1414(1418) and 1516(1512) switches. Expected action(s): CHECKS CLOSED 4kV START BUS 1B ALT FDR BKR 1414. CHECKS OPEN 4kV START BUS 1B NORM FDR B1516. RELEASES breakers 1414 and 1516 switches.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 6: **[3.7] CHECK 4kV START BUS 1B(2B) AC volts is between 3950 and 4400 Volts as indicated on 0-EI-204-1/6A(0-EI-204-2/5A).** **Expected action(s):** **Determines** that 4kV START BUS 1B AC volts is between 3950 and 4400 Volts as indicated on 0-EI-204-1/6A.	_____ SAT _____UNSAT _____N/A
Step 7: **[3.8] CHECK LOCALLY 4kV START BUS 1B(2B) ALT FDR BKR 1414(1418**) closing spring target indicates charged and the amber breaker spring charged light is on. Expected action(s): Communicates with local AUO and determines that 4kV START BUS 1B ALT FDR BKR 1414 closing spring target indicates charged and the amber breaker spring charged light is on.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: When Examinee dispatches AUO to check breakers locally, inform the candidate "Your task is complete.".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

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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. Preparations are underway to unload CSST B for scheduled maintenance.

- All Precautions and Limitations have been reviewed and are being addressed by another Operator.
- AUO reports from the field that the Start Bus area is clear of all personnel and all pre-start checks are complete.

INITIATING CUES:

The Unit Supervisor directs you transfer 4kV Start Bus 1B to alternate in accordance with 0-OI-57A, Switchyard and 4160V AC Electrical System, Section 8.21.

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Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place a Second RHRSW Pump in Service in accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW)	
JPM NUMBER:		758-U2	REVISION:	0

TASK APPLICABILITY:	☐SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-023-NO-01/ Start an RHRSW/EECW Pump		
K/A RATINGS:	RO: 3.8		
K/A No. &STATEMENT:	400000 SF8 CCW Component Cooling Water System K1.01: Knowledge of the physical connections and/or cause and effect relationships between the Component Cooling Water System and the following systems: Service water system		
RELATED PRA INFORMATION:	SSCs in the Top 90% for CDF		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 758-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- STARTS RHRSW PUMP A1 using 0-HS-23-1A/2
- THROTTLES RHR HX 2A RHRSW OUTLET VALVE, 2-FCV-23-34, to maintain total A RHRSW flow between 4000 and 4500 gpm.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 0-OI-23

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	12/15/23	All	Initial issue

Procedure Revisions

Procedure	Revision
0-OI-23	109

SIMULATOR SETUP

IC	N/A
Exam IC	264

Console Operator Instructions	• **Reset to IC 264** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Schedule Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Event Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	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. Special testing is in progress for the RHRSW System, with the following plant conditions:

- All Precautions and Limitations have been reviewed and are being addressed by another Operator.
- The Outside AUO reports from the field that the RHRSW pump area is clear of all personnel and all pre-start checks are complete.

INITIATING CUES:

The Unit Supervisor directs you to place the A1 RHRSW Pump in Service in accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW), Section 6.1.1, Starting Second RHRSW Pump 'A' RHRSW Header.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] ENSURE the A2 or A1 RHRSW pump is in service in accordance with Section 5.1. Expected action(s): Determines by visual observation on panel 2-9-3 that the A2 RHRSW pump is in service.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Examinee may elect to notify Unit 1 and Unit 3 Operators before step [1] is performed.	
EXAMINER CUE: When the Examinee reaches for a phone or radio, notify the Examinee, "All communications will be through the Examiner".	
Step 2: [2] NOTIFY other Unit Operators that additional RHRSW pump(s) are to be started. Expected action(s): NOTIFIES other Unit Operators that additional RHRSW pump(s) are to be started.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: When the Examinee notifies another Unit Operator, notify the Examinee, "Notification understood".	
Step 3: [3] CHECK the upper and lower bearing oil level in the proper range for the RHRSW Pump A1(A2) motor to be started. Expected action(s): Requests status of upper and lower bearing oil level in the proper range for the RHRSW Pump A1.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: When the Examinee contacts the Outside AUO, notify the Examinee, "Upper and lower bearing oil level in the proper range for RHRSW Pump A1".	

STEP / STANDARD	SAT / UNSAT
Step 4: [4] START the RHRSW PUMP A1 using one of the following: □ 0-HS-23-1A/1 for Unit 1 □ 0-HS-23-1A/2 for Unit 2 □ 0-HS-23-1A/3 for Unit 3 Expected action(s): STARTS RHRSW PUMP A1 using 0-HS-23-1A/2	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Once RHRSW PUMP A1 is started header flow will be out of band high, the examinee must complete the below action.	
Step 5: [5] IF necessary, THEN THROTTLE the RHR HX 1(2)(3)A RHRSW OUTLET VLV(s), 1(2)(3)-FCV-23-34, using 1(2)(3)-HS-23-34A to maintain a cumulative RHRSW flow through the A RHR in-service heat exchangers greater than 4000 gpm (sum of the values on 1, 2, and 3-FI-23-36), and less than or equal to 4500 gpm through any one heat exchanger. Expected action(s): THROTTLES RHR HX 2A RHRSW OUTLET VALVE, 2-FCV-23-34, to maintain total A RHRSW flow to between 4000 and 4500 gpm.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 6: [6] **CHECK** RHRSW Pump A1 and A2 motor amps less than 61 amps indicated on RHRSW PUMP A1(A2) Amps: □ 0-EI-23-1/1(5/1) on Panel 1-9-3 □ 0-EI-23-1/2(5/2) on Panel 2-9-3 □ 0-EI-23-1/3(5/3) on Panel 3-9-3 Expected action(s): CHECKS RHRSW Pump A1 and A2 motor amps less than 61 amps indicated on RHRSW PUMP A1(A2) Amps, 0-EI-23-1/2(5/2), on Panel 2-9-3	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [7] **CHECK** at 4160V Shutdown Board A that RHRSW Pump A1(A2) charging spring recharged by observing amber breaker spring charged light on and closing spring target indicates charged. Expected action(s): Dispatches AUO to CHECK at 4160V Shutdown Board A that RHRSW Pump A1(A2) charging spring recharged by observing amber breaker spring charged light on and closing spring target indicates charged.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: When the Examinee dispatches another Operator to check local RHRSW pump breakers, inform the Examinee, "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 an Operator on Unit 2. Special testing is in progress for the RHRSW System, with the following plant conditions:

- All Precautions and Limitations have been reviewed and are being addressed by another Operator.
- The Outside AUO reports from the field that the RHRSW pump area is clear of all personnel and all pre-start checks are complete.

INITIATING CUES:

The Unit Supervisor directs you to place the A1 RHRSW Pump in Service in accordance with 0-OI-23, Residual Heat Removal Service Water (RHRSW), Section 6.1.1, Starting Second RHRSW Pump 'A' RHRSW Header.

Job Performance Measure (JPM)

SITE: BFN		JPM TITLE:	Emergency Venting Primary Containment
JPM NUMBER:	55A-U2	REVISION:	3

TASK APPLICABILITY:	☐SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-63 / Emergency Vent Primary Containment in Accordance with EOI Appendix 13		
K/A RATINGS:	RO 3.5 SRO 3.6		
K/A No. &STATEMENT:	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		
RELATED PRA INFORMATION:	SSCs in the Top 90% for CDF		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 55A-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Places 2-HS-64-221B in the PERM position
- Places 2-HS-64-221 in the OPEN position and recognizes that the valve did NOT open. The Operator may dispatch an Assistant Unit Operator (AUO) to locally operate 2-FCV-64-221. Reports that the Suppression Chamber cannot be vented and proceeds to Step 9.

Alternate Path:

- Contacts an AUO or an extra operator to perform step [3.8].
- Places keylock switch 2-HS-84-36 in the DRYWELL position.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI Appendix-13

VALIDATION TIME: 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
1	3/29/17	All	JPM Update
2	11/08/21	All	JPM Update
3	12/16/23	All	JPM Update

Procedure Revisions

Procedure	Revision
2-EOI Appendix 13	11

SIMULATOR SETUP

IC	N/A
Exam IC	265

Console Operator Instructions	• **Reset to IC 265** • **Place the simulator in RUN to ensure stable conditions**

Malfunctions	Description	Event	Severity	Delay	Initial set
T33D	MSL D BREAK IN DRYWELL	2	0.05	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
HS-64-221A	HARDENED SUPPRESSION CHAMBER VENT INBOARD ISOLATION VALVE	N/A	N/A	N/A	CLOSE

Schedule Files	Description	Event	Severity	Delay	Initial set
PC04	JUMPER 64-30 TO ALLOW DRYWELL VENT	1	N/A	N/A	N/A

Event Files	Description	Event	Severity	Delay	Initial set
ZDIHS8436(2) ==1	84-36 HS TO DW	2	N/A	N/A	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, with the following plant conditions:

- A large leak inside Primary Containment has developed
- The Reactor has been SCRAMMED
- Another Operator has been assigned to control Reactor Water Level and Reactor Pressure

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) directs you to vent Primary Containment in accordance with 2-EOI Appendix-13, Emergency Venting Primary Containment.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] NOTIFY Shift Manager / SED of the following: • Emergency Venting of Primary Containment is in progress. • Off-Gas Release Rate Limits will be exceeded. Expected Action(s): Notifies Shift Manager/SED.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: As Shift Manager/SED acknowledge report that Emergency Primary Containment Venting is in progress and Off-Gas release limits will be exceeded	
Step 2: **NOTES** 1) HARDENED CONTAINMENT VENT VALVES 2-FCV-64-221 and 222 may be operated locally with handwheels (U3 RB el. 580, west of clean room). 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] Expected Action(s): Verifies Suppression Pool level below 26 ft using 2-LI-64-159A and makes note of action required if the Suppression Pool cannot be vented.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [2.2] PLACE keylock switch 2-HS-64-222B, HARDENED SUPPRESSION CHAMBER VENT OUTBOARD PERMISSIVE, in PERM Expected Action(s): Places 2-HS-64-222B in the PERM position.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 4: [2.3] CHECK blue indicating light above 2-HS-64-222B, HARDENED SUPPRESSION CHAMBER VENT OUTBOARD PERMISSIVE, illuminated. Expected Action(s): Verifies BLUE indicating lamp above 2-HS-64-222B illuminated.	_____ SAT _____UNSAT _____N/A
Step 5: [2.4] OPEN 2-FCV-64-222, HARDENED SUPPRESSION CHAMBER VENT OUTBOARD ISOLATION VALVE. Expected Action(s): Places 2-HS-64-222A in the OPEN position and verifies 2-FCV-64-222 OPEN.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 6: [2.5] PLACE keylock switch 2-HS-64-221B, HARDENED SUPPRESSION CHAMBER VENT INBOARD PERMISSIVE, in PERM. Expected Action(s): Places 2-HS-64-221B in the PERM position.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: [2.6] CHECK blue indicating light above 2-HS-64-221B, HARDENED SUPPRESSION CHAMBER VENT INBOARD PERMISSIVE, illuminated. Expected Action(s): Verifies BLUE indicating lamp above 2-HS-64-221B illuminated.	_____ SAT _____UNSAT _____N/A
EXAMINER NOTE: Alternate path starts with Step 8.	
Step 8: [2.7] OPEN 2-FCV-64-221, HARDENED SUPPRESSION CHAMBER VENT INBOARD ISOL VALVE. Expected Action(s): Determines that 2-FCV-64-221will not OPEN and that the Suppression Chamber cannot be vented and proceeds to Step 3. If the Operator dispatches an Assistant Unit Operator (AUO) to perform any outside actions to open 2-FCV-64-221 then see cue below.	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER CUE: If the Operator dispatches an AUO to locally perform actions to open 2-FCV-64-221, report that "Outside actions to open 2-FCV-64-221 have failed".	
EXAMINER CUE: Acknowledge that the Suppression Chamber cannot be vented, and that Operator is continuing to Step 9.	

STEP / STANDARD	SAT / UNSAT
Step 9: [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 (RP) 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. Expected Action(s): Notifies the SM/SED that containment integrity failure is possible, notifies RP that the Reactor Building will be evacuated due to imminent failure of Primary Containment vent ducts, and makes P.A. announcement to evacuate Reactor Building.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: As SM/SED acknowledge report of possible containment integrity failure, as RP acknowledge report of Reactor Building evacuation due to imminent failure of Primary Containment vent ducts.	
Step 10: [3.4] START ALL available SGTS trains. Expected Action(s): Determines that all trains of SGTS are already in service by observing SGT OPERATING lights on 2-9-20.	_____ SAT _____UNSAT _____N/A
Step 11: [3.5] ENSURE CLOSED 2-FCV-64-36, DRYWELL/SUPPRESSION CHAMBER VENT TO SGT (Panel 2-9-3). Expected Action(s): Verifies 2-FCV-64-36 is closed on Panel 9-3.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 12: [3.6] ENSURE OPEN the following dampers (Panel 2-9-25): • 2-FCO-64-40, REACTOR ZONE EXHAUST TO SGTS • 2-FCO-64-41, REACTOR ZONE EXHAUST TO SGTS Expected Action(s): Verifies dampers 2-FCO-64-40 and 2-FCO-64-41 are open on Panel 2-9-25.	_____ SAT _____UNSAT _____N/A
Step 13: [3.7] ENSURE CLOSED 2-FCV-64-29, DRYWELL VENT INBOARD ISOLATION VALVE (Panel 2-9-3 or Panel 2-9-54). Expected Action(s): Verifies 2-FCV-64-29 is closed on Panel 9-3 or Panel 9-54.	_____ SAT _____UNSAT _____N/A
Step 14: [3.8] DISPATCH personnel to Unit 2 Auxiliary Instrument Room to perform the following: [3.8.1] REFER TO Attachment 1 and OBTAIN one 12-in. Banana Jack Jumper from EOI Equipment Storage Box. [3.8.2] LOCATE terminal strip DD in 2-PNLA-009-0043, Front. [3.8.3] JUMPER DD-76 to DD-77 (2-PNLA-009-0043). [3.8.4] NOTIFY Unit Operator that jumper for 2-FCV-64-30, DRYWELL VENT OUTBOARD ISOLATION VALVE, is in place. Expected Action(s): Contacts an AUO or an extra operator to perform step [3.8].	**Critical Step** _____ SAT _____UNSAT _____N/A
DRIVER: When called to install jumper for 2-FCV-64-30 • **Insert Event 1** • **Inform the Operator that you are using time compression, and report that the jumper for 2-FCV-64-30 is in place.**	

STEP / STANDARD	SAT / UNSAT
Step 15: [3.9] ENSURE OPEN 2-FCV-64-30, DRYWELL VENT OUTBOARD ISOLATION VALVE (Panel 2-9-3). Expected Action(s): Verifies open 2-FCV-64-30 on Panel 2-9-3.	_____ SAT _____UNSAT _____N/A
Step 16: **CAUTION** 1. The following step will fail ductwork inside Secondary Containment and may fail Secondary Containment Integrity. 2. Off-Gas Release Rate Limits will be exceeded. [3.10] PLACE keylock switch 2-HS-84-36, SUPPRESSION CHAMBER/DRYWELL VENT ISOLATION BYPASS SELECT, to DRYWELL (Panel 2-9-54). Expected Action(s): Places keylock switch 2-HS-84-36 in the DRYWELL position.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 17: [3.11] ENSURE OPEN 2-FCV-64-29, DRYWELL VENT INBOARD ISOLATION VALVE (Panel 2-9-54). Expected Action(s): Verifies that 2-FCV-64-29 opens on Panel 2-9-54.	_____ SAT _____UNSAT _____N/A
CUE: "Another Operator will continue this procedure; this completes your task".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

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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:

- A large leak inside Primary Containment has developed
- The Reactor has been SCRAMMED
- Another Operator has been assigned to control Reactor Water Level and Reactor Pressure

INITIATING CUE:

The Nuclear Unit Senior Operator (NUSO) directs you to vent Primary Containment in accordance with 2-EOI Appendix-13, Emergency Venting Primary Containment.

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Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Prepare IRMs for Reactor Startup per 2-OI-92A, Intermediate Range Monitors (Pre-brief)	
JPM NUMBER:		706-U2	REVISION:	1

TASK APPLICABILITY:	☐SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-092A-NO-01 / Prepare IRMs for Reactor Startup		
K/A RATINGS:	RO 3.9		
K/A No. &STATEMENT:	215003 SF7 Intermediate Range Monitor System A4.03 Ability to manually operate and/or monitor in the control room: IRM range switches		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 706-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Depresses the IRM 'H' IN OUT SELECT pushbutton and observes proper light indications.
- Depresses and holds 2-HS-92-7C/S3, DRIVE OUT, and observes the IRM 'H' IN backlight extinguishes.
- Releases 2-HS-92-7C/S3, DRIVE OUT, when the IRM 'H' OUT backlight illuminates.
- Depresses 2-HS-92-7C/S2, DRIVE IN, pushbutton and observes the IRM 'H' OUT backlight extinguishes.
- Depresses the IN OUT CH H IRM SELECT pushbutton and observes that the CH H IRM SELECT backlight extinguishes.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-OI-92A, Intermediate Range Monitors

VALIDATION TIME: 10 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	08/07/17	All	Initial issue
1	12/16/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-OI-92A	29

SIMULATOR SETUP

IC	N/A
Exam IC	278

Console Operator Instructions	• **Ensure the candidate has been pre-briefed on 2-OI-92A, Intermediate Range Monitors, Section 5.1, Preparing Intermediate Range Monitors for Reactor Startup** • **Reset the simulator to IC 278** • **Place the simulator in RUN to ensure stable conditions** • **Bypass IRM 'B'** • **Ensure De-selected all IRMs** • **De-select SRM/IRM Detector Position pushbutton** • **Set ICS Screen to IRMs**

Malfunctions	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Overrides	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Schedule Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	N/A

Event Files	Description	Event	Severity	Delay	Initial set
N/A	N/A	N/A	N/A	N/A	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 a Reactor Operator on Unit 2. Preparations for Reactor Startup are in progress.

INITIATING CUE:

You have been assigned to take over the procedure for preparing the IRMs for Reactor Startup in accordance with 2-OI-92A, Intermediate Range Monitors. The previous shift completed the procedure on all IRMs except IRM 'H'. The Unit Supervisor directs you to continue with 2-OI-92A, Intermediate Range Monitors, Section 5.1, Preparing Intermediate Range Monitors for Reactor Startup. 2-OI-92A, Intermediate Range Monitors, Section 4.0, Prestartup/Standby Readiness Requirements, is complete.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
NOTE: Ensure the candidate has been pre-briefed on 2-OI-92A, Intermediate Range Monitors, Section 5.1, Preparing Intermediate Range Monitors for Reactor Startup. Step 1: [2] REVIEW precautions and limitations in Section 3.0. Standard: Reviews precautions and limitations.	_____ SAT _____UNSAT _____N/A
Step 2: [3] CHECK that no more than one IRM Channel per trip system is bypassed. Standard: Notes that IRM 'B' is bypassed and verifies that no more than one IRM Channel per trip system is bypassed.	_____ SAT _____UNSAT _____N/A
Step 3: [4] VERIFY all operable IRM drives are working properly as follows: [4.1] DEPRESS the SRM/IRM DETECTOR POSITION pushbutton and OBSERVE the pushbutton illuminates. Standard: Depresses 2-HS-92-7C/S1, SRM/IRM DETECTOR POSITION, and observes that the pushbutton illuminates.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [4.2] DEPRESS the IN OUT CH H IRM SELECT pushbutton and OBSERVE the CH H IRM SELECT and IN backlights illuminate. Standard: Depresses the IRM 'H' IN OUT SELECT pushbutton and observes proper light indications.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 5: [4.3] DEPRESS and HOLD the DRIVE OUT pushbutton and OBSERVE the IRM 'H' IN backlight extinguishes. Standard: Depresses and holds 2-HS-92-7C/S3, DRIVE OUT, and observes the IRM 'H' IN backlight extinguishes.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 6: [4.4] WHEN the IRM 'H' OUT backlight illuminates, THEN: RELEASE the DRIVE OUT pushbutton. Standard: Releases 2-HS-92-7C/S3, DRIVE OUT, when the IRM 'H' OUT backlight illuminates.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 7: **NOTE** To stop the insertion of the monitor probe at any intermediate point, DEPRESS the DRIVE IN pushbutton a second time. Depressing the DRIVE IN pushbutton again will recommence inward motion. [4.5] DEPRESS the DRIVE IN pushbutton and OBSERVE the IRM 'H' OUT backlight extinguishes. Standard: Depresses 2-HS-92-7C/S2, DRIVE IN, pushbutton and observes the IRM 'H' OUT backlight extinguishes.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 8: [4.6] WHEN the IRM 'H' IN backlight illuminates, THEN DEPRESS the IN OUT CH H IRM SELECT pushbutton and OBSERVE the CH H IRM SELECT backlight extinguishes. Standard: Depresses the IN OUT CH H IRM SELECT pushbutton and observes that the CH H IRM SELECT backlight extinguishes.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 9: [5] CHECK all IN lights for un-bypassed IRMs are illuminated. Standard: Checks IN lights for all IRMs except IRM 'B'.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 10: **NOTE** The IRMs may or may not be downscale when in position 1. However, the downscale rod block (but not the alarm or status light) is bypassed in position 1 and operation of the system may continue. [6] On all unbypassed IRM channels, PLACE each Range Switch to position 1. Standard: Places/checks all un-bypassed IRM Range Switches are in Position 1.	_____ SAT _____UNSAT _____N/A
Step 11: [7] CHECK the following annunciators on 2-XA-55-5A are clear: A. IRM HIGH (Window 26) B. IRM CH A, C, E, G HI-HI/INOP (Window 33) C. IRM B, D, F, H HI-HI/INOP (Window 34) Standard: Verifies alarms clear.	_____ SAT _____UNSAT _____N/A
Step 12: [8] CHECK the IRM HI-HI or INOP and HIGH lights for all un-bypassed IRM channels are extinguished. Standard: Checks IRM HI-HI or INOP and HIGH lights for all IRMs except IRM 'B' are extinguished.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 13: **NOTE** IRM ranging and positioning is controlled by 3-GOI-100-1A. [9] Verify each IRM SELECT pushbutton is extinguished. Standard: Verifies each IRM SELECT pushbutton is extinguished.	_____ SAT _____UNSAT _____N/A
Step 14: Notifies the SRO that 2-OI-92A, Intermediate Range Monitors, Section 5.1, Preparing Intermediate Range Monitors for Reactor Startup, is complete.	_____ SAT _____UNSAT _____N/A
Examiner Cue: Acknowledge the report of task completion and inform the Examinee "Your task is complete".	

STOP TIME: _____

Job Performance Measure (JPM)

Provide to Applicant

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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 a Reactor Operator on Unit 2. Preparations for Reactor Startup are in progress.

INITIATING CUE:

You have been assigned to take over the procedure for preparing the IRMs for Reactor Startup in accordance with 2-OI-92A, Intermediate Range Monitors. The previous shift completed the procedure on all IRMs except IRM 'H'. The Unit Supervisor directs you to continue with 2-OI-92A, Intermediate Range Monitors, Section 5.1, Preparing Intermediate Range Monitors for Reactor Startup. 2-OI-92A, Intermediate Range Monitors, Section 4.0, Prestartup/Standby Readiness Requirements, is complete

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Vent and Re-pressurize the SCRAM Pilot Air Header
JPM NUMBER:	312-U2	REVISION:	6

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-20/ Vent and Repressurize the SCRAM Pilot Air Header In Accordance With EOI Appendix 1B		
K/A RATINGS:	RO: 3.6		
K/A No. &STATEMENT:	201001 SF1 CRDH Control Rod Drive Hydraulic System; K1.09; Knowledge of the physical connections and/or cause and effect relationships between the Control Rod Drive Hydraulic System and the following systems: IA		
RELATED PRA INFORMATION:	Risk Significant		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 312-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- SIMULATES closing 2-SHV-085-0331, CONTROL AIR SUPPLY HEADER SHUT OFF VALVE.
- SIMULATES opening 2-DRIV-085-0038A (2-PS-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS)
- SIMULATES opening 2-DRIV-085-0038B0038B (2-PI-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS)

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI Appendix-1B

VALIDATION TIME: 10 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	10/17/05	All	Initial issue
1	11/09/97	All	JPM update
2	09/22/02	All	JPM update
3	05/31/06	All	JPM update
4	05/31/08	All	JPM update
5	9/20/18	All	JPM update
6	12/17/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-EOI Appendix-1B	5

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 Operator on Unit 2.

- A Reactor SCRAM has occurred
- Control Rods are out
- The Unit Supervisor has entered 2-EOI-1A, ATWS RPV Control

INITIATING CUES:

The Unit Supervisor directs you to perform 2-EOI-Appendix-1B, Venting and Repressurizing the SCRAM Pilot Air Header.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: LOCATION: Unit 2 RB NE, Elevation 565', Panel 2-25-18 ATTACHMENTS: 1. Tools and Equipment [1] NOTIFY Unit Operator and CONTINUE in this procedure. Expected Action(s): Notifies Unit 2 Unit Operator and continues in the procedure. **CUE: If contacted as the Unit 2 Unit Operator, acknowledge the report given.**	_____ SAT _____UNSAT _____N/A
Step 2: [2] CLOSE 2-SHV-085-0331, CONTROL AIR SUPPLY HEADER SHUT OFF VALVE (RB North wall, near Scram Air Header Pressure Regulators). Expected Action(s): SIMUILATES closing 2-SHV-085-0331, CONTROL AIR SUPPLY HEADER SHUT OFF VALVE. **CUE: (As the candidate simulates closing the valves) Inform the examinee "The handwheel is turning", and after several seconds "The valve is closed".**	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [3] OPEN the following instrument drain valves (located on Panel 2-25-18, east end): • 2-DRIV-085-0038A (2-PS-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS) • 2-DRIV-085-0038B (2-PI-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS) Expected Action(s): • SIMUILATES opening 2-DRIV-085-0038A (2-PS-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS) • SIMUILATES opening 2-DRIV-085-0038B0038B (2-PI-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS)	**Critical Step** _____ SAT _____UNSAT _____N/A
CUES: • **(As the candidate simulates opening the valves) Inform the examinee "The handwheel is turning", and after several seconds "the valve is open".** • **(When both Drain Valves are open and the candidate checks Air Pressure on 2-PI-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS, Inform the examinee "SCRAM Valve Pilot Air Header Pressure is lowering".** • **Time compression: After 15 seconds, inform the examinee "SCRAM Valve Pilot Air Header Pressure is Zero PSIG".**	
Step 4: [4] WHEN 2-PI-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS, indicates 0 psig, THEN NOTIFY Unit 2 Operator. Expected Action(s): When 2-PI-085-0038, CRD SCRAM VALVE PILOT AIR HEADER PRESS, indicates 0 PSIG, notifies the Unit Operator.	_____ SAT _____UNSAT _____N/A
CUE: As the Unit 2 Operator, when informed by Examinee that SCRAM Valve Pilot Air Header Pressure is Zero PSIG, acknowledge the report.	
CUE: "Another Operator will perform the remaining steps. Task is complete."	

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 Operator on Unit 2.

- A Reactor SCRAM has occurred
- Control Rods are out
- The Unit Supervisor has entered 2-EOI-1A, ATWS RPV Control

INITIATING CUES:

The Unit Supervisor directs you to perform 2-EOI-Appendix-1B, Venting and Repressurizing the SCRAM Pilot Air Header.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Defeat Alternate Rod Insertion (ARI) Logic Trips	
JPM NUMBER:		314-U2	REVISION:	3

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-000-EM-26 / Defeat ARI Logic Trips in accordance with EOI Appendix-2		
K/A RATINGS:	RO: 4.2		
K/A No. &STATEMENT:	295037 SF7 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown EA1.01; Ability to operate and/or monitor the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Reactor Protection System.		
RELATED PRA INFORMATION:	Risk Significant		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 314-U2

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Refers to Attachment 1 and states that two keys will be retrieved from the Unit 2 Control Room.
- SIMULATES placing 2-HS-068-0118B, ATWS MODE SWITCH, in TEST
- SIMULATES placing 2-HS-068-0118A, ATWS MODE SWITCH, in TEST.

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 2-EOI Appendix-2

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	12/3/19	All	Initial issue
1	1/14/20	All	JPM update
2	11/16/21	All	JPM update
3	12/17/23	All	JPM update

Procedure Revisions

Procedure	Revision
2-EOI Appendix-2	4

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.

- A Reactor SCRAM has occurred
- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1A, ATWS RPV Control

INITIATING CUES:

The NUSO directs you to defeat Alternate Rod Insertion (ARI) Logic trips in accordance with 2-EOI Appendix-2, Defeating ARI Logic Trips.

CAUTION:

DO NOT OPEN PANELS!
DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: 2-EOI-APPENDIX-2, Defeating ARI Logic Trips [1] REFER to Attachment 1 and OBTAIN two keys. Expected action(s): **EXAMINER CUE: When the candidate begins to go to the Unit 2 Control Room to obtain the keys, state "You have the two required keys".** Refers to Attachment 1 and states that two keys will be retrieved from the Unit 2 Control Room.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [2] PLACE 2-HS-68-118B, ATWS MODE SWITCH, in TEST position on 2-LPNL-925-0717, ATWS PANEL CHANNEL-B. Expected action(s): **EXAMINER CUE: When the candidate simulates placing 2-HS-068-0118B in TEST, state "The ATWS Mode Switch is in the TEST position".** SIMULATES placing 2-HS-068-0118B, ATWS MODE SWITCH, in TEST.	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 3: [3] **PLACE** 2-HS-068-0118A, ATWS MODE SWITCH, in TEST position on 2-LPNL-925-0716, ATWS PANEL CHANNEL-A. Expected action(s): **EXAMINER CUE: When the candidate simulates placing 2-HS-068-0118A in TEST, state "The ATWS Mode Switch is in the TEST position".** **SIMULATES** placing 2-HS-068-0118A, ATWS MODE SWITCH, in TEST.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 4: [4] **NOTIFY** Unit Operator that ARI logic trips are defeated and ARI is reset. Expected action(s): **EXAMINER CUE: Acknowledge any report given to the Control Room/Unit Operator.** Informs the Unit 2 Control Room/Unit Operator that ARI logic trips are defeated, and that ARI is reset.	_____ SAT _____UNSAT _____N/A
EXAMINER CUE: Acknowledge the candidate's report that the task is complete.	

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.

- A Reactor SCRAM has occurred
- The Nuclear Unit Senior Operator (NUSO) has entered 2-EOI-1A, ATWS RPV Control

INITIATING CUES:

The NUSO directs you to defeat Alternate Rod Insertion (ARI) Logic trips in accordance with 2-EOI Appendix-2, Defeating ARI Logic Trips.

CAUTION:

DO NOT OPEN PANELS!
DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Emergency Shutdown at Diesel Generator (U3 ONLY)	
JPM NUMBER:	211A-U3	REVISION:	2	

TASK APPLICABILITY:	☐SRO-U	☒SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-082-NO-06 Emergency Shut Down Diesel Generator		
K/A RATINGS:	RO: 3.5		
K/A No. &STATEMENT:	264000 SF6 EGE Emergency Generators (Diesel/Jet) K4.07 Knowledge of Emergency Generators design features and/or interlocks that provide for the following: Local operation and control		
RELATED PRA INFORMATION:	Risk Significant		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 211-U3

RO _____ SRO _____ DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:
Alternate Path:

- SIMULATES placing 3-BKR-254-000A/06, DSL GEN 3A LOGIC RELAY PANEL (LOGIC BREAKER), in OFF on 3-BDGG-254-0003A, 125 VDC DSL SYS BAT BOARD A.
- SIMULATES depressing both ENGINE STOP push-buttons simultaneously on the Engine Control Cabinet.
- SIMULATES fully pulling out the red stopper for 0-SHV-018-0664-3A

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-2 Controlling Plant Evolutions Precisely.

REFERENCES/PROCEDURES NEEDED: 3-OI-82

VALIDATION TIME: 20 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	5/27/10	All	Initial Issue
1	11/30/16	All	JPM update
2	12/17/23	All	JPM update

Procedure Revisions

Procedure	Revision
3-OI-82	163

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 Auxiliary Reactor Operator.
- The 3A Diesel Generator is running at 900 rpm, unloaded.
- The 3A Diesel Generator is vibrating excessively.
- Control Room attempts to shut down the 3A Diesel Generator failed.

INITIATING CUES:

The Unit Supervisor directs you to shut down the 3A Diesel Engine locally in accordance with 3-OI-82, Standby Diesel Generator System, section 7.5.

CAUTION: DO NOT OPERATE ANY PLANT EQUIPMENT!

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [1] PULL OUT and HOLD the Injector Control Lever in the No Fuel position until the diesel engine comes to a complete stop. Expected Action(s): SIMULATES Injector Control Lever held out in the No Fuel position until the cue is given that the diesel engine continues at speed.	**Critical Step** _____SAT _____UNSAT _____N/A
EXAMINER NOTE: Begin Alternate Path **EXAMINER CUE: After thirty seconds tell the Operator that "The Diesel Engine continues at the same speed.".**	
Step 2: [2] IF the diesel engine does NOT stop, THEN PERFORM the following: [2.1] PLACE 3-BKR-254-000A(B,C,D)/06, DSL GEN3A(3B,3C,3D) LOGIC RELAY PANEL (LOGIC BREAKER), in OFF on 3-BDGG-54-0003A(3B,3C,3D), 125 VDC DSL SYS BAT BOARD A(B,C,D). Expected Action(s): SIMULATES placing 3-BKR-254-000A/06, DSL GEN 3A LOGIC RELAY PANEL (LOGIC BREAKER), in OFF on 3-BDGG-254-0003A, 125 VDC DSL SYS BAT BOARD A.	**Critical Step** _____SAT _____UNSAT _____N/A
Step 3: [2.2] DEPRESS both ENGINE STOP push-buttons simultaneously on the Engine Control Cabinet to initiate the shutdown sequence (this will also stop the Priming Fuel Pump). Expected Action(s): SIMULATES depressing both ENGINE STOP push-buttons simultaneously on the Engine Control Cabinet.	**Critical Step** _____SAT _____UNSAT _____N/A
EXAMINER CUE: After a few seconds tell the Operator "Diesel Engine speed starts lowering."	

STEP / STANDARD	SAT / UNSAT
Step 4: [2.3] CLOSE ENGINE DRIVEN FUEL PMP SUCT, 0-SHV-018-0664-3A (3B, 3C, 3D). Expected Action(s): SIMULATES fully pulling out the red stopper for 0-SHV-018-0664-3A.	**Critical Step** _____SAT _____UNSAT _____N/A
NOTE: Instructions are posted locally on a placard for closing 0-SHV-018-0664-3A.	
EXAMINER CUE: 0-SHV-018-0664-3A red stopper has moved from the full in position to the full out position and has stopped moving. After about thirty seconds notify the Examinee "The Diesel has come to a stop.".	
Step 5: [2.4] PLACE both DG 3A(3B,3C,3D) START CIRCUIT 1 CONT POWER BKR, 3-BKR-082-003A(B,C,D) 35W1 and DG 3A(3B,3C,3D) START CIRCUIT 2 CONT POWER BKR, 3-BKR-082-003A (3B,3C,3D) 35W2 in the OFF position at the Electrical Control Cabinet. Expected Action(s): SIMULATES placing DG 3A START CIRCUIT 1 CONT POWER BKR, 3-BKR-082-003A 35W1 and DG 3A START CIRCUIT 2 CONT POWER BKR, 3-BKR-082-003A 35W2 in the OFF position.	_____SAT _____UNSAT _____N/A
EXAMINER CUE: "Another Operator will continue form here; your task is complete".	

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 Auxiliary Reactor Operator.
- The 3A Diesel Generator is running at 900 rpm, unloaded.
- The 3A Diesel Generator is vibrating excessively.
- Control Room attempts to shut down the 3A Diesel Generator failed.

INITIATING CUES:

The Unit Supervisor directs you to shut down the 3A Diesel Engine locally in accordance with 3-OI-82, Standby Diesel Generator System, section 7.5.

CAUTION:
DO NOT OPERATE ANY PLANT EQUIPMENT!

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards
JPM NUMBER:	500	REVISION:	2

TASK APPLICABILITY:	☐SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-082-NO-10, Perform a Verification of Offsite Power Availability		
K/A RATINGS:	RO 3.7		
K/A No. &STATEMENT:	2.2.12 Knowledge of surveillance procedures		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 500

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Marks the 500 KV and 161 KV Sources as Qualified
- Marks Attachment 1 to match any of the Keys
- Marks A. 4KV SD BD 3EA 4300 VOLTS (+ or – 50)
- Marks B. 4KV SD BD 3EB 4300 VOLTS (+ or – 50)
- Marks C. 4KV SD BD 3EC 4200 VOLTS (+ or – 50)
- Marks D. 4KV SD BD 3ED 4250 VOLTS (+ or – 50)
- Determines that Acceptance Criteria is met

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely

REFERENCES/PROCEDURES NEEDED: 0-SR-3.8.1.A.1, PIP-02-03 AC Distribution

VALIDATION TIME: 20 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	08/28/05	All	Initial Issue
1	02/08/17	All	Procedure Revision, New format
2	2/7/24	All	Procedure Revision, Modified JPM

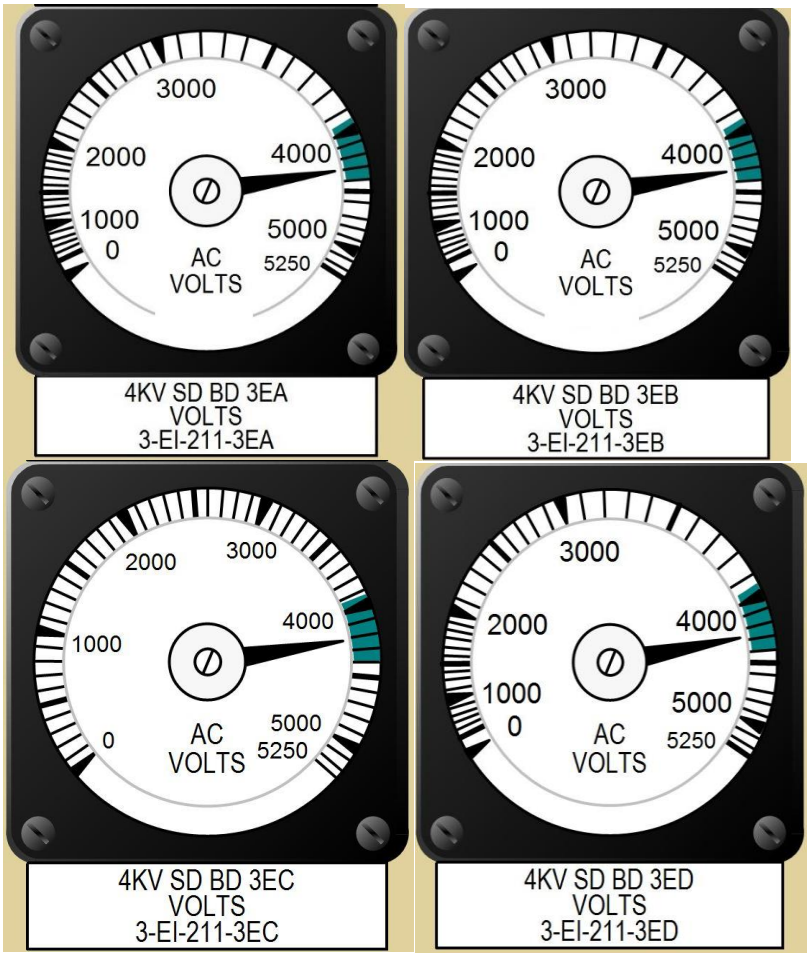
Procedure Revisions

Procedure	Revision
0-SR-3.8.1.A.1	20
PIP-02-03 AC Distribution	Revised 4/6/22

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 a Unit 3 Operator.

- Breaker 1312, 4KV Unit BD 3A Normal Feeder Breaker is out of service for a breaker swap.
- 3A Diesel Generator was removed from service and declared INOPERABLE 15 minutes ago and Tech Spec 3.8.1 Condition B has been entered.
- Units 1, 2 and 3 are at 100% power.
- The TVA Transmission Operator reports the following qualification status:
  - 500 kV Source to Browns Ferry Nuclear Plant is Qualified
  - 161 kV Source to Browns Ferry Nuclear Plant is Qualified
- The following voltages are indicated in the Unit 3 Control Room:



INITIATING CUES: Perform 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards for **Unit 3**. The Unit 1 and 2 portions have already been completed.

NOTE: The STS sheet is not required for this task.

KEY

Key

BFN Unit 0	Verification of Offsite Power Availability to 4.16 kV Shutdown Boards	0-SR-3.8.1.A.1 Rev. 0020 Page 21 of 22

Attachment 1
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Table of Offsite and Onsite Circuits Assigned to Units 1, 2 and 3

UNIT NO.			1 and 2		3	
QUALIFIED CIRCUIT			1	2	1	2
COMPONENT DESCRIPTION	USST/*CSST*		USST 1B	USST 1B	USST 3B JD	USST 3B
			USST 2B	USST 2B	------	------
			------	*CSST A*	*CSST A*	*CSST A* JD
			------	*CSST B*	*CSST B*	*CSST B*
	FOR CSST ONLY	A	------	BKR 1412	BKR 1412	BKR 1412 JD
			------	BKR 1414	BKR 1414	BKR 1414
		B	------	BKR 1518	BKR 1518	BKR 1518
			------	BKR 1516	BKR 1516	BKR 1516
	START BUS		------	*START BUS 1A*	*START BUS 1A*	*START BUS 1A* JD
			------	*START BUS 1B*	*START BUS 1B*	*START BUS 1B*
	FOR USST ONLY	1B	BKR 1112	BKR 1112	------	------
			BKR 1114	BKR 1114	------	------
		2B	BKR 1212	BKR 1212	------	------
			BKR 1214	BKR 1214	------	------
		3B	------	------	BKR 1312	BKR 1312
			------	------	BKR 1314 JD	BKR 1314
	FOR CSST ONLY	*1A*	------	*BKR 1424*	*BKR 1432*	*BKR 1432* JD
			------	*BKR 1428*	------	------
		1B	------	*BKR 1524*	*BKR 1528*	*BKR 1528*
			------	*BKR 1526*	------	------
	4KV UNIT BD		4KV UNIT BD 1A	4KV UNIT BD 1A	4KV UNIT BD 3A	4KV UNIT BD 3A JD
			4KV UNIT BD 1B	4KV UNIT BD 1B	4KV UNIT BD 3B JD	4KV UNIT BD 3B
			4KV UNIT BD 2A	4KV UNIT BD 2A	------	------
			4KV UNIT BD 2B	4KV UNIT BD 2B	------	------
	UNIT 1 & 2 ONLY		BKR 1126 & 1612	BKR 1126 & 1612	------	------
			BKR 1132 & 1712	BKR 1132 & 1712	------	------
			BKR 1226 & 1722	BKR 1226 & 1722	------	------
			BKR 1232 & 1622	BKR 1232 & 1622	------	------
	SHUTDOWN (SD) BUS		SD BUS 1	SD BUS 1	------	------
			SD BUS 2	SD BUS 2	------	------
	UNIT 3 ONLY		------	------	BKR 1326	BKR 1326 JD
			------	------	BKR 1332 JD	BKR 1332
	SD BD FDR BKR		BKR 1614 & BKR 1616	BKR 1614 & BKR 1616	BKR 1334 & BKR 1336	BKR 1334 & BKR 1336 JD
			BKR 1718 & BKR 1724	BKR 1718 & BKR 1724	BKR 1338 & BKR 1342 JD	BKR 1338 & BKR 1342
	4 KV SHUTDOWN BD		4KV SD BD A & B	4KV SD BD A & B	4KV SD BD 3EA & 3EB	4KV SD BD 3EA & 3EB JD
			4KV SD BD C & D	4KV SD BD C & D	4KV SD BD 3EC & 3ED JD	4KV SD BD 3EC & 3ED

See Attachment 1 page 2 for explanation and information for completing the above Table.

KEY

Key

BFN Unit 0	Verification of Offsite Power Availability to 4.16 kV Shutdown Boards	0-SR-3.8.1.A.1 Rev. 0020 Page 21 of 22

Attachment 1
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Table of Offsite and Onsite Circuits Assigned to Units 1, 2 and 3

UNIT NO.			1 and 2		3	
QUALIFIED CIRCUIT			1	2	1	2
COMPONENT DESCRIPTION	USST/CSST		USST 1B	USST 1B	USST 3B	USST 3B JD
			USST 2B	USST 2B	------	------
			------	CSST A	JD CSST A	CSST A
			------	CSST B	CSST B	CSST B
	FOR CSST ONLY	A	------	BKR 1412	JD BKR 1412	BKR 1412
			------	BKR 1414	BKR 1414	BKR 1414
		B	------	BKR 1518	BKR 1518	BKR 1518
			------	BKR 1516	BKR 1516	BKR 1516
	START BUS		------	START BUS 1A	JD START BUS 1A	START BUS 1A
			------	START BUS 1B	START BUS 1B	START BUS 1B
	FOR USST ONLY	1B	BKR 1112	BKR 1112	------	------
			BKR 1114	BKR 1114	------	------
		2B	BKR 1212	BKR 1212	------	------
			BKR 1214	BKR 1214	------	------
		3B	------	------	BKR 1312	BKR 1312
			------	------	BKR 1314	BKR 1314 JD
	FOR CSST ONLY	1A	------	BKR 1424	JD BKR 1432	BKR 1432
			------	BKR 1428	------	------
		1B	------	BKR 1524	BKR 1528	BKR 1528
			------	BKR 1526	------	------
	4KV UNIT BD		4KV UNIT BD 1A	4KV UNIT BD 1A	JD 4KV UNIT BD 3A	4KV UNIT BD 3A
			4KV UNIT BD 1B	4KV UNIT BD 1B	4KV UNIT BD 3B	4KV UNIT BD 3B JD
			4KV UNIT BD 2A	4KV UNIT BD 2A	------	------
			4KV UNIT BD 2B	4KV UNIT BD 2B	------	------
	UNIT 1 & 2 ONLY		BKR 1126 & 1612	BKR 1126 & 1612	------	------
			BKR 1132 & 1712	BKR 1132 & 1712	------	------
			BKR 1226 & 1722	BKR 1226 & 1722	------	------
			BKR 1232 & 1622	BKR 1232 & 1622	------	------
	SHUTDOWN (SD) BUS		SD BUS 1	SD BUS 1	------	------
			SD BUS 2	SD BUS 2	------	------
	UNIT 3 ONLY		------	------	JD BKR 1326	BKR 1326
			------	------	BKR 1332	BKR 1332 JD
	SD BD FDR BKR		BKR 1614 & BKR 1616	BKR 1614 & BKR 1616	JD BKR 1334 & BKR 1336	BKR 1334 & BKR 1336
			BKR 1718 & BKR 1724	BKR 1718 & BKR 1724	BKR 1338 & BKR 1342	BKR 1338 & BKR 1342 JD
	4 KV SHUTDOWN BD		4KV SD BD A & B	4KV SD BD A & B	JD 4KV SD BD 3EA & 3EB	4KV SD BD 3EA & 3EB
			4KV SD BD C & D	4KV SD BD C & D	4KV SD BD 3EC & 3ED	4KV SD BD 3EC & 3ED JD

See Attachment 1 page 2 for explanation and information for completing the above Table.

KEY

BFN Unit 0	Verification of Offsite Power Availability to 4.16 kV Shutdown Boards	0-SR-3.8.1.A.1 Rev. 0020 Page 21 of 22

Attachment 1
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Table of Offsite and Onsite Circuits Assigned to Units 1, 2 and 3

	UNIT NO.		1 and 2		3	
	QUALIFIED CIRCUIT		1	2	1	2
COMPONENT DESCRIPTION	USST/*CSST*		USST 1B	USST 1B	USST 3B	USST 3B
			USST 2B	USST 2B	------	------
			------	*CSST A*	*CSST A*	*CSST A*
			------	*CSST B*	*CSST B*	*CSST B*
	FOR CSST ONLY	A	------	BKR 1412	BKR 1412	BKR 1412
			------	BKR 1414	BKR 1414	BKR 1414
		B	------	BKR 1518	BKR 1518	BKR 1518
			------	BKR 1516	BKR 1516	BKR 1516
	START BUS		------	*START BUS 1A*	*START BUS 1A*	*START BUS 1A*
			------	*START BUS 1B*	*START BUS 1B*	*START BUS 1B*
	FOR USST ONLY	1B	BKR 1112	BKR 1112	------	------
			BKR 1114	BKR 1114	------	------
		2B	BKR 1212	BKR 1212	------	------
			BKR 1214	BKR 1214	------	------
		3B	------	------	BKR 1312	BKR 1312
			------	------	BKR 1314	BKR 1314
	FOR CSST ONLY	*1A*	------	*BKR 1424*	*BKR 1432*	*BKR 1432*
			------	*BKR 1428*	------	------
		1B	------	*BKR 1524*	*BKR 1528*	*BKR 1528*
			------	*BKR 1526*	------	------
	4KV UNIT BD		4KV UNIT BD 1A	4KV UNIT BD 1A	4KV UNIT BD 3A	4KV UNIT BD 3A
			4KV UNIT BD 1B	4KV UNIT BD 1B	4KV UNIT BD 3B	4KV UNIT BD 3B
			4KV UNIT BD 2A	4KV UNIT BD 2A	------	------
			4KV UNIT BD 2B	4KV UNIT BD 2B	------	------
	UNIT 1 & 2 ONLY		BKR 1126 & 1612	BKR 1126 & 1612	------	------
			BKR 1132 & 1712	BKR 1132 & 1712	------	------
			BKR 1226 & 1722	BKR 1226 & 1722	------	------
			BKR 1232 & 1622	BKR 1232 & 1622	------	------
	SHUTDOWN (SD) BUS		SD BUS 1	SD BUS 1	------	------
			SD BUS 2	SD BUS 2	------	------
	UNIT 3 ONLY		------	------	BKR 1326	BKR 1326
			------	------	BKR 1332	BKR 1332
	SD BD FDR BKR		BKR 1614 & BKR 1616	BKR 1614 & BKR 1616	BKR 1334 & BKR 1336	BKR 1334 & BKR 1336
			BKR 1718 & BKR 1724	BKR 1718 & BKR 1724	BKR 1338 & BKR 1342	BKR 1338 & BKR 1342
	4 KV SHUTDOWN BD		4KV SD BD A & B	4KV SD BD A & B	4KV SD BD 3EA & 3EB	4KV SD BD 3EA & 3EB
			4KV SD BD C & D	4KV SD BD C & D	4KV SD BD 3EC & 3ED	4KV SD BD 3EC & 3ED

See Attachment 1 page 2 for explanation and information for completing the above Table.

KEY

Key

BFN Unit 0	Verification of Offsite Power Availability to 4.16 kV Shutdown Boards	0-SR-3.8.1.A.1 Rev. 0020 Page 21 of 22

Attachment 1
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Table of Offsite and Onsite Circuits Assigned to Units 1, 2 and 3

	UNIT NO.		1 and 2		3	
	QUALIFIED CIRCUIT		1	2	1	2
COMPONENT DESCRIPTION	USST/CSST		USST 1B	USST 1B	USST 3B	USST 3B
			USST 2B	USST 2B	------	------
			------	CSST A	JD CSST A	CSST A
			------	CSST B	CSST B	CSST B JD
	FOR CSST ONLY	A	------	BKR 1412	BKR 1412	BKR 1412
			------	BKR 1414	JD BKR 1414	BKR 1414
		B	------	BKR 1518	BKR 1518	BKR 1518 JD
			------	BKR 1516	BKR 1516	BKR 1516
	START BUS		------	START BUS 1A	START BUS 1A	START BUS 1A JD
			------	START BUS 1B	JD START BUS 1B	START BUS 1B
	FOR USST ONLY	1B	BKR 1112	BKR 1112	------	------
			BKR 1114	BKR 1114	------	------
		2B	BKR 1212	BKR 1212	------	------
			BKR 1214	BKR 1214	------	------
		3B	------	------	BKR 1312	BKR 1312
			------	------	BKR 1314	BKR 1314
	FOR CSST ONLY	1A	------	BKR 1424	BKR 1432	BKR 1432 JD
			------	BKR 1428	------	------
		1B	------	BKR 1524	JD BKR 1528	BKR 1528
			------	BKR 1526	------	------
	4KV UNIT BD		4KV UNIT BD 1A	4KV UNIT BD 1A	4KV UNIT BD 3A	4KV UNIT BD 3A JD
			4KV UNIT BD 1B	4KV UNIT BD 1B	JD 4KV UNIT BD 3B	4KV UNIT BD 3B
			4KV UNIT BD 2A	4KV UNIT BD 2A	------	------
			4KV UNIT BD 2B	4KV UNIT BD 2B	------	------
	UNIT 1 & 2 ONLY		BKR 1126 & 1612	BKR 1126 & 1612	------	------
			BKR 1132 & 1712	BKR 1132 & 1712	------	------
			BKR 1226 & 1722	BKR 1226 & 1722	------	------
			BKR 1232 & 1622	BKR 1232 & 1622	------	------
	SHUTDOWN (SD) BUS		SD BUS 1	SD BUS 1	------	------
			SD BUS 2	SD BUS 2	------	------
	UNIT 3 ONLY		------	------	BKR 1326	BKR 1326 JD
			------	------	JD BKR 1332	BKR 1332
	SD BD FDR BKR		BKR 1614 & BKR 1616	BKR 1614 & BKR 1616	BKR 1334 & BKR 1336	BKR 1334 & BKR 1336 JD
			BKR 1718 & BKR 1724	BKR 1718 & BKR 1724 JD	BKR 1338 & BKR 1342	BKR 1338 & BKR 1342
	4 KV SHUTDOWN BD		4KV SD BD A & B	4KV SD BD A & B	4KV SD BD 3EA & 3EB	4KV SD BD 3EA & 3EB JD
			4KV SD BD C & D	4KV SD BD C & D JD	4KV SD BD 3EC & 3ED	4KV SD BD 3EC & 3ED

See Attachment 1 page 2 for explanation and information for completing the above Table.

KEY

Key

BFN Unit 0	Verification of Offsite Power Availability to 4.16 kV Shutdown Boards	0-SR-3.8.1.A.1 Rev. 0020 Page 21 of 22

Attachment 1
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Table of Offsite and Onsite Circuits Assigned to Units 1, 2 and 3

UNIT NO.			1 and 2		3	
QUALIFIED CIRCUIT			1	2	1	2
COMPONENT DESCRIPTION	USST/CSST		USST 1B	USST 1B	USST 3B	USST 3B
			USST 2B	USST 2B	------	------
			------	CSST A	CSST A	CSST A JD
			------	CSST B	JD CSST B	CSST B
	FOR CSST ONLY	A	------	BKR 1412	BKR 1412	BKR 1412
			------	BKR 1414	BKR 1414	BKR 1414 JD
		B	------	BKR 1518	JD BKR 1518	BKR 1518
			------	BKR 1516	BKR 1516	BKR 1516
	START BUS		------	START BUS 1A	JD START BUS 1A	START BUS 1A
			------	START BUS 1B	START BUS 1B	START BUS 1B JD
	FOR USST ONLY	1B	BKR 1112	BKR 1112	------	------
			BKR 1114	BKR 1114	------	------
		2B	BKR 1212	BKR 1212	------	------
			BKR 1214	BKR 1214	------	------
		3B	------	------	BKR 1312	BKR 1312
			------	------	BKR 1314	BKR 1314
	FOR CSST ONLY	1A	------	BKR 1424	JD BKR 1432	BKR 1432
			------	BKR 1428	------	------
		1B	------	BKR 1524	BKR 1528	BKR 1528 JD
			------	BKR 1526	------	------
	4KV UNIT BD		4KV UNIT BD 1A	4KV UNIT BD 1A	JD 4KV UNIT BD 3A	4KV UNIT BD 3A
			4KV UNIT BD 1B	4KV UNIT BD 1B	4KV UNIT BD 3B	4KV UNIT BD 3B JD
			4KV UNIT BD 2A	4KV UNIT BD 2A	------	------
			4KV UNIT BD 2B	4KV UNIT BD 2B	------	------
	UNIT 1 & 2 ONLY		BKR 1126 & 1612	BKR 1126 & 1612	------	------
			BKR 1132 & 1712	BKR 1132 & 1712	------	------
			BKR 1226 & 1722	BKR 1226 & 1722	------	------
			BKR 1232 & 1622	BKR 1232 & 1622	------	------
	SHUTDOWN (SD) BUS		SD BUS 1	SD BUS 1	------	------
			SD BUS 2	SD BUS 2	------	------
	UNIT 3 ONLY		------	------	JD BKR 1326	BKR 1326
			------	------	BKR 1332	BKR 1332 JD
	SD BD FDR BKR		BKR 1614 & BKR 1616	BKR 1614 & BKR 1616	JD BKR 1334 & BKR 1336	BKR 1334 & BKR 1336
			BKR 1718 & BKR 1724	BKR 1718 & BKR 1724	BKR 1338 & BKR 1342	BKR 1338 & BKR 1342 JD
	4 KV SHUTDOWN BD		4KV SD BD A & B	4KV SD BD A & B	JD 4KV SD BD 3EA & 3EB	4KV SD BD 3EA & 3EB
			4KV SD BD C & D	4KV SD BD C & D	4KV SD BD 3EC & 3ED	4KV SD BD 3EC & 3ED JD

See Attachment 1 page 2 for explanation and information for completing the above Table.

KEY

Key

BFN Unit 0	Verification of Offsite Power Availability to 4.16 kV Shutdown Boards	0-SR-3.8.1.A.1 Rev. 0020 Page 21 of 22

Attachment 1
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Table of Offsite and Onsite Circuits Assigned to Units 1, 2 and 3

UNIT NO.			1 and 2		3	
QUALIFIED CIRCUIT			1	2	1	2
COMPONENT DESCRIPTION	USST/CSST		USST 1B	USST 1B	USST 3B	USST 3B
			USST 2B	USST 2B	------	------
			------	CSST A	CSST A	CSST A JD
			------	CSST B	JD CSST B	CSST B
	FOR CSST ONLY	A	------	BKR 1412	BKR 1412	BKR 1412 JD
			------	BKR 1414	BKR 1414	BKR 1414
		B	------	BKR 1518	BKR 1518	BKR 1518
			------	BKR 1516	JD BKR 1516	BKR 1516
	START BUS		------	START BUS 1A	START BUS 1A	START BUS 1A JD
			------	START BUS 1B	JD START BUS 1B	START BUS 1B
	FOR USST ONLY	1B	BKR 1112	BKR 1112	------	------
			BKR 1114	BKR 1114	------	------
		2B	BKR 1212	BKR 1212	------	------
			BKR 1214	BKR 1214	------	------
		3B	------	------	BKR 1312	BKR 1312
			------	------	BKR 1314	BKR 1314
	FOR CSST ONLY	1A	------	BKR 1424	BKR 1432	BKR 1432 JD
			------	BKR 1428	------	------
		1B	------	BKR 1524	JD BKR 1528	BKR 1528
			------	BKR 1526	------	------
	4KV UNIT BD		4KV UNIT BD 1A	4KV UNIT BD 1A	4KV UNIT BD 3A	4KV UNIT BD 3A JD
			4KV UNIT BD 1B	4KV UNIT BD 1B	JD 4KV UNIT BD 3B	4KV UNIT BD 3B
			4KV UNIT BD 2A	4KV UNIT BD 2A	------	------
			4KV UNIT BD 2B	4KV UNIT BD 2B	------	------
	UNIT 1 & 2 ONLY		BKR 1126 & 1612	BKR 1126 & 1612	------	------
			BKR 1132 & 1712	BKR 1132 & 1712	------	------
			BKR 1226 & 1722	BKR 1226 & 1722	------	------
			BKR 1232 & 1622	BKR 1232 & 1622	------	------
	SHUTDOWN (SD) BUS		SD BUS 1	SD BUS 1	------	------
			SD BUS 2	SD BUS 2	------	------
	UNIT 3 ONLY		------	------	BKR 1326	BKR 1326 JD
			------	------	JD BKR 1332	BKR 1332
	SD BD FDR BKR		BKR 1614 & BKR 1616	BKR 1614 & BKR 1616	BKR 1334 & BKR 1336	BKR 1334 & BKR 1336 JD
			BKR 1718 & BKR 1724	BKR 1718 & BKR 1724	JD BKR 1338 & BKR 1342	BKR 1338 & BKR 1342
	4 KV SHUTDOWN BD		4KV SD BD A & B	4KV SD BD A & B	4KV SD BD 3EA & 3EB	4KV SD BD 3EA & 3EB JD
			4KV SD BD C & D	4KV SD BD C & D JD	4KV SD BD 3EC & 3ED	4KV SD BD 3EC & 3ED

See Attachment 1 page 2 for explanation and information for completing the above Table.

START TIME_____

STEP / STANDARD	SAT / UNSAT
Step 1: **Section 6.3 Verification of Offsite 500kv and 161kv Sources** [2] DOCUMENT the source qualification below. Expected Action(s): Marks the 500 KV and 161 KV Sources as Qualified	**Critical Step** ___SAT ___UNSAT ___N/A
Step 2: [3] COMPLETE Attachment 1, to document which subcomponents are credited as satisfying the requirements of a qualified offsite circuit for Unit 1, Unit 2, and Unit 3 by marking the subcomponent with a circle around it and placing an initial next to the subcomponent. Expected Action(s): Marks Attachment 1 to match any of the Keys	**Critical Step** ___SAT ___UNSAT ___N/A
EXAMINER NOTE: Any combination that matches a column from any of the 4 Attachment 1 Keys is satisfactory provided that each component is only selected once. For example, USST 3B and any beneath subcomponent can only be selected once on Attachment 1.	
Step 3: [4] IF 4KV SD BD A(B,C,D) VOLTS, 0-EI-211-A(B,C,D) is used to obtain Shutdown board voltage, THEN ENSURE 4KV SD BD A(B,C,D) VOLTAGE SELECT, 0-XS-211-A(B,C,D) positioned to desired Shutdown Board while checking voltages in following step. (Otherwise mark N/A) [5] IF performing this procedure to satisfy Technical Specification 3.8.1 Required Action A.1 and/or B.1 for Unit 1 and Unit 2, THEN RECORD voltages for the following boards and CHECK proper voltage is available. (Otherwise N/A) Expected Action(s): Marks both steps as N/A	___SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [6] IF 4KV SD BD 3EA(3EB,3EC,3ED) VOLTS, 3-EI-2113-EA (3EB,3EC,3ED) is used to obtain Shutdown board voltage, THEN ENSURE 4KV SD BD 3EA(3EB,3EC,3ED) VOLTAGE SELECT, 3-XS-211-3EA(3EB,3EC,3ED) positioned to desired Shutdown Board while checking voltages in following step. (Otherwise mark N/A) Expected Action(s) Marks as N/A	_____ SAT _____UNSAT _____N/A
Step 5: [7] IF performing this procedure to satisfy Technical Specification 3.8.1 Required Action A.1 and/or B.1 for Unit 3, THEN RECORD voltages for the following boards and CHECK proper voltage is available. (Otherwise N/A) A. 4KV SD BD 3EA VOLTAGE Voltage_____(3950 - 4400 VOLTS) B. 4KV SD BD 3EB VOLTAGE Voltage_____(3950 - 4400 VOLTS) C. 4KV SD BD 3EC VOLTAGE Voltage_____(3950 - 4400 VOLTS) D. 4KV SD BD 3ED VOLTAGE Voltage_____(3950 - 4400 VOLTS) Expected Action(s): Marks A. 4KV SD BD 3EA 4300 VOLTS (+ or – 50) Marks B. 4KV SD BD 3EB 4300 VOLTS (+ or – 50) Marks C. 4KV SD BD 3EC 4200 VOLTS (+ or – 50) Marks D. 4KV SD BD 3ED 4250 VOLTS (+ or – 50)	**Critical Step** ___SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
NOTE **Steps 6.4[1], 6.4[2] and 6.4[3] may be marked N/A if this procedure is NOT required to satisfy Unit 1 and 2 Technical Specification 3.8.1 Required Action A.1 or B.1.**	
Step 6: **6.4 Verification of Unit 1 and or Unit 2 Offsite AC Power Circuits** [1] IF performing this procedure to satisfy Unit 1(2) Technical Specification 3.8.1 Required Action B.1 (one required Unit 1 and 2 DG inoperable) AND Unit 1(2) Technical Specification 3.8.1 Required Action A.1 (one required offsite circuit inoperable) concurrently, with only one 4.16 KV Shutdown Board affected THEN (Otherwise, N/A) [2] IF performing this procedure to satisfy Unit 1(2) Technical Specification 3.8.1 Required Action B.1 (one required Unit 1 and 2 DG inoperable) only, THEN CHECK at least TWO Unit 1/2 Offsite AC circuits are determined Operable in Attachment 1 and the Shutdown Boards included in the Operable circuits have proper voltage verified in Step 6.3[5]. (Otherwise, N/A) [3] IF performing this procedure to satisfy Unit 1(2) Technical Specification 3.8.1 Required Action A.1 (one required offsite circuit inoperable) only, THEN CHECK at least ONE Unit 1/2 Offsite AC circuit is determined Operable in Attachment 1 and the Shutdown Boards included in the Operable circuit have proper voltage verified in Step 6.3[5]. (Otherwise, N/A) Expected Action(s): Marks Steps 6.4 [1] [2] and [3] as N/A	___SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 7: **6.5 Verification of Unit 3 Offsite AC Power Circuits** [1] IF performing this procedure to satisfy Unit 3 Technical Specification 3.8.1 Required Action B.1 (one required Unit 3 DG inoperable) AND Unit 3 Technical Specification 3.8.1 Required Action A.1 (one required offsite circuit inoperable) concurrently, with only one 4.16 KV Shutdown Board affected THEN (Otherwise N/A) [2] IF performing this procedure to satisfy Unit 3 Technical Specification 3.8.1 Required Action B.1 (one required Unit 3 DG inoperable) only, THEN CHECK at least TWO Unit 3 Offsite AC circuits are determined Operable in Attachment 1 and the Shutdown Boards included in the Operable circuits have proper voltage verified in Step 6.3[7]. _____(AC) [3] IF performing this procedure to satisfy Unit 3 Technical Specification 3.8.1 Required Action A.1 (one required offsite circuit inoperable) only, THEN (Otherwise N/A) Expected Action(s): 6.5[2] Determines that Acceptance Criteria is met (**Critical Step)** Marks Steps 6.5 [1] and [3] as N/A (not a Critical Step)	**Critical Step** ___SAT ___UNSAT ___N/A
EXAMINER NOTE: • **When the Candidate identifies that the Acceptance Criteria is met, the JPM can be ended.** • **If Candidate enquires about Section 7.0 on page 20, reply "Section 7 on page 20 is not required for this task".**	

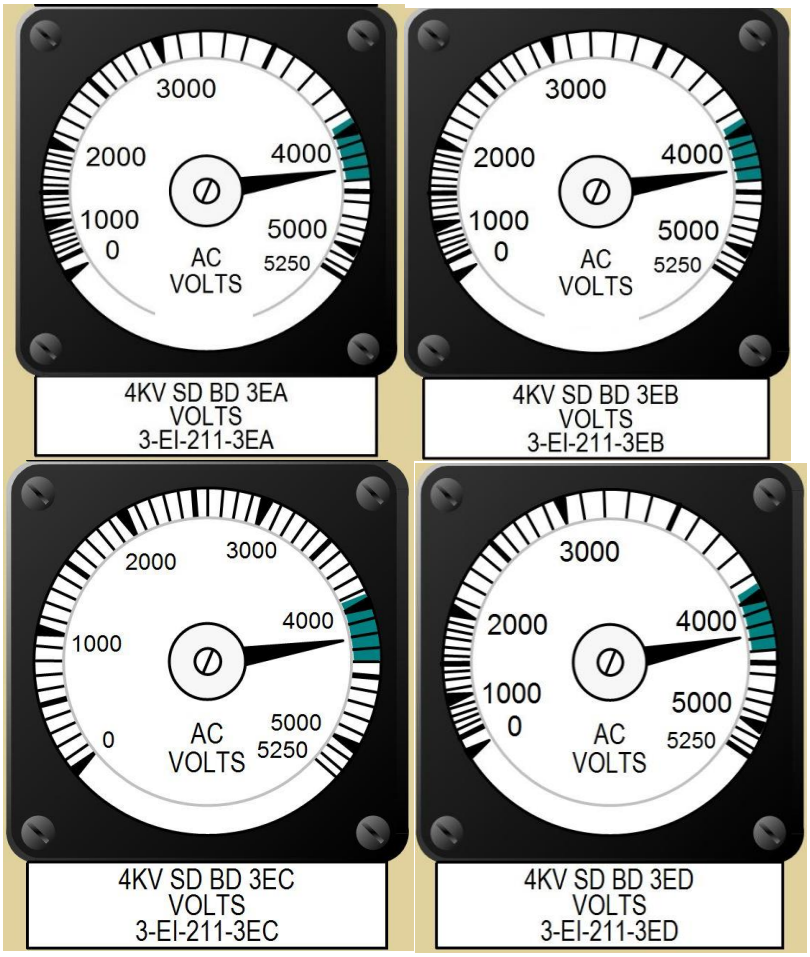
STOP TIME _____

END OF TASK

Provide to Applicant

INITIAL CONDITIONS: You are a Unit 3 Operator.

- Breaker 1312, 4KV Unit BD 3A Normal Feeder Breaker is out of service for a breaker swap.
- 3A Diesel Generator was removed from service and declared INOPERABLE 15 minutes ago.
- Units 1, 2 and 3 are at 100% power.
- The TVA Transmission Operator reports the following qualification status:
  - 500 kV Source to Browns Ferry Nuclear Plant is Qualified
  - 161 kV Source to Browns Ferry Nuclear Plant is Qualified
- The following voltages are indicated in the Unit 3 Control Room



INITIATING CUES: Perform 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards for Unit 3. The Unit 1 and 2 portions have already been completed.

NOTE: The STS sheet is not required for this task.

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place an RPS Channel in Trip
JPM NUMBER:	745	REVISION:	2

TASK APPLICABILITY:	☐SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	RO: 3.9		
K/A No. &STATEMENT:	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.		
RELATED PRA INFORMATION:	Risk Significant RPS Scram Reduction		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 745

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Locates 2-PIS-3-22AA, Reactor High Pressure A1 Channel, on Print 2-730E915-9 **AND/OR** in 2-OI-99, Reactor Protection System, Attachment 3
- Determines that fuse 2-FU1-3-22AA (5A-F5A) is required to be removed

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-3 Operating the Plant with a Conservative Bias

REFERENCES/PROCEDURES NEEDED: 2-OI-99, Prints 2-730E915-9, 2-45E671-26

VALIDATION TIME: 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
1	02/01/2022	ALL	Updated JPM task standard and initiating cue.
2	03/12/2021	All	Updated formatting

Procedure Revisions

Procedure	Revision
2-OI-99	101
2-730E915-9	30
2-45E671-26	5

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).
The Instrument Mechanics conducted testing on 2-PIS-3-22AA, Reactor High Pressure A1 Channel to support a surveillance. The instrument, 2-PIS-3-22AA, failed downscale.

INITIATING CUES: You are a Unit Operator; the Nuclear Unit Senior Operator (NUSO) has directed you to place 2-PIS-3-22AA, Reactor High Pressure A1 Channel, in trip in accordance with plant procedures.

Determine **ALL** of the following:

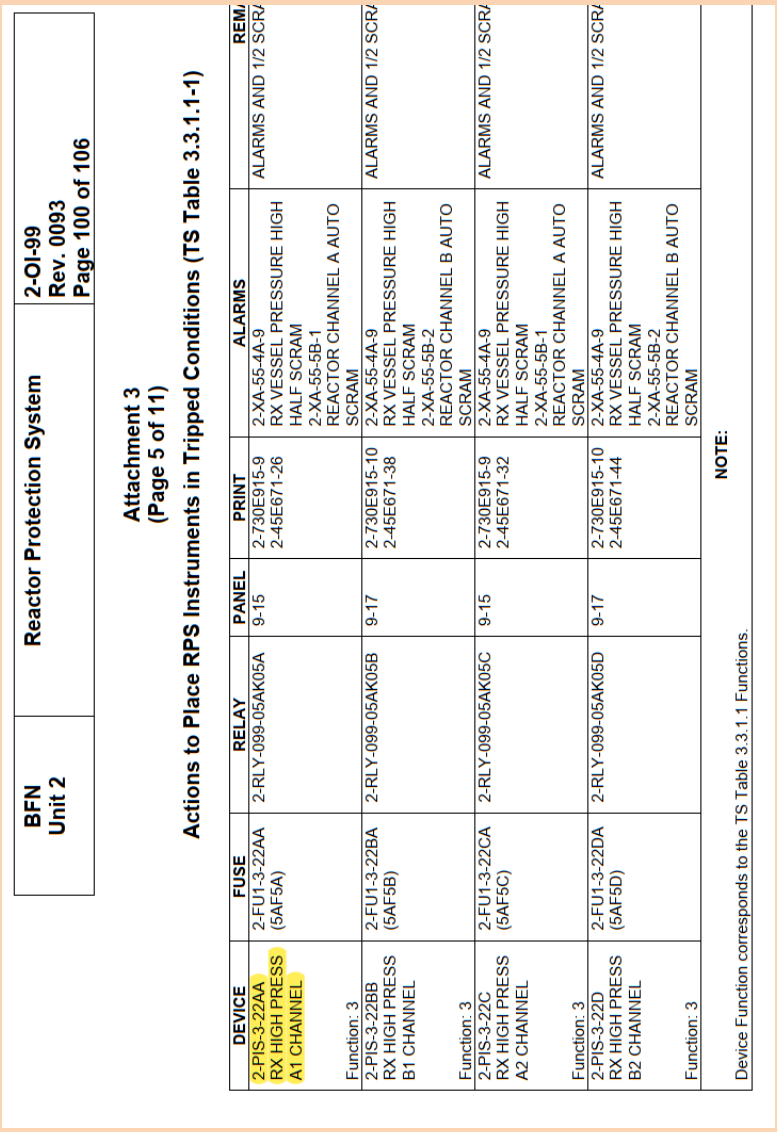
- List the plant procedure(s)/document(s) used to perform this task
- In accordance with the respective plant procedure(s)/document(s) selected, identify how this task is performed

Answer:

_____
_____
_____
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_____
_____
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_____
_____
_____

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: Refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) **AND/OR** Print 2-730E915-9 (see next page) for 2-PIS-3-22AA, Reactor High Pressure A1 Channel.	**Critical Step** _____ SAT _____UNSAT _____N/A



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Attachment 3 (Page 5 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REM
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCR

NOTE:

Device Function corresponds to the TS Table 3.3.1.1 Functions.

Expected Action(s):

Examinee refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9 (see next page) to reference the respective failed instrument 2-PIS-3-22AA, Reactor High Pressure A1 Channel.

STEP / STANDARD	SAT / UNSAT
Step 1 (continued): Print 2-730E915-9 (2-PIS-3-22AA is located between A-3 and E-3 coordinates).  	

STEP / STANDARD	SAT / UNSAT
Step 2: Determine how the Required Action is performed to place 2-PIS-3-22AA, Reactor High Pressure A1 Channel in trip in accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9. Expected Action(s): In accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9, determines fuse **2-FU1-3-22AA (5A-F5A)** is required to be removed in order for 2-PIS-3-22AA, Reactor High Pressure A1 Channel to be placed in trip as directed.	**Critical Step** _____ SAT _____UNSAT _____N/A

2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11)

BFN Unit 2 | Reactor Protection System | 2-OI-99 Rev. 0093 Page 100 of 106

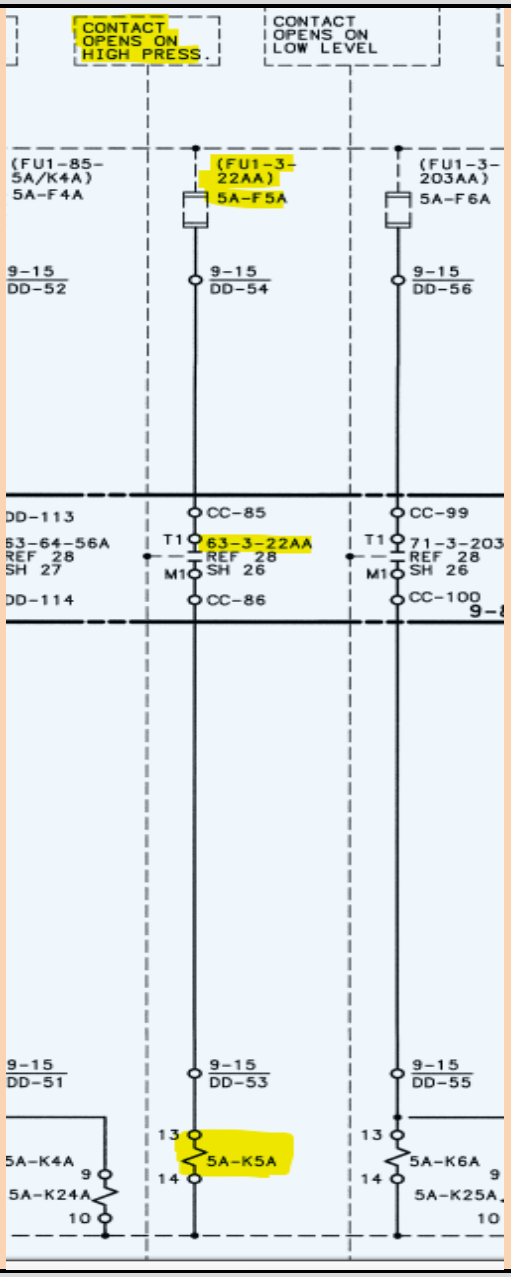
Attachment 3
(Page 5 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL

NOTE:
Device Function corresponds to the TS Table 3.3.1.1 Functions.

STEP / STANDARD	SAT / UNSAT
Step 2 (continued): Print 2-730E915-9	



EXAMINER CUE: Once the Operator identifies that 2-FU1-3-22AA(5A-F5A) has to be pulled in accordance with 2-OI-99, Reactor Protection System, Attachment 3 and/or print, 2-730E915-9 for the failed instrument (2-PIS-3-22AA, Reactor High Pressure A1 Channel) Inform the candidate "Another Operator will finish this procedure. 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:

Unit 2 is operating at 100% Rated Thermal Power (RTP).
The Instrument Mechanics conducted testing on 2-PIS-3-22AA, Reactor High Pressure A1 Channel to support a surveillance. The instrument, 2-PIS-3-22AA, failed downscale.

INITIATING CUES: You are a Unit Operator; the Nuclear Unit Senior Operator (NUSO) has directed you to place 2-PIS-3-22AA, Reactor High Pressure A1 Channel, in trip in accordance with plant procedures.

Determine **ALL** of the following:

- List the plant procedure(s)/document(s) used to perform this task
- In accordance with the respective plant procedure(s)/document(s) selected, identify how this task is performed

Answer:

_____
_____
_____
_____
_____
_____
_____
_____
_____

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	3-SR-3.4.9.3&4 Reactor Recirculation Pump Start Limitations
JPM NUMBER:	639	REVISION:	5

TASK APPLICABILITY:	☐SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	U-068-SU-04 Perform Reactor Recirculation Pump Start Limitations SR		
K/A RATINGS:	RO: 3.7		
K/A No. &STATEMENT:	2.2.12 Knowledge of surveillance procedures		
RELATED PRA INFORMATION:	N/A		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 639

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Records the Recirc Loop 3A Temperature of 172.2 °F and/or 164.4°F in table 1, records Recirc Loop 3B Temperature of 160 °F and/or 151.8°F and records current time.
- 212°F in Table 1.
- Records Bottom Head Drain Temperature of 123.2°F in Table 1 using the Integrated Computer System (ICS) Point 56-8.
- Subtracts 123.2°F from 212°F and records 88.8°F in Table 1
- Verifies the temperature difference is less than 145°F
- Subtracts Recirc Loop 3A Temperature of either 172.2°F or 164.4°F from 212°F and records 39.8°F or 47.6°F for Loop A and then subtracts loop 3B Temperature of either 160°F or 151.8°F from 212°F and records 52°F or 60.2°F for Loop B.
- Determines that the **3A** Recirc Pump can be started **(< 50°F Δ** Temp) and that the **3B** Recirc Pump can NOT be started **(> 50°F Δ** Temp).

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-3 Operating the Plant with a Conservative Bias

REFERENCES/PROCEDURES NEEDED: 3-SR-3.4.9.3&4

VALIDATION TIME: 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
2	08/16/17	All	Converted JPM to new format
3	9/11/18	All	Updated JPM
4	9/17/19	All	New revision of 3-SR-3.4.9.3&4
5	1/04/24	All	New revision of 3-SR-3.4.9.3&4

Procedure Revisions

Procedure	Revision
3-SR-3.4.9.3&4	18

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 a Unit 3 Unit Operator. Unit 3 is in MODE 4 preparing for a Reactor Startup. Forced Reactor Coolant circulation has been lost and is required to be restored. Flow has been verified in the Reactor Water Cleanup (RWCU) System.

INITIATING CUES:

You are a Unit Operator; the Nuclear Unit Senior Operator (NUSO) has directed you to Perform 3-SR-3.4.9.3&4 Reactor Recirculation Pump Start Limitations and determine which, if any, Recirc Pump(s) can be started.

NOTE: The STS sheet is not required for this task.

Answer:

KEY

Table 1
(Page 1 of 1)
Reactor Recirculation Pump Start Limitations Data

NOTE

Where applicable, circle the instrument used. May mark the step **N/A** where a loop is **NOT** being started.

STEP	PARAMETER	Panel NO.	INSTRUMENT **OR** COMPUTER LOG NO.	UNITS
7.0[3]	Recirc Loop A Temp.	Computer Computer Computer 3-9-4	68-6A, Recirc Pmp A Suct Temp **OR** 68-6B, Recirc Pmp A Suct Temp **OR** 68-2, Recirc Pmp A Discharge Temp **OR** 3-XR-68-2/5, 3-TE-68-2 (Point 003) RED - Pump A	**172.2 or 164.4** °F
	Recirc Loop B Temp.	Computer Computer Computer 3-9-4	68-83A, Recirc Pmp B Suct Temp **OR** 68-83B, Recirc Pmp B Suct Temp **OR** 68-78, Recirc Pmp B Discharge Temp **OR** 3-XR-68-2/5, 3-TE-68-78 (Point 004) - Pump B	**160 or 151.8** °F
7.0[4]	Reactor Steam Dome Press.	3-9-5 3-9-5 Computer	3-PI-3-54, Press A **OR** 3-PI-3-61, Press B **OR** 3-54, Reactor Press	__**0**__ psig
7.0[5]	RPV Coolant Temp.	As DETERMINED in Step 7.0[5]		__**212**__ °F
7.0[6]	Reactor Bottom Head Drain Temp.	As DETERMINED in Step 7.0[6]		_**123.2**__ °F
7.0[7] **AND** 7.0[8]	Dome/Drain △T (≤145°F)	Step (7.0[5] - 7.0[6])		_**88.8**_ °F(AC)
7.0[9] **AND** 7.0[10]	Dome/Loop △T ≤50°F [In MODE 2 with **BOTH** pumps **NOT** in operation (≤75°F)]	Step (7.0[5] - 7.0[3])		**39.8 or 47.6** °F(AC) Loop A **52 or 60.2** °F(AC) Loop B

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: **NOTE** The following steps are required to be completed within 15 minutes of starting a Recirc Pump. On occasions where RWCU is out of service, the Bottom Head Drain temperature is required to be obtained using a Contact Pyrometer (Step 7.0[6.2.2]). [3] RECORD Recirc Loop temperature(s) in Table 1 AND [3.1] ENSURE that the data taker is in place AND ready to record the temperature data. [3.2] RECORD the time the Recirc Loop temperature(s) data was taken (IF applicable, THEN mark the loop temperature N/A for the loop NOT being started). Record time _____ Standard: Records the Recirc Loop 3A Temperature of 172.2 °F and/or 164.4°F in table 1, records Recirc Loop 3B Temperature of 160 °F and/or 151.8°F and records current time.	**Critical Step** _____ SAT _____UNSAT _____N/A
Step 2: [4] RECORD Reactor Steam Dome Pressure in Table 1. Standard: Records Reactor Steam Dome Pressure of 0 psig in Table 1.	_____ SAT _____UNSAT _____N/A
Step 3: [5] DETERMINE Reactor Coolant Temperature as follows using the first step that satisfies its conditional requirement: (mark N/A for the steps NOT used.): [5.1] If the Reactor Coolant System is producing steam, (in a saturated steam condition). RECORD value in Table 1. Standard: Marks step as N/A.	_____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
Step 4: [5.2] IF one Recirculation Loop is in operation AND the Reactor is NOT in MODE 1, THEN Standard: Marks step as N/A.	_____ SAT _____UNSAT _____N/A
Step 5: [5.3] IF Shutdown Cooling is in service, THEN Standard: Marks step as N/A.	_____ SAT _____UNSAT _____N/A
Step 6: [5.4] IF the Reactor is in cold shutdown at zero pressure with NO forced circulation, THEN RECORD 212°F in Table 1. Standard: Records 212°F in Table 1.	**Critical Step** ___ SAT ___UNSAT ___N/A
CUE: IF the candidate references Section 4.0, Prerequisites, Step [2], and requests to know if there are any outstanding work orders against the Bottom Head Drain Thermocouple, state "Refer to the cue sheet".	
Step 7: [6] DETERMINE Reactor Bottom Head Drain Temperature as follows using EITHER step that satisfies its conditional requirement (mark N/A for the steps NOT used): [6.1] IF RWCU is in service with Bottom Head Drain thermocouple indicating accurately, THEN RECORD 3-TR-56-4, Point 7 (3-TE-56-8) OR RECORD Integrated Computer System (ICS) Point 56-8 as the bottom Head Drain Temperature in Table 1. Standard: Records Bottom Head Drain Temperature of 123.2°F in Table 1 using the Integrated Computer System (ICS) Point 56-8.	**Critical Step** ___ SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 8: [6.2] IF RWCU is out of service, or Step [6.1] cannot be performed, THEN PERFORM the following (OTHERWISE, N/A): Standard: Marks sub steps [6.2.1], [6.2.2], [6.2.3] as N/A.	___ SAT ___UNSAT ___N/A
Step 9: [7] CALCULATE the difference between Bottom Head Coolant Temperature AND the RPV Coolant Temperature by SUBTRACTING Reactor Bottom Head Drain Temperature from RPV Coolant Temperature AND RECORD this value in Table 1. Standard: Subtracts 123.2°F from 212°F and records 88.8°F in Table 1.	**Critical Step** ___ SAT ___UNSAT ___N/A
Step 10: [8] ENSURE the difference between the Bottom Head Coolant Temperature AND the RPV Coolant Temperature is less than or equal to 145°F, AND RECORD the time of verification. Record time _____ Standard: Verifies the temperature difference is less than 145°F.	**Critical Step** ___ SAT ___UNSAT ___N/A

STEP / STANDARD	SAT / UNSAT
Step 11: [9] CALCULATE the difference between the Coolant Temperature in the Recirculation Loop to be started AND the RPV Coolant Temperature by SUBTRACTING the Recirculation Loop to be started from the RPV Coolant Temperature AND RECORD this value in Table 1. (N/A the other loop on occasions where only one loop is being started). Standard: Subtracts Recirc Loop 3A Temperature of either 172.2°F or 164.4°F from 212°F and records 39.8°F or 47.6°F for Loop A and then subtracts loop 3B Temperature of either 160°F or 151.8°F from 212°F and records 52°F or 60.2°F for Loop B.	**Critical Step** ___ SAT ___UNSAT ___N/A
Step 12: [10] ENSURE the difference between the Coolant Temperature in the Recirculation Loop to be started the AND the RPV Coolant is less than or equal to 50°F. OR In MODE 2 AND Both Recirc Pumps are NOT in operation, the difference may be less than or equal to 75°F. Standard: Determines that the 3A Recirc Pump **can be** started (< 50°F Δ Temp) and that the 3B Recirc Pump **can NOT be** started (> 50°F Δ Temp).	**Critical Step** ___ SAT ___UNSAT ___N/A
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:

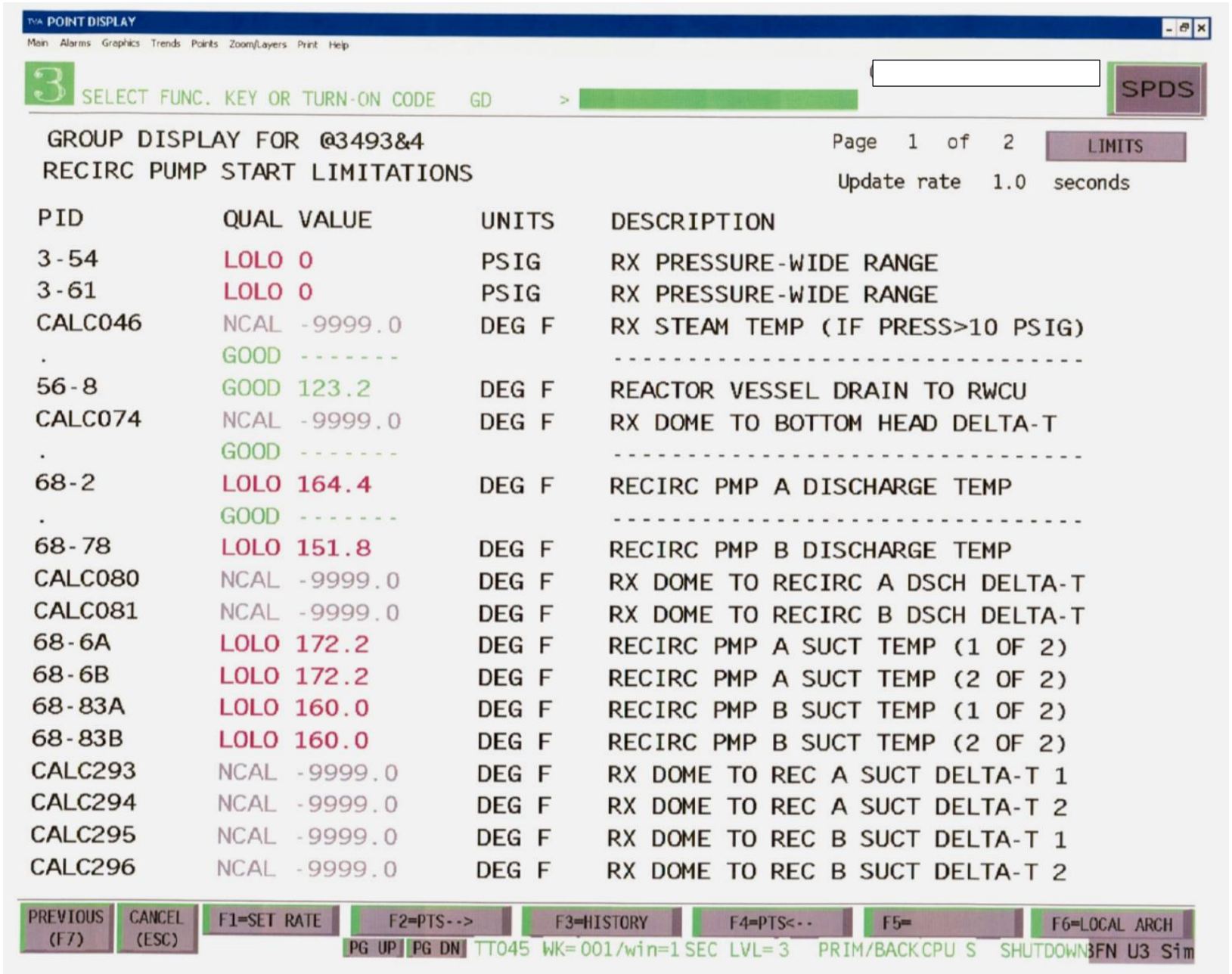
You are a Unit 3 Unit Operator. Unit 3 is in MODE 4 preparing for a Reactor Startup. Forced Reactor Coolant circulation has been lost and is required to be restored. Flow has been verified in the Reactor Water Cleanup (RWCU) System.

INITIATING CUES:

You are a Unit Operator; the Nuclear Unit Senior Operator (NUSO) has directed you to Perform 3-SR-3.4.9.3&4 Reactor Recirculation Pump Start Limitations and determine which, if any, Recirc Pump(s) can be started.

NOTE: The STS sheet is not required for this task.

Answer:


TVA POINT DISPLAY
Main Alarms Graphics Trends Points Zoom/Layers Print Help
3 SELECT FUNC. KEY OR TURN-ON CODE GD >
SPDS
GROUP DISPLAY FOR @3493&4
RECIRC PUMP START LIMITATIONS
Page 1 of 2
LIMITS
Update rate 1.0 seconds
PID QUAL VALUE UNITS DESCRIPTION
3-54 LOLO 0 PSIG RX PRESSURE-WIDE RANGE
3-61 LOLO 0 PSIG RX PRESSURE-WIDE RANGE
CALC046 NCAL -9999.0 DEG F RX STEAM TEMP (IF PRESS>10 PSIG)
. GOOD ------- ----------
56-8 GOOD 123.2 DEG F REACTOR VESSEL DRAIN TO RWCU
CALC074 NCAL -9999.0 DEG F RX DOME TO BOTTOM HEAD DELTA-T
. GOOD ------- ----------
68-2 LOLO 164.4 DEG F RECIRC PMP A DISCHARGE TEMP
. GOOD ------- ----------
68-78 LOLO 151.8 DEG F RECIRC PMP B DISCHARGE TEMP
CALC080 NCAL -9999.0 DEG F RX DOME TO RECIRC A DSCH DELTA-T
CALC081 NCAL -9999.0 DEG F RX DOME TO RECIRC B DSCH DELTA-T
68-6A LOLO 172.2 DEG F RECIRC PMP A SUCT TEMP (1 OF 2)
68-6B LOLO 172.2 DEG F RECIRC PMP A SUCT TEMP (2 OF 2)
68-83A LOLO 160.0 DEG F RECIRC PMP B SUCT TEMP (1 OF 2)
68-83B LOLO 160.0 DEG F RECIRC PMP B SUCT TEMP (2 OF 2)
CALC293 NCAL -9999.0 DEG F RX DOME TO REC A SUCT DELTA-T 1
CALC294 NCAL -9999.0 DEG F RX DOME TO REC A SUCT DELTA-T 2
CALC295 NCAL -9999.0 DEG F RX DOME TO REC B SUCT DELTA-T 1
CALC296 NCAL -9999.0 DEG F RX DOME TO REC B SUCT DELTA-T 2
PREVIOUS (F7)
CANCEL (ESC)
F1=SET RATE
F2=PTS-->
F3=HISTORY
F4=PTS<--
F5=
F6=LOCAL ARCH
PG UP PG DN TT045 WK=001/win=1 SEC LVL=3 PRIM/BACKCPU S SHUTDOWN3FN U3 Sim

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Determine ALARA Path
JPM NUMBER:	688	REVISION:	4

TASK APPLICABILITY:	☐SRO-U	☐SRO-I	☒UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	RO: 3.2		
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:	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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 688

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Calculates Path 1 round-trip dose of **575 mr (570 to 580 mr)**
- Calculates Path 2 round-trip dose of **1100 mr (1090 to 1110 mr)**
- Calculates Lifesaving activity work dose of **9.1 R (9.0 to 9.2 R OR 9100 mr)**
- Determines Path 1 total TEDE is the lowest at **9675 mr (9575 to 9775)**

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-3 Operating the Plant with a Conservative Bias

REFERENCES/PROCEDURES NEEDED: N/A

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
3	02/22/23	All	Revised JPM to be SRO only
4	01/04/24	All	Revised JPM to be RO only

Procedure Revisions

Procedure	Revision

Examiner Copy

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:

A General Emergency has been declared on Unit 2. Drywell entry is required to perform a lifesaving event with the following conditions:

- Dose rates for this evolution are **NOT** due to airborne activity

INITIATING CUES:

Given the conditions on the following page, calculate the **Total TEDE** for each path and determine which path maintains dose **ALARA**.

Answer:

KEY

Dose Rate	Path 1 Transit time	Dose Rate	Path 2 Transit time
300 mr/hr	From staging area to point A requires: 10 minutes =___50 mr____	300 mr/hr	From staging area to point C requires: 10 minutes =___50 mr ____
150 mr/hr	From point A to point B requires: 15 minutes =___37.5 mr____	1.5 R/hr	From point C to the work location requires: 20 minutes =__500 mr ____
2 R/hr	From point B to the work location requires: 6 minutes =__200 mr ____		
Arrives at work location with =__287.5 mr ____		Arrives at work location with =__550 mr ____	
Round Trip TEDE	=__575 mr ____	**Round Trip TEDE**	=__1100 mr ____
18.2 R/hr	Lifesaving activity 30 minutes =__9100 mr ____	18.2 R/hr	Lifesaving activity 30 minutes =__9100 mr ____
Total TEDE	=__9,675 mr ____	**Total TEDE**	=__10,200 mr ____

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: Determine Path 1 transit dose. Expected Action(s): Calculates Path 1round-trip dose of **575 mr** 287.5 X 2 = **575 mr (570 to 580 mr)**	**Critical Step** ___SAT ___UNSAT ___N/A
Step 2: Determine Path 2 transit dose. Expected Action(s): Calculates Path 2 round-trip dose of **1100 mr** 550 X 2 = **1100 mr (1090 to 1110 mr)**	**Critical Step** ___SAT ___UNSAT ___N/A
Step 3: Determines Lifesaving activity work dose. Expected Action(s): Calculates Lifesaving activity work dose of **9.1 R** 18.2 R/hr x 0.5 hr = **9.1 R (9.0 to 9.2 R OR 9100 mr)**	**Critical Step** ___SAT ___UNSAT ___N/A
Step 4: Determines the transit Path with the lowest dose option. Expected Action(s): Determines Path 1 total TEDE is the lowest at **9675 mr (9575 to 9775)**	**Critical Step** ___SAT ___UNSAT ___N/A
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:

A General Emergency has been declared on Unit 2. Drywell entry is required to perform a lifesaving event with the following conditions:

- Dose rates for this evolution are **NOT** due to airborne activity

INITIATING CUES:

Given the conditions on the following page, calculate the **ROUND TRIP** TEDE for each path and determine which path maintains dose ALARA.

Answer:

Dose Rate	Path 1 Transit time	Dose Rate	Path 2 Transit time
300 mr/hr	From staging area to point A requires: 10 minutes =_____	300 mr/hr	From staging area to point C requires: 10 minutes =_____
150 mr/hr	From point A to point B requires: 15 minutes =_____	1.5 R/hr	From point C to the work location requires: 20 minutes =_____
2 R/hr	From point B to the work location requires: 6 minutes =_____		
Arrives at work location with =_____		Arrives at work location with =_____	
Round Trip TEDE	=_____	**Round Trip TEDE**	=_____
18.2 R/hr	Lifesaving activity 30 minutes =_____	18.2 R/hr	Lifesaving activity 30 minutes =_____
Total TEDE	=_____	**Total TEDE**	=_____

Name: _____ Date: _____

Job Performance Measure (JPM)

SITE: BFN		JPM TITLE:	0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards
JPM NUMBER:	500-SRO	REVISION:	2

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	U-082-NO-10, Perform a Verification of Offsite Power Availability		
K/A RATINGS:	SRO 4.1		
K/A No. &STATEMENT:	2.2.12 Knowledge of surveillance procedures		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 500-SRO

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Determines Acceptance Criteria (AC) is met
- Determines no Technical Specification Required Action(s) must be taken

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely

REFERENCES/PROCEDURES NEEDED: 0-SR-3.8.1.A.1, PIP-02-03 AC Distribution

VALIDATION TIME: 20 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	08/28/05	All	Initial Issue
1	02/08/17	All	Procedure Revision, New format
2	2/7/24	All	Procedure Revision, Modified JPM

Procedure Revisions

Procedure	Revision
0-SR-3.8.1.A.1	20
PIP-02-03 AC Distribution	Revised 4/6/22

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 Unit 3 NUSO.

- Breaker 1312, 4KV Unit BD 3A Normal Feeder Breaker is out of service for a breaker swap.
- 3A Diesel Generator was removed from service and declared INOPERBLE 15 minutes ago and Tech Spec 3.8.1 Condition B has been entered.
- Units 1, 2 and 3 are at 100% power.
- The Unit 3 BOP Operator has completed 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 kV Shutdown Boards, for your review.

INITIATING CUES: Review 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards, for **Unit 3**. The Unit 1 and 2 portions have already been completed.

- Determine if Acceptance Criteria (AC) is met
- Determine what Technical Specification Required Action(s) must be taken (if any)?

Answer:

_____
_____
_____
_____
_____
_____
_____
_____

KEY

BFN Unit 0	Verification of Offsite Power Availability to 4.16 kV Shutdown Boards	0-SR-3.8.1.A.1 Rev. 0020 Page 15 of 22

Date today

6.3 Verification of Offsite 500kv and 161kv Sources

NOTES

1) TVA's Transmission Operator (TOp) is responsible for determining if the transmission grid is configured and operating within established limits that ensure the grids ability to provide QUALIFIED offsite power to Browns Ferry Nuclear Plant.
2) Southwest TOp phone number is 9-1-888-882-4009.
3) Loss of Supervisory Control and Data Acquisition (SCADA) is NOT reportable. (Refer to CR 1333068 and 1333603)

[1] **CONTACT** the TVA Southwest TOp to verify the transmission grid is able to provide QUALIFIED offsite power to BFN, in accordance with TRO-EA-SOP-30.405, Nuclear Offsite Power Operating Requirements. A2

[2] **DOCUMENT** the source qualification below.

	QUALIFIED	UNQUALIFIED
500KV SOURCE	☒	☐
161KV SOURCE	☒	☐

A2

[3] **COMPLETE** Attachment 1, to document which subcomponents are credited as satisfying the requirements of a qualified offsite circuit for Unit 1, Unit 2, and Unit 3 by marking the subcomponent with a circle around it. A2

[4] **IF** performing this procedure to satisfy Technical Specification 3.8.1 Required Action A.1 and/or B.1 for Unit 3, **THEN**

RECORD voltages for the following boards and **CHECK** proper voltage is available. (Otherwise N/A)

A. 4KV SD BD 3EA VOLTAGE

Voltage 4300 (3950 - 4400 VOLTS) A2

B. 4KV SD BD 3EB VOLTAGE

Voltage 4300 (3950 - 4400 VOLTS) A2

C. 4KV SD BD 3EC VOLTAGE

Voltage 4200 (3950 - 4400 VOLTS) A2

D. 4KV SD BD 3ED VOLTAGE

Voltage 4250 (3950 - 4400 VOLTS) A2

KEY

Attachment 1
(Page 1 of 2)

Table of Offsite and Onsite Circuits Assigned to Units 1, 2 and 3

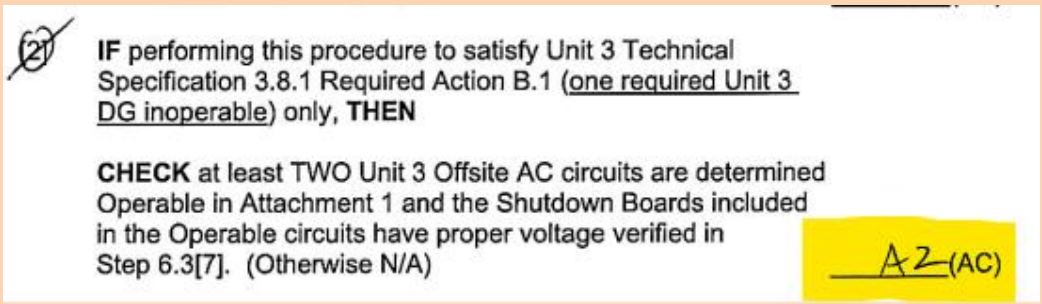
			UNIT NO.	1 and 2		3	
			QUALIFIED CIRCUIT	1	2	1	2
COMPONENT DESCRIPTION	USST/CSST			USST 1B A2	USST 1B	USST 3B	USST 3B A2
				USST 2B	USST 2B A2	-----	-----
				-----	CSST A	A2 CSST A	CSST A
				-----	CSST B	CSST B	CSST B
	FOR CSST ONLY	A		-----	BKR 1412	A2 BKR 1412	BKR 1412
				-----	BKR 1414	BKR 1414	BKR 1414
		B		-----	BKR 1518	BKR 1518	BKR 1518
				-----	BKR 1516	BKR 1516	BKR 1516
	START BUS			-----	START BUS 1A	A2 START BUS 1A	START BUS 1A
				-----	START BUS 1B	START BUS 1B	START BUS 1B
	FOR USST ONLY	1B		BKR 1112 A2	BKR 1112	-----	-----
				BKR 1114	BKR 1114	-----	-----
		2B		BKR 1212	BKR 1212 A2	-----	-----
				BKR 1214	BKR 1214	-----	-----
		3B		-----	-----	BKR 1312	BKR 1312
				-----	-----	BKR 1314	BKR 1314 A2
	FOR CSST ONLY	1A		-----	BKR 1424	A2 BKR 1432	BKR 1432
				-----	BKR 1428	-----	-----
		1B		-----	BKR 1524	BKR 1528	BKR 1528
				-----	BKR 1526	-----	-----
	4KV UNIT BD			4KV UNIT BD 1A A2	4KV UNIT BD 1A	A2 4KV UNIT BD 3A	4KV UNIT BD 3A
				4KV UNIT BD 1B	4KV UNIT BD 1B	4KV UNIT BD 3B	4KV UNIT BD 3B A2
				4KV UNIT BD 2A	4KV UNIT BD 2A A2	-----	-----
				4KV UNIT BD 2B	4KV UNIT BD 2B	-----	-----
	UNIT 1 & 2 ONLY			BKR 1126 & 1612 A2	BKR 1126 & 1612	-----	-----
				BKR 1132 & 1712	BKR 1132 & 1712	-----	-----
				BKR 1226 & 1722	BKR 1226 & 1722 A2	-----	-----
				BKR 1232 & 1622	BKR 1232 & 1622	-----	-----
	SHUTDOWN (SD) BUS			SD BUS 1 A2	SD BUS 1	-----	-----
				SD BUS 2	SD BUS 2 A2	-----	-----
	UNIT 3 ONLY			-----	-----	A2 BKR 1326	BKR 1326
				-----	-----	BKR 1332	BKR 1332 A2
	SHUTDOWN BD FDR BKR			BKR 1614 & BKR 1616 K3	BKR 1614 & BKR 1616	K3 BKR 1334 & BKR 1336	BKR 1334 & BKR 1336
				BKR 1718 & BKR 1724	BKR 1718 & BKR 1724 A2	BKR 1338 & BKR 1342	BKR 1338 & BKR 1342 A2
	4 KV SHUTDOWN BD			4KV SD BD A & B K3	4KV SD BD A & B	K3 4KV SD BD 3EA & 3EB	4KV SD BD 3EA & 3EB
				4KV SD BD C & D	4KV SD BD C & D A2	4KV SD BD 3EC & 3ED	4KV SD BD 3EC & 3ED K3

21 **IF** performing this procedure to satisfy Unit 3 Technical Specification 3.8.1 Required Action B.1 (<u>one required Unit 3 DG inoperable</u>) only, **THEN**

CHECK at least TWO Unit 3 Offsite AC circuits are determined Operable in Attachment 1 and the Shutdown Boards included in the Operable circuits have proper voltage verified in Step 6.3[7]. (Otherwise N/A)

A2 (AC)

START TIME_____

Examiner Note: A completed copy of 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards is required to be issued to the examinee for review	
STEP / STANDARD	**SAT / UNSAT**
Step 1: **Section 6.3 Verification of Offsite 500kv and 161kv Sources**  (2) **IF** performing this procedure to satisfy Unit 3 Technical Specification 3.8.1 Required Action B.1 (one required Unit 3 DG inoperable) only, **THEN** **CHECK** at least TWO Unit 3 Offsite AC circuits are determined Operable in Attachment 1 and the Shutdown Boards included in the Operable circuits have proper voltage verified in Step 6.3[7]. (Otherwise N/A) A2 (AC) Expected Action(s): Determines Acceptance Criteria (AC) is met	**Critical Step** ___SAT ___UNSAT ___N/A
Examiner Note: **Pertinent information reviewed by the examinee is referenced above in the key**	
Step 2: No Technical Specification Required Action(s) must be taken Expected Action(s): Determines no Technical Specification Required Action(s) must be taken	**Critical Step** ___SAT ___UNSAT ___N/A
Examiner Note: **If Candidate asks about RICT, state "RICT does not apply".** **The cue sheet states that Tech Spec 3.8.1 Condition B has been entered for the 3A Diesel Generator.**	

STOP TIME _____

END OF TASK

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 Unit 3 NUSO.

- Breaker 1312, 4KV Unit BD 3A Normal Feeder Breaker is out of service for a breaker swap.
- 3A Diesel Generator was removed from service and declared INOPERBLE 15 minutes ago and Tech Spec 3.8.1 Condition B has been entered.
- Units 1, 2 and 3 are at 100% power.
- The Unit 3 BOP Operator has completed 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 kV Shutdown Boards for your review.

INITIATING CUES: Review 0-SR-3.8.1.A.1, Verification of Offsite Power Availability to 4.16 KV Shutdown Boards for Unit 3. The Unit 1 and 2 portions have already been completed.

- Determine if Acceptance Criteria (AC) is met
- Determine what Technical Specification Required Action(s) must be taken (if any)?

Answer:

_____
_____
_____
_____
_____
_____
_____
_____

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Place an RPS Channel in Trip
JPM NUMBER:	745-SRO	REVISION:	2

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	N/A		
K/A RATINGS:	SRO: 4.2		
K/A No. &STATEMENT:	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.		
RELATED PRA INFORMATION:	Risk Significant RPS Scram Reduction		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 745-SRO

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Determines Tech Spec 3.3.1.1 CONDITION A is NOT met, and the REQUIRED ACTION is to Place the channel in trip in 12 hours OR place the associated trip system in trip in 12 hours.
- Locates 2-PIS-3-22AA, Reactor High Pressure A1 Channel, on Print 2-730E915-9 **AND/OR** in 2-OI-99, Reactor Protection System, Attachment 3
- Determines that fuse 2-FU1-3-22AA (5A-F5A) is required to be removed

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-3 Operating the Plant with a Conservative Bias

REFERENCES/PROCEDURES NEEDED:

- Unit 2 TS 3.3.1.1, 2-OI-99, Prints 2-730E915-9, 2-45E671-26

VALIDATION TIME: 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
1	02/01/2022	ALL	Updated JPM task standard and initiating cue.
2	03/12/2021	All	Updated formatting

Procedure Revisions

Procedure	Revision
2-OI-99	101
Unit 2 TS 3.3.1.1	Amendment 351
2-730E915-9	30
2-45E671-26	5

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). The Instrument Mechanics conducted testing on 2-PIS-3-22AA, Reactor High Pressure A1 Channel, to support a surveillance. The instrument, 2-PIS-3-22AA, failed downscale.

INITIATING CUES: As the Nuclear Unit Senior Operator (NUSO), you are required to determine **ALL** of the following:

- What is/are the Technical Specification Condition(s) and Required Action(s) (if any)?
- List the plant procedure(s)/document(s) used to perform the Technical Specification(s) Required Action(s) (if any)?
- Identify how the Required Action(s) is/are performed in accordance with the respective plant procedure(s)/document(s) from above?

Answer:

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: Refers to Unit 2 Technical Specification 3.3.1.1, RPS Instrumentation	**Critical Step** _____ SAT _____UNSAT _____N/A

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 In accordance with the Risk Informed Completion Time Program
	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 OR In accordance with the Risk Informed Completion Time Program

(continued)

Examiner Note:
If Candidate asks about RICT, state "RICT does not apply".

STEP / STANDARD	SAT / UNSAT

Step 1 (cont):

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
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

Expected Action(s):

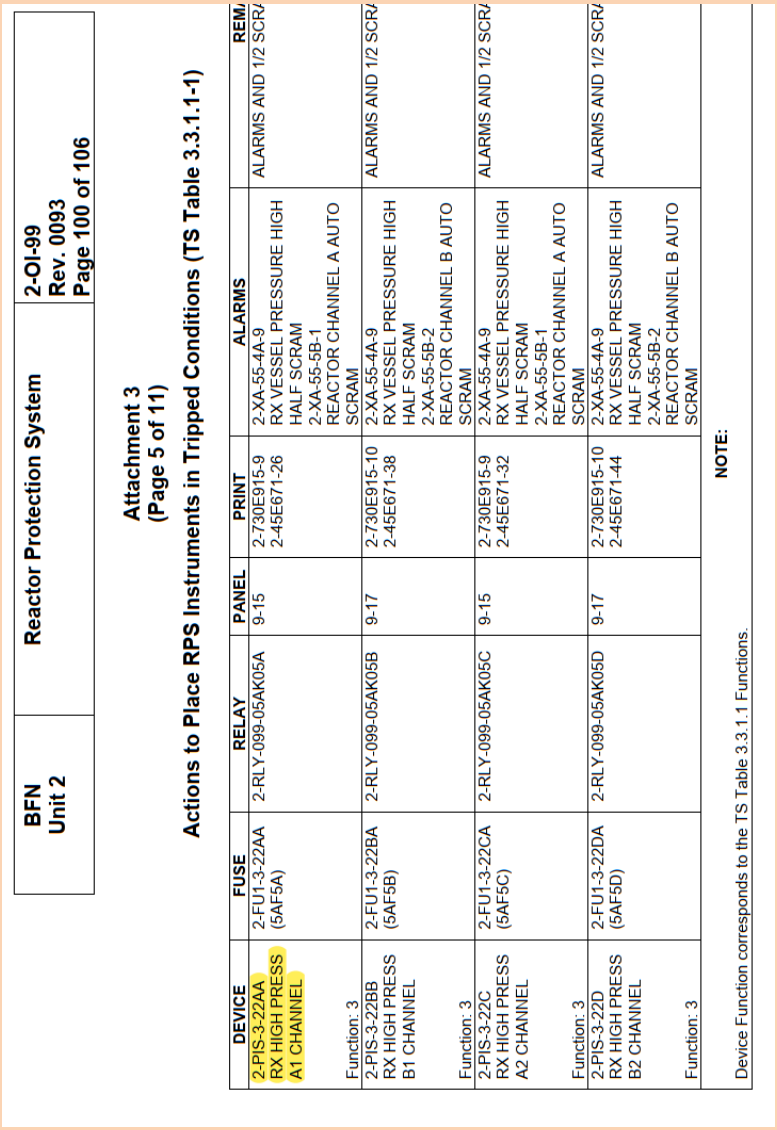
Determines Tech Spec 3.3.1.1 CONDITION A is NOT met, and the REQUIRED ACTION is to Place the channel in trip in 12 hours OR place the associated trip system in trip in 12 hours.

EXAMINER NOTE: The Examinee may elect to first refer to Unit 2 Tech Spec 3.3.1.1, RPS Instrumentation, and/or refer to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 13).

2-OI-99, Reactor Protection System, Attachment 3 lists the respective failed instruments (2-PIS-3-22AA, Reactor High Pressure A1 Channel) fuse, relay, prints and remarks/results.

If candidate elects to place the Channel in trip by other than pulling fuse, follow up questions may be warranted.

STEP / STANDARD	SAT / UNSAT
Step 2: Refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) **AND/OR** Print 2-730E915-9 (see next page) for 2-PIS-3-22AA, Reactor High Pressure A1 Channel.	**Critical Step** _____ SAT _____UNSAT _____N/A



BFN Unit 2 | Reactor Protection System | 2-OI-99 Rev. 0093 Page 100 of 106

Attachment 3
(Page 5 of 11)
Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

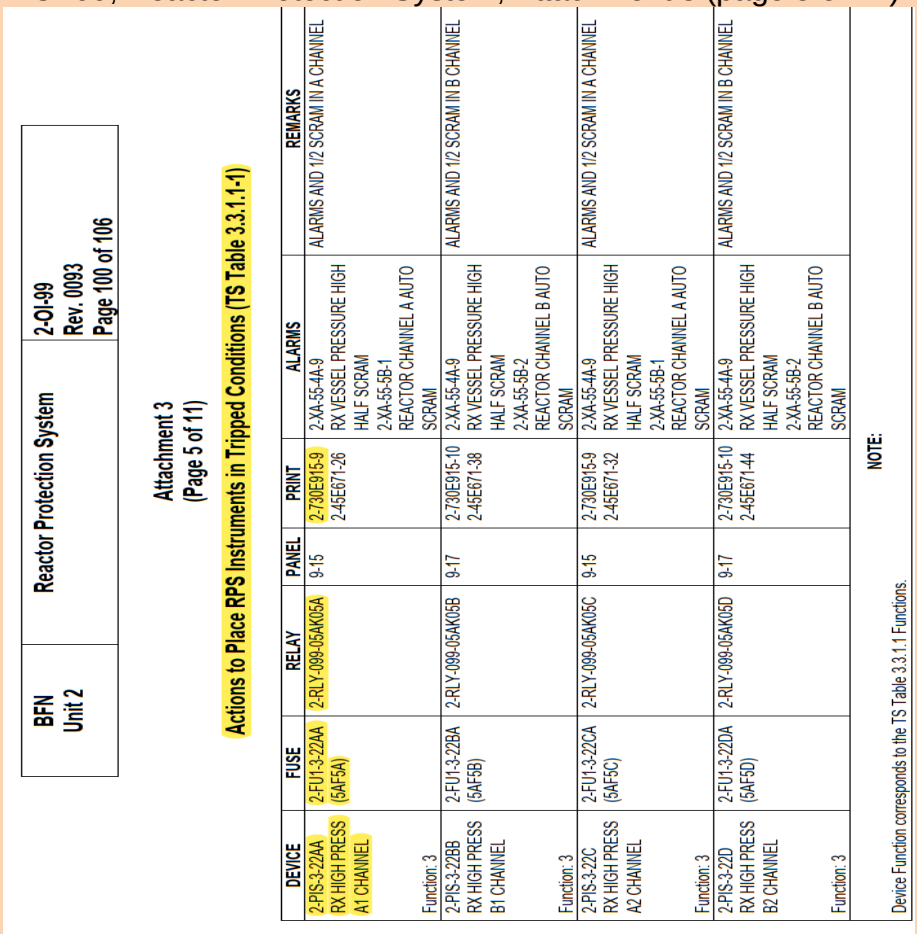
DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REM
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCR
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCR

NOTE:
Device Function corresponds to the TS Table 3.3.1.1 Functions.

Expected Action(s):

Examinee refers to 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9 (see next page) to reference the respective failed instrument 2-PIS-3-22AA, Reactor High Pressure A1 Channel.

STEP / STANDARD	SAT / UNSAT
Step 2 (continued): Print 2-730E915-9 (2-PIS-3-22AA is located between A-3 and E-3 coordinates).  	
EXAMINER NOTE: Hard copies of print 2-45E671-26 are available upon candidate request	

STEP / STANDARD	SAT / UNSAT
Step 3: Determine how the Required Action is performed to place 2-PIS-3-22AA, Reactor High Pressure A1 Channel, in trip in accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9. Expected Action(s): In accordance with 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) and/or Print 2-730E915-9, determines fuse **2-FU1-3-22AA (5A-F5A)** is required to be removed in order for 2-PIS-3-22AA, Reactor High Pressure A1 Channel, to be placed in trip as directed. 2-OI-99, Reactor Protection System, Attachment 3 (page 5 of 11) 	**Critical Step** _____ SAT _____UNSAT _____N/A

BFN Unit 2 | Reactor Protection System | 2-OI-99 Rev. 0093 Page 100 of 106

Attachment 3
(Page 5 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

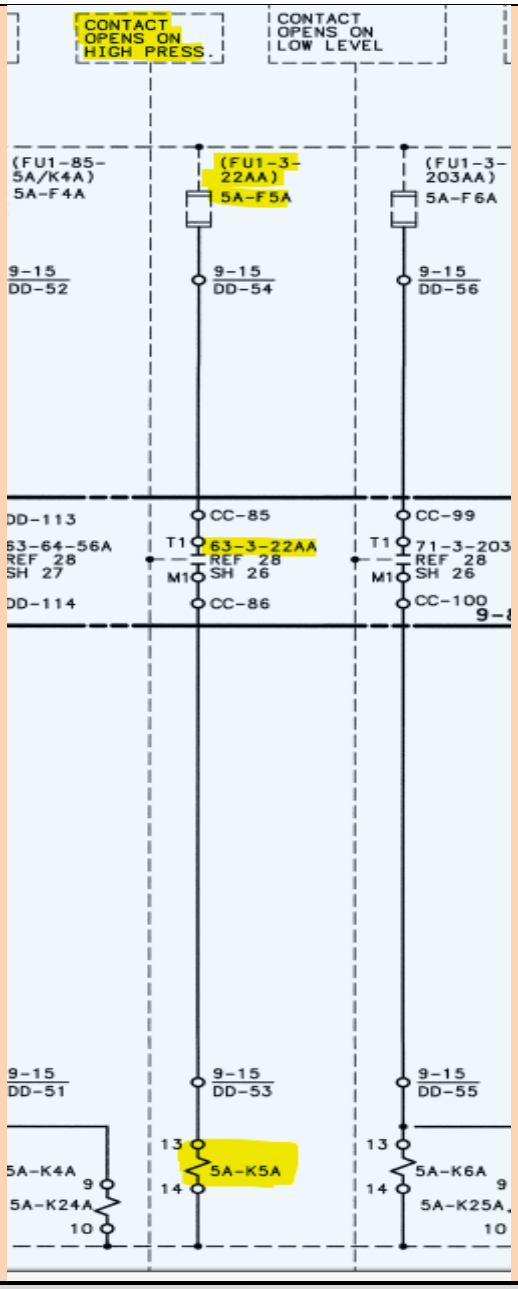
DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
2-PIS-3-22AA RX HIGH PRESS A1 CHANNEL Function: 3	2-FU1-3-22AA (5AF5A)	2-RLY-099-05AK05A	9-15	2-730E915-9 2-45E671-26	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22BB RX HIGH PRESS B1 CHANNEL Function: 3	2-FU1-3-22BA (5AF5B)	2-RLY-099-05AK05B	9-17	2-730E915-10 2-45E671-38	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL
2-PIS-3-22C RX HIGH PRESS A2 CHANNEL Function: 3	2-FU1-3-22CA (5AF5C)	2-RLY-099-05AK05C	9-15	2-730E915-9 2-45E671-32	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN A CHANNEL
2-PIS-3-22D RX HIGH PRESS B2 CHANNEL Function: 3	2-FU1-3-22DA (5AF5D)	2-RLY-099-05AK05D	9-17	2-730E915-10 2-45E671-44	2-XA-55-4A-9 RX VESSEL PRESSURE HIGH HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN B CHANNEL

NOTE:
Device Function corresponds to the TS Table 3.3.1.1 Functions.

STEP / STANDARD	SAT / UNSAT

Step 3 (continued):

Print 2-730E915-9



EXAMINER CUE: Once the Operator identifies that 2-FU1-3-22AA(5A-F5A) has to be pulled in accordance with 2-OI-99, Reactor Protection System, Attachment 3, and/or print 2-730E915-9 for the failed instrument (2-PIS-3-22AA, Reactor High Pressure A1 Channel), Inform the candidate "Another Operator will continue. 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:

Unit 2 is operating at 100% Rated Thermal Power (RTP). The Instrument Mechanics conducted testing on 2-PIS-3-22AA, Reactor High Pressure A1 Channel, to support a surveillance. The instrument, 2-PIS-3-22AA, failed downscale.

INITIATING CUES: As the Nuclear Unit Senior Operator (NUSO), you are required to determine **ALL** of the following:

- What is/are the Technical Specification Condition(s) and Required Action(s) (if any)?
- List the plant procedure(s)/document(s) used to perform the Technical Specification(s) Required Action(s) (if any)?
- Identify how the Required Action(s) is/are performed in accordance with the respective plant procedure(s)/document(s) from above?

Answer:

_____
_____
_____
_____
_____
_____
_____
_____
_____

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Review 3-SR-3.3.8.2.1(A), RPS Circuit Protector Calibration/Functional Test for 3A1 and 3A2
JPM NUMBER:	759-SRO	REVISION:	0

TASK APPLICABILITY:	☒SRO-U	☒SRO-I	☐UO
TASK NUMBER / TASK TITLE(S):	S-000-AD-10 Review Completed Surveillance Test Package		
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:	Risk Significant RPS Scram Reduction		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 759-SRO

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Determines that section 6.2.7 step [6] Acceptance Criteria is NOT met
- Determines Technical Specification (Tech Spec) 3.3.8.2 CONDITION A is NOT met and the REQUIRED ACTION is to Remove associated inservice power supply(s) from service in 72 hours

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-3 Operating the Plant with a Conservative Bias

REFERENCES/PROCEDURES NEEDED: 3-SR-3.3.8.2.1(A), Unit 3 TS 3.3.8.2

VALIDATION TIME: 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
1	01/04/24	All	Initial

Procedure Revisions

Procedure	Revision
3-SR-3.3.8.2.1(A)	18
Unit 3 TS 3.3.8.2	Amendment 253

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 3 is operating at 100% Rated Thermal Power (RTP). During the previous shift, Electrical Maintenance performed 3-SR-3.3.8.2.1(A), RPS Circuit Protector Calibration/Functional Test For 3A1 And 3A2. Unit 3 operators have placed the 3A1 and 3A2 RPS Circuit Protectors in service. You have been directed to relieve the Unit 3 Nuclear Unit Senior Operator (NUSO) and continue to perform an independent review of 3-SR-3.3.8.2.1(A).

INITIATING CUES: As the Unit 3 (NUSO), you are required to:

- Continue reviewing 3-SR-3.3.8.2.1(A) beginning at section 6.2.7, Time Delay Testing
- Determine Technical Specification or TRM Condition(s) and Required Action(s) (if any)

Answer:

KEY

(5) **RECORD** As Found time delay of TD1 relays.

As Found 3A1 TD1 (3-02-099-003A1TD1) Time Delay:
4.20 seconds — A2

As Found 3A2 TD1 (3-02-099-003A2TD1) Time Delay:
4.00 seconds — A2

(6) **CHECK** As Found 3A1 TD1 (3-02-099-003A1TD1) Time Delay less than or equal to 4.00 seconds — A2 (AC)

(7) **CHECK** As Found 3A2 TD1 (3-02-099-003A2TD1) Time Delay less than or equal to 4.00 seconds — A2 (AC)

(10) **RECORD** As Left 3A1 TD1 (3-02-099-003A1TD1) time delay.

As Left 3A1 TD1 (3-02-099-003A1TD1) Time Delay:
4.10 seconds — A2

(11) **RECORD** As Left 3A2 TD1 (3-02-099-003A2TD1) time delay.

As Left 3A2 TD1 (3-02-099-003A2TD1) Time Delay:
2.99 seconds — A2

BFN Unit 3	RPS Circuit Protector Calibration/Functional Test For 3A1 And 3A2	3-SR-3.3.8.2.1(A) Rev. 0018 Page 52 of 59

Date today

6.2.7 **Time Delay Testing (continued)**

(12) **ENSURE** As Left 3A1 TD1 (3-02-099-003A1TD1) Time Delay between 2.85 and 3.15 seconds. — A2 (AC)

(13) **ENSURE** As Left 3A2 TD1 (3-02-099-003A2TD1) time delay between 2.85 and 3.15 seconds. — A2 (AC)

KEY

LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each inservice RPS motor generator set or alternate power supply.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both inservice power supplies with one electric power monitoring assembly inoperable.	A.1 Remove associated inservice power supply(s) from service.	72 hours

SR 3.3.8.2.2 Perform CHANNEL CALIBRATION. The Allowable Values shall be:

a. Overvoltage ≤ 132 V, with time delay set to ≤ 4 seconds.

b. Undervoltage ≥ 108.5 V, with time delay set to ≤ 4 seconds.

c. Underfrequency ≥ 56 Hz, with time delay set to ≤ 4 seconds.

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: [6] CHECK As Found 3A1 TD1 (3-02-099-003A1TD1) Time Delay less than or equal to 4.00 seconds [7] CHECK As Found 3A2 TD1 (3-02-099-003A2TD1) Time Delay less than or equal to 4.00 seconds Expected Action(s): Determines that section 6.2.7 step [6] Acceptance Criteria is NOT met (Section 6.2.7 Step [5], As Found 3A1 TD1 (3-02-099-003A1TD1) Time Delay: 4.20 seconds)	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: The examinee may review the first 5 steps or may continue at the first Acceptance Criteria (AC) step of 3-SR-3.3.8.2.1	
Step 2: Refers to Unit 3 Tech Spec 3.3.8.2. Expected Action(s): Determines Tech Spec 3.3.8.2 CONDITION A is NOT met and the REQUIRED ACTION is to Remove associated inservice power supply(s) from service in 72 hours	**Critical Step** _____ SAT _____UNSAT _____N/A
Examiner Note: **If Candidate asks about RICT, state "RICT does not apply".**	
EXAMINER NOTE: The examinee may immediately enter Tech Spec 3.3.8.2 once section 6.2.7 step [6] is reviewed or continue to the end of section 6.2.7 before entering the Tech Spec.	
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:

Unit 3 is operating at 100% Rated Thermal Power (RTP). During the previous shift Electrical Maintenance performed 3-SR-3.3.8.2.1(A), RPS Circuit Protector Calibration/Functional Test For 3A1 And 3A2. Unit three operators have placed the 3A1 and 3A2 RPS Circuit Protectors in service. You have been directed to relieve the Unit 3 Nuclear Unit Senior Operator (NUSO) and continue to perform an independent review of 3-SR-3.3.8.2.1(A).

INITIATING CUES: As the unit 3 (NUSO), you are required to:

- Continue reviewing 3-SR-3.3.8.2.1(A) beginning at section 6.2.7, Time Delay Testing
- Determine Technical Specification or TRM Condition(s) and Required Action(s) (if any)

Answer:

_____
_____
_____
_____
_____
_____
_____
_____
_____
_____
_____

Job Performance Measure (JPM)

SITE:	BFN	JPM TITLE:	Emergency Action Level Classification
JPM NUMBER:	760-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. (SRO Only)		
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		

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 760-SRO

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Declares a GENERAL EMERGENCY – FG1
- Completes EPIP-5 Attachment 1, including:
  - Page 1 (Step 1) Selects Recommendation 3
  - Page 1 (Step 2) Wind Direction from 7 degrees
  - Page 2 (Step 3) Initiating Condition (IC) Designator – FG1
  - Page 2 (Step 4) Selects **NO** Radiological release
  - Page 2 (Step 6) Wind Direction and Wind Speed
  - Page 2 (Step 7) Evacuate 2 mile, **A2, B2, F2, G2** Radius and Evacuate 5 miles Downwind **E5, F5, G5**

Operator Fundamental evaluated:
OF-1 Monitoring Plant Indications and Conditions Closely

REFERENCES/PROCEDURES NEEDED: EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3
EPIPs 1, 2, 3, 4, and 5

VALIDATION TIME**:** 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

JPM Revision Summary

Rev No.	Effective Date	Pages Affected	Description
0	01/05/24	ALL	Initial issue

Procedure Revisions

Procedure	Revision
EPIP-1, Attachment 1, HOT INITIATING CONDTIONS-MODES 1-2-3	67
EPIP-1	67
EPIP-2	46
EPIP-3	50
EPIP-4	50
EPIP-5	65

EXAMINER NOTE:
Initially handout each 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, 2 and 3 are operating at 100% Reactor Power.

Continue the INITIAL CONDITIONS on the following page:

This JPM is TIME CRITICAL

The following event quickly develops on Unit 3:

- Unit 3 SCRAMS
- All MSIVs are closed
- All ADS Valves are open
- Reactor water level indicates (-)110" and is slowly lowering
- Reactor pressure indicates 65 psig
- Electrical failures result in a loss of all Suppression Pool Cooling and Containment Spray Subsystems
- Suppression Pool temperature indicates 250°F
- Suppression Pool level indicates 16 feet
- Primary Containment Pressure indicates 56 psig
- Drywell Radiation monitors are as follows:
  - 3-RM-90-272A = 3265 R/hr and slowly rising
  - 3-RM-90-273A = 3200 R/hr and slowly rising
- Chemistry has not completed a Reactor coolant sample
- Dose assessment at site boundary is in progress
- Primary Containment Venting (EOI Appendix 13) could not be aligned
- 91 meter Met Tower Data:
  - 15 minute average Wind Direction is from 7 degrees.
  - 15 minute average Wind Speed is 2 mph.

NOTE:

- **NO** previous Events have been classified; therefore, Emergency Facilities (CECC, TSC, OSC) are **NOT** currently 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

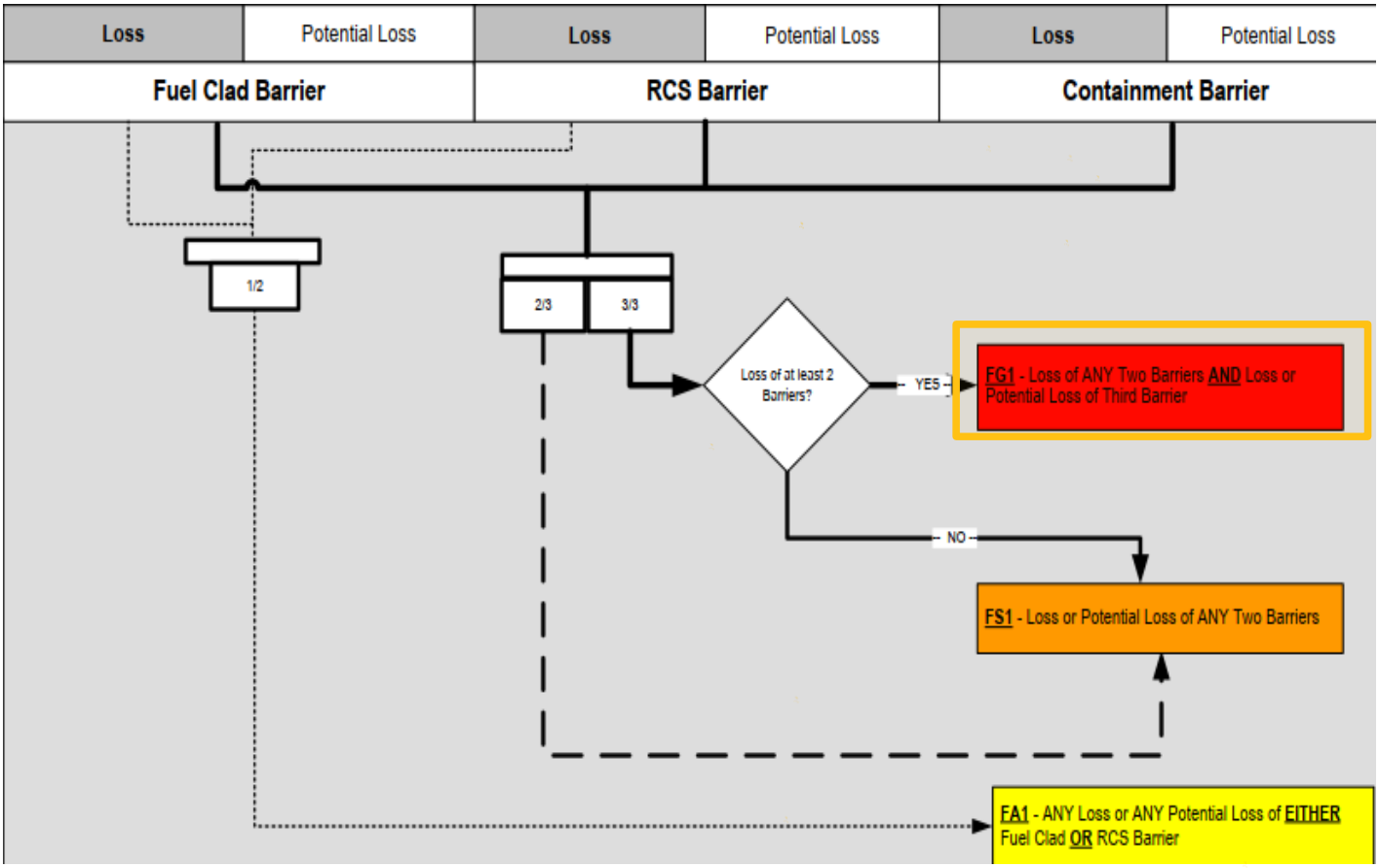
KEY

FG1: **1.** Loss of Fuel Clad Barrier 4. A;
2. Loss of RCS Barrier 4. A; OR (Loss of RCS Barrier may be determined due to 1.A)
3. Potential Loss of Containment Barrier 1. A

A. **ANY** primary containment radiation monitor reading greater than:	A. **ANY** primary containment radiation monitor reading greater than:
UNIT 1 1-RM-90-272A >3.17E+03 R/HR	**UNIT 1** 1-RM-90-272A >3.38E+01 R/HR
1-RM-90-273A >3.15E+03 R/HR	1-RM-90-273A >3.36E+01 R/HR
UNIT 2 2-RM-90-272A >2.42E+03 R/HR	**UNIT 2** 2-RM-90-272A >2.58E+01 R/HR
2-RM-90-273A >3.15E+03 R/HR	2-RM-90-273A >3.36E+01 R/HR
UNIT 3 3-RM-90-272A >3.17E+03 R/HR	**UNIT 3** 3-RM-90-272A >3.38E+01 R/HR
3-RM-90-273A >3.15E+03 R/HR	3-RM-90-273A >3.36E+01 R/HR

A. Drywell pressure greater than 55 psig.
OR
B. H_2 greater than 6% AND O_2 greater than 5% exists inside primary containment
OR
C. HCTL exceeded.

A. Drywell pressure at or above 2.45 psig due to RPV leakage.



KEY

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Attachment 1
(Page 1 of 4)

General Emergency Initial Notification Form

Protective Action Recommendation (PAR) Worksheet

1. Determine the correct PAR using the PAR diagram (Attachment 8)

☐ **Recommendation 1**
- Shelter 2 mile radius
- Shelter 5 miles downwind

☐ **Recommendation 2**
- Shelter 5 mile radius

X **Recommendation 3**
- Evacuate 2 mile radius
- Evacuate 5 miles downwind

☐ **Recommendation 4**
- Evacuate 2 mile radius
- Evacuate 5 miles downwind
- Shelter 10 miles downwind

☐ **Recommendation 5**
- Evacuate 2 mile radius and all sectors greater than Protective Action Guidelines (PAG)

2. Identify the affected sectors using the table below (15 minute average)

Wind direction From (91 meter): 7 degrees

2 Mile Radius Sectors	5 Mile Radius Sectors	5 Mile Downwind Sectors	Wind Direction From	10 Mile Downwind Sectors
A2, B2, F2, G2	A2, B2, F2, G2, A5, B5, E5, F5, G5	E5, F5, G5	4°-40°	E10, F10, G10
		F5, G5	41°-73°	F10, G10, H10
		G5	74°-92°	G10, H10, I10
		A5, G5	93°-137°	H10, I10, J10, K10
		A5	138°-203°	A10, I10, J10, K10
		A5, B5	204°-282°	A10, B10
		B5, E5	283°-326°	B10, C10, D10, E10
		E5, F5	327°-3°	C10, D10, E10, F10

3. Use the above information to complete page 2 of this attachment.

KEY

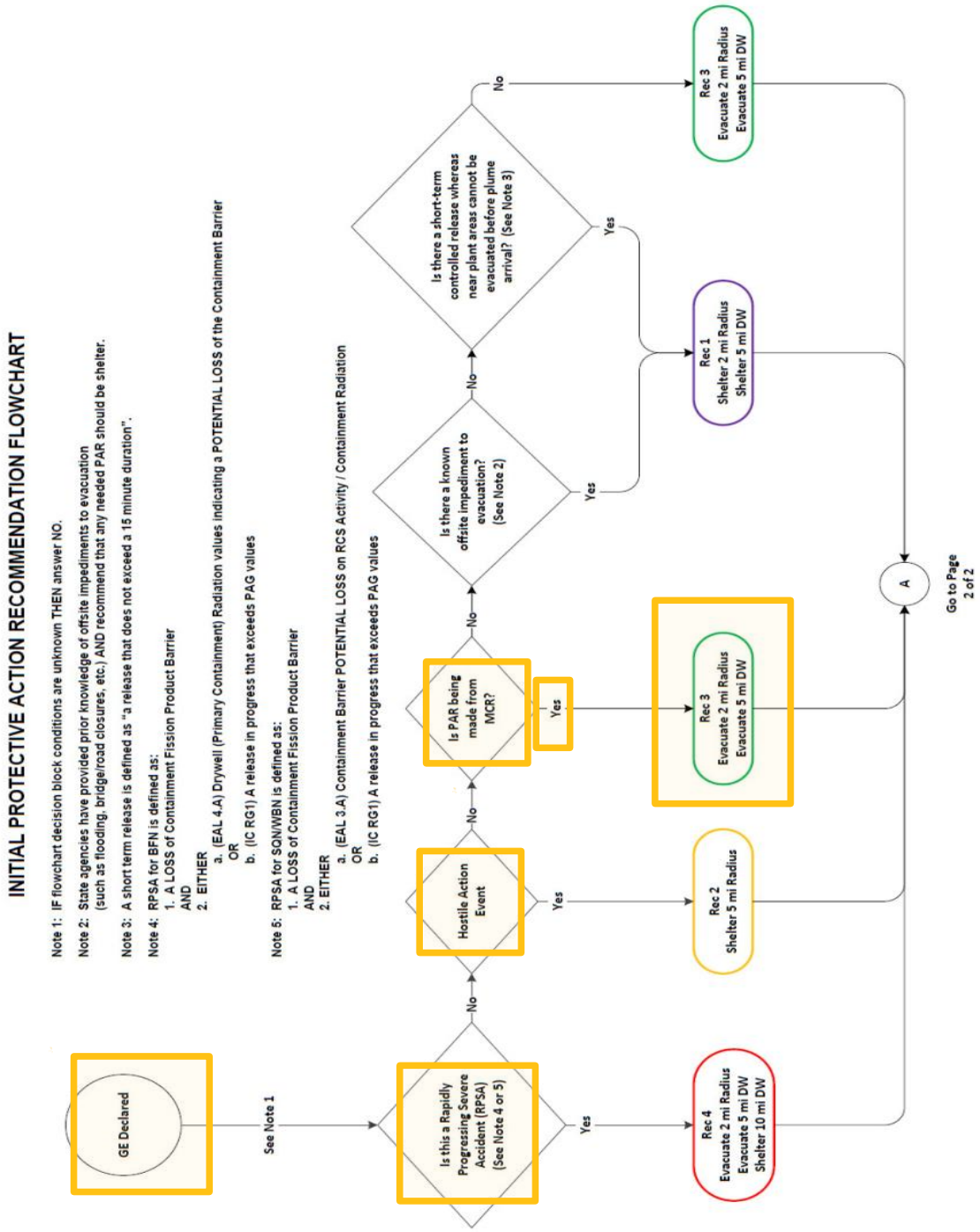
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Attachment 1
(Page 2 of 4)

General Emergency Initial Notification Form

1. [X] 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 a **GENERAL EMERGENCY**											
3. **Initiating Condition (IC) Designator:** EG1 (Use only one IC)											
4. Has a Radiological release occurred or is in progress attributable to the event? [X] NO ☐ YES, AIRBORNE Radiological Release ☐ YES, LIQUID Radiological Release											
5. **Event Declared:** Time within 15 minutes (Central Time) Date: todays date											
6. **The Meteorological Conditions are:** (Use 91 meter data from the Met Tower. If data is not available from the MET tower, contact the National Weather Service by dialing 1-256-890-8507 or 1-205-621-5650. The National Weather Service will provide wind direction and wind speed.) Wind Direction is FROM: 7 degrees (15 minute average) Wind Speed: 2 mph (15 minute average)											
7. Provide Protective Action Recommendation utilizing Attachment 8: (Check all Sectors as appropriate)											
A) AFFECTED SECTORS											
Evacuate											
2 Mile	[X] A2	[X] B2	[X] F2	[X] G2							
5 Mile	☐ A5	☐ B5	[X] E5	[X] F5	[X] G5						
10 Mile	☐ A10	☐ B10	☐ C10	☐ D10	☐ E10	☐ F10	☐ G10	☐ H10	☐ I10	☐ J10	☐ K10
Shelter (do not check sectors evacuated above)											
2 Mile	☐ A2	☐ B2	☐ F2	☐ G2							
5 Mile	☐ A5	☐ B5	☐ E5	☐ F5	☐ G5						
10 Mile	☐ A10	☐ B10	☐ C10	☐ D10	☐ E10	☐ F10	☐ G10	☐ H10	☐ I10	☐ J10	☐ K10
B) ALL OTHER SECTORS MONITOR AND PREPARE.											
C) CONSIDER ISSUANCE OF KI IN ACCORDANCE WITH THE STATE PLAN.											
Completed by (SED): _____ **Peer Reviewed by:** _____											

KEY



From EPIP-5 Attachment 8 (Page 1 of 2)

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.**	
Step 1: Classifies the Event using EPIP-1, Emergency Classification Procedure **3.1 Precautions/Limitations** (page 5 of 142) 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 **GENERAL EMERGENCY – FG1** – Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier **TIME CLASSIFICATION COMPLETE:** _____	**Critical Step** _____ SAT _____UNSAT _____N/A

STEP / STANDARD	SAT / UNSAT
EXAMINER CUE: **Direct Examinee to complete the next Time Critical portion of the JPM for the appropriate INITIAL NOTIFICATION using appropriate EPIP 2, EPIP 3, EPIP 4 or EPIP 5, respectively (handouts already provided), then IMMEDIATELY bring completed form(s), to the Examiner.** **Record the 2nd TIME START for the Examinee's last 15 minute required time in JPM Step 2 below.**	
Step 2: **TIME START** _____ Implements EPIP-5, GENERAL EMERGENCY. Expected Action(s): Examinee implements EPIP-5, GENERAL EMERGENCY.	____ SAT ____UNSAT ____N/A

STEP / STANDARD	SAT / UNSAT
Step 3:	

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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 the classification declaration) and Section 3.5 (as soon as possible, not to exceed 60 minutes from classification declaration) are time critical.
- A Shift Technical Advisor or Senior Reactor Operator (SRO) should peer review Vaporstream Notification or 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.
- Step 3.1[1.1], Attachment 1, Step 7 and Attachment 7, Steps 1.1[1],and 1.1[5.1.1] CANNOT be delegated.

[1] **WHEN** the Technical Support Center (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-5, GENERAL EMERGENCY, because the TSC has not yet been staffed as given in the Initial Conditions.

STEP / STANDARD	SAT / UNSAT
Step 4: **3.1 BFN Risk Counties and State of Alabama Notification** **NOTE** Notification of the Risk Counties/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 GENERAL EMERGENCY Event Classification:_____ Expected Action(s): Examinee enters the time of GENERAL 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 step. 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: [2.1.1] COMPLETE Attachment 1, "General 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-5, Attachment 1 to the Examiner. The following items are **Critical Steps** on Attachment 1: • Page 1 (Step 1) Selects **Recommendation 3** • Page 1 (Step 2) Wind Direction **from 7 degrees** • Page 2 (Step 3) Initiating Condition (IC) Designator **– FG1** • Page 2 (Step 4) Selects **NO Radiological Release** • Page 2 (Step 6) Wind Direction from **7 degrees** and Wind Speed **2 mph** • Page 2 (Step 7) Evacuate 2 mile, **A2, B2, F2, G2** Radius and Evacuate 5 miles Downwind **E5, F5, G5**	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: The following are not Critical Steps on Attachment 1; however, the Examinee may elect to complete them: Page 2 (Step 1) – This is a Drill, Page 2 (Step 5) - Time/Date and/or Page 2 Step (7) – Signature.	
Record the 2nd STOP TIME for the Examinee's second 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:_____

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, 2 and 3 are operating at 100% Reactor Power.

Continue the INITIAL CONDITIONS on the following page:

This JPM is TIME CRITICAL

Job Performance Measure (JPM)

The following event quickly develops on Unit 3:

- Unit 3 SCRAMS
- All MSIVs are closed
- All ADS Valves are open
- Reactor water level indicates (-)110” and is slowly lowering
- Reactor pressure indicates 65 psig
- Electrical failures result in a loss of all Suppression Pool Cooling and Containment Spray Subsystems
- Suppression Pool temperature indicates 250°F
- Suppression Pool level indicates 16 feet
- Primary Containment Pressure indicates 56 psig
- Drywell Radiation monitors are as follows:
    - 3-RM-90-272A = 3265 R/hr and slowly rising
    - 3-RM-90-273A = 3200 R/hr and slowly rising
- Chemistry has not completed a Reactor coolant sample
- Dose assessment at site boundary is in progress
- Primary Containment Venting (EOI Appendix 13) could not be aligned
- 91 meter Met Tower Data:
    - 15 minute average Wind Direction is from 7 degrees.
    - 15 minute average Wind Speed is 2 mph.

NOTE:

- **NO** previous Events have been classified; therefore, Emergency Facilities (CECC, TSC, OSC) are **NOT** currently 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 Radiation Exposure Limits Under Emergency Conditions
JPM NUMBER:	732-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: 3.8		
K/A No. &STATEMENT:	2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits		
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	

Job Performance Measure (JPM)

OPERATOR: _____ JPM Number: 732-SRO

DATE: _____

TASK STANDARD: The task is satisfactorily met when the candidate:

- Calculates that plant worker will receive 25.5 REM
- Determines that Emergency Dose **CAN** be authorized

Operator Fundamental evaluated:
OF-1 Monitoring plant indications and conditions closely.
OF-3 Operating the Plant with a Conservative Bias

REFERENCES/PROCEDURES NEEDED: EPIP-15

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
1	08/22/17	All	Converted JPM to new format
2	10/02/19	All	Updated JPM
3	01/05/23	All	Updated formatting

Procedure Revisions

Procedure	Revision
EPIP-15	13

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:

A General Emergency has been declared on Unit 2. You are the Shift Manager, and the Technical Support Center (TSC) has **NOT** yet been staffed.

An Auxiliary Unit Operator (AUO), Brian Smith, has volunteered to go into the Unit 2 Reactor Building and manually close 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE, to prevent a large off-site release and extensive exposures to large populations. The Radiation Protection Supervisor has briefed you on the following information:

- The travel path dose rate is 3 REM/hr to the valve in question
- Dose rate is 50 REM/hr at 2-FCV-73-3
- It is estimated that it will take Brian 5 minutes to travel (each way) to and from the valve
- It is estimated that it will take Brian 30 minutes to close the valve
- Brian currently has zero dose to date with NO previous emergency dose received
- Brian has been briefed on the radiological hazards associated with this task

.

INITIATING CUE:

Calculate the dose that the plant worker would receive for performing this task.

Determine if authorization CAN or CANNOT be given.

Answer:

_____
_____
_____
_____
_____
_____
_____
_____

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.

3.5 Post-Exposure Activities

A. Personnel receiving emergency doses should be restricted from further occupational exposures pending the outcome of exposures evaluations, and if necessary, medical surveillance.

B. RP shall conduct post exposure dose assessments for exposed individuals, with particular attention to determining the adequacy of administrative dosimeter correction factors for TEDE doses resulting from internal and external exposures.

C. Any exposures above 5 rem TEDE shall be reported to a TVA physician. It is the responsibility of the physician to determine appropriate medical evaluations and required care. Cross reference guidance is found in EPIP-10.

D. Any emergency exposure shall be reported to the RP Manager as soon as possible.

KEY

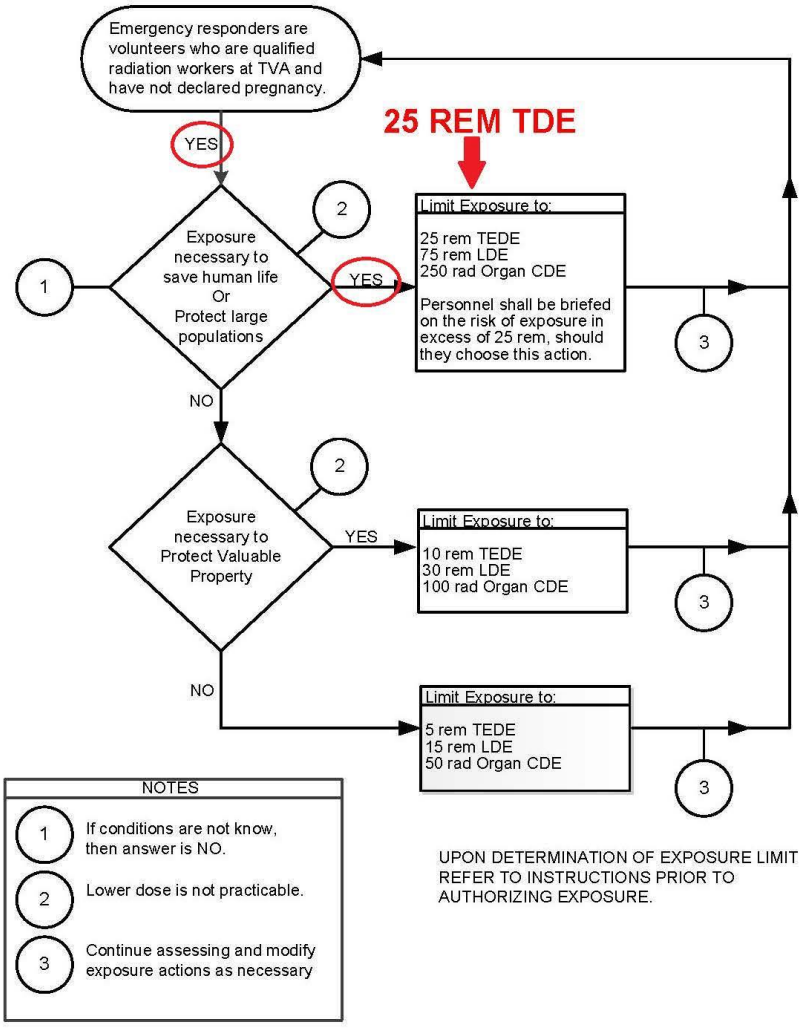
KEY

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Appendix B
(Page 2 of 2)

Acknowledgement and Authorization to Exceed Occupational Dose Limits

Emergency Exposure Reference Guide



KEY

START TIME:_____

STEP / STANDARD	SAT / UNSAT
Step 1: Calculate the dose that the plant worker will receive. Expected Action(s): Calculate that the plant worker will receive 25.5 REM from this task. (Travel: (10 Minutes) *(3 R/HR)/60 minutes) = 0.5 REM (Valve Operation: (30 minutes) *(50 R/HR) / (60 minutes)) = 25 REM 25 REM + 0.5 REM = 25.5 REM	**Critical Step** _____ SAT _____UNSAT _____N/A
EXAMINER NOTE: The Critical Step is to determine that Brian Smith will receive 25-26 REM from the task.	
Step 2: Refers to EPIP 15, Emergency Exposure to determine if authorization **CAN or CANNOT** be given. Expected Action(s): Determines that Emergency Dose **CAN** be authorized.	**Critical Step** ____ SAT ____UNSAT ____N/A
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:

A General Emergency has been declared on Unit 2. You are the Shift Manager, and the Technical Support Center (TSC) has **NOT** yet been staffed.

An Assistant Unit Operator (AUO), Brian Smith, has volunteered to go into the Unit 2 Reactor Building and manually close 2-FCV-73-3, HPCI STEAM LINE OUTBOARD ISOLATION VALVE, to prevent a large off-site release and extensive exposures to large populations. The Radiation Protection Supervisor has briefed you on the following information:

- The travel path dose rate is 3 REM/hr to the valve in question
- Dose rate is 50 REM/hr at 2-FCV-73-3
- It is estimated that it will take Brian 5 minutes to travel (each way) to and from the valve
- It is estimated that it will take Brian 30 minutes to close the valve
- Brian currently has zero dose to date with NO previous emergency dose received
- Brian has been briefed on the radiological hazards associated with this task

.

INITIATING CUE:

Calculate the dose that the plant worker would receive for performing this task.

Determine if authorization CAN or CANNOT be given.

Answer:

_____
_____
_____
_____
_____
_____
_____