Facility: Waterford **K/A Catalog Rev. 3** **Rev.** dd/mm/yyyy **Date of Exam:** 8/30/2023

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

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

Tier	Reactor Theory	Thermodynamics	Total
4. Theory	3	3	6

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

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

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

ES-4.1-PWR Waterford
Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/**SRO**)

Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
1	(000007) (EPE 7; BW E02 & E10; CE E02) Reactor Trip, Stabilization, Recovery						X	(000007) (EPE 7; BW E02 & E10; CE E02) Reactor Trip, Stabilization, Recovery (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)	3.5	1
2	(000008) (APE 8) Pressurizer Vapor Space Accident		X					(000008AK2.07) Knowledge of the relationship between (APE 8) PRESSURIZER VAPOR Space Accident and the following systems or components (CFR: 41.8 / 41.10 / 45.3): PZR LCS	3.6	2
3	(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer						**X**	**(040) (CE E05)** EXCESS STEAM DEMAND **(G2.1.9) CONDUCT OF OPERATIONS: Ability to direct licensed personnel activities inside the control room(CFR: 43.1 / 45.5 / 45.12 / 45.13)$**	**4.5**	**76**
4	(000009) (EPE 9) Small Break LOCA				X			(000009EA1.07) Ability to operate and/or monitor the following as they apply to (EPE 9) SMALL-Break LOCA (CFR: 41.5 / 41.7 / 45.5 to 45.8): CCS	3.5	3
5	(000011) (EPE 11) Large Break LOCA		X					(000011EK2.03) Knowledge of the relationship between (EPE 11) LARGE-Break LOCA and the following systems or components (CFR: 41.8 / 41.10 / 45.3): RCS	4.3	4
6	(000015) (APE 15) Reactor Coolant Pump Malfunctions	X						(000015AK1.02) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 15) REACTOR COOLANT Pump Malfunctions (CFR: 41.5 / 41.7 / 45.7 / 45.8): Consequences of an RCP failure$	3.9	5
7	**(000022) (APE 22) Loss of Reactor Coolant Makeup**					**X**		**(000022AA2.02) Ability to determine and/or interpret the following as they apply to (APE 22) LOSS OF REACTOR Coolant Makeup (CFR: 41.10 / 43.5 / 45.13): Charging pump problems$**	**3.7**	**77**
8	**(000025) (APE 25) Loss of Residual Heat Removal System**						**X**	**(000025) (APE 25) Loss of Residual Heat Removal System (G2.4.50) EMERGENCY PROCEDURES/PLAN: Ability to verify system alarm setpoints and operate controls identified in the alarm response procedure (CFR: 41.10 / 43.5 / 45.3)**	**4.0**	**78**
9	(000026) (APE 26) Loss of Component Cooling Water			X				(000026AK3.08) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 26) LOSS OF Component Cooling Water (CFR: 41.5 / 41.10 / 45.6 / 45.13): Verifying CCW adequate surge tank level	3.4	6
10	(000027) (APE 27) Pressurizer Pressure Control System Malfunction				X			(000027AA1.01) Ability to operate and/or monitor the following as they apply to (APE 27) PRESSURIZER PRESSURE Control System Malfunction (CFR: 41.5 / 41.7 / 45.5 to 45.8): PZR heaters, sprays, and PORVs	3.8	7
11	(000029) (EPE 29) Anticipated Transient Without Scram	X						(000029EK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (EPE 29) ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) (CFR: 41.5 / 41.7 / 45.7 / 45.8): Addition of negative reactivity$	4.2	8
12	(000038) (EPE 38) Steam Generator Tube Rupture					X		(000038EA2.15) Ability to determine and/or interpret the following as they apply to (EPE 38) STEAM GENERATOR Tube Rupture (CFR: 41.10 / 43.5 / 45.13): RCS pressure	4.0	9

13	(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer			X				(CE05EK3.07) Knowledge of the reasons for the following responses and/or actions as they apply to (CE E05) EXCESS STEAM DEMAND (CFR: 41.5 / 41.10 / 45.6 / 45.13): Operating RCPs within operating limits	3.7	10
14	(000054) (APE 54; CE E06) Loss of Main Feedwater						X	(000054) (APE 54; CE E06) Loss of Main Feedwater (G2.1.20) CONDUCT OF OPERATIONS: Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)$	4.6	11
15	(000055) (EPE 55) Station Blackout					X		(000055EA2.01) Ability to determine and/or interpret the following as they apply to (EPE 55) Station Blackout (CFR: 41.10 / 43.5 / 45.13): Existing valve positioning	3.5	12
16	(000056) (APE 56) Loss of Offsite Power	X						(000056AK1.07) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 56) Loss of Offsite Power (CFR: 41.5 / 41.7 / 45.7 / 45.8): Long term core cooling	4.0	13
17	**(000056) (APE 56) Loss of Offsite Power**						**X**	**(000056) (APE 56) Loss of Offsite Power (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**	**79**
18	(000057) (APE 57) Loss of Vital AC Instrument Bus				X			(000057AA1.06) Ability to operate and/or monitor the following as they apply to (APE 57) LOSS OF VITAL AC ELECTRICALINSTRUMENT BUS (CFR: 41.5 / 41.7 / 45.5 to 45.8): Manual control of components for which automatic control is lost	3.8	14
19	(000058) (APE 58) Loss of DC Power						X	(000058) (APE 58) Loss of DC Power (G2.1.19) CONDUCT OF OPERATIONS: Ability to use available indications to evaluate system or component status (CFR: 41.10 / 45.12)	3.9	15
20	**(000058) (APE 58) Loss of DC Power**					**X**		**(000058AA2.03 Ability to determine and/or interpret the following as they apply to (APE 58) LOSS OF DC Power (CFR: 41.10 / 43.5 / 45.13): Impact on ability to operate and monitor plant systems**	4.0	**80**
21	(000062) (APE 62) Loss of Nuclear Service Water		X					(000062AK2.09) Knowledge of the relationship between (APE 62) LOSS OF SERVICE WATER and the following systems or components (CFR: 41.8 / 41.10 / 45.3): AFW System$	3.3	16
22	(000065) (APE 65) Loss of Instrument Air			X				(000065AK3.03) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 65) LOSS OF Instrument Air (CFR: 41.5 / 41.10 / 45.6 / 45.13): Knowing effects on plant operation of isolating certain equipment from instrument air	3.6	17
23	(000077) (APE 77) Generator Voltage and Electric Grid Disturbances					X		(000077AA2.04) Ability to determine and/or interpret the following as they apply to (APE 77) GENERATOR VOLTAGE AND ELECTRIC Grid Disturbances (CFR: 41.10 / 43.5 / 45.13): VAR$	3.6	18
24	**(000077) (APE 77) Generator Voltage and Electric Grid Disturbances**					**X**		**(000077AA2.05) Ability to determine and/or interpret the following as they apply to (APE 77) GENERATOR VOLTAGE AND ELECTRIC Grid Disturbances (CFR: 41.5 / 43.5 / 45.5 / 45.7 / 45.8): Status of grid$**	3.6	**81**
	(W E04) LOCA Outside Containment / 3									
	(W E11) Loss of Emergency Coolant Recirculation / 4									

		K1	K2	K3	A1	A2	G*			
	(BW E04; W E05) Inadequate Heat Transfer – Loss of Secondary Heat Sink / 4									
K/A Category Totals:		3	3	3	3	6	6	Group Point Total:		24

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

Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
25	(000001) (APE 1) Continuous Rod Withdrawal						X	(000001) (APE 1) Continuous Rod Withdrawal (G2.4.12) Knowledge of operating crew responsibilities during emergency and abnormal operations (CFR: 41.10 / 45.12)$	4.0	19
26	(000003) (APE 3) Dropped Control Rod					X		(000003AA2.09) Ability to determine and/or interpret the following as they apply to (APE 3) DROPPED Control Rod (CFR: 41.10 / 43.5 / 45.13): Reactor Power$	3.8	20
	000005 (APE 5) Inoperable/Stuck Control Rod / 1									
32	000024 (APE 24) Emergency Boration / 1					X		**(000024AA2.05) Ability to determine and/or interpret the following as they apply to (024) Emergency Boration (CFR: 43.5 / 45.13): Amount of boric acid to achieve required SDM$**	3.9	84
27	**(000028) (APE 28) Pressurizer (PZR) Level Control Malfunction**					**X**		**(000028AA2.06) Ability to determine and/or interpret the following as they apply to (APE 28) PRESSURIZER (PZR) Level Control Malfunction (CFR: 41.10 / 43.5 / 45.13): Letdown flow**	**3.6**	**82**
28	(000032) (APE 32) Loss of Source Range Nuclear Instrumentation			X				(000032AK3.02) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 32) LOSS OF SOURCE RANGE Nuclear Instrumentation (CFR: 41.5 / 41.10 / 45.6 / 45.13): Guidance contained in procedures for loss of source range NI$	3.5	21
29	(000033) (APE 33) Loss of Intermediate Range Nuclear Instrumentation				X			(000033AA1.02) Ability to operate and/or monitor the following as they apply to (APE 33) LOSS OF INTERMEDIATE RANGE Nuclear Instrumentation (CFR: 41.5 / 41.7 / 45.5 to 45.8): Level trip bypass$	3.3	22
	000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8									
30	(000037) (APE 37) Steam Generator Tube Leak						X	(000037) (APE 37) Steam Generator Tube Leak (G2.2.22) EQUIPMENT CONTROL: Knowledge of limiting condition for operation and safety limits (CFR: 41.5 / 43.2 / 45.2)$	4.0	23
	000051 (APE 51) Loss of Condenser Vacuum / 4									
	000059 (APE 59) Accidental Liquid Radwaste Release / 9									

	000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
	000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
31	**(000067) (APE 67) Plant Fire On Site**						**X**	**(000067) (APE 67) Plant Fire On Site (G2.1.8) CONDUCT OF OPERATIONS: Ability to coordinate activities outside the control room (CFR: 41.10 /43.1 / 45.5/ 45.12 /45.13)$**	4.1	**83**

	(000068) (APE 68; BW A06) Control Room Evacuation									
	000069 (APE 69; W E14) Loss of Containment Integrity / 5									
33	(000074) (EPE 74; W E06 & E07) Inadequate Core Cooling		X					(000074EK2.07) Knowledge of the relationship between (EPE 74) Inadequate Core Cooling and the following systems or components (CFR: 41.8 / 41.10 / 45.3): Main Steam system$	3.4	24
	(000076) (APE 76) High Reactor Coolant Activity									
	000078 (APE 78*) RCS Leak / 3									
	(W E01 & E02) Rediagnosis & SI Termination / 3									
	(W E13) Steam Generator Overpressure / 4									
	(W E15) Containment Flooding / 5									
	(W E16) High Containment Radiation /9									
	(BW A01) Plant Runback / 1									
	(BW A02 & A03) Loss of NNI-X/Y/7									
	(BW A04) Turbine Trip / 4									
	(BW A05) Emergency Diesel Actuation / 6									
	(BW A07) Flooding / 8									
	(BW E03) Inadequate Subcooling Margin / 4									
	(BW E08; W E03) LOCA Cooldown – Depressurization / 4									
	(BW E09; CE A13**; W E09 & E10) Natural Circulation/4									
	(BW E13 & E14) EOP Rules and Enclosures									
	(CE A11**; W E08) RCS Overcooling – Pressurized Thermal Shock / 4									

35	(CE A16) Excess RCS Leakage	X						(CA16AK1.06) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (CE A16) EXCESS RCS Leakage (CFR: 41.5 / 41.7 / 45.7 / 45.8): Methods for quantifying an RCS leak$	3.8	25
34	**(CE E09) Functional Recovery**						X	**(CE E09) Functional Recovery (G2.4.22) EMERGENCY PROCEDURES/PLAN: Knowledge of the basis for prioritizing safety functions during abnormal and emergency operations (CFR: 41.7 / 41.10 / 43.5 to 45.12$**	**4.4**	**85**
36	(CE E13*) Loss of Forced Circulation/LOOP/Blackout				X			(CE13EA1.14) Ability to operate and/or monitor the following as they apply to (CE E13) LOSS OF FORCED CIRCULATION AND/OR LOOP AND/OR A BLACKOUT (CFR: 41.5 / 41.7 / 45.5 to 45.8): MFW system	2.7	26
K/A Category Totals:		1	1	1	2	3	4	Group Point Total:		12

Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
37	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM							X					(003A1.05) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4P RCP) REACTOR COOLANT PUMP SYSTEM, including (CFR: 41.5 / 45.5): RCS flow	3.4	27
38	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM											X	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM (G2.1.28) CONDUCT OF OPERATIONS: Knowledge of the purpose and function of major system components and controls (CFR: 41.7)	4.1	28
39	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM								X				(004A2.22) Ability to (a) predict the impacts of the following on the (SF1; SF2 CVCS) CHEMICAL AND VOLUME 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): Mismatch of letdown and charging flows$	3.4	29
40	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM						X						(004K6.07) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM (CFR: 41.7 / 45.7): Regenerative and nonregenerative heat exchangers	3.4	30
41	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM						X						(005K6.16) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM (CFR: 41.7 / 45.7): Injection and/or recirculation valves	3.9	31
42	**(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM**								**X**				**(005A2.02) Ability to (a) predict the impacts of the following on the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Pressure transient protection during cold shutdown**	**3.9**	**86**

43	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM								X				(006A2.02) Ability to (a) predict the impacts of the following on the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Loss of flowpath$	3.9	32
44	(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM								X				(007A2.02) Ability to (a) predict the impacts of the following on the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK 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): Abnormal pressure in the PRT/quench tank	3.6	33
45	(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM									X			(007A3.01) Ability to monitor automatic features of the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM, including (CFR: 41.7 / 45.7): Components that discharge to the PRT/quench tank	3.4	34
46	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM										X		(008A4.04) Ability to manually operate and/or monitor the (SF8 CCW) COMPONENT COOLING WATER SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Startup of a CCW pump when the system is shut down	2.8	35
47	(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM									X			(010A3.03) Ability to monitor automatic features of the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM, including (CFR: 41.7 / 45.7): PZR heater operation	3.3	36
48	**(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM**											**X**	**(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM (G2.4.31) EMERGENCY PROCEDURES/PLAN: Knowledge of annunciator alarms, indications, or response procedures (CFR: 41.10 / 45.3)**	**4.1**	**87**
49	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM		X										(012K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF7 RPS) REACTOR PROTECTION SYSTEM RPS channels, components, and interconnections	4.0	37

50	**(012) (SF7 RPS) REACTOR PROTECTION SYSTEM**								**X**				**(012A2.01) Ability to (a) predict the impacts of the following on the (SF7 RPS) REACTOR 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): Faulty bistable operation$**	**3.6**	**88**
51	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM							X					(013A1.16) Ability to predict and/or monitor changes in parameters associated with operation of the (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, including (CFR: 41.5 / 45.5): Auxiliary building HVAC system status	2.9	38
52	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM				X								(013K4.06) Knowledge of (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Recirculation actuation/reset	4.1	39
53	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM							X					(022A1.01) Ability to predict and/or monitor changes in parameters associated with operation of the (SF5 CCS) CONTAINMENT COOLING SYSTEM, including (CFR: 41.5 / 45.5): Containment temperature	3.8	40
	025 (SF5 ICE) ICE CONDENSER SYSTEM														
54	(026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM		X										(026K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF5 CSS) CONTAINMENT SPRAY SYSTEM Containment spray pumps	3.9	41
55	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM											X	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM (G2.1.23) CONDUCT OF OPERATIONS: Ability to perform general and/or normal operating procedures during any plant condition$	4.3	42
	053 (SF1; SF4P ICS*) INTEGRATED														
56	CONTROL (059) (SF4S MFW) MAIN FEEDWATER SYSTEM	X											(059K1.14) Knowledge of the physical connections and/or cause and effect relationships between the (SF4S MFW) MAIN FEEDWATER SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): ESFAS	3.8	43

57	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM			X									(059K3.05) Knowledge of the effect that a loss or malfunction of the (SF4S MFW) MAIN FEEDWATER SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): Extraction steam	2.7	44
58	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM				X								(061K4.11) Knowledge of (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Automatic level control$	3.6	45
59	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM			X									(062K3.02) Knowledge of the effect that a loss or malfunction of the (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): EDG	4.4	46
60	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM										X		(063A4.03) Ability to manually operate and/or monitor the (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Battery discharge rate	3.5	47
61	**(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM**											**X**	**(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM (G2.1.32) CONDUCT OF OPERATIONS: Ability to explain and apply system precautions, limitations, notes, or cautions (CFR: 41.10 / 43.2 / 45.12)**	**4.0**	**89**
62	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM					X							(064K5.07) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM (CFR: 41.5 / 45.3): Loading the EDG$	3.7	48
63	**(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM**								**X**				**(064A2.24) Ability to (a) predict the impacts of the following on the (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Fuel oil storage system failure**	**3.7**	**90**

64	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM	X											(073K1.05) Knowledge of the physical connections and/or cause and effect relationships between the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): CCWS$	3.1	49
65	(076) (SF4S SW) SERVICE WATER SYSTEM					X							(076K5.05) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4S SW) SERVICE WATER SYSTEM (CFR: 41.5 / 45.3): Radiation alarms on SWS	3.1	50
66	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM									X			(078A3.03) Ability to monitor automatic features of the (SF8 IAS) INSTRUMENT AIR SYSTEM, including (CFR: 41.7 / 45.7): Isolation of instrument air to containment$	3.3	51
67	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM						X						(078K6.06) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF8 IAS) INSTRUMENT AIR SYSTEM (CFR: 41.7 / 45.7): Cross-tie valve	3.0	52
68	(103) (SF5 CNT) CONTAINMENT SYSTEM											X	(103) (SF5 CNT) CONTAINMENT SYSTEM (G2.4.49) EMERGENCY PROCEDURES/PLAN: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls (CFR: 41.10 / 43.2 / 45.6)	4.4	53
69	(103) (SF5 CNT) CONTAINMENT SYSTEM			X									(103K3.04) Knowledge of the effect that a loss or malfunction of the (SF5 CNT) CONTAINMENT SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): Shield building vent system	3.0	54
K/A Category Totals:		2	2	3	2	2	3	3	6	3	2	5	Group Point Total:		33

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Waterford
Plant Systems—Tier 2/Group 2 (RO/**SRO**)

Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
70	(001) (SF1 CRDS) CONTROL ROD DRIVE SYSTEM										X		(001A4.01) Ability to manually operate and/or monitor the (SF1 CRDS) CONTROL ROD DRIVE SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): CRDM cooling	2.9	55

71	(002) (SF2; SF4P RCS) REACTOR COOLANT SYSTEM														
72	**(011) (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM**								**X**				**(011A2.01) Ability to (a) predict the impacts of the following on the (SF2 PZR LCS) PRESSURIZER 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): Excessive Letdown$**	**3.5**	**91**
73	**(014) (SF1 RPI) ROD POSITION INDICATION SYSTEM**											**X**	**(014) (SF1 RPI) ROD POSITION INDICATION SYSTEM (G2.4.4) EMERGENCY PROCEDURES/PLAN: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures$ (CFR: 41.10 / 43.2 / 45.6)**	**4.7**	**92**
	015 (SF7 NI) NUCLEAR INSTRUMENTATIO N SYSTEM														
74	(016) (SF7 NNI) NONNUCLEAR INSTRUMENTATIO N SYSTEM									X			(016A3.01) Ability to monitor automatic features of the (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM, including (CFR: 41.7 / 45.7): Automatic selection of NNI system inputs to control systems	3.3	57
	017 (SF7 ITM) IN CORE TEMPERATURE MONITOR														
	SYSTEM 027 (SF5 CIRS) CONTAINMENT IODINE REMOVAL SYSTEM														
75	(028) (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM			X									(028K3.01) Knowledge of the effect that a loss or malfunction of the (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): Hydrogen concentration in containment	3.1	58

76	**(029) (SF8 CPS) CONTAINMENT PURGE SYSTEM**								**X**				**(029A2.06) Ability to (a) predict the impacts of the following on the (SF8 CPS) CONTAINMENT PURGE 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): CPS component malfunction$**	**3.0**	**93**
	033 (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM					X							(033K5.06) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF8) SPENT FUEL POOL COOLING SYSTEM (CFR: 41.5 / 45.7): Shielding (water level)$	3.7	56
77	(034) (SF8 FHS) FUEL HANDLING EQUIPMENT SYSTEM								X				(034A2.07) Ability to (a) predict the impacts of the following on the (SF8 FHS) FUEL HANDLING EQUIPMENT 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 refueling cavity or Spent Fuel pool level.$	3.7	59
	035 (SF4P SG) STEAM GENERATOR SYSTEM														
	041 (SF4S SDS) STEAM DUMP / TURBINE BYPASS CONTROL														
	SYSTEM 045 (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM														
	050 (SF9 CRV*) CONTROL ROOM VENTILATION														
78	(055) (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM	X											(055K1.02) Knowledge of the physical connections and/or cause and effect relationships between the (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): Main condenser	3.5	60
79	(056) (SF4S CDS) CONDENSATE SYSTEM				X								(056K4.14) Knowledge of (SF4S CDS) CONDENSATE SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): MFW pump NPSH$	3.6	61

80	(068) (SF9 LRS) LIQUID RADWASTE SYSTEM											X	(068) (SF9 LRS) LIQUID RADWASTE SYSTEM (G2.3.11) Radiation Control: Ability to control radiation releases (41.11 /43.4 /45.10)$	3.8	62
81	(071) (SF9 WGS) WASTE GAS DISPOSAL SYSTEM							X					(071A1.09) Ability to predict and/or monitor changes in parameters associated with operation of the (SF9 WGS) WASTE GAS DISPOSAL SYSTEM, including (CFR: 41.5 / 45.5): Waste gas tank discharge radiation levels	3.2	63
	072 (SF7 ARM) AREA RADIATION MONITORING SYSTEM														
	075 (SF8 CW) CIRCULATING WATER SYSTEM														
	079 (SF8 SAS**) STATION AIR SYSTEM														
	086 (SF8 FPS) FIRE PROTECTION														
SYSTEM K/A Category Totals:		1	0	1	1	1	0	1	3	1	1	2	Group Point Total:		12

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Waterford)
Facility: Waterford	Date of Exam: 8/30/2023

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

Category	K/A #	Topic	Item #	RO IR	RO Q#	SRO-Only IR	SRO-Only Q#
1. Conduct of Operations	**G2.1.17**	**(G2.1.17) CONDUCT OF OPERATIONS: Ability to make accurate, clear, and concise verbal reports (CFR: 41.10 / 45.12 / 45.13)**	**82**			**4.0**	**94**
	G2.1.31	**(G2.1.31) CONDUCT OF OPERATIONS: Ability to locate control room switches, controls, and indications and to determine whether they correctly reflect the desired plant lineup (CFR: 41.10 / 45.12)**	**83**			**4.3**	**95**
	G2.1.47	(G2.1.47) CONDUCT OF OPERATIONS: Ability to direct nonlicensed personnel activities inside the control room (CFR: 41.10 / 43.5 / 45.5 / 45.12 / 45.13)	84	3.2	65		
	G2.1.8	(G2.1.8) CONDUCT OF OPERATIONS: Ability to coordinate personnel activities outside the control room (CFR: 41.10 / 43.1 / 45.5 / 45.12 / 45.13)	85	3.4	64		
	Subtotal			N/A	2	N/A	**2**
2. Equipment Control	G2.2.12	(G2.2.12) EQUIPMENT CONTROL: Knowledge of surveillance procedures (CFR: 41.10 / 43.2 / 45.13)	86	3.7	66		
	G2.2.13	(G2.2.13) EQUIPMENT CONTROL: Knowledge of tagging and clearance procedures (CFR: 41.10 / 43.1 / 45.13)	87	4.1	67		
	G2.2.25	**(G2.2.25) EQUIPMENT CONTROL: Knowledge of the bases in TS for limiting conditions for operation and safety limits (CFR: 43.2)$**	**88**			**4.2**	**96**
	G2.2.37	**(G2.2.37) EQUIPMENT CONTROL: Ability to determine operability or availability of safety-related equipment(CFR: 43.2 / 43.5 / 45.12)$**	**89**			**4.6**	**97**
	Subtotal			N/A	2	N/A	**2**
3. Radiation Control	G2.3.5	(G2.3.5) RADIATION CONTROL: Ability to use RMSs, such as fixed radiation monitors and alarms or personnel monitoring equipment (CFR: 41.11 / 41.12 / 43.4 / 45.9)	90	2.9	68		
	G2.3.14	**(G2.3.14) RADIATION CONTROL: 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 to analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only) (CFR: 43.4 / 45.10)**	**91**			**3.8**	**98**
	Subtotal			N/A	1	N/A	**1**
4. Emergency Procedures / Plan	G2.4.21	(G2.4.21) EMERGENCY PROCEDURES/PLAN: Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions (CFR: 41.7 / 43.5 / 45.12)	92	4	69		

	K/A #	Topic	Item #				
	G2.4.29	**(G2.4.29) EMERGENCY PROCEDURES/PLAN: Knowledge of the emergency plan implementing procedures (CFR: 41.10 / 43.5 / 45.11)**	**93**			**4.4**	**99**
	G2.4.40	**(G2.4.40) EMERGENCY PROCEDURES/PLAN: Knowledge of SRO responsibilities in emergency plan implementing procedures (SRO Only) (CFR: 43.5 / 45.11)**	**94**			**4.5**	**100**
	Subtotal			N/A	1	N/A	**2**
Tier 3 Point Total				**N/A**	**6**	**N/A**	**7**

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Waterford)
Facility: Waterford	Date of Exam: 8/30/2023

Theory (Tier 4) (RO)

Category	K/A #	Topic	Item #	RO IR	RO Q#
Reactor Theory	192006	- (192006K1.04) FISSION PRODUCT POISONS (CFR: 41.1): Describe the removal of xenon 135	95	2.8	70
	192007	(192007K1.04) FUEL DEPLETION AND BURNABLE POISONS (CFR: 41.1): Describe how and why boron concentration changes over core life	96	3.4	71
	192008	(192008K1.13) REACTOR OPERATIONAL PHYSICS (CFR: 41.1): (INTERMEDIATE RANGE OPERATION) Discuss the concept of the POAH and its impact on reactor power	97	3.6	72
	Subtotal			N/A	3
Thermodynamics	193003	(193003K1.24) STEAM (CFR: 41.14): Explain the usefulness of steam tables to the control room operator	98	3.1	73
	193004	(193004K1.15) THERMODYNAMIC PROCESS (CFR: 41.14): (THROTTLING AND THE THROTTLING PROCESS) Determine the exit conditions for a throttling process based on the use of steam and/or water	99	2.8	74
	193009	(193009K1.01) CORE THERMAL LIMITS (CFR: 41.14): Explain radial peaking factor	100	2.8	75
	Subtotal			N/A	3
Tier 4 Point Total				**N/A**	**6**

Form 4.1-1 Record of Rejected Knowledge and Abilities

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

Tier/Group	Randomly Selected K/A	Reason for Rejection
Tier 1 Group 1 Question #5	015 AK1.02	015 AK1.01 rejected. The only question that could be written regarding natural circulation criteria was too close to the question written for RO13. One of these two K/As had to be rejected. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #8	029 EK1.03	029 EK1.01 rejected. No guidance found on reactor nucleonic and thermo hydraulic behavior during an ATWS that could be developed into an RO question that did not resemble a theory question. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #11	054 G2.1.20	An RO question involving the criteria for plant announcements that involve a loss of Main Feedwater could not be developed with existing W3 procedures. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #16	062 AK2.09	062 AK2.08 rejected. Three different K/As in this outline involve the relationship between service water and radiation monitoring. They are questions 16, 49 and 50. Question on this subject was written for question 50 and the K/As are rejected for questions 16 and 49. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #18	077 A2.04	077 A2.03 rejected. There is no guidance in any of the W3 procedures on the effect of an Electrical Grid Disturbance and Generator Current. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #19	001 G2.4.12	001 G2.4.18 rejected. Continuous rod withdrawal is not discussed in the W3 EOPs and W3 abnormal operating procedures do not have basis documents. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #20	003 A2.09	003 A2.08 rejected. W3 abnormal operating procedures for a dropped rod do not discuss incore temperature monitoring. All actions are based on cold leg temperatures. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #21	032 AK3.02	032 AK3.01 rejected. All TS actions and procedure guidance for the startup channels at W3 pertain to core alterations. The startup channels de-energize early in the startup (5.3x10-6 power). This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #22	033 A1.02	033 A1.03 rejected. There is no guidance in any W3 procedure for manual restoration of power to the intermediate range nuclear instrumentation. This change does not affect any of the numbers on the first page of the outline.

Tier 1 Group 2 Question #23	037 G2.2.22	037 2.2.35 rejected. Could not develop a question that would link a Steam Generator Tube leak with the ability to determine TS mode for operation This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #24	074 EK2.07	074 EK2.03 rejected. The discussion in W3 Tech Spec basis for inadequate core cooling involves HPSI flow and ADVs. (not EFW flow). This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #25	CE A16 AK1.06	CE A16 AK1.05 rejected. The guidance for detecting and isolating an RCS leak inside containment only involves taking temperatures of inlet piping to the reactor drain tank. An RO question could not be developed from this. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 1 Question #29	004 A2.22	004 A2.23 rejected. The only guidance for high filter DP is in RCS watch logs. A reading is taken on this D/P and the RCA watch must inform the CRS if it is too high. An RO question could not be developed on this information.
Tier 2 Group 1 Question #32	006 A2.01	006 A2.02 rejected. There is no guidance in the W3 procedures for a high bearing temperature on ECCS pumps. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 1 Question #42	039 G2.1.23	039 191008K1.09 rejected. This is a theory question K/A. An RO question could not be developed on losing control power to main or reheat steam components. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 1 Question #45	061 K4.11	061 K4.08 rejected. Not enough guidance or information in W3 procedures on EFW pump recirc to develop an RO question. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 1 Question #48	064 K5.07	064 K5.13 rejected. Not enough guidance on the consequences of a high VAR on EDG integrity to develop an RO question. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 1 Question #49	073 K1.05	073 K1.08 rejected. Three different K/As in this outline involve the relationship between service water and radiation monitoring. They are questions 16, 49 and 50. Question on this subject was written for question 50 and the K/As are rejected for questions 16 and 49. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 1 Question #51	078 A3.03	078 A3.04 rejected. Instrument air is always isolated from station air at W3. This made it impossible to write a question on isolating instrument air from station air. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 2 Question #56		002 K5.05 rejected. Could not develop an RO level question on rise in reactor drain tank pressure during water inventory operations with the RCS system. This is not an evolution that is performed at W3.

Tier 2 Group 2 Question #59	034 A2.07	034 A2.05 rejected. There is no impact on Fuel Handling equipment due to high radiation at W3, other than leaving the area. No question could be written for this K/A. This change does not affect any of the numbers on the first page of the outline.
Tier Group 2 Question #61	056 K4.14	056 K4.01 rejected. There are no design features or interlocks that provide for feedwater heating at W3. No question could be written for this K/A. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 2 Question #62	068 G2.3.11	068 191006K1.09 rejected. This is a theory question K/A. An RO question could not be developed on the effects of water hammer in the heat exchangers of the LWM system. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #76	CE E05 G2.1.9	008 2.1.37 rejected. Could not develop and SRO question that related a Pressurizer Vapor Accident with reactivity management that seemed valid. W3 does not have specific guidance for vapor space accident. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #77	022 AA2.09	022 AA2.06 rejected. No guidance in W3 procedures for monitoring charging pump ammeters. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #81	077 AA2.05	077 AA2.11 rejected. No guidance in W3 procedures for interpret excessive step up transformer neutral DC ground current. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #83	067 G2.1.8	067 G2.1.44 rejected. This K/A is N/A for an SRO. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 1 Question #84	024 A2.05	068 AA2.01 rejected. Question 83 is a Control Room Evacuation question. This would have made back to back Control Room Evacuation questions. Also, could not write a SRO level question pertaining to CRE and steam generator level. This change does not affect any of the numbers on the first page of the outline.
Tier1 Group 1 Question #85	CE09 G2.4.22	076 G2.4.35 rejected. Could not write an SRO level question on non-licensed operator duties during High RCS activity. This change does not affect any of the numbers on the first page of the outline.
Tier 1 Group 2 Question #88	012 A2.01	012 A2.04 rejected. Could not develop and SRO question on erratic power supply operation to the RPS due to limited guidance. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 2 Question #91	011 A2.01	011 A2.08 rejected. No guidance at W3 for loss of level compensation to the pressurizer level control system. This change does not affect any of the numbers on the first page of the outline.
Tier 2 Group 2 Question #92	014 G2.4.4	014 G2.4.49 rejected. Could not develop an SRO question that involved immediate actions as required by this K/A. Additionally, there are no immediate actions at W3 for the Rod Position indicating system. This change does not affect any of the numbers on the first page of the outline.

Tier 2 Group 2 Question #93	029 A2.06	029 A2.02 rejected. An SRO question could not be developed for the effects of adverse environmental conditions on containment purge. It would involve giving the SRO a flowchart but would then be considered a direct lookup. This change does not affect any of the numbers on the first page of the outline.
Tier 3 Question #96	G2.2.25	G2.2.2 rejected. Could not write an SRO question on manipulating the console controls. It is also hard to keep this K/A in the generic category. This change does not affect any of the numbers on the first page of the outline.
Tier 3 Question # 97	G2.2.37	2.2.44 rejected. Could not write an SRO question on interpreting control indications to verify the status and operation of a system while keeping the question generic. This change does not affect any of the numbers on the first page of the outline.

Form 3.2-1 Administrative Topics Outline

Facility: Waterford 3		Date of Examination: 8/21/2023
Examination Level: RO ☒ SRO ☐		Operating Test Number: 1
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
A1 Conduct of Operations	Complete calculations to determine available Feedwater and maximum time remaining to Shutdown Cooling in accordance with OP-902-009, Attachment 2-G. 2.1.25 Ability to interpret reference materials, such as graphs, curves, and tables RO-3.9	R,M(2018)
A2 Conduct of Operations	Determine Acceptability of Containment Temperature in accordance with OP-903-001, Technical Specifications Surveillance Logs 2.1.19 Ability to use available indications to evaluate system or component status RO-3.9	R,D(2015)
A3 Equipment Control	Calculate Keff in accordance with OP-903-090, Shutdown Margin, Section 7.5, Keff Calculation. 2.2.12 Knowledge of surveillance procedures RO-3	R,M(2017)

A4 Radiation Control	Evaluate Meteorological conditions for gaseous release from the Gaseous Waste Management System in accordance with OP-007-003, Gaseous Waste Management. 2.3.11 Ability to control radiation releases RO-3.8	R,D(2017)

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)

Form 3.2-1 Administrative Topics Outline

Facility: Waterford 3		Date of Examination: 8/21/2023
Examination Level: RO ☐ SRO ☒		Operating Test Number: 1
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
A5 Conduct of Operations	Review and approve a completed calculation for determining the amount of pure water that may be added to the Refuel Cavity without dilution to below shutdown margin requirements in accordance with OP-010-006, Outage Operations, section 9.24, Refueling Cavity Boron Concentration. . 2.1.23, Ability to perform specific system and integrated plant procedures during any plant condition SRO-4.4	R,N
A6 Conduct of Operations	Perform SM/CRS review of OP-901-501, PMC or Core Operating Limit Supervisory System Malfunction, Attachments 1, 2 and 3 following a PMC failure. 2.1.20 Ability to interpret and execute procedure steps. SRO-4.6	P,R,D(2020)

A7 Equipment Control	Review Keff Calculation in accordance with OP-903-090, Shutdown Margin, Section 7.5, Keff Calculation. Applicant determines Keff does not meet Tech Spec 3.1.2.9 requirements and identifies required corrective actions. 2.2.12, Knowledge of Surveillance Procedures SRO-4.1	R,M
A8 Radiation Control	Authorize Emergency Exposure as the Emergency Director in accordance with EP-002-030, Emergency Radiation Exposure Guidelines and Controls. 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 to analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits(SRO Only) SRO-3.8	R,M
A9 Emergency Plan	Determine the appropriate emergency action level in accordance with EP-001-001, Recognition and Classification of Emergency Conditions(SRO Only) SRO-4.6 2.4.41, Knowledge of the emergency action level thresholds and classifications	R,N

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)

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

Facility: Waterford 3

Date of Examination: 8/21/2023

Operating Test Number: 1

Exam Level: ☒ RO ☐ SRO-I ☐ SRO-U

System/JPM Title	Type Code	Safety Function
Control Room Systems		
S1 APE 001 Continuous Rod Withdrawal Operator will withdraw CEA's to support plant startup per OP-010-003, Plant Startup. Alt. Path: During CEA withdrawal, the selected CEA group will continue to withdraw after the CEA Shim switch is released. The operator is required to manually trip the reactor per OP-901-102, CEA or CEDMCS Malfunction. AA2.05 Uncontrolled rod withdrawal RO-4.2 SRO-4.1	N,A,E,S	1
S2 006 ECCS Emergency Core Cooling System Adjust HPSI Flow to Maintain Pressurizer Level after HPSI Throttle Criteria Met per OP-902-004, Excess Steam Demand Recovery A1.18 PZR level and pressure RO/SRO-3.8	N,EN,L,S	2
S3 010 PZR RCS Pressurizer Pressure Control System Operate Pressurizer Vents to lower RCS Pressure to maintain P-T Limits per Appendix 38, Pressurizer Vent Operations A2.04 Loss of charging flow to auxiliary spray valves RO-3.4 SRO-3.1	E,L,N,S	3

S4 003 Reactor Coolant Pump System (**2017 Exam**) Perform a Reactor Coolant Pump Shutdown in accordance with OP-001-002, Reactor Coolant Pump Operation. (2014 NRC Exam) Alt. Path: Reactor Coolant pump reverse rotates requiring stopping of remaining Reactor Coolant Pumps. (W3 OE) A2.02 Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP RO – 3.5 SRO – 3.8	A,D,L,S	4P
S5 059 MFW Main Feedwater System Emergency Feedwater Actuation Signal reset in accordance with OP-902-006, Loss of Feedwater, step 29.b.2) and OP-902-009, Appendix 5C. A2.01 Actuation of AFW system RO-4.0 SRO-4.1	N,EN,S	4S
S6 022 CCS Containment Cooling System Change running Containment Cooling Fan configuration per OP-008-003, Containment Cooling System, sections 6.1 and 7.1. A4.01 Ability to manually operate and/or monitor in the control room: CCS fans RO/SRO-3.7	N,S	5
S7 062 ED AC AC Electrical Distribution System Transfer 3A from EDG A to 2A in accordance with OP-902-009, Standard Appendices, Attachment 12C A4.07 Synchronizing and paralleling of different AC supplies RO/SRO-3.7	N,E,L,S	6
S8 068 Liquid Radwaste System Discharging a Boric Acid Condensate Tank to Circulating Water in accordance with OP-007-001, Boron Management Alt. Path: After flow is established, the flow controller will fail and flow will exceed the release permit allowed value. A4.03 Stoppage of release if limits exceeded RO/SRO-3.5	M,A,S	9

In-Plant Systems		
P1 013 Engineered Safety Features Actuation System (ESFAS) (**2020 Exam**) Actuate a Recirculation Actuation Signal (RAS) manually in accordance with OP-902-009 Appendix 34, RAS Manual Actuation. Fault: The RAS will not actuate using the manual pushbuttons and will actuate by opening the breakers. A4.03 ESFAS Initiation RO/SRO-4.4	A,D,EN,P	2
P2 045 MT/G Main Turbine Generator System Transfer Operating Stator Cooling Water Pump in accordance with OP-003-022, Stator Cooling Water, section 6.2. Fault: The stopped stator cooling water pump will experience reverse rotation requiring the pump to be restarted to prevent loss of stator cooling water flow to the main generator. A2.02 Loss of Stator Cooling Water RO – 3.4 SRO – 3.6	A,N	4S
P3 064 Electrical Diesel Generators (**2017 Exam**) Reset EDG A following an overspeed trip with a LOOP in accordance with OP-009-002, Emergency Diesel Generator, section 8.8. EPE 055 EA1.06 Restoration of power with one EDG RO/SRO-4.3	D,E,L,R	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			

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

Facility: Waterford 3

Date of Examination: 8/21/2023

Operating Test Number: 1

Exam Level: ☐ RO ☒ SRO-I ☐ SRO-U

System/JPM Title	Type Code	Safety Function
Control Room Systems		
S1 APE 001 Continuous Rod Withdrawal Operator will withdraw CEA's to support plant startup per OP-010-003, Plant Startup. Alt. Path: During CEA withdrawal, the selected CEA group will continue to withdraw after the CEA Shim switch is released. The operator is required to manually trip the reactor per OP-901-102, CEA or CEDMCS Malfunction. AA2.05 Uncontrolled rod withdrawal RO-4.2 SRO-4.1	N,A,E,S	1
S2 006 ECCS Emergency Core Cooling System Adjust HPSI Flow to Maintain Pressurizer Level after HPSI Throttle Criteria Met per OP-902-004, Excess Steam Demand Recovery. A1.18 PZR level and pressure RO/SRO-3.8	N,EN,L,S	2
S3 010 PZR RCS Pressurizer Pressure Control System Operate Pressurizer Vents to lower RCS Pressure to maintain P-T Limits per Appendix 38, Pressurizer Vent Operations A2.04 Loss of charging flow to auxiliary spray valves RO-3.4 SRO-3.1	E,L,N,S	3

S4 003 Reactor Coolant Pump System (**2017 Exam**) Perform a Reactor Coolant Pump Shutdown in accordance with OP-001-002, Reactor Coolant Pump Operation. (2014 NRC Exam) Alt. Path: Reactor Coolant pump reverse rotates requiring stopping of remaining Reactor Coolant Pumps. (W3 OE) A2.02 Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP RO – 3.5 SRO – 3.8	A,D,L,S	4P
S5 059 MFW Main Feedwater System Emergency Feedwater Actuation Signal reset in accordance with OP-902-006, Loss of Feedwater, step 29.b.2) and OP-902-009, Appendix 5C. A2.01 Actuation of AFW system RO-4.0 SRO-4.1	N,EN,S	4S
S6 022 CCS Containment Cooling System Change running Containment Cooling Fan configuration per OP-008-003, Containment Cooling System, sections 6.1 and 7.1. A4.01 Ability to manually operate and/or monitor in the control room: CCS fans RO/SRO-3.7	N,S	5
S8 068 Liquid Radwaste System Discharging a Boric Acid Condensate Tank to Circulating Water in accordance with OP-007-001, Boron Management Alt. Path: After flow is established, the flow controller will fail and flow will exceed the release permit allowed value. A4.03 Stoppage of release if limits exceeded RO/SRO-3.5	M,A,S	9

In-Plant Systems		
P1 013 Engineered Safety Features Actuation System (ESFAS) (**2020 Exam**) Actuate a Recirculation Actuation Signal (RAS) manually in accordance with OP-902-009 Appendix 34, RAS Manual Actuation. Fault: The RAS will not actuate using the manual pushbuttons and will actuate by opening the breakers. A4.03 ESFAS Initiation RO/SRO-4.4	A,D,EN,P	2
P2 045 MT/G Main Turbine Generator System Transfer Operating Stator Cooling Water Pump in accordance with OP-003-022, Stator Cooling Water, section 6.2. Fault: The stopped stator cooling water pump will experience reverse rotation requiring the pump to be restarted to prevent loss of stator cooling water flow to the main generator. A2.02 Loss of Stator Cooling Water RO – 3.4 SRO – 3.6	A,N	4S
P3 064 Electrical Diesel Generators (**2017 Exam**) Reset EDG A following an overspeed trip with a LOOP in accordance with OP-009-002, Emergency Diesel Generator, Section 8.8. EPE 055 EA1.06 Restoration of power with one EDG RO – 4.1 SRO – 4.5	D,E,L,R	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			

Form 3.3-1 Scenario Outline

Facility:	Waterford 3	Scenario #:	1
Scenario Source:		Op. Test #:	1
Examiners:		Applicants/ Operators:	

Initial Conditions: 1% Power, Plant Startup in progress, PTED not available

Turnover: Dilute to raise power to 5% to 10%.

Critical Tasks: CT-1: Trip RCP's upon a loss of CCW; CT-2: Establish RCS Pressure Control

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R-ATC N-SRO	Dilute to continue plant startup.
2	CH08E1	I – BOP I – SRO TS – SRO	Plant Protection System Channel D Containment Pressure (CIAS), CB-IPI-6701SMD, fails high requiring Technical Specification entry and bypass of channel trip bistables. (TS 3.3.1 action 2 & 3.3.2 action 13)
3	RC15A2	MC-ATC TS-SRO	Pressurizer Level Control Channel Level Transmitter, RC-ILT-0110X (selected channel), fails low requiring the "Y" channel to be selected per OP-901-110, Pressurizer Level Control Malfunction. (TS 3.3.3.5 action a)
4	TP01A TP08B	MC – BOP C – SRO	Running Turbine Cooling Water Pump A trips and the standby pump does not auto start resulting in manual action to start TCCW Pump B in accordance with OP-901-512, Loss of Turbine Cooling Water
5	MS11A	M-All	Excess Steam Demand Event will occur on #1 Steam Generator inside Containment. resulting in a manual or automatic reactor trip, CSAS actuation, (**Critical Task 1, Trip RCPs) (Critical Task 2, Establish RCS Pressure Control)**
6	RP08C	C – ATC C – BOP I – SRO	Relay K202A fails, CVC-401, CVC-109, IA-909, and FP-601A fail to close automatically
7	CS01A	C – BOP C – SRO	Containment Spray Pump A trips requiring action to override close CS-125A, Containment Spray Header A Isolation

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Scenario Quantitative Attributes

1. Malfunctions after EOP entry (1–2)	2
2. Abnormal events (2–4) *[Events 3 and 4 credited]*	2
3. Major transients (1–2)	1
4. EOPs entered/requiring substantive actions (1–2)	1
5. Entry into a contingency EOP with substantive actions ($\geq$ 1 per scenario set)	0
6. Preidentified critical tasks ($\geq$ 2)	2

The crew assumes the shift with the reactor at POAH following a forced outage. The turnover will include instructions to continue the startup by diluting to the charging pump suction in accordance with the reactivity plan.

Event 1: The reactivity plan will include instructions to add primary water RCS. Once enough of a reactivity change is seen by the examiners the next event can be triggered.

Event 2: After the first event is complete, CB-IPI-6701SMD, Containment Pressure (CIAS) fails high. The SRO should review Technical Specifications 3.3.1 and 3.3.2. Per Table 3.3-1 under Containment Pressure – High (Functional Unit 6) the SRO should enter Technical Specification 3.3.1 action 2. Per Table 3.3-3 under Functional Units 1b (Safety Injection, Containment Pressure-High), 3b (Containment Isolation, Containment Pressure-High), and 4c (Main Steam Line Isolation, Containment Pressure High) the SRO should enter Tech 3.3.2 action 13. The SRO should direct the BOP to bypass the Containment Pressure High (RPS) and Containment Pressure High (ESF) trip bistables (13&16) in PPS Channel D within 1 hour. The BOP should bypass the trip bistables in accordance with OP-009-007, Plant Protection System.

Event 3: Once TS have been entered and bypasses are complete, pressurizer Level Control Channel Level Transmitter, RC-ILT-0110X, fails low. The SRO should enter OP-901-110, Pressurizer Level Control Malfunction and implement Section E1. The crew should take manual control of the Pressurizer Level Controller and/or operate Charging Pumps to restore Pressurizer level, swap control to the Channel Y level channel, and return the Pressurizer Level Controller back to AUTO. The SRO should review Technical Specifications 3.3.3.5 and 3.3.3.6 and OP-903-013, Monthly Channel Checks. Tech Spec 3.3.3.5 action “a” should be entered with the action to restore the inoperable channel within 7 days. The SRO should determine that TS 3.3.3.6 requirements are met after verifying QSPDS value for PZR level is within the channel check limit of 5%.

Event 4: After event 3 is complete, the running Turbine Cooling Water Pump A will trip and the standby pump will fail to start automatically. The CRS should direct the BOP to start the standby pump in accordance with OP-901-512, Loss of Turbine Cooling Water and OP-500-005, Control Room Cabinet E.

Event 5: After TCCW Pump B is manually started, an Excess Steam Demand Event will occur on Steam generator #1 inside containment. This will result in an automatic reactor trip and containment pressure exceeding 17.7 psia (the setpoint for automatic Containment Spray Actuation). The ATC should manually stop all RCPs within 3 minutes of the loss of CCW to the RCPs (CRITICAL TASK 1). Once SG #1 has blown dry, the BOP should be directed to Establish RCS Pressure Control to prevent lifting the RCS Safety Valves (Critical Task 2).

Event 6: On actuation of the Safety Injection signal, Relay K202A will not actuate and CVC-401, CVC-109, IA-909, and FP-601A fail to close automatically. The ATC and BOP should position these valves to ensure Containment Isolation.

Event 7: After to crew diagnose to and enter OP-902-004, Excess Steam Demand Recovery, Containment Spray Pump A trips requiring the BOP to manually close CS-125A meet Containment Isolation.

The scenario can be terminated once all CT’s are complete AND event 7 is completed or at the lead examiner’s discretion.

Critical Task	Safety Significance	Cueing	Measurable Performance Indicator	Performance Feedback
<u>CT-1:</u> Trip any RCP exceeding operating limits or after 3 min without CCW flow This task is satisfied by manually tripping all 4 Reactor Coolant Pumps within **3 minutes** of a loss of CCW flow to the RCPs. This task becomes applicable following the actuation of CSAS.	This step is performed for protection of the RCPs, since CCW, which provides cooling to the RCPs, is isolated upon CSAS actuation.	CCW flow low/lost to RCPs alarms on CP-2 and CP-18 CCW valve status CP-2 CSAS initiated CP-8 Procedurally driven from OP-902-000 step 3.b.1 and 9.3	Stops RCPs using control switch	RCP off light illuminated RCP indicated flow lowering
<u>CT-2:</u> Establish RCS Pressure Control. This task is satisfied by performing any of the following: securing pressurizer heaters, reducing charging flow, raising letdown flow, manually initiating pressurizer spray flow, or raising steam generator feedwater flow or steaming rate. This task becomes applicable following the affected steam generator reaching dry out condition.	Prevents RCS pressure from exceeding brittle fracture specifications or reaching safety valve lift setpoint. Safety valves lifting or exceeding brittle fracture limitations could result in a LOCA and degradation of the RCS fission product barrier.	Affected SG Level approx. 0% WR Affected SG pressure is approx. atmospheric /containment pressure CET Temperature rising Pressurizer Pressure rising Procedurally driven from OP-902-000 step 7.d.2 and Appendix 13	ADV opened on least affected SG EFW flow established to least affected SG	SG ADV Open annunciator EFW flow indicated RCS Pressure stable or lowering CET Temperature stable or lowering

REFERENCES

Event	Procedures
1	OP-010-003, Plant Startup, Rev. 363
2	OP-009-007, Plant Protection System, Rev. 20 OP-903-013, Monthly Channel Checks, Rev. 23 OP-500-009, Ann. Response Procedure Cabinet K, Rev. 023 Technical Specifications 3.3.1, 3.3.3.2, Rev. 366
3	OP-901-110, Pressurizer Level Control Malfunction, Rev 011 OP-500-008, Ann. Response Procedure Cabinet H, Rev. 048 Technical Specifications 3.3.3.5, Rev. 366
4	OP-901-512, Loss of Turbine Cooling Water, Rev. 003 OP-500-005, Ann. Response Procedure Cabinet E, Rev. 034
5/6/7	OP-902-000, Standard Post Trip Actions, Rev. 17 OP-902-004, Excess Steam Demand Recovery, Rev. 17 OP-902-009, Standard Appendices, Rev. 323
GEN	EN-OP-115, Conduct of Operations, Rev. 31 EN-OP-115-08, Annunciator Response, Rev. 7 OI-038-000, EOP Operations Expectations / Guidance, Rev. 21 OP-100-017, EOP Implementation Guide, Rev 6

Form 3.3-1 Scenario Outline

Facility:	Waterford 3	Scenario #:	2
Scenario Source:		Op. Test #:	1
Examiners:		Applicants/ Operators:	

Initial Conditions: 100% Power, EDG A OOS, PTED not available

Turnover:

Critical Tasks: CT-1: Manually trip the reactor; CT-2: Establish SI Flow

Event No.	Malf. No.	Event Type*	Event Description
1	RC19C	I–BOP I-SRO TS–SRO	Safety Channel C RCS Cold Leg instrument RC-ITI-0112CC fails high requiring TS 3.3.1 entry and bypassing affected bistables.
2	CV30A2	C – ATC C – SRO	Letdown Flow Control Valve, CVC-113A, fails closed requiring entry into OP-901-112, Charging or Letdown Malfunction.
3	FW51A	TS – SRO	Condensate Storage Pool level instrument EFW-ILI-9013A fails low. (TS 3.3.3.5, TS 3.3.3.6)
4	FW21A FW21AA	R- ATC R/N-BOP N-SRO	Lowering Main Condenser vacuum requiring implementation of OP-901-220, Loss of Condenser Vacuum and a plant power reduction in accordance with OP-901-212, Rapid Plant Power Reduction.
5	RP02A-D RC03C	C – ATC M – All	RCP 2A sustains a locked rotor and an automatic reactor trip does not occur. Manual action is needed to trip the reactor (**Critical Task 1, manually trip the reactor**)
6	SG01B	M – ALL	SGTR SG #2.
7	SI02A SI02B	C/MC – BOP	Both HPSI pumps fail to start requiring manual action (**Critical Task 2, establish SI flow**)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Scenario Quantitative Attributes

1. Malfunctions after EOP entry (1–2)	1
2. Abnormal events (2–4) *[Events 3 and 4 credited]*	2
3. Major transients (1–2)	2
4. EOPs entered/requiring substantive actions (1–2)	1
5. Entry into a contingency EOP with substantive actions (≥ 1 per scenario set)	0
6. Preidentified critical tasks (≥ 2)	2

The crew assumes the shift with the reactor at 100% power. Emergency Diesel Generator A is danger tagged out of service and the PTED is not available.

Event 1: Safety Channel C RCS Cold Leg instrument RC-ITI-0102CC (Loop T112C) fails high requiring TS 3.3.1 entry and bypassing affected bistables. The SRO should review and enter Technical Specification 3.3.1 Action 2 and bypass 3-HI LOCAL POWER and 4-LOW DNBR bistables within 1 hour in accordance with OP-009-007, Plant Protection System.

Event 2: The in-service Letdown flow control valve, CVC-113A, fails closed. The SRO should enter OP-901-112, Charging or Letdown Malfunction and implement Section E2, Letdown Malfunction, and place the backup flow control valve, CVC-113B, in-service. Manual control of letdown flow is required to restore letdown to service.

Event 3: Condensate Storage Pool level indicator EFW-ILI-9013 A will fail low. The SRO should use OP-903-013, Monthly channel Checks, and enter Tech Spec 3.3.3.5 and 3.3.3.6.

Event 4: A leak in the Main Condenser develops and Main Condenser vacuum begins to drop. The SRO will enter OP-901-220, Loss of Condenser Vacuum. Main Condenser vacuum will drop below 25 inches, requiring a rapid plant power reduction. The SRO will enter OP-901-212, Rapid Plant Power Reduction. For the power reduction, the ATC will perform direct boration to the RCS as well as ASI control with CEAs and Pressurizer boron equalization. The BOP will manipulate the controls to reduce Main Turbine load. Vacuum will stabilize during the power reduction.

Event 5: Reactor Coolant Pump 2A rotor seizes and the RCP breaker trips. The Reactor Protection System fails to open the required Reactor Trip Breakers and an ATWS condition exists. The ATC should recognize that an automatic protection system has failed to occur and attempt to manually trip the reactor. Manual trip buttons on CP-2 (including Diverse Rx Trip) will trip the reactor. (**Critical Task 1)**

Event 6: After the reactor is tripped and during Standard Post Trip Actions, a Steam Generator tube rupture will occur on SG #2. The CRS should direct a manual Safety Injection (SIAS) and Containment Isolation (CIAS). The SRO should enter OP-902-007, Steam Generator Tube Rupture Recovery Procedure following diagnoses. The crew should perform a rapid RCS cooldown to < 520 °F and isolate Steam Generator #2.

Event 7: After the crew initiates SIAS, both HPSI Pumps will fail to autostart. The BOP should manually start the at least one HPSI Pump to meet SI flow requirements (**CRITICAL TASK 2**). Both pumps should be started once identified.

The scenario can be terminated once all CT's are complete () AND event 7 is completed or at the lead examiner's discretion.

Critical Task	Safety Significance	Cueing	Measurable Performance Indicator	Performance Feedback
CT-1: Establish Reactivity Control This task is satisfied by manually tripping the Reactor by using the manual trip pushbuttons, Diverse Reactor trip pushbuttons, or by de-energizing busses 32A and 32B within 1 minute of exceeding a Plant Protection System (PPS) limit. This task becomes applicable following the trip of RCP 2A.	Failure to trip the Reactor when an automatic PPS signal has failed to actuate can lead to a degradation of fission product barriers.1 minute is determined to be a reasonable time limit to identify and take action for satisfactory performance. OPS management standard documented in WTRN-OPS-CRITTASK. (WTRN-OPS-CRITTASK, CT-1)	RCP off light illuminated Trips and pre-trips on SG lo flow on CP-7 All CEA rod bottom lights extinguished Procedurally driven from OP-902-000 step 1.a.1.1)	Depresses two Reactor Trip pushbuttons on CP-2 or CP-8 within 1 minute.	Reactor Trip breakers open All CEA rod bottom lights illuminated Reactor power lowering
CT-1: Establish Required Safety Injection Flow This task is satisfied by starting either High Pressure Safety Injection Pump such that Safety Injection flow is greater than or equal to the minimum required per the flow delivery curve in OP-902-009, Appendix 2-E prior to exiting the step (step 7) to establish adequate HPSI flow in OP-902-007. This task becomes applicable following the Safety Injection Actuation Signal. (OP-902-007, 7.a.)	Based on Emergency Operating Procedure Required flow. Failure to take action to maintain the core covered demonstrates a significant performance deficiency regarding the maintenance of the RCS inventory Control Safety Function and jeopardizes the Fuel clad fission product barrier. OPS management Standard documented in WTRN-OPS-CRITTASK. (WTRN-OPS-CRITTASK, CT-16)	HPSI Flow on CP-8 less than required HPSI Pump running light extinguished Procedurally driven from OP-902-007 step 7	Starts either HPSI Pump using control switch	Adequate HPSI Flow on CP-8 indicated HPSI Pump running light illuminated

REFERENCES

Event	Procedures
1	OP-009-007, Plant Protection System, Rev. 20 OP-903-013, Monthly Channel Checks, Rev. 23 Technical Specifications 3.3.1, Rev. 366 OP-500-009, Ann. Response Procedure Cabinet K, Rev. 023
2	OP-901-112, Charging or Letdown Malfunction, Rev. 011 OP-500-007, Annun. Response Procedure Cabinet G, Rev 025
3	OP-903-013, Monthly Channel Checks, Rev. 18 Technical Specifications 3.3.3.5, 3.3.3.6, Rev. 366
4	OP-901-220, Loss of Condenser Vacuum, Rev. 306 OP-901-212, Rapid Plant Power Reduction, Rev. 20
5/6/7	OP-902-000, Standard Post Trip Actions, Rev. 17 OP-902-007, Steam Generator Tube Rupture, Rev. 18 OP-902-009, Standard Appendices, Rev. 323
GEN	EN-OP-115, Conduct of Operations, Rev. 31 EN-OP-115-08, Annunciator Response, Rev. 7 OI-038-000, EOP Operations Expectations / Guidance, Rev. 21 OP-100-017, EOP Implementation Guide, Rev 6

Form 3.3-1 Scenario Outline

Facility:	Waterford 3	Scenario #:	3
Scenario Source:		Op. Test #:	1
Examiners:		Applicants/ Operators:	

Initial Conditions: 100% Power, EDG A OOS, PTED not available

Turnover: Maintain 100% power

Critical Tasks: CT-1: Emergency Borate; CT-2: Energize at least 1 vital AC Bus

Event No.	Malf. No.	Event Type*	Event Description
1	FW06A	N – BOP TS – SRO	Manually start EFW Pump A. EFW Pump A fails during operability check. (TS 3.7.1.2)
2	RX14A	I/MC – ATC I – SRO	PZR pressure control channel failure low (RC-IPIC-0100X)
3	RD02A11	R-ATC R-BOP TS-SRO	CEA 11 drops into the core. The crew will enter Rapid plant power reduction required IAW OP-901-212.
4	DI-01A07A2S01-1	C/MC-BOP C-SRO	During power reduction, the main turbine will revert to manual control.
5	RD02A57 ED01A ED01B ED01C ED01D	M-All	CEA 57 drops into the core resulting in an automatic Reactor Trip and LOOP
6	RD11A28 RD11A40	C-ATC	Two CEAs stuck out **(Critical Task 1, emergency boration required)**
7	EG08B	C – BOP	EDG B does not auto start (**Critical Task 2, Energize at least 1 vital AC bus)**.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Scenario Quantitative Attributes

1. Malfunctions after EOP entry (1–2)	2
2. Abnormal events (2–4) *[Events 3 and 4 credited]*	2
3. Major transients (1–2)	1
4. EOPs entered/requiring substantive actions (1–2)	1
5. Entry into a contingency EOP with substantive actions (≥ 1 per scenario set)	0
6. Preidentified critical tasks (≥ 2)	2

The crew assumes the shift with the reactor at 100% power. Emergency Diesel Generator A is danger tagged out of service and the PTED is not available.

Event 1: The crew turnover includes instructions to complete OP-903-046, Emergency Feed Pump Operability, for Emergency Feedwater (EFW) Pump A. EFW pump A will trip on overcurrent shortly after it is started. The SRO should declare EFW pump A inoperable and enter Tech Spec 3.7.1.2.d.

Event 2: After Technical Specifications have been addressed, Pressurizer Pressure Control Channel RC-IPIC-0100X will fail low. The crew will observe pressurizer pressure alarms and that all PZR heaters are energized. The SRO will implement OP-901-120, Pressurizer Pressure Control Malfunction, Section E_1, Pressurizer Pressure Control Channel Instrument Failure. The SRO should direct the ATC to align the alternate pressurizer pressure channel and verify correct Pressurizer pressure control response.

Event 3: After the alternate pressurizer pressure channel has been placed in service, CEA 11 (Reg. Group 4) drops into the core. The SRO should enter procedure OP-901-102, CEA or CEDMCS Malfunction and proceed to section E_1, CEA Misalignment Greater than 7 inches. The SRO will direct the BOP to adjust turbine load to match T_{AVG} to T_{REF} initially and then perform a rapid plant down power in accordance with OP-901-212, Rapid Plant Power Reduction. RCS direct boration must commence within 15 minutes of the dropped CEA to comply with Technical Specifications and the COLR. The SRO should enter procedure OP-901-501, PMC or COLSS Malfunction. Actions in OP-901-501 are normally performed by the STA. The SRO should evaluate and enter TS 3.1.3.1 action c and TS 3.2.3 action a., and b.

Event 4: During the power reduction, the Main Turbine controls will shift from automatic to manual mode. The SRO will direct the BOP to maintain the turbine load reduction by controlling the turbine in manual mode in accordance with OP-901-212, Rapid Plant Power Reduction. The BOP will be manually changing governor valve position instead of controlling at a certain megawatt per minute rate..

Event 5: After the BOP has lowered Turbine load in manual, CEA 57 will drop into the core. An automatic reactor trip and Loss of Offsite Power will occur simultaneously. The crew will respond using OP-902-000, Standard Post Trip Actions.

Event 6: The ATC should notice that 2 CEAs (40 & 28) have remained stuck out on the reactor trip and Emergency Boration is warranted (**CRITICAL TASK 1**).

Event 7: During the loss of offsite power the B Emergency Diesel Generator will fail to auto start. The BOP should manually start the B EDG using the control switch to prevent the crew from entering the station blackout procedure (**CRITICAL TASK 2**).

The scenario can be terminated after the crew has performed actions in OP-902-003 or at the lead examiner's discretion.

Critical Task	Safety Significance	Cueing	Measurable Performance Indicator	Performance Feedback
CT-1: Establish Reactivity Control This task is satisfied by manually initiating emergency boration using the gravity feed flowpath prior to entering OP-902-003, Loss of Offsite Power / Loss of Forced Circulation. This is accomplished by opening BAM-113A or BAM-113B and closing CVC-183. This task becomes applicable following the Reactor Trip and the Loss of Offsite Power.	Failure to initiate emergency boration would result in a condition that is not allowed by the facility license as analysis assumes that all CEAs are fully inserted during a reactor trip with the exception of the most reactive rod.	Rod bottom lights extinguished CEAs 40 & 28 CEA indicates withdrawn on CEAC Procedurally driven from OP-902-000, Standard Post Trip Actions	Opens BAM-113A or BAM-113B using control switch Closes CVC-183 using control switch	Charging flow $\geq$ 40 gpm on CP-4 BAM-113A or BAM-113B open indicating light illuminated CVC-183 closed indicating light illuminated
CT-2: Energize at least 1 vital AC Bus This task is satisfied by manually starting Emergency Diesel generator B using the control switch prior to performing actions in OP-902-005, Station Blackout Recovery. This task becomes applicable once the Loss of Offsite power occurs.	Failure to energize at least one emergency bus will result in the plant remaining in a configuration that will not support protection if a subsequent event would occur. This lowers the mitigative capability of the plant. Once the crew transitions to and begins taking actions in an inappropriate procedure without taking action to establish power to at least 1 safety bus when one is available demonstrates a significant performance deficiency regarding protecting critical safety functions.	Control room lighting reduces EDG B start light extinguished Bus 3B is de-energized Procedurally driven from OP-902-000, Standard Post Trip Actions	Starts EDG B using the control switch	Control room lighting brightens EDG B start light illuminated Bus 3B is energized EDG B sequencer is timing out

REFERENCES

Event	Procedures
1	OP-903-046, Emergency Feed Pump Operability Check, Rev. 325 Technical Specification 3.7.1.2
2	OP-901-120, Pressurizer Pressure Control Malfunction, Rev 303
3	OP-901-102, CEA or CEDMCS Malfunction, Rev. 308 OP-901-212, Rapid Plant Power Reduction, Rev.20 OP-901-501, PMC or Core Operating Limit Supervisory System Malfunction, Rev. 20 Technical Specification 3.1.3.1 Technical Specification 3.2.3
4	OP-901-212, Rapid Plant Power Reduction, Attachment 4, Rev.20
5/6/7	OP-902-000, Standard Post Trip Actions, Rev. 17 OP-902-003, Loss of Offsite Power / Loss of Forced Circulation, Rev. 11 OP-902-009, Standard Appendices, Rev. 323
GEN	EN-OP-115, Conduct of Operations, Rev. 31 EN-OP-115-08, Annunciator Response, Rev. 7 OI-038-000, EOP Operations Expectations / Guidance, Rev. 21 OP-100-017, EOP Implementation Guide, Rev 6

Form 3.3-1 Scenario Outline

Facility:	Waterford 3	Scenario #:	4
Scenario Source:		Op. Test #:	1
Examiners:		Applicants/ Operators:	

Initial Conditions: 100% Power, EDG A OOS, PTED not available

Turnover: Maintain 100% Power

Critical Tasks: CT-1: Establish Reactivity Control; CT-2: LPSI Pump to replace CS Pump

Event No.	Malf. No.	Event Type*	Event Description
1	RP04B5 AO-07A2M11-1	I – BOP I – SRO TS – SRO	RWSP Level Instrument, SI-ILI-0305B, fails low and generates an RAS trip requiring TS 3.3.2 entry and bypassing the affected trip bistable.
2	CV01B	C – ATC TS – SRO	Charging Pump B trips on overcurrent requiring implementation of OP-901-112, Charging or Letdown malfunction.
3	TC03D2	C – BOP R–ATC N – SRO	Turb Control VLV MS-156 Fail Closed requiring implementation of OP-901-221 and Rapid plant power reduction per OP-901-212
4	TC03C2	C-BOP	Turb Control VLV MS-155 Fail Closed requiring manual Turb Trip and auto Rx Cutback.
5	RD02A01	C–ATC	On Rx Cutback, CEA 1 falls into core resulting in an unanalyzed rod pattern requiring a manual RX trip (**Critical Task 1, establish Rx control)**
6	RC23A CS04A	M – ALL	Loss of Coolant Accident occurs inside containment and valve CS-125A (Containment Spray Hdr A Isolation) fails closed
7	DI-08A04S22-1 CS01B	C/MC – BOP C – SRO	Containment Spray Pump B trips and cannot be restarted requiring entry into OP-902-008, Functional Recovery and action taken to align Low Pressure Safety Injection pump to provide Containment Spray (**Critical Task 2, align LPSI pump to replace CS pump prior to exiting Appendix 28 of OP-902-009**)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

Scenario Quantitative Attributes

1. Malfunctions after EOP entry (1–2)	2
2. Abnormal events (2–4) *[Events 3 and 4 credited]*	2
3. Major transients (1–2)	1
4. EOPs entered/requiring substantive actions (1–2)	2
5. Entry into a contingency EOP with substantive actions (≥ 1 per scenario set)	1
6. Preidentified critical tasks (≥ 2)	2

The crew assumes the shift with the reactor at 100% power. Emergency Diesel Generator A is danger tagged out of service and the PTED is not available.

Event 1: RWSP Level instrument, SI-ILI-0305B, fails low and generates an RAS trip on channel B. The ATC operator will review the annunciators for this failure. The CRS should evaluate Tech Specs and enter Tech Spec 3.3.2 and determine that the Plant Protection System bistable (18) for Low RWSP Level must be bypassed within 1 hour on Channel B. Tech Spec 3.3.3.5 and 3.3.3.6 should be referenced but not entered.

Event 2: Charging Pump B trips. Per the Annunciator Response Procedure, the SRO should direct the ATC to start a standby charging pump after verifying a suction path available or isolate Letdown using CVC-101, Letdown Stop Valve. The SRO will implement OP-901-112, Charging or Letdown Malfunction, Section E_1, Charging Malfunction. If Letdown is isolated, Charging and Letdown will be re-initiated using Attachment 2 of OP-901-112. The SRO should review and enter Technical Specification 3.4.3.2 and TRM 3.13.3, TRM 3.1.2.4. TRM 3.1.2.4 and Technical Specification 3.4.3.2 may be exited after aligning Charging Pump AB to replace Charging Pump B.

Event 3: Turb Control VLV MS-156 Fails Closed. The SRO will Enter and implement OP-901-221, Secondary System Transient, Section E_0 Step 7. The SRO will enter OP-901-212, Rapid Plant Power Reduction and the crew will begin a rapid plant power reduction to reduce Turbine power to less than or equal to 927 MWe. For the power reduction, the ATC will perform direct boration to the RCS as well as ASI control with CEA's and Pressurizer boron equalization. The BOP will manipulate the controls to reduce Main Turbine Load.

Event 4: Turb Control VLV MS-155 Fails Closed. The SRO will implement OP-901-221, Secondary System Transient, Section E_0, Step 6. Reactor Cutback will be in service, SRO will direct the BOP to Trip the Main Turbine and the Plant will get an automatic Reactor Power Cutback.

Event 5: Following the turbine trip, an automatic Reactor Power Cutback will occur. CEA #1 will also drop on the RXC. This requires the ATC to Manually trip the Reactor due to an unanalyzed rod pattern. (**Critical Task 1)**

Event 6: After the Reactor trip, a large Break LOCA is inserted on a ramp RCS break will exceed Charging, the SRO will direct manual initiation of SIAS/CIAS. Containment pressure will rise and exceed 17.7 psia. CS-125A is failed closed. The crew will perform Standard Post Trip Actions using OP-902-000, SPTAs and diagnose to OP-902-002, Loss of Coolant Accident Recovery.

Event 7: After the crew diagnoses to OP-902-002, Loss of Coolant Accident Recovery, Containment Spray Pump B trips and cannot be restarted requiring entry into OP-902-008, Functional Recovery due to not meeting Safety Function in OP-902-002. The SRO will prioritize and take action taken to align Low Pressure Safety Injection pump to provide Containment Spray. BOP will perform the steps in OP-902-009, Standard Appendices, Attachment 28, to align LPSI Pump B to replace CS Pump B. (**Critical Task 2, align LPSI pump B to replace CS pump B prior to exiting Appendix 28 of OP-902-009**)

The scenario can be terminated once all CT's are complete (Reactor Tripped and LPSI Pump B aligned to replace CS pump B) AND event 7 is completed or at the lead examiner's discretion.

Critical Task	Safety Significance	Cueing	Measurable Performance Indicator	Performance Feedback
<u>CT-1:</u> Establish Reactivity Control This task is satisfied by manually tripping the Reactor by using the manual trip pushbuttons, Diverse Reactor trip pushbuttons, or by de-energizing busses 32A and 32B within 1 minute of recognizing an incorrect CEA pattern. This task becomes applicable following the Reactor Power Cutback and CEA # 1 dropping.	Failure to trip the reactor when an unanalyzed CEA pattern exists will result in operating the plant in a manner that would violate a condition of the facility license. OPS management standard documented in WTRN-OPS-CRITTASK. (WTRN-OPS-CRITTASK, CT-1)	CEA rod bottom lights extinguished CEAC positions OP-901-101 IOA verifies correct rod pattern for a cutback	Depresses two Reactor Trip pushbuttons on CP-2 or CP-8 within 1 minute of recognition of unanalyzed rod pattern.	Reactor Trip breakers open All CEA rod bottom lights illuminated Reactor power lowering
<u>CT-2</u>: Establish Containment Temperature and Pressure Control This task is satisfied by aligning LPSI Pump B to replace CS Pump B. This task becomes applicable following CS Pump B tripping.	Failure to take action to establish containment pressure and temperature control may result in containment pressure exceeding maximum design and therefore exceed design leakage of containment. This would lead to a degradation of a fission product barrier. OPS management Standard documented in WTRN-OPS-CRITTASK. (WTRN-OPS-CRITTASK, CT-15)	LPSI pump light status on CP-8 CS Pump light status on CP-8 Containment pressure OP-902-008, Functional recovery CPTC continuing actions OP-902-009, Standard Appendices.	The crew takes action to manually align an available LPSI pump to replace a CS pump.	Adequate CS flow indication on CP-8.

REFERENCES

Event	Procedures
1	OP-009-007, Plant Protection System, Rev. 22 OP-903-013, Monthly Channel Checks, Rev. 23 Technical Specification 3.3.2
2	OP-500-007, Control Cabinet G, Rev. 25 OP-901-112, Charging or Letdown Malfunction, Rev 11 Technical Specification 3.4.3.2 Technical Requirement 3.1.2.4 Technical Requirement 3.13.3
3	OP-901-221, Secondary System Transient, Rev. 11 OP-901-212, Rapid Plant Power Reduction, Rev. 20
4	OP-901-221, Secondary System Transient Rev. 11
5	OP-901-101, Reactor Power Cutback, Rev. 10
6/7	OP-902-000, Standard Post Trip Actions, Rev. 17 OP-902-002, Loss of Coolant Accident Recovery, Rev. 21 OP-902-008, Functional Recovery Procedure, Rev. 31 OP-902-009, Standard Appendices, Rev. 323
Gen	EN-OP-115, Conduct of Operations, Rev. 31 EN-OP-115-08, Annunciator Response, Rev. 7 OI-038-000, EOP Operations Expectations / Guidance, Rev. 21 OP-100-017, EOP Implementation Guide, Rev 6

Facility: Waterford 3 Date of Exam: 8/21/2023 Operating Test No.: 1

APPLICANT	EVENT TYPE	Scenarios															
		1			2			3			4			TOTAL	MINIMUM*		
		POSITION			POSITION			POSITION			POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
RO ☐ SRO-I ☐ SRO-U ☐	RX		1												1	1	0
	NOR	1													1	1	1
	I/C	2,4,6,7	6	2,6,7											4	4	2
	MAJ	5	5	5											2	2	1
	Man. Ctrl		3	4											1	1	0
	TS	2,3													0	2	2
RO ☐ SRO-I ☐ SRO-U ☐	RX					4	4								1	1	0
	NOR				4										1	1	1
	I/C				1,2	2,5	1,7								4	4	2
	MAJ				5,6	5,6	5,6								2	2	1
	Man. Ctrl						7								1	1	0
	TS				1,3										0	2	2
RO ☐ SRO-I ☐ SRO-U ☐	RX								3	3					1	1	0
	NOR									1					1	1	1
	I/C							2,4	2,6	7					4	4	2
	MAJ							5	5	5					2	2	1
	Man. Ctrl								2						1	1	0
	TS							1,3							0	2	2
RO ☐ SRO-I ☐ SRO-U ☐	RX											3			1	1	0
	NOR										3				1	1	1
	I/C										1,7	2,5	1,3,4		4	4	2
	MAJ										6	6	6		2	2	1
	Man. Ctrl												7		1	1	0
	TS										1,2				0	2	2

Form 3.4-1 Events and Evolutions Checklist

Form 3.4-1 Instructions for the Events and Evolutions Checklist

1. Mark the applicant license level for each simulator operating test number.

2. For the set of scenario columns, fill in the associated event number from Form 3.3-1, “Scenario Outline,” to show the specific event types being used for the applicant while in the assigned crew position for that scenario.

* Minimums are subject to the instructions in Section C.2, “License Level Criteria.”
KEY: RX = Reactivity Manipulation; NOR = Normal Evolution; I/C = Instrument/Component Failure; MAJ = Major Transient; Man. Ctrl = Manual Control of Automatic Function; TS = Technical Specification Evaluation; RO = Reactor Operator; SRO-I or I = Instant Senior Reactor Operator; SRO-U or U = Upgrade Senior Reactor Operator; SRO = Senior Reactor Operator; ATC = At the Controls; and BOP = Balance of Plan