Form 4.1-PWR Pressurized-Water Reactor Examination Outline

Facility: Comanche Peak	**K/A Catalog Rev. 3**	**Rev.** 2	**Date of Exam:** 6/12/2023

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

3. Generic Knowledge and Abilities Categories	CO	EC	RC	EM	6	**CO**	**EC**	**RC**	**EM**	**7**
	2	2	1	1		**2**	**2**	**1**	**2**	

4. Theory	Reactor Theory	Thermodynamics	6	
	3	3		

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

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

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

ES-4.1-PWR PWR Examination Outline (Comanche Peak)

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

Item #	E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	Q#
1	(000007) (EPE 7; BW E02 & E10; CE E02) Reactor Trip, Stabilization, Recovery		X					(000007EK2.13) Knowledge of the relationship between (EPE 7) REACTOR TRIP, STABILIZATION, RECOVERY and the following systems or components (CFR: 41.8 / 41.10 / 45.3): CVCS	3.5	11
2	(000008) (APE 8) Pressurizer Vapor Space Accident				X			(000008AA1.02) Ability to operate and/or monitor the following as they apply to (APE 8) PRESSURIZER VAPOR Space Accident (CFR: 41.5 / 41.7 / 45.5 to 45.8): CVCS System to control PZR level/pressure	3.6	12
3	(000009) (EPE 9) Small Break LOCA			X				(000009EK3.22) Knowledge of the reasons for the following responses and/or actions as they apply to (EPE 9) SMALL-Break LOCA (CFR: 41.5 / 41.10 / 45.6 / 45.13): Maintenance of heat sink	3.9	13
4	(000011) (EPE 11) Large Break LOCA					X		(000011EA2.05) Ability to determine and/or interpret the following as they apply to (EPE 11) LARGE-Break LOCA (CFR: 41.10 / 43.5 / 45.13): Significance of ECCS pump operation	4.4	14
5	(000015) (APE 15) Reactor Coolant Pump Malfunctions						X	(000015) (APE 15) Reactor Coolant Pump Malfunctions (G2.4.2) Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedures entry conditions (CFR 41.7)	4.5	15
6	(000022) (APE 22) Loss of Reactor Coolant Makeup	X						(000022AK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 22) LOSS OF REACTOR Coolant Makeup (CFR: 41.5 / 41.7 / 45.7 / 45.8): Relationship between charging flow and PZR level	3.6	16
7	(000025) (APE 25) Loss of Residual Heat Removal System	X						(000025AK1.06) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 25) LOSS OF RESIDUAL Heat Removal System (CFR: 41.5 / 41.7 / 45.7 / 45.8): Time to boiling determinations	3.8	17
8	(000026) (APE 26) Loss of Component Cooling Water				X			(000026AA1.05) Ability to operate and/or monitor the following as they apply to (APE 26) LOSS OF Component Cooling Water (CFR: 41.5 / 41.7 / 45.5 to 45.8): The CCWS surge tank, including level control, level alarms, and a radiation alarm	3.5	18
9	(000027) (APE 27) Pressurizer Pressure Control System Malfunction					X		(000027AA2.09) Ability to determine and/or interpret the following as they apply to (APE 27) PRESSURIZER PRESSURE Control System Malfunction (CFR: 41.10 / 43.5 / 45.13): Reactor power	4.0	19
10	(000029) (EPE 29) Anticipated Transient Without Scram		X					(000029EK2.15) Knowledge of the relationship between (EPE 29) ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) and the following systems or components (CFR: 41.8 / 41.10 / 45.3): MFW system	3.3	20
11	(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer			X				(WE12EK3.07) Knowledge of the reasons for the following responses and/or actions as they apply to (W E12) UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS (CFR: 41.5 / 41.10 / 45.6 / 45.13): Controlling S/G feed flow	4.1	21
12	(000054) (APE 54; CE E06) Loss of Main Feedwater				X			(000054AA1.07) Ability to operate and/or monitor the following as they apply to (APE 54) LOSS OF Main Feedwater (CFR: 41.5 / 41.7 / 45.5 to 45.8): MFW pumps	3.4	22
13	(000058) (APE 58) Loss of DC Power	X						(000058AK1.04) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 58) LOSS OF DC Power (CFR: 41.5 / 41.7 / 45.7 / 45.8): Loss of breaker protection	3.7	23

14	(000062) (APE 62) Loss of Nuclear Service Water		X					(000062AK2.02) Knowledge of the relationship between (APE 62) LOSS OF SERVICE WATER and the following systems or components (CFR: 41.8 / 41.10 / 45.3): IAS	3.0	24
15	(000065) (APE 65) Loss of Instrument Air						X	(000065) (APE 65) Loss of Instrument Air (G2.1.7) Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR 41.5)	4.4	25
16	(W E04) LOCA Outside Containment						X	(W E04) LOCA Outside Containment (G2.4.35) EMERGENCY PROCEDURES/PLAN: Knowledge of non-licensed operator responsibilities during an emergency (CFR: 41.10 / 43.1 / 43.5 / 45.13)	3.8	26
17	(W E11) Loss of Emergency Coolant Recirculation			X				(WE11EK3.15) Knowledge of the reasons for the following responses and/or actions as they apply to (W E11) LOSS OF EMERGENCY Coolant Recirculation (CFR: 41.5 / 41.10 / 45.6 / 45.13): Isolating or venting SI accumulators	3.3	27
18	(BW E04; W E05) Inadequate Heat Transfer – Loss of Secondary Heat Sink					X		(WE05EA2.15) Ability to determine and/or interpret the following as they apply to (W E05) Loss of Secondary Heat Sink (CFR: 41.10 / 43.5 / 45.13): CCW flow to RHR heat exchangers	2.9	28
19	**(000008) (APE 8) Pressurizer Vapor Space Accident**					**X**		**(000008A2.27) (APE 8) Pressurizer Vapor Space Accident Ability to determine and/or interpret the following as they apply to PZR vapor space accident: PZR pressure and/or level due to sensing line leakage (CFR 43.5)**	**3.6**	**84**
20	**(000038) (EPE 38) Steam Generator Tube Rupture**						**X**	**(000038) (EPE 38) Steam Generator Tube Rupture (G2.4.18) Knowledge of the specific bases for emergency and abnormal operating procedures (CFR: 43.1)**	**4.0**	**85**
21	**(000055) (EPE 55) Station Blackout**					**X**		**(000055EA2.04) Ability to determine and/or interpret the following as they apply to (EPE 55) Station Blackout (CFR: 41.10 / 43.5 / 45.13): Instruments and controls operable with only DC battery power available**	**3.9**	**86**
22	**(000056) (APE 56) Loss of Offsite Power**						**X**	**(000056) (APE 56) Loss of Offsite Power (G2.4.16) Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines, such as operating procedures, abnormal operating procedures, or severe accident management guidelines (CFR: 41.10 / 43.5 / 45.13)**	**4.4**	**87**
23	**(000057) (APE 57) Loss of Vital AC Instrument Bus**						**X**	**(000057) (APE 57) Loss of Vital AC Instrument Bus (G2.2.25) Knowledge of the bases in TS for limiting conditions for operation and safety limits (SRO Only) (CFR: 43.2)**	**4.2**	**88**
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: 41.10 / 43.5 / 45.13): Status of grid**	**3.6**	**89**
K/A Category Totals:		3	3	3	3	6	6	Group Point Total:		24

ES-4.1-PWR	PWR Examination Outline (Comanche Peak)									
	Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/**SRO**)									
Item #	E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	Q#
25	(000005) (APE 5) Inoperable/Stuck Control Rod						X	(000005) (APE 5) Inoperable/Stuck Control Rod (G2.4.3) EMERGENCY PROCEDURES/PLAN: Ability to identify post-accident instrumentation (CFR: 41.6 / 45.4)	3.7	47
26	(000033) (APE 33) Loss of Intermediate Range Nuclear Instrumentation		X					(000033AK2.03) Knowledge of the relationship between (APE 33) LOSS OF INTERMEDIATE RANGE Nuclear Instrumentation and the following systems or components (CFR: 41.8 / 41.10 / 45.3): RPS	3.8	48
27	(000036) (APE 36; BW/A08) Fuel-Handling Incidents			X				(000036AK3.04) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 36) FUEL HANDLING INCIDENTS (CFR: 41.5 / 41.10 / 45.6 / 45.13): Establishing containment isolation or closure	3.9	49
28	(000037) (APE 37) Steam Generator Tube Leak			X				(000037AK3.10) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 37) STEAM Generator Tube Leak (CFR: 41.5 / 41.10 / 45.6 / 45.13): Automatic actions associated with high radioactivity in S/G sample lines	3.5	50
29	(000068) (APE 68; BW A06) Control Room Evacuation				X			(000068AA1.33) Ability to operate and/or monitor the following as they apply to (APE 68) Control Room Evacuation (CFR: 41.5 / 41.7 / 45.5 to 45.8): RPS	3.5	51
30	(BW E08; W E03) LOCA Cooldown – Depressurization	X						(WE03EK1.06) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (W E03) LOCA COOLDOWN AND Depressurization (CFR: 41.5 / 41.7 / 45.7 / 45.8): Blocking low steamline pressure SI when the PZR pressure lowers less than P-11 setpoint during cooldown	3.9	52
31	(BW E09; CE A13**; W E09 & E10) Natural Circulation				X			(WE09EA1.20) Ability to operate and/or monitor the following as they apply to (W E09) NATURAL CIRCULATION OPERATIONS (CFR: 41.5 / 41.7 / 45.5 to 45.8): CRDM Cooling fans	3.0	53
32	(CE A11**; W E08) RCS Overcooling – Pressurized Thermal Shock					X		(WE08EA2.10) Ability to determine and/or interpret the following as they apply to (W E08) Pressurized Thermal Shock (CFR: 41.10 / 43.5 / 45.13): VCT level and/or pressure	3.0	54
33	**(000024) (APE 24) Emergency Boration**					**X**		**(000024AA2.04) Ability to determine and/or interpret the following as they apply to (APE 24) EMERGENCY Boration (CFR: 41.10 / 43.5 / 45.13): Availability of the RWST**	**3.7**	**90**
34	**(W E15) Containment Flooding / 5**						**X**	**(W E15) Containment Flooding (G2.4.47) EMERGENCY PROCEDURES/PLAN: Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential) (CFR: 43.5)**	**4.2**	**91**
35	**(000069) (APE 69; W E14) Loss of Containment Integrity**						**X**	**(000069) (APE 69; W E14) Loss of Containment Integrity (G2.3.11) RADIATION CONTROL: Ability to control radiation releases (CFR: 41.11 / 43.4 / 45.10)**	**4.3**	**92**
36	**(000074) (EPE 74; W E06 & E07) Inadequate Core Cooling**					**X**		**(WE07 EA2.05) Ability to determine and/or interpret the following as they apply to Saturated Core Cooling: ECCS flow (CFR 43.5)**	**3.7**	**93**
	000001 (APE 1) Continuous Rod Withdrawal / 1									

	000003 (APE 3) Dropped Control Rod / 1									
	000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
	000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
	(000051) (APE 51) Loss of Condenser Vacuum									
	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									
	000067 (APE 67) Plant Fire On Site / 8									
	000076 (APE 76) High Reactor Coolant Activity / 9									
	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 E13 & E14) EOP Rules and Enclosures									
	(CE A16) Excess RCS Leakage / 2									
	(CE E09) Functional Recovery									
	(CE E13*) Loss of Forced Circulation / LOOP / Blackout / 4									
K/A Category Totals:		1	1	2	2	3	3	Group Point Total:		12

ES-4.1-PWR	PWR Examination Outline (Comanche Peak) Plant Systems—Tier 2/Group 1 (RO/**SRO**)														
Item #	System / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	Q#
37	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM									X			(003A3.05) Ability to monitor automatic features of the (SF4P RCP) REACTOR COOLANT PUMP SYSTEM, including (CFR: 41.7 / 45.7): RCP lube oil and bearing lift pumps	3.0	1
38	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM		X										(003K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF4P RCP) REACTOR COOLANT PUMP SYSTEM RCPS	3.7	2
39	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM	X											(004K1.25) Knowledge of the physical connections and/or cause and effect relationships between the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): Interface between HPI flowpath and excess letdown flowpath	3.5	3
40	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM			X									(004K3.06) Knowledge of the effect that a loss or malfunction of the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): RCS	3.9	4
41	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM		X										(005K2.02) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM Containment isolation valves	3.3	5

42	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM			X									(005K3.05) Knowledge of the effect that a loss or malfunction of the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): ECCS	4.3	6
43	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM							X					(006A1.02) Ability to predict and/or monitor changes in parameters associated with operation of the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM, including (CFR: 41.5 / 45.5): Boron concentration in safety injection tank and BAT	3.4	7
44	(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	8
45	(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM										X		(007A4.03) Ability to manually operate and/or monitor the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Nitrogen block valve	2.6	9
46	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM								X				(008A2.05) Ability to (a) predict the impacts of the following on the (SF8 CCW) COMPONENT COOLING WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Effect of loss of instrument and control air on the position of air-operated CCW valves	3.2	10
47	(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM						X						(010K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM (CFR: 41.7 / 45.7): PZR pressure channels	3.9	29
48	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM											X	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM (G2.1.31) 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)	4.6	30
49	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM				X								(013K4.08) Knowledge of (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Redundancy	3.7	31

50	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM					X							(013K5.02) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (CFR: 41.5 / 45.3): Safety system logic and reliability	3.7	32
51	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM									X			(022A3.01) Ability to monitor automatic features of the (SF5 CCS) CONTAINMENT COOLING SYSTEM, including (CFR: 41.7 / 45.7): Initiation of ESFAS mode of operation	4.2	33
52	(026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM				X								(026K4.07) Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Adequate level in containment sump for suction (CFR: 41.7)	3.8	34
53	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM						X						(039K6.12) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM (CFR: 41.7 / 45.7): MSIVs	3.8	35
54	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM	X											(059K1.13) 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): S/GB system	2.9	36
55	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM					X							(061K5.07) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM (CFR: 41.5 / 45.3): Back leakage through discharge check valves	3.1	37
56	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM					X							(061K5.08) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM (CFR: 41.5 / 45.3): Expected AFW flow rates based on plant conditions	3.9	38
30	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM		X										(062K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM Major bus or motor control center power supplies	3.8	39

58	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM								X				(063A2.03) Ability to (a) predict the impacts of the following on the (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Malfunction of battery chargers	3.9	40
59	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM							X					(064A1.09) Ability to predict and/or monitor changes in parameters associated with operation of the (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM, including (CFR: 41.5 / 45.5): Diesel engine operating parameters	3.6	41
60	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM										X		(073A4.02) Ability to manually operate and/or monitor the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): RMS control panel	3.6	42
61	(076) (SF4S SW) SERVICE WATER SYSTEM											X	(076) (SF4S SW) SERVICE WATER SYSTEM (G2.1.30) CONDUCT OF OPERATIONS: Ability to locate and operate components, including local controls (CFR: 41.7 / 45.7)	4.4	43
62	(076) (SF4S SW) SERVICE WATER SYSTEM	X											(076K1.05) Knowledge of the physical connections and/or cause and effect relationships between the (SF4S SW) SERVICE WATER SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): EDGs	4.1	44
63	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM					X							(078K5.03) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF8 IAS) INSTRUMENT AIR SYSTEM (CFR: 41.5 / 45.3): Loss of instrument air	3.9	45
64	(103) (SF5 CNT) CONTAINMENT SYSTEM				X								(103K4.04) Knowledge of (SF5 CNT) CONTAINMENT SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Personnel access hatch and emergency access hatch	3.3	46
65	**(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM**											**X**	**(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM (G2.2.3) (Multi-unit license) Knowledge of the design, procedural, and/or operational differences between units (CFR:43.5)**	**3.8**	**83**

66	**(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM**								**X**				**(010A2.09) Ability to (a) predict the impacts of the following on the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 43.5): RPS failure**	**3.8**	**76**
67	**(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM**											**X**	**(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM (G2.2.12) Knowledge of surveillance procedures (CFR: 41.10 / 43.2 / 45.13)**	**4.1**	**77**
68	**(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM**								**X**				**(073A2.01) Ability to (a) predict the impacts of the following on the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 43.5 / 45.6): PRM component failures**	**3.1**	**78**
69	**(103) (SF5 CNT) CONTAINMENT SYSTEM**											**X**	**(103) (SF5 CNT) CONTAINMENT SYSTEM (G2.3.14) 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) (CFR: 43.4 / 45.10)**	**3.8**	**79**
	025 (SF5 ICE) ICE CONDENSER SYSTEM														
	053 (SF1; SF4P ICS*) INTEGRATED CONTROL SYSTEM														
K/A Category Totals:		3	3	2	3	4	2	2	4	3	2	5	Group Point Total:		33

ES-4.1-PWR	PWR Examination Outline (Comanche Peak) Plant Systems—Tier 2/Group 2 (RO/**SRO**)														
Item #	System / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	Q#
70	(011) (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM						X						(011K6.16) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM (CFR: 41.7 / 45.7): PZR PCS	3.6	55

71	(015) (SF7 NI) NUCLEAR INSTRUMENTATION SYSTEM					X							(015K5.06) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF7 NI) NUCLEAR INSTRUMENTATION SYSTEM (CFR: 41.5 / 45.3): Subcritical multiplications and NIS indications	3.7	56
72	(016) (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM			X									(016K3.07) Knowledge of the effect that a loss or malfunction of the (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): ECCS	3.7	57
73	(029) (SF8 CPS) CONTAINMENT PURGE SYSTEM	X											(029K1.06) Knowledge of the physical connections and/or cause and effect relationships between the (SF8 CPS) CONTAINMENT PURGE SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): CNT	3.3	58
75	(045) (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM									X			(045A3.04) Ability to monitor automatic features of the (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM, including (CFR: 41.7 / 45.7): Main turbine trip	3.8	59
76	(071) (SF9 WGS) WASTE GAS DISPOSAL SYSTEM										X		(071A4.06) Ability to manually operate and/or monitor the (SF9 WGS) WASTE GAS DISPOSAL SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Meteorological data	2.7	60
77	(072) (SF7 ARM) AREA RADIATION MONITORING SYSTEM							X					(072A1.01) Ability to predict and/or monitor changes in parameters associated with operation of the (SF7 ARM) AREA RADIATION MONITORING SYSTEM, including (CFR: 41.5 / 45.5): Radiation levels	3.4	61
78	(086) (SF8 FPS) FIRE PROTECTION SYSTEM								X				(086A2.02) Ability to (a) predict the impacts of the following on the (SF8 FPS) FIRE PROTECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Low FPS header pressure	3.4	62
79	**(014) (SF1 RPI) ROD POSITION INDICATION SYSTEM**											**X**	**(014) (SF1 RPI) ROD POSITION INDICATION SYSTEM (G2.2.7) Knowledge of the process for conducting infrequently performed tests or evolutions (CFR: 41.10 / 43.3 / 45.13)**	**3.6**	**81**

80	**(027) (SF5 CIRS) CONTAINMENT IODINE REMOVAL SYSTEM**											**X**	**(027) (SF5 CIRS) CONTAINMENT IODINE REMOVAL SYSTEM (G2.2.38) Knowledge of conditions and limitations in the facility license (CFR 43.1)**	**4.5**	**82**
81	**(028) (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM**								**X**				**(028A2.02) Ability to (a) predict the impacts of the following on the (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE 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): LOCA with significant hydrogen production**	**3.4**	**80**
	001 (SF1 CRDS) CONTROL ROD DRIVE SYSTEM														
74	002 (SF2; SF4P RCS) REACTOR COOLANT SYSTEM				X								(002) Knowledge of (SF2; SF4P RCS) REACTOR COOLANT SYSTEM as it relates to (K4.10) overpressure protection	4.1	63
	017 (SF7 ITM) IN CORE TEMPERATURE MONITOR SYSTEM														
	~~(034) (SF8 FHS) FUEL HANDLING EQUIPMENT SYSTEM~~														
	033 (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM														
	035 (SF4P SG) STEAM GENERATOR SYSTEM														
	041 (SF4S SDS) STEAM DUMP / TURBINE BYPASS CONTROL SYSTEM														
	050 (SF9 CRV*) CONTROL ROOM VENTILATION														
	055 (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM														
	056 (SF4S CDS) CONDENSATE SYSTEM														
	068 (SF9 LRS) LIQUID RADWASTE SYSTEM														

	075 (SF8 CW) CIRCULATING WATER SYSTEM														
	079 (SF8 SAS**) STATION AIR SYSTEM														
K/A Category Totals:		1	0	1	1	1	1	1	2	1	1	2	Group Point Total:		12

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Comanche Peak)	
Facility: Comanche Peak		Date of Exam: 6/12/2023

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

Category	K/A #	Topic	Item #	RO IR	RO Q#	SRO-Only IR	SRO-Only Q#
1. Conduct of Operations	G2.1.4	(G2.1.4) Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, and maintenance of active license status, 10 CFR Part 55 (CFR: 41.10 / 43.2)	82	3.3	64		
	G2.1.29	(G2.1.29) CONDUCT OF OPERATIONS: Knowledge of how to conduct system lineups, such as valves, breakers, or switches (CFR: 41.10 / 45.1 / 45.12)	83	4.1	65		
	G2.1.5	**(G2.1.5) Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential) (CFR: 41.10 / 43.5 / 45.12)**	**84**			**3.9**	**94**
	G2.1.34	**(G2.1.34) CONDUCT OF OPERATIONS: Knowledge of RCS or balance-of-plant chemistry controls, including parameters measured and reasons for the control (CFR: 41.10 / 43.5 / 45.12)**	**85**			**3.5**	**95**
	Subtotal			N/A	2	N/A	**2**
2. Equipment Control	G2.2.39	(G2.2.39) EQUIPMENT CONTROL: Knowledge of less than or equal to 1 hour TS action statements (does not include action statements of 1 hour or less that follow the expiration of a completion time for a TS condition for which an action statement has already been entered) (CFR: 41.7 / 41.10 / 43.2 / 45.13)	86	3.9	66		
	G2.2.13	(G2.2.13) Knowledge of tagging and clearance procedures (CFR: 41.10 / 43.1 / 45.13)	87	4.1	67		
	G2.2.6	**(G2.2.6) EQUIPMENT CONTROL: Knowledge of the process for making changes to procedures (CFR: 41.10 / 43.3 / 45.13)**	**88**			**3.6**	**96**
	G2.2.19	**(G2.2.19) EQUIPMENT CONTROL: Knowledge of maintenance work order requirements (CFR: 41.10 / 43.5 / 45.13)**	**89**			**3.4**	**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	G2.4.34	(G2.4.34) EMERGENCY PROCEDURES/PLAN: Knowledge of RO responsibilities outside the main control room during an emergency (CFR: 41.10 / 43.5 / 45.13)	92	4.2	69		
	G2.4.14	**(G2.4.14) EMERGENCY PROCEDURES/PLAN: Knowledge of general guidelines for emergency and abnormal operating procedures usage (CFR: 41.10 / 43.1 / 45.13)**	**93**			**4.5**	**99**
	G2.4.21	**(G2.4.21) EMERGENCY PROCEDURES/PLAN: Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions (CFR: 41.7 / 43.5 / 45.12)**	**94**			**4.6**	**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 (Comanche Peak)	
Facility: Comanche Peak		Date of Exam: 6/12/2023

Theory (Tier 4) (RO)

Category	K/A #	Topic	Item #	RO IR	RO Q#
Reactor Theory	192006	(192006K1.03) FISSION PRODUCT POISONS (CFR: 41.1): Describe the production of xenon-135	95	2.8	70
	192007	(192007K1.01) FUEL DEPLETION AND BURNABLE POISONS (CFR: 41.1): Define burnable poison and state its use in the reactor	96	2.5	71
	192008	(192008K1.21) REACTOR OPERATIONAL PHYSICS (CFR: 41.1): (POWER OPERATION) Explain the relationship between steam flow and reactor power given specific conditions	97	3.8	72
	Subtotal			N/A	3
Thermodynamics	193003	(193003K1.16) STEAM (CFR: 41.14): Define the following term: -- subcooled and compressed liquids	98	2.7	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	(193009K1.07) CORE THERMAL LIMITS (CFR: 41.14): Describe factors that affect peaking and hot channel factors	100	3.3	75	
	Subtotal			N/A	3	
Tier 4 Point Total				N/A	**6**	

Facility: Comanche Peak		**Date of Examination: June-12-2023**
Examination Level: RO ■ SRO □		**Operating Test Number: NRC 2023 Exam**
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations A1	Perform Manual Calorimetric (NRC 2018 RO1804) 2.1.23 RO 4.3 Ability to perform general and/or normal operating procedures during any plant condition	D, R
Conduct of Operations A2	Determine Alternate Level for the 2-3 SI Accumulator (New - RO1502) 2.1.19 RO 3.9 Ability to use available indications to evaluate system or component status	N, R
Equipment Control A3	Perform 1/M plot and Predict Criticality (NRC 2016 RO1003) 2.2.1 RO 4.5 Ability to perform pre-startup procedures for the facility, including those controls associated with plant equipment that could affect reactivity	D, R
Radiation Control A4	Determine Stay time Requirements (RWT029E modified from June 2019) 2.3.12 RO 3.2 Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters	M, R

Facility: Comanche Peak		**Date of Examination: 6-12-2023**
Examination Level: RO □ SRO ■		**Operating Test Number: NRC 2023 Exam**
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations SA5	Determine Fuel Classification and Position in Spent Fuel Pool (New - SO1035) 2.1.42 SRO 3.4 Knowledge of new and spent fuel movement procedures.	N, R
Conduct of Operations SA6	Determine Alternate Level for the 2-3 SI Accumulator and Determine TS (New - SRO1502) 2.1.19 SRO 3.8 Ability to use available indications to evaluate system or component status	N, R
Equipment Control SA7	Perform AFD calculation and Determine TS (NRC 2015 SO1808) 2.2.12 SRO 4.1 Knowledge of surveillance procedures.	D, R
Radiation Control SA8	Review a Completed Liquid Waste Release Permit (mod from April 2013 NRC SO1039) 2.3.6 SRO 3.8 Ability to approve release permits.	M, R
Emergency Plan SA9	Classify an Event (New - SO1136K) 2.4.41 SRO 4.6 Knowledge of the emergency action level thresholds and classifications	N, R

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)

CP 2023 NRC Exam
NRC Admin JPM Description

RO

RA1 **Perform Manual Calorimetric (RO1804)**
Demonstrate the ability to properly perform a heat balance calculation IAW OPT-309, starting at step 8.2.2.9 C, calculate Unit 1 power level per Step 8.2.2.11.
Direct from Bank (June 2018 NRC Exam)

RA2 **Alternate Level Determination for the Safety Injection Accumulator**
Demonstrate the ability to use procedure SOP-202B, Safety Injection Accumulators Attachment 1, Alternate Level Determination, to determine the current level and the proposed final 7% drain level for accumulator 2-3.
New – RO1502

RA3 **Perform 1/M plot and Predict Criticality**
Demonstrate the ability to perform Inverse Count Rate Ratio (ICRR) Calculation for each of the four rod withdrawal readings per IPO-002B Attachment 2, Create 1/M plot per IPO 002B, Attachment 2. Determine criticality predicted is below Rod Insertion Limit and conclude that all Control Banks should be inserted to CBO position.
Direct from Bank (July 2016 NRC Exam)

RA4 **Determine Stay Time Requirements**
Demonstrate the ability to use procedure STI-211.07, Heat Stress Management, and general radiation principles in order to 1) calculate gamma and neutron dose stay times, 2) calculate heat stress stay time, and 3) determine limiting containment entry time.
Modified from Bank (NRC June 2019 RWT029E)– **The math is different for dose and heat stress stay time and conclusion in June 2019 JPM was that neutron dose was most limiting. In this JPM the heat stress time is most limiting so the answers and conclusion are different.**

SRO

SA5 **Determine Fuel Classification and Position in Spent Fuel Pool**

Demonstrate the ability to calculate the required Minimum Burnup value for Fuel Category 3 IAW TS 3.7.17 and determine fuel assembly X74 fuel burnup meets criteria for Fuel Category 3.

New – SO1035

SA6 **Alternate Level Determination for Safety Injection Accumulator and Apply Tech Specs**

Demonstrate the ability to use procedure SOP-202B, Safety Injection Accumulators Attachment 1, Alternate Level Determination, to determine the current level and the proposed final 7% drain level for accumulator 2-3. SRO must also determine that TS LCO 3.5.1 CONDITION B was not met and REQUIRED ACTION B.1 required the accumulator to be restored to operable status within 24 hours.

New – SO1502

SA7 **Perform Axial Flux Difference Surveillance and determine TS**

Demonstrate ability to perform Axial Flux Difference surveillance and determine that 3 of 4 PR Δ FLUX channels were not within Acceptable Operation region. SRO applicant must also determine that TS LCO 3.2.3 CONDITION A was applicable and REQUIRED ACTION A.1 required THERMAL POWER be restored to < 50% RTP within 30 minutes.

Direct from Bank (June 2015 SO1808)

SA8 **Approve Liquid Release Permit**

Demonstrate the ability to review a Liquid Waste Release Permit IAW STA-603, Control of Station Radioactive Effluents

Modified from Bank (April 2013 SO1039) – **Different math, different tanks being used for release, rad monitor out of service for the 2023 JPM versus in service for 2013 JPM, and added pH assessment and sample ID evaluations for 2023 JPM and these aspects were not part of the 2013 JPM.**

SA9 **Determine Emergency Plan Protective Action Recommendations**

Demonstrate the ability to correctly determine an Emergency Action level using EPP-201, Assessment of Emergency Action Levels, within 15 minutes.

(New - SO1136K)

Facility: Comanche Peak Date of Examination: 6-12-2023

Exam Level: RO ■ SRO-I ■ SRO-U ■ Operating Test No.: NRC

System / JPM Title	Type Code*	Safety Function
Control Room Systems		
a. (S1) Place RCS Makeup in Automatic – uncontrolled dilution occurs (Direct from bank - RO1324B) 004.A2.06 RO 3.9 (SRO-U)	D, A, S	1
b. (S2) Secure from Safety Injection following LOCA (New - RO1519) WE02 EA2.09 RO 4.5	N, ENS, S	2
c. (S3) Wide Range Pressure Channel Failure in Mode 5 (LTOP is active) (RO1824-direct from 2013 NRC Exam) 010 A2.03 RO 4.3 (RO-Only)	D, L, S	3
d. (S4) Transfer to Cold Leg Recirculation from Hot Leg Recirculation (RO1507C -direct from 2017 NRC exam) WE11 EA1.12 RO 4.1 (SRO-U)	D, ENS, L, S	4P
e. (S5) Lower PRT Pressure (New – RO7003) 007 A2.02 RO 3.6	N, S	5
f. (S6) Realign Safeguards Bus 1EA2 to Offsite Power (New – RO4203) 064 A4.07 RO 3.8	N, S	6
g. (S7) Respond to CR Rad Monitor Alarm X-RE-5895 - manually align CR HVAC (Modified from RO4103) 072 A3.01 RO 2.9	M, A, ENS, S	7
h. (S8) Respond to loss of all CW pumps at low power-manually trip turbine and close MSIVs (New – RO3809) 075 A2.02 RO 3.9 (SRO-U)	N, A, L	8
In-Plant Systems		
i. (P1) Local start of TDAFWP during Loss of all Feedwater (New – AO6405) 061 A2.04 RO 4.1 (SRO-U)	N, E, L, R	4S
j. (P2) Transfer Inverter IV2PC1 from Bypass to Normal Operation (AO4204D Direct from 2015 NRC Exam) 063 A3.03 RO 3.3	D	6
k. (P3) Terminate Release of Radioactive Liquid (Modified from RO8063) 068 A2.04 RO 3.3 (SRO-U)	M, A, E, R	9

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

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:

Form 3.2-2 Instructions for Control Room/In-Plant Systems Outline (continued)

- 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			

Control Room Systems JPMs

(S1) **Place RCS Makeup in Automatic – uncontrolled dilution occurs (RO1324B)**
Demonstrate the ability to realign the Reactor Makeup Control System for Automatic operation using SOP-104A. Once the applicant places switch 1/1-MU, RCS MU MAN ACT in START, an uncontrolled dilution occurs and requires manual action to stop the dilution.
Alternate path.
Direct from Bank **(RO1324B)**

(S2) **Secure from SI following a LOCA**
Demonstrate the ability to secure SI pumps following a LOCA in accordance with EOS-1.2A. Want to see the applicant use Table 1 and/or Table 2 to evaluate SCM and determine pumps to secure.
Normal path
(New - RO1519)

(S3) **Wide Range Pressure Channel Failure in Mode 5 (LTOP active)**
Demonstrate the ability to respond to a Wide Range Pressure Transmitter failure while in MODE 5, Utilizing ALM-0053A or ABN-715, and closed the Block Valve associated with the failed open PORV within 8 minutes (2X validation).
Direct from Bank. **(NRC Exam 2013 RO1824)**
RO-only JPM.

(S4) **Transfer to Cold leg Recirculation from Hot leg Recirculation**
During LOCA, transfer back from hot leg injection to cold leg injection IAW EOP-1.4A, Attachment 2.
Normal path.
Direct from bank **(NRC Exam 2017 RO1507C**).

(S5) **Lower PRT Pressure**
Demonstrate the ability to lower PRT pressure using SOP-109A to clear the high-level alarm and reach approximately 3 psig in the PRT.
Normal path.
(New – RO7003)

(S6) **Realign Safeguards Bus 1EA2 to Offsite Power**
Transfer 1EA2 bus from loaded EDG back to offsite power IAW SOP-609A, section 5.7.
(New – RO4203)

(S7) **Respond to CR Radiation Monitor Alarm X-RE-5895(A/B)**
Demonstrate the ability to properly align Control Room HVAC after a high radiation actuation in accordance with ALM-3200, ABN-902 and SOP-902. After valid high radiation alarm occurs on ALM-3200, the applicant recognizes that Emergency Recirculation Automatic Initiation did not occur per step 1.c of ABN-902 and won't work with switch X-ZL-5877A/B, CR EMER RECIRC, so manual alignment per SOP-902 is required.
This is an Alternate Path JPM.
Modified from (**RO4103)**. **In the original JPM, when applicant placed hand switch X-HS-5877 in EMER RECIRC MAN ACT position, the actuation switch actually works and**

repositions dampers and fans. In this JPM the dampers must be manually realigned and fans manually started (the switch does not work).

(S8) **Respond to Loss of all CW Pumps at Low power**

With reactor power at 8% and two Circ water pumps running, applicant is directed to start a third CW pump and when attempted all running CW pumps trip. Applicant must use ABN-304 and trip the turbine and close the MSIV's.

Alternate path.

(New - RO3809)

In Plant Systems JPMs

(P1) **Local start of TDAFWP during Loss of all Feedwater**

Reset and open 1-HV-2452, AFWPT 1-01 TRIP AND THROT VLV per ABN-305, Auxiliary Feedwater System Malfunction, Attachment 1, HV 2452 TDAFW Pump Trip and Throttle Valve

Normal path.

(New – AO6405)

(P2) **Transfer Inverter IV2PC1 from Bypass to Normal Operation**

Utilize SOP-607B, 118 VAC Distribution System and Inverters, Section 5.5.9,

and transfer Inverter IV2PC1 from Bypass to Normal Operation.

Normal path.

Direct from bank **(NRC Exam 2015 AO4204D**).

(P3) **Terminate Release of Radioactive Liquid**

Demonstrate ability to Stop 1-01 RCDT Pump and Start 1-02 RCDT Pump per SOP-110A, Reactor Coolant Drain Tank System, Section 5.8 Alternating Reactor Coolant Drain Tank Pumps. During the surveillance the alarm window 2.6 at rad waste panel locks in high and the applicant must stop the release (either ABN-903 or ALM procedure ALM-301 window 2.6) by locally closing valve XWP-0117, LWPS DISCH HDR VLV 5253 UPSTRM ISOL VLV.

This is an Alternate Path JPM.

Modified from **(RO8063) – In the original JPM it was not started as an RCDT pump swap and the first valve to close (XHS-5253) with the key switch actually worked. In this JPM XHS-5253 fails to close even with close switch in close. Applicant will have to go to the upstream manual valve (valve XWP-117) and close it to terminate the release. This requires walking away from Rad waste panel and locating valve.**

NRC EXAM

Facility:	CPNPP 1 & 2	Scenario No.:	1
Examiners:		Operators:	

Initial Conditions: 100% power MOL.

Turnover: Maintain steady-state power conditions.

Critical Tasks:

- **Manually Insert Rods When Rod Speed Falls Below 48 Steps/Minute Until Reactor Power is Below 5% Power Range Indication and Intermediate Range Channels Indicate Negative Startup Rate Per FRS-0.1A, Response to Nuclear Power Generation/ATWT.**
- **Initiate Emergency Boration During Anticipated Transient Without Trip Prior to Exiting FRS-0.1A, Response to Nuclear Power Generation/ATWT.**
- **Dispatch Operator to Locally Trip Reactor Trip Breakers During Anticipated Transient Without Trip Prior to Exiting FRS-0.1A, Response to Nuclear Power Generation/ATWT.**

Event No.	Malf. No.	Event Type*	Event Description
1	RP05B	(I, MC) RO (I) BOP (TS) SRO	Loop 2 T_C Fails High
2	CV16A	(I, MC) RO	VCT Level Channel LT-112 Fails Low
3	ED05I	(C) BOP, RO (TS) SRO	86-1 Lockout on 1EA2
4	RC03B RP01 RP13C DIED1B41 DIEDT1B4	(C) RO (C) RO, BOP (M) RO, BOP, SRO	1-02 RCP High Shaft Vibration Reactor Fails to Auto Trip (ATWT) Manual Trip Failure from CB-02 and CB-07 Override 1B4-1 Handswitch to Close Override T1B4-1 Handswitch to Close
5	RP14A	(C) RO, BOP	Safety Injection Actuation - Spurious Train A Safety Injection
6	SI04C	(C, MC) BOP	1-01 SI Pump SI Sequencer Fail to Start
7	CS02G	(C, MC) RO	1-03 CS Pump SI Sequencer Fail to Start

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

Table 3.4-1

Actual	Initial Target Attributes
3	Events after EOP entry (1-2)
4	Abnormal events (2-4)
1	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
1	Entry into a contingency EOP with substantial actions – 1 per scenario set
3	Preidentified CTs (2 or more)

Table 3.4-2

Event/Evolution Type	Applicant Position			
	RO	BOP	SRO-I	SRO-U
Reactivity Manipulation	0	0	0	0
Normal Evolution	0	0	0	0
Instrument/Component (I/C) Failure	5	4	7	7
Major Transient	1	1	1	1
Manual Control of Automatic Function	3	1	na	na
TS Evaluation	na	na	2	2

Scenario Objectives:

Demonstrate control of Loop 2 Tcold failure (high)

Demonstrate control of VCT LT-112 failure (low)

Demonstrate control of 86-1 electrical lockout on 1EA2 safety bus

Demonstrate control of RCP 1-02 shaft vibration

Demonstrate control of major event (ATWT)

Demonstrate control of CT 1, 2, and 3 (see CT tables)

Demonstrate control of post-major malfunctions by manually starting 1-01 SI pump and 1-03 CS pump.

SCENARIO SUMMARY

OBJECTIVE: Demonstrate crew competency in responding to abnormal and emergency events which are potential contributors to challenging fission product barriers necessary for the protection of the health and safety of the public.

The crew will assume the watch at 100% power with no scheduled activities per IPO-003A, Power Operations.

The first event is a failure high of Loop 2 T_{Cold} resulting in automatic rod insertion. Crew actions are per ABN-704, Tc/N-16 Instrumentation Malfunction and will include placing rod control in manual and restoring Reactor Coolant System temperature with manual rod withdrawal. The failed channel will be defeated, and rod control should remain in manual. The SROs will refer to Technical Specifications.

The second event is VCT level channel LT-112 failure low. This will result in an automatic makeup. The RO will respond in accordance with ALM-0061A, Alarm Procedure 1-ALB-6A to stop automatic makeup. The crew will refer to ABN-105, Chemical and Volume Control System Malfunction and SOP-104A, Reactor Make-up and Chemical Control System to restore makeup system alignment.

The third event is an 86-1 lockout relay actuation resulting in a loss of 6.9 KV Safeguards Bus 1EA2. The crew will respond per ABN-602, Response to a 6900/480V System Malfunction. Actions include placing Emergency Diesel Generator 1-02 (without Station Service Water flow) in Pull-Out. Additionally, the crew will perform actions per ABN-602 to ensure necessary plant equipment is operating and affected equipment is placed in PULL OUT. The SRO will refer to Technical Specifications.

The major event is initiated by 1-02 RCP high vibration. The crew will initially respond per ABN-101, RCP Trip/Malfunction, which will direct reactor trip. The event is complicated when the reactor fails to trip from either hand switch on the main control board and one of the non-safeguard buses cannot be de-energized to permit tripping the associated rod drive motor generator. The crew will respond per EOP-0.0A, Reactor Trip or Safety Injection, attempt to momentarily de-energize non-safety 480-volt buses, and then transition to FRS-0.1A, Response to Nuclear Power Generation/ATWT. The crew will then insert control rods, emergency borate, start AFW, and dispatch an operator to locally open the reactor trip breakers.

After the reactor is tripped and FRS-0.1A steps are completed, the crew will return to EOP-0.0A. The crew will then transition to EOS-0.1A, Reactor Trip Response when it is determined that a Safety Injection is not required.

After actions are taken in EOS-0.1A, the abnormal event after EOP entry is triggered for a spurious Train A safety injection, which requires re-entry into EOP-0.0A. The response is complicated by failure of 1-01 Safety Injection Pump and 1-03 Containment Spray Pump to start.

Scenario will be terminated when 1-01 SI Pump and 1-03 Containment Spray Pump are restarted or at the discretion of the Lead Examiner.

NRC EXAM
Critical Task Determination

Critical Task	Safety Significance	Initiating Cue	Measurable Performance Standard	Performance Feedback
Ensure Control Rods are inserting at greater than or equal to 48 Steps / Minute until Reactor Power is below 5% Power Range indication and Intermediate Range Channels indicate Negative Startup Rate, per FRS-0.1A, Response to Nuclear Power Generation / ATWT.	Take actions to insert Control Rods at the fastest rate speed available to prevent challenges to plant's heat removal capability and fuel integrity.	Procedurally driven from FRS-0.1A to reduce the heat input to the RCS to only that of decay heat and reactor coolant pump heat.	If the reactor is not tripped, the intent of the RNO actions is to insert rods at fastest rate available. If RCS temperature increased above current reference temperature, then rods should automatically be driven in by Rod Control System. When Rod Control System signal decreases to less than speed for manual rod insertion, operator is required to manually insert rods to maximize negative reactivity addition.	Neutron flux lowering and indication of rods driving into the core by DRPI and step counters.
Initiate Emergency Boration During Anticipated Transient Without Trip Prior to Exiting FRS-0.1A, Response to Nuclear Power Generation / ATWT.	Take one or more actions that would prevent challenges to plant's heat removal capability and fuel integrity.	Procedurally driven from FRS-0.1A to add negative reactivity to the core.	The operator will realign charging pump suction to the RWST and raise charging flow to maximum.	Valve position indication and charging flow indication on CB-06.
Dispatch Operator to Locally Trip Reactor Trip Breakers During Anticipated Transient Without Trip Prior to Exiting FRS-0.1A, Response to Nuclear Power Generation / ATWT.	Recognize a failure or an incorrect automatic actuation of an ESF system or component.	Procedurally driven from FRS-0.1A to add negative reactivity to the core and properly align the P-4 signal.	The operator will contact an operator outside the control room to locally trip the reactor trip breakers.	Indication of reactor trip breaker position on CB--07.

NOTE: If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.

SIMULATOR OPERATOR INSTRUCTIONS for SIMULATOR SETUP

Initialize to IC 18 and 2023 NRC Simulator Scenario 1 Scenario File.

EVENT	TYPE	MALF #	DESCRIPTION	DEMAND VALUE	INITIATING PARAMETER
SETUP					
2	COND		Remove LT-112 Failure {Key[15]!=0} DMF CV16A		K15
3	RF	EAR101	De-energize Train B Sequencer	f:0	K16
3	RF	EAR498	De-energize Train B Sequencer	f:0	K16
4	IMF	RP01	Reactor Fails to Auto Trip	f:1	K0
4	IMF	RP13C	Manual Trip Failure from CB-02 and CB-07	f:1	K0
4	RF	RPR112	Open Reactor Trip Breakers A	f:2	K17
4	RF	RPR113	Open Reactor Trip Breakers B	f:2	K17
4	IOR	DIED1B41	Override 1B4-1 Handswitch to Close	f:3	K0
4	IOR	DIEDT1B4	Override T1B4 Handswitch to Close	f:3	K0
6	IMF	SI04C	1-01 SI Pump SI Sequencer Fail to Start	f:1	K0
7	IMF	CS02G	1-03 CS Pump SI Sequencer Fail to Start	f:1	K0
1	IMF	RP05B	Loop 2 T_c NR Temperature (TE-421B) Fails High	f:630	K1
2	IMF	CV16A	VCT Level Channel LT-112 Fails Low	f:0	K2
3	IMF	ED05I	86-1 Lockout on 1EA2	f:1	K3
4	IMF	RC03B	1-02 RCP High Shaft Vibration	f:21 r:120	K4
5	IMF	RP14A	Spurious Train A Safety Injection	f:1	K5

Simulator Operator: **Initialize to IC 18 and LOAD 2023 NRC Simulator Scenario 1 Scenario File.**

ENSURE procedures in progress are on SRO desk:
- **COPY of IPO-003A, Power Operations, Section 5.5 Operating at Constant Turbine Load.**

Control Room Annunciators in Alarm:

PCIP-1.1 – SR TRN A RX TRIP BLK
PCIP-1.2 – IR TRN A RX TRIP BLK
PCIP-1.4 – CNDSR AVAIL STM DMP ARMED C-9
PCIP-1.6 – RX ≥ 10% PWR P-10
PCIP-2.1 – SR TRN B RX TRIP BLK
PCIP-2.2 – IR TRN B RX TRIP BLK
PCIP-2.5 – SR RX TRIP BLK PERM P-6
PCIP-3.2 – PR TRN A LO SETPT RX TRIP BLK
PCIP-4.2 – PR TRN B LO SETPT RX TRIP BLK

NRC EXAM

Facility:	CPNPP 1 & 2	Scenario No.:	2
Examiners:		Operators:	

Initial Conditions: 100% power MOL.

Turnover: Maintain steady-state power conditions. Place Train B Control Room Ventilation in service.

Critical Tasks:

- **Manually Align Containment Spray Due to Failure to Automatically Align Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.**
- **Dispatch Operator to Locally Align Train B Containment Isolation Phase B RCP Thermal Barrier CCW Return Valve Due to Failure to Automatically Align Prior To Exiting EOP-0.0A, Reactor Trip.**
- **Manually Start 1-02 RHR pump prior to completing Attachment 2 of EOP-0.0A, Reactor Trip or Safety Injection.**

Event No.	Malf. No.	Event Type*	Event Description
1		(N) BOP	Place Train B Control Room Ventilation in service.
2	CV37	(C, MC) RO	1-PK-0131 Fails Causing PCV-0131 to Open
3	RP03H	(I, MC) BOP (TS) SRO	1-03 Steam Line Pressure Transmitter PT-535 Channel II Fails Low
4	CV31A	(C) RO (TS) SRO	1-01 Centrifugal Charging Pump Sheared Shaft
5	RC09A2 ED05H	(M) RO, BOP, SRO (C) BOP	RCS CL Loop 1 LBLOCA Double Ended Loss of 6.9kV Bus 1EA1 (86-1)
6	CS07B	(C) RO	1-02 Containment Spray Heat Exchanger Outlet Valve 1-HV-4777 Fails to Automatically Open
7	RH01D	(C, MC) BOP	1-02 RHR Pump SI Sequencer Fail to Start
8	IOR DICCHS4709 CC05A	(C, MC) RO	Override1-HS-4709 Thermal Barrier Cooler CCW Return Isolation Valve Handswitch to Open Thermal Barrier Cooler Leakage Loop 1

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

Table 3.4-1

Actual	Initial Target Attributes
3	Events after EOP entry (1-2)
2	Abnormal events (2-4)
1	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
0	Entry into a contingency EOP with substantial actions – 1 per scenario set
3	Preidentified CTs (2 or more)

Table 3.4-2

Event/Evolution Type	Applicant Position			
	RO	BOP	SRO-I	SRO-U
Reactivity Manipulation	0	0	0	0
Normal Evolution	0	1	1	1
Instrument/Component (I/C) Failure	4	3	7	7
Major Transient	1	1	1	1
Manual Control of Automatic Function	2	2	na	na
TS Evaluation	na	na	2	2

Scenario Objectives:

Demonstrate control of shifting control room ventilation

Demonstrate control of failure of PK-0131

Demonstrate control of 1-01 CCP shaft shear

Demonstrate control of Loss of 1EA1 bus

Demonstrate control of major (LBLOCA)

Demonstrate control of CT 1, 2, and 3 (see CT tables)

Demonstrate control of post-major malfunctions as required by CT tables since they are incorporated into these malfunctions.

SCENARIO SUMMARY

OBJECTIVE: Demonstrate crew competency in responding to abnormal and emergency events which are potential contributors to challenging fission product barriers necessary for the protection of the health and safety of the public.

The crew will assume the watch at 100% power per IPO-003A, Power Operations with instructions to place Train B Control Room Ventilation in service.

The first event is realignment of Control Room Ventilation system to satisfy equipment rotation requirements per OWI-409, Equipment Rotation Program. Crew actions will be to shift alignment of Control Room Ventilation from Train A in service to Train B in service per SOP-802, Control Room Ventilation System.

The second event is failure of 1-PCV-131, U1 LTDN HX OUT PRESS CTRL VLV. The control valve slowly fails open resulting in loss of letdown pressure control. The crew will respond per ALM-0061A, Alarm Procedure 1-ALB-6A. Actions include placing 1-PK-131, LTDN HX OUT PRESS CTRL in manual, adjusting letdown pressure to approximately 310 psig, and dispatching personnel to determine cause of the failure. Once the cause is corrected, 1-PK-131 is restored to automatic control.

The third event is a failure low of 1-03 Main Steam Line Pressure Instrument 1-PT-0535. Crew actions are in accordance with ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, and Feed Header Pressure Instrument Malfunction, Section 2.0. Actions include taking manual control of 1-FCV-0530, stabilizing SG 1-03 level, and transferring to an alternate steam flow controlling channel. SG level control will be restored to automatic control. The SRO will refer to Technical Specifications.

The next event is a sheared shaft of the running Centrifugal Charging Pump (CCP). When low flow alarms are received, initial operator actions of ABN-105, Chemical and Volume Control System Malfunction, Section 3.0 will be performed to start the standby CCP. Any delay in determining cause of the low charging flow will result in letdown isolation, which will be restored prior to proceeding. The SRO will refer to Technical Specifications.

The major event is a Loop 1 RCS cold leg large break LOCA. The response is complicated by a loss of Train A electrical bus, failure of 1-02 RHR pump automatic start, failure of 1-02 Containment Spray heat exchanger outlet valve to automatically open, and failure of Phase B to isolate RCP thermal barrier CCW return Containment isolation valve. The crew will initially respond per EOP-0.0A, Reactor Trip or Safety Injection and transition to EOP-1.0A, Loss of Reactor or Secondary Coolant. The Unit Supervisor will address safety functions by entry into and out of FRP-0.1A, Response to Imminent Pressurized Thermal Shock Condition, depending on plant conditions, and entry into FRZ-0.1A, Response to High Containment Pressure.

Scenario will be terminated when the crew has transitioned to EOP-1.0A or at the discretion of the Lead Examiner.

NRC EXAM
Critical Task Determination

Critical Task	Safety Significance	Initiating Cue	Measurable Performance Standard	Performance Feedback
Manually Align Containment Spray Due to Failure to Automatically Align Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.	Recognize failure of an automatic actuation of an ESF component. Containment Spray is required to reduce challenge to Containment integrity.	Procedurally driven from EOP-0.0A, which provides instructions for manual alignment of Containment Spray upon failure of automatic actuation.	The operator will manually open 1-HV-4777 1-02 Containment Spray Heat Exchanger Outlet Valve.	Indication of valve position including light indication, Containment Spray flow, and lowering Containment pressure.
Dispatch Operator to Locally Align Train B Containment Isolation Phase B RCP Thermal Barrier CCW Return Valve Due to Failure to Automatically Align Prior To Exiting EOP-0.0A, Reactor Trip or Safety Injection.	Recognize failure of an automatic actuation of an ESF component. Phase B isolation is required to isolate Containment atmosphere from the environment to preclude radioactive release.	Procedurally driven from EOP-0.0A, which provides instructions for manual alignment of Phase B valves upon failure of automatic actuation.	The operator will dispatch an operator to locally close 1-HV-4709 Thermal Barrier Cooler CCW Return Isolation Valve.	Report from the field that the valve is closed and MLB indication for valve position on CB-02.
Manually Start 1-02 RHR pump prior to completing Attachment 2 of EOP-0.0A, Reactor Trip or Safety Injection.	Recognize a failure of automatic actuation of an ESF component. RHR flow is required to remove decay heat following a large break LOCA.	Procedurally driven from EOP-0.0A, Attachment 2 to provide adequate ECCS flow during accident conditions.	The operator will start 1-02 RHR Pump using the handswitch on CB-04.	Indication of pump start including light indication, flow and discharge pressure on CB-04.

NOTE: If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.

SIMULATOR OPERATOR INSTRUCTIONS for SIMULATOR SETUP

Initialize to IC 18 and 2023 NRC Simulator Scenario 2 Scenario File.

EVENT	TYPE	MALF #	DESCRIPTION	DEMAND VALUE	INITIATING PARAMETER
ETUP					
2	COND		Remove 1-PK-0131 Failure {Key[11]!=0} DMF CV37		K11
4	RF	CVR05	Stop CCP 1-01 Auxiliary Lube Oil Pump	f:0	K12
4	RF	CVR06	CCP 1-02 Auxiliary Lube Oil Pump to AUTO	f:1	K12
6	IMF	CS07B	1-02 Containment Spray Heat Exchanger Outlet Valve Fails to Automatically Open	f:1	K0
7	IMF	RH01D	1-02 RHR Pump SI Sequencer Fail to Start	f:1	K0
8	IOR	DICCHS4709	Override1-HS-4709 Handswitch to Open	f:2	K0
8	IMF	CC05A	Thermal Barrier Cooler Leakage Loop 1	f:5	K5
8	IRF	CCR43	1-HV-4709 Thermal Barrier Cooler CCW Return Isolation Valve Local Breaker Off	f:0	K13
8	IRF	CCR33	1-HV-4709 Thermal Barrier Cooler CCW Return Isolation Valve Manually Close	f:0	K13
2	IMF	CV37	1-PK-0131 Fails Causing PCV-0131 to Open	i:17 f:100 r:400	K2
3	IMF	RP03H	1-03 Steam Line Pressure Transmitter PT-535 Channel II fails low	f:0	K3
4	IMF	CV31A	1-01 Centrifugal Charging Pump Sheared Shaft	f:1	K4
5	IMF	RC09A2	RCS CL Loop 1 LBLOCA Double Ended	f:1	K5
5	IMF	ED05H	Loss of 6.9kV Bus 1EA1 (86-1)	f:1	K5 + 5

Simulator Operator: INITIALIZE to IC 18 and LOAD 2023 NRC Simulator Scenario 2 Scenario File. **ENSURE procedures in progress are on SRO desk:** **- COPY of IPO-003A, Power Operations, Section 5.5 Operating at Constant Turbine Load.** **- Ensure Train A Control Room Ventilation System is in service per SOP-802, Control Room Ventilation System.** **- Perform light check on CR HVAC panel to ensure no fan and damper lights are burned out.** **- Ensure a copy of SOP-802 Section 5.3.10 is available for markup.**
Control Room Annunciators in Alarm: **CIP-1.1 – SR TRN A RX TRIP BLK** **CIP-1.2 – IR TRN A RX TRIP BLK** **CIP-1.4 – CNDSR AVAIL STM DMP ARMED C-9** **CIP-1.6 – RX ≥ 10% PWR P-10** **CIP-2.1 – SR TRN B RX TRIP BLK** **CIP-2.2 – IR TRN B RX TRIP BLK** **CIP-2.5 – SR RX TRIP BLK PERM P-6** **CIP-3.2 – PR TRN A LO SETPT RX TRIP BLK** **CIP-4.2 – PR TRN B LO SETPT RX TRIP BLK**

NRC EXAM

Facility: CPNPP 1 & 2 Scenario No.: 3

Examiners: _____ Operators: _____

_____ _____

_____ _____

Initial Conditions: 100% power EOL.

Turnover: Maintain steady-state power conditions.

Critical Tasks:

- Place EHC Pumps in PULL OUT or Manually Close MSIVs Upon Failure of Main Turbine Trip Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.
- Manually close Main Steam Isolation Valves Upon Failure of Automatic Closure Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.
- Identify and Isolate 1-03 Steam Generator Prior to Exiting EOP-2.0, Faulted Steam Generator Isolation.

Event No.	Malf. No.	Event Type*	Event Description
1	FW16 FW17A IOR DIFWHS2958	(R) RO, SRO (C, MC) BOP, SRO	Condenser Vacuum Leak 1-01 CEV Pump Trip 1-03 CEV Pump Fails to Auto Start
2	RX05A	(I, MC) RO, SRO (TS) SRO	Pressurizer LT-459 Fails Low
3	NI06E	(I, MC) RO, SRO (C) BOP (TS) SRO	PR Channel NI-44 Fails High
4	FW14B TC09I	(C, MC) BOP (R) RO (TS) SRO	1-02 Heater Drain Pump Trips Auto Turbine Runback Failure
5	MS01C	(M) RO, BOP, SRO	1-03 SG Fault IRC
6	TC07C	(C, MC) BOP, SRO	Main Turbine Fails to Auto Trip and Fails to Manually Trip from Pushbutton, Requires Stopping EHC Pumps
7	IOR DIFWFK2454A_1	(C) BOP	1-FK-2454A MD AFWP 2 SG 3 FLO CTRL Fails in AUTO (will not close)
8	SS02A1 SS02A2	(C, MC) RO	SSPS Train A Master Relay K504 Failure (MSIV fails to auto close) SSPS Train B Master Relay K504 Failure (MSIV fails to auto close)

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

Table 3.4-1

Actual	Initial Target Attributes
3	Events after EOP entry (1-2)
4	Abnormal events (2-4)
1	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
0	Entry into a contingency EOP with substantial actions – 1 per scenario set
3	Preidentified CTs (2 or more)

Table 3.4-2

Event/Evolution Type	Applicant Position			
	RO	BOP	SRO-I	SRO-U
Reactivity Manipulation	2	2	2	2
Normal Evolution	0	0	0	0
Instrument/Component (I/C) Failure	4	5	7	7
Major Transient	1	1	1	1
Manual Control of Automatic Function	3	3	na	na
TS Evaluation	na	na	3	3

Scenario Objectives:

Demonstrate control of condenser vacuum leak

Demonstrate control of PZR LT-459 failure (low)

Demonstrate control PR NI-44 failure (high)

Demonstrate control of 1-02 HTR DRN Pump trip and auto runback failure on main turbine

Demonstrate control of major (SG Fault IRC)

Demonstrate control of CT 1, 2, and 3 (see CT tables)

Demonstrate control of post-major malfunctions by stopping EHC pumps, isolating AFW flow for faulted AFW flow control valve, and manually closing MSIVs when SSPS relays fail to auto close them

NRC EXAM
SCENARIO SUMMARY

OBJECTIVE: Demonstrate crew competency in responding to abnormal and emergency events which are potential contributors to challenging fission product barriers necessary for the protection of the health and safety of the public.

The crew will assume the watch at 100% power with no scheduled activities per IPO-003A, Power Operations.

The first event is trip of one of the operating Condenser Exhaust Vacuum (CEV) pumps and a failure of the standby CEV to automatically start. A condenser vacuum leak will cause lowering megawatts and lowering condenser vacuum. The crew will enter ABN-304, Main Condenser and Circulating Water System Malfunction Section 3.0 and manually start the standby CEV pump. The crew will reduce turbine load to ensure Reactor Power is maintained less than 100% power.

The second event is a failure low of Pressurizer Level channel. Crew actions are per ABN-706, Pressurizer Level Instrumentation Malfunction and include manual control of pressurizer level, bypassing the failed channel, and restoring pressurizer level control to automatic. The crew will be required to restore normal letdown to service due to letdown isolation. The SRO will refer to Technical Specifications.

The third event is a failure high of Power Range Nuclear Instrument NI-44. The Reactor Operator will verify no transient in progress and place Rod Control in manual to stop unnecessary rod motion. The crew will take the actions of ABN-703, Power Range Instrumentation Malfunction. The SRO will refer to Technical Specifications.

The fourth event is a trip of Heater Drain Pump 1-02 with an automatic turbine runback failure. Crew actions are per ABN-302, Feedwater, Condensate, Heater Drain System Malfunction Section 4.0 and will include placing control rods in automatic, manually initiating a turbine runback, and initiation of a boration. The SROs will refer to Technical Specifications when control rods insert below the Rod Insertion Limit.

The major event is a SG fault inside Containment. Response is complicated by failure of automatic and manual main turbine trip, failure of MSIVs to automatically close (SSPS relay failure, manually closed), and failure of 1-03 SG MD AFW flow control valve (won't control in manual and requires shutting AFW isolation valve for SG 1-03). The crew will enter EOP-0.0A, Reactor Trip or Safety Injection, determine SG 1-03 is faulted, and transition to EOP-2.0A, Faulted Steam Generator Isolation. Once the SG is isolated, the crew will transition to EOS-1.1A, Safety Injection Termination.

Scenario will be terminated when EOS-1.1A, Safety Injection Termination Attachment 1.J is complete or at the discretion of the Lead Examiner.

NRC EXAM
Critical Task Determination

Critical Task	Safety Significance	Initiating Cue	Measurable Performance Standard	Performance Feedback
Place EHC Pumps in PULL OUT or Manually Close MSIVs Upon Failure of Main Turbine Trip Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.	Recognize a failure or an incorrect automatic actuation of an ESF system or component.	Procedurally driven from EOP-0.0A. Tripping the main turbine following a reactor trip prevents an uncontrolled cooldown of the RCS.	The operator will place all EHC pump handswitches on CB-10 in PULL OUT and verify turbine HP stop valves close. If HP stop valves fail to close, the operator will close all MSIVs.	Indication of pump stop including light indication and EHC fluid pressure, HP stop valve light indication or MSIV light indication.
Manually close Main Steam Isolation Valves Upon Failure of Automatic Closure Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.	Recognize a failure or an incorrect automatic actuation of an ESF system or component.	Procedurally driven from EOP-0.0A. MSIVs are closed to minimize mass and energy release associated with a high energy secondary line break inside Containment.	The operator will close the MSIVs from CB-08 using the MSL isolation manual actuation handswitches or the individual MSIV handswitches.	Indication of MSIV position including light indication, Steam Generator steam flow, and Steam Generator pressure trend.
Identify and Isolate 1-03 Steam Generator Prior to Exiting EOP-2.0, Faulted Steam Generator Isolation.	Take one or more actions that would prevent a challenge to plant safety.	Procedurally driven from EOP-2.0A to isolate the faulted SG to prevent further RCS cooldown and mass and energy release. AFW flow indicated to SG 1-03 which is faulted.	The operator will close the AFW motor operated isolation valve to SG 1-03 OR stop 1-02 MDAFW pump and close 1-FCV-2461A, TD AFWP SG 3 FLO CTRL VLV.	Valve position will change and AFW flow to SG 1-03 will reduce to zero.
NOTE: If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.				

SIMULATOR OPERATOR INSTRUCTIONS for SIMULATOR SETUP

Initialize to IC 20 and 2023 NRC Simulator Scenario 3 Scenario File.

EVENT	TYPE	MALF #	DESCRIPTION	DEMAND VALUE	INITIATING PARAMETER
SETUP					
1	IOR	DIFWHS2958	1-03 CEV Pump Fails to Auto Start	f:0	K0 Note 1 and 2
1	COND		Allow Manual Start of 1-03 CEV Pump {DIFWHS2958.Value=2} MOR DIFWHS2958 f:2		K0
1	COND		Reduce Condenser Vacuum Leak When 1-03 CEV Pump is Started {DIFWHS2958.Value=2} MMF FW16 f:8		K0 Note 2
1	COND		Remove Condenser Vacuum Leak {Key[11]!=0} DMF FW16		K11
4	IMF	TC09I	Auto Turbine Runback Failure	f:1	K0
5	IRF	EDR73	Reset 1-01 IAC	f:2	K12
5	IRF	EDR74	Reset 1-02 IAC	f:2	K13
5	IRF	IAR07	Reset X-01 IAC	f:2	K14
6	IMF	TC07C	Main Turbine Automatic and Manual Trip Fails (Pushbutton will not trip)	f:1	K0
7	COND		1-FK-2454A MD AFWP 2 SG 3 FLO CTRL Fails in AUTO (will not close) {LOFWFK2454A_1.Value=1} IOR DIFWFK2454A_1 f:1		K0
8	IMF	SS02A1	SSPS Train A Master Relay K504 Failure (MSIVs fail to auto close)	f:1	K0
8	IMF	SS02A2	SSPS Train A Master Relay K504 Failure (MSIVs fail to auto close)	f:1	K0
1	IMF	FW16	Main Condenser Vacuum Leak	f:16	K1 Note 2
1	IMF	FW17A	1-01 CEV Pump Trip	f:1	K1
2	IMF	RX05A	Pressurizer LT-459 Fails Low	f:0	K2
3	IMF	NI06E	PR N44 Channel Fails High	f:200	K3

4	IMF	FW14B	1-02 Heater Drain Pump Trip	f:1	K4
5	IMF	MS01C	1-03 SG Fault IRC	f:0.4	K5

Note 1: {DIFWHS2958.Value=2} MOR DIFWHS2958 f:2 – Allow 1-03 CEV Pump to be manually started with the handswitch
Note 2: {DIFWHS2958.Value=2} MMF FW16 f:8 – Reduce condenser vacuum leak when 1-03 CEV Pump is manually started to allow CEVs to maintain vacuum

Simulator Operator: **Initialize to IC 20 and LOAD 2023 NRC Simulator Scenario 3 Scenario File.**

ENSURE procedures in progress are on SRO desk:
- **COPY of IPO-003A, Power Operations, Section 5.5 Operating at Constant Turbine Load.**

ENSURE EOL Reactivity Briefing Sheet available for crew

Control Room Annunciators in Alarm:

PCIP-1.1 – SR TRN A RX TRIP BLK
PCIP-1.2 – IR TRN A RX TRIP BLK
PCIP-1.4 – CNDSR AVAIL STM DMP ARMED C-9
PCIP-1.6 – RX ≥ 10% PWR P-10
PCIP-2.1 – SR TRN B RX TRIP BLK
PCIP-2.2 – IR TRN B RX TRIP BLK
PCIP-2.5 – SR RX TRIP BLK PERM P-6
PCIP-3.2 – PR TRN A LO SETPT RX TRIP BLK
PCIP-4.2 – PR TRN B LO SETPT RX TRIP BLK