Facility: Diablo Canyon   Date of Exam: April, 2016

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	2	3	3	N/A			4	3	N/A		3	18			6
	2	2	1	1				1	2			2	9			4
	Tier Totals	4	4	4				5	5			5	27			10
2. Plant Systems	1	3	2	3	2	2	2	2	3	3	3	3	28			5
	2	1	1	1	1	0	1	2	1	1	1	0	10			3
	Tier Totals	4	3	3	3	3	3	4	4	4	4	3	38			8

3. Generic Knowledge and Abilities Categories	1	2	3	4	Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	3	3	2	2	10					7

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401 PWR Examination Outline Form ES-401-2

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

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1						X	G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	39.
~~000008 Pressurizer Vapor Space Accident / 3~~									
000009 Small Break LOCA / 3					X		EA2.11 Ability to determine or interpret the following as they apply to a small break LOCA: Containment temperature, pressure, and humidity	3.8	40.
000011 Large Break LOCA / 3				X			EA1.04 Ability to operate and monitor the following as they apply to a Large Break LOCA: ESF actuation system in manual	4.4	41.
000015/17 RCP Malfunctions / 4		X					AK2.07 Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP seals	2.9	42.
000022 Loss of Rx Coolant Makeup / 2	X						AK1.02 Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Relationship of charging flow to pressure differential between charging and RCS	2.7	43.
000025 Loss of RHR System / 4		X					AK2.05 Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Reactor building sump	2.6	44.
000026 Loss of Component Cooling Water / 8			X				AK3.04 Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Effect on the CCW flow header of a loss of CCW	3.5	45.
000027 Pressurizer Pressure Control System Malfunction / 3			X				AK3.03 Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Actions contained in EOP for PZR PCS malfunction	3.7	46.
000029 ATWS / 1				X			EA1.02 Ability to operate and monitor the following as they apply to a ATWS: Charging pump suction valves from RWST operating switch	3.6	47.
000038 Steam Gen. Tube Rupture / 3						X	G2.1.28 Knowledge of the purpose and function of major system components and controls.	4.1	48.
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4					X		AA2.02 Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring a reactor trip	4.6	49.
000054 (CE/E06) Loss of Main Feedwater / 4 L081 #49				X			AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): AFW controls, including the use of alternate AFW sources	4.5	50.

000055 Station Blackout / 6	X						EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout : Natural circulation cooling	4.1	51.
~~000056 Loss of Off-site Power / 6~~									
000057 Loss of Vital AC Inst. Bus / 6			X				AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus	4.1	52.
000058 Loss of DC Power / 6				X			AA1.02 Aility to operate and / or monitor the following as they apply to the Loss of DC Power: Static inverter dc input breaker, frequency meter, ac output breaker, and ground fault detector	3.1	53.
~~000062 Loss of Nuclear Svc Water / 4~~									
~~000065 Loss of Instrument Air / 8~~									
W/E04 LOCA Outside Containment / 3		X					EK2.1 Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.5	54.
W/E11 Loss of Emergency Coolant Recirc. / 4					X		EA2.2 Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation). Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.4	55.
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4						X	G2.4.18 Knowledge of the specific bases for EOPs.	3.3	56.
~~000077 Generator Voltage and Electric Grid Disturbances / 6~~									
K/A Category Totals:	2	3	3	4	3	3	Group Point Total:		18

ES-401 PWR Examination Outline Form ES-401-2
Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
~~000001 Continuous Rod Withdrawal / 1~~									
~~000003 Dropped Control Rod / 1~~									
~~000005 Inoperable/Stuck Control Rod / 1~~									
000024 Emergency Boration / 1						X	G2.4.6 Knowledge of EOP mitigation strategies.	3.7	57.
000028 Pressurizer Level Malfunction / 2					X		AA2.09 Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Charging and letdown flow capacities	2.9	58.
~~000032 Loss of Source Range NI / 7~~									
~~000033 Loss of Intermediate Range NI / 7~~									
~~000036 (BW/A08) Fuel Handling Accident / 8~~									
~~000037 Steam Generator Tube Leak / 3~~									
~~000051 Loss of Condenser Vacuum / 4~~									
~~000059 Accidental Liquid RadWaste Rel. / 9~~									
~~000060 Accidental Gaseous Radwaste Rel. / 9~~									
~~000061 ARM System Alarms / 7~~									
~~000067 Plant Fire On-site / 8~~									
~~000068 (BW/A06) Control Room Evac. / 8~~									
~~000069 (W/E14) Loss of CTMT Integrity / 5~~									
000074 (W/E06&E07) Inad. Core Cooling / 4			X				**E06:** Degraded core cooling, EK3.3Knowledge of the reasons for the following responses as they apply to degraded core cooling: Manipulation of controls required to obtain the desired operating results during abnormal and emergency situations.	4.0	59.
000076 High Reactor Coolant Activity / 9					X		AA2.01 Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Location or process point that is causing an alarm	2.7	60.
W/EO1 & E02 Rediagnosis & SI Termination / 3		X					E02 SI termination: EK2.2 Knowledge of the interrelations between SI termination and the following: Facility's heat removal systems, incl primary coolant, emergency coolant, decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	3.5	61.
W/E13 Steam Generator Over-pressure / 4	X						EK1.3 Knowledge of the operational implications of the following concepts as they apply to the (Steam Generator Overpressure). Annunciators and conditions indicating signals, and remedial actions associated with the (Steam Generator Overpressure).	3.0	62.

W/E15 Containment Flooding / 5 Bank P-85720	X						EK1.2 Knowledge of the operational implications of the following concepts as they apply to the (Containment Flooding): Normal, abnormal, and emergency operating procedures associated with containment flooding.	2.7	63.
W/E16 High Containment Radiation / 9						X	G2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	3.8	64.
~~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 - Depress. / 4~~									
~~BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4~~									
~~BW/E13&E14 EOP Rules and Enclosures~~									
CE/A11; W/E08 RCS Overcooling - PTS / 4				X			AA1.2 Ability to operate and / or monitor the following as they apply to the (RCS Overcooling). Operating behavior characteristics of the facility.	3.2	65.
~~CE/A16 Excess RCS Leakage / 2~~									
~~CE/E09 Functional Recovery~~									
K/A Category Point Totals:	2	1	1	1	2	2			9

ES-401 PWR Examination Outline Form ES-401-2
Plant Systems - Tier 2/Group 1 (RO / SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump									X			A3.05 Ability to monitor automatic operation of the RCPS, including: RCP lube oil and bearing lift pumps	2.7	1.
004 Chemical and Volume Control							X					A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: VCT level	3.0	2.
005 Residual Heat Removal	X					X						K1.12 Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: Safeguard pumps	3.1	3.
												K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger	2.5	4.
006 Emergency Core Cooling									X	X		A3.08 Ability to monitor automatic operation of the ECCS, including: Automatic transfer of ECCS flowpaths	4.2	5.
												A4.08 Ability to manually operate and/or monitor in the control room: ESF system, including reset	4.2	6.
007 Pressurizer Relief/Quench Tank											X	G2.1.20 Ability to interpret and execute procedure steps.	4.6	7.
008 Component Cooling Water			X									K3.01 Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS	3.4	8.
010 Pressurizer Pressure Control								X				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures	3.9	9.
012 Reactor Protection					X							K5.01 Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB	3.3	10.
013 Engineered Safety Features Actuation						X				X		K6.01 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors	2.7	11.

												A4.02 Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels	4.3	12.
022 Containment Cooling	X											K1.01 Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system	3.5	13.
~~025 Ice Condenser~~														
026 Containment Spray				X							X	K4.06 Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Iodine scavenging via the CSS	2.8	14.
												G2.4.11 Knowledge of abnormal condition procedures.	4.0	15.
039 Main and Reheat Steam – steam dump?	X											K1.04 Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: RCS temperature monitoring and control	3.1	16.
059 Main Feedwater							X					A1.03 Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: Power level restrictions for operation of MFW pumps and valves	2.7	17.
061 Auxiliary/Emergency Feedwater					X				X			K5.01 Knowledge of the operational implications of the following concepts as the apply to the AFW: Relationship between AFW flow and RCS heat transfer	3.6	18.
												A3.04 Ability to monitor automatic operation of the AFW, including: Automatic AFW isolation	4.1	19.
062 AC Electrical Distribution								X				A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Methods for energizing a dead bus	2.9	20.
063 DC Electrical Distribution		X										K2.01 Knowledge of bus power supplies to the following: Major DC loads	2.9	21.

064 Emergency Diesel Generator New higher								X		X		A2.16 Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of offsite power during full-load testing of ED/G	3.3	22.
												A4.07 Ability to manually operate and/or monitor in the control room: Transfer ED/G (with load) to grid	3.4	23.
073 Process Radiation Monitoring			X									K3.01 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases	3.6	24.
076 Service Water		X										K2.01 Knowledge of bus power supplies to the following: Service water	2.7	25.
078 Instrument Air				X								K4.01 Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control	2.7	26.
103 Containment			X								X	K3.03 Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under refueling operations	3.7	27.
												G2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	3.8	28.
K/A Category Point Totals:	3	2	3	2	2	2	2	3	3	3	3	Group Point Total:		28

ES-401 PWR Examination Outline Form ES-401-2
Plant Systems - Tier 2/Group 2 (RO / SRO)

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	#
001 Control Rod Drive									X			A3.02 Ability to monitor automatic operation of the CRDS, including: Rod height	3.7	29.
002 Reactor Coolant								X				A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of forced circulation	4.1	30.
011 Pressurizer Level Control										X		A4.04 Ability to manually operate and/or monitor in the control room: Transfer of PZR LCS from automatic to manual control	3.2	31.
014 Rod Position Indication														
015 Nuclear Instrumentation						X						K6.02 Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Discriminator/compensation circuits	2.6	32.
016 Non-nuclear Instrumentation														
017 In-core Temperature Monitor			X									K3.01 Knowledge of the effect that a loss or malfunction of the ITM system will have on the following: Natural circulation indications	3.5	33.
027 Containment Iodine Removal		X										K2.01 Knowledge of bus power supplies to the following: Fans	3.1	34.
028 Hydrogen Recombiner and Purge Control							X					A1.02 Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Containment pressure	3.4	35.
029 Containment Purge														
033 Spent Fuel Pool Cooling														
034 Fuel Handling Equipment														
035 Steam Generator	X											K1.01 Knowledge of the physical connections and/or cause-effect relationships between the S/GS and the following systems: MFW/AFW systems	4.2	36.
041 Steam Dump/Turbine Bypass Control														
045 Main Turbine Generator														
055 Condenser Air Removal														
056 Condensate														
068 Liquid Radwaste														

071 Waste Gas Disposal														
072 Area Radiation Monitoring							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: Radiation levels	3.4	37.
075 Circulating Water														
079 Station Air				X								K4.01 Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: Cross-connect with IAS	2.9	38.
086 Fire Protection														
K/A Category Point Totals:	1	1	1	1	0	1	2	1	1	1	0	Group Point Total:		10

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations redsox16	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	2.9			66
	2.1.19	Ability to use plant computers to evaluate system or component status.	3.9			67
	2.1.29	Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	4.1			68
	2.1.					
	2.1.					
	2.1.					
2. Equipment Control	2.2.3	Knowledge of the design, procedural, and operational differences between units.	3.8			69
	2.2.12	Knowledge of surveillance procedures.	3.7			70
	2.2.22	Knowledge of limiting conditions for operations and safety limits.	4.0			71
	2.2.					
	2.2.					
	2.2.					
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2			72
	2.3.12	Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc	3.2			73
4. Emergency Procedures / Plan	2.4.2	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	4.5			74
	2.4.3	Ability to identify post-accident instrumentation.	3.7			75
	2.4.					
	Subtotal					
Tier 3 Point Total				10		

Tier / Group	Randomly Selected K/A	Reason for Rejection

Facility: Diablo Canyon Date of Exam: April, 2016

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 & Abnormal Plant Evolutions	1												18	3	3	6
	2				N/A					N/A			9	2	2	4
	Tier Totals												27	5	5	10
2. Plant Systems	1												28	3	2	5
	2												10	1 1	1	3
	Tier Totals												38	5	3	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
														2 1	2 2	

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the “Tier Totals” in each K/A category shall not be less than two).
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4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics’ importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401 PWR Examination Outline Form ES-401-2
Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1									
000008 Pressurizer Vapor Space Accident / 3									
000009 Small Break LOCA / 3						X	G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4. 7	76
000011 Large Break LOCA / 3									
000015/17 RCP Malfunctions / 4									
000022 Loss of Rx Coolant Makeup / 2									
000025 Loss of RHR System / 4									
000026 Loss of Component Cooling Water / 8					X		AA2.06 Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The length of time after the loss of CCW flow to a component before that component may be damaged	3. 1	77
000027 Pressurizer Pressure Control System Malfunction / 3									
000029 ATWS / 1						X	G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4. 2	78
000038 Steam Gen. Tube Rupture / 3									
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4									
000054 (CE/E06) Loss of Main Feedwater / 4									
000055 Station Blackout / 6									
000056 Loss of Off-site Power / 6									
000057 Loss of Vital AC Inst. Bus / 6					X		AA2.20 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: Interlocks in effect on loss of ac vital electrical instrument bus that must be bypassed to restore normal equipment operation	3. 9	79
000058 Loss of DC Power / 6									
000062 Loss of Nuclear Svc Water / 4									
000065 Loss of Instrument Air / 8									
W/E04 LOCA Outside Containment / 3						X	G2.4.23 Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.	4. 4	80

W/E11 Loss of Emergency Coolant Recirc. / 4					X		EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	4.2	81
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4									
000077 Generator Voltage and Electric Grid Disturbances / 6									
K/A Category Totals:					3	3	Group Point Total:		6

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)		Form ES-401-2
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000001 Continuous Rod Withdrawal / 1									
000003 Dropped Control Rod / 1									
000005 Inoperable/Stuck Control Rod / 1									
000024 Emergency Boration / 1					X		AA2.04 Ability to determine and interpret the following as they apply to the Emergency Boration: Availability of BWST (RWST)	4.2	82
000028 Pressurizer Level Malfunction / 2									
000032 Loss of Source Range NI / 7									
000033 Loss of Intermediate Range NI / 7									
000036 (BW/A08) Fuel Handling Accident / 8						X	AA2.02 Occurrence of a fuel handling incident	4.1	83
000037 Steam Generator Tube Leak / 3									
000051 Loss of Condenser Vacuum / 4									
000059 Accidental Liquid RadWaste Rel. / 9									
000060 Accidental Gaseous Radwaste Rel. / 9					X		G2.4.44 Knowledge of emergency plan protective action recommendations.	4.4	84
000061 ARM System Alarms / 7									
000067 Plant Fire On-site / 8									
000068 (BW/A06) Control Room Evac. / 8									
000069 (W/E14) Loss of CTMT Integrity / 5									
000074 (W/E06&E07) Inad. Core Cooling / 4									
000076 High Reactor Coolant Activity / 9									
W/EO1 & E02 Rediagnosis & SI Termination / 3									
W/E13 Steam Generator Over-pressure / 4									
W/E15 Containment Flooding / 5									
W/E16 High Containment Radiation / 9						X	G2.4.41 Knowledge of the emergency action level thresholds and classifications.	4.6	85
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 - Depress. / 4									
BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4									
BW/E13&E14 EOP Rules and Enclosures									
CE/A11; W/E08 RCS Overcooling - PTS / 4									
CE/A16 Excess RCS Leakage / 2									
CE/E09 Functional Recovery									

K/A Category Point Totals:					2	2	Group Point Total:	4

ES-401 PWR Examination Outline Form ES-401-2
Plant Systems - Tier 2/Group 1 (RO / SRO)

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	#
003 Reactor Coolant Pump														
004 Chemical and Volume Control														
005 Residual Heat Removal											X	G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.2	86
006 Emergency Core Cooling								X				A2.06 Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Water hammer	3.5	87
007 Pressurizer Relief/Quench Tank														
008 Component Cooling Water								X				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low surge tank level	3.5	88
010 Pressurizer Pressure Control														
012 Reactor Protection											X	G2.2.22 Knowledge of limiting conditions for operations and safety limits.	4.7	89
013 Engineered Safety Features Actuation														
022 Containment Cooling														
025 Ice Condenser														
026 Containment Spray														
039 Main and Reheat Steam														
059 Main Feedwater														
061 Auxiliary/Emergency Feedwater														

062 AC Electrical Distribution								X				A2.11 Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Aligning standby equipment with correct power source (D/G)	4.1	90
063 DC Electrical Distribution													
064 Emergency Diesel Generator													
073 Process Radiation Monitoring													
076 Service Water													
078 Instrument Air													
103 Containment													
K/A Category Point Totals:								3			2	Group Point Total:		5

ES-401 PWR Examination Outline Form ES-401-2
Plant Systems - Tier 2/Group 2 (RO / SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
001 Control Rod Drive														
002 Reactor Coolant														
011 Pressurizer Level Control														
014 Rod Position Indication														
015 Nuclear Instrumentation														
016 Non-nuclear Instrumentation														
017 In-core Temperature Monitor											X	G2.2.23 Ability to track Technical Specification limiting conditions for operations.	4.6	91
027 Containment Iodine Removal														
028 Hydrogen Recombiner and Purge Control														
029 Containment Purge														
033 Spent Fuel Pool Cooling								X				A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal spent fuel pool water level or loss of water level	3.5	92
034 Fuel Handling Equipment							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Load limits	3.2	93
035 Steam Generator														
041 Steam Dump/Turbine Bypass Control														
045 Main Turbine Generator														
055 Condenser Air Removal														
056 Condensate														
068 Liquid Radwaste														
071 Waste Gas Disposal														
072 Area Radiation Monitoring														
075 Circulating Water														
079 Station Air														
086 Fire Protection														

K/A Category Point Totals:							1	1			1	Group Point Total:		3

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.			3.9	95
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management.			4.6	94
	2.1.					
	2.1.					
	2.1.					
2. Equipment Control	2.2.7	Knowledge of the process for conducting special or infrequent tests.			3.6	96
	2.2.12	Knowledge of surveillance procedures. (Replacement for 2.3.5)			4.1	97
	2.2.					
	2.2.					
	2.2.					
	2.2.					
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.			3.7	98
	2.3.					
	2.3.					
	2.3.					
	2.3.					
	2.3.					
4. Emergency Procedures / Plan	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.			4.3	99
	2.4.11	Knowledge of abnormal condition procedures.			4.2	100
	2.4.					
	2.4.					
	Subtotal					
Tier 3 Point Total					7	

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO T2/G1	073 K5.03	Typo on outline for K5 KA. Question was actually written to K3.01. Outline was updated during draft outline comment review period.
SRO T2/G1	062 A2.03	Randomly selected to replace A2.03. 062 A2.03 is very nearly the same as the KA for APE 057 AA2.20 (question 79) and RO knowledge. Randomly replaced with A2.11 – Aligning standby equipment with correct power source (D/G)
SRO T3/G3	2.3.5	randomly replaced with 2.2.12. 2.3.5 is RO knowledge Updated ES-401-2
SRO T3/G1	2.1.3	randomly replaced with 2.1.37 (reactivity management). Orginal KA (shift turnover) too close to KA 2.1.5 (shift staffing) and utilized same procedure.
RO T1/G1	EPE 029 EA1.07	Unable to write a question for these valves during an ATWS. Valves have no auto features or interlocks and are not operated for an ATWS. Randomly replaced with EPE 029 EA1.02

Facility: Diablo Canyon	Date of Examination: 04/18/2016
Examination Level: RO ☒ SRO ☐	Operating Test Number: L141

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations (NRCL141-A1)	M, R	**Perform STP I-1A Channel Checks** 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (4.4) (modified from NRCL061-A1)
Conduct of Operations (NRCL141-A2)	M, R	**Determine Spent Fuel Pool Heat Load/Removal Parameters** 2.1.42 Knowledge of new and spent fuel movement procedures. (2.5) (modified from NRCADM061C-A1)
Equipment Control (NRCL141-A3)	M, R	**Determine Clearance Points and Tagging Requirements** 2.2.13 Knowledge of tagging and clearance procedures. (4.1) (modified from NRCL061-EC-SRO)
Radiation Control (NRCL141-A4)	N, R	**Evaluate Does Limits and Margin for RHR System Work** 2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions. (3.5) (New)
Emergency Procedures/Plan		

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when 5 are required.

* Type Codes & Criteria:
(C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1)
(P)revious 2 exams (≤ 1; randomly selected)

Facility: Diablo Canyon	Date of Examination: 04/18/2016
Examination Level: RO ☐ SRO ☒	Operating Test Number: L141

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations (NRCL141-A5)	M, R	**Determine Decay Heat and Heatup Rate** 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (4.2) (modified from Bank LJC-014)
Conduct of Operations (NRCL141-A6)	M, R	**Review Determination of Spent Fuel Pool Heat Load/Removal Parameters** 2.1.42 Knowledge of new and spent fuel movement procedures (3.4) (modified from NRCADM061C-A1)
Equipment Control (NRCL141-A7)	M, R	**Determine Clearance Points and Tagging Requirements** 2.2.13 Knowledge of tagging and clearance procedures. (4.3) (modified from NRCL061-EC-SRO)
Radiation Control (NRCL141-A8)	M, R	**Authorize Gas Decay Tank Discharge** 2.3.6 Ability to approve release permits. (3.8) (from NRCADM061-RC-SRO)
Emergency Procedures/Plan (NRCL141-A9)	N, R	**Classify Inability to Establish Control of the Plant Following a Control Room Evacuation** 2.4.41 Emergency Procedures/Plan. (4.6) (New)

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when 5 are required.

* Type Codes & Criteria:
(C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank ($\leq$ 3 for ROs; $\leq$ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank ($\geq$ 1)
(P)revious 2 exams ($\leq$ 1; randomly selected)

Facility: Diablo Canyon | Date of Examination: 04/18/2016

Exam Level: RO ☒ SRO-I ☐ SRO-U ☐ | Operating Test Number: L141

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U)

System / JPM Title	Type Code*	Safety Function
a. (S1) (004.A2.25) Makeup Control – Dilute (LJC-075)	A,M,S	1
b. (S2) (013.A2.01) Reinitiate ECCS Flow Following SI Termination	A,E,EN,L,N,S	2
c. (S3) (006.A4.02) Isolate Accumulators Following a LOCA (LJC-048)	A,D,E,L,S	3
d. (S4) (074.EA2.02) Restore Temporary Core Cooling During ICC Event	A,E,L,N,S	4P
e. (S5) (076.A2.02) Respond to ASW System Heat Exchanger Low Pressure	N,S	4S
f. (S6) (062.A4.07) Transfer Vital 4kV Buses from Auxiliary to Start-Up Power	A,L,N,S	6
g. (S7) (015.A2.02) Remove Power Range Channel N42 from Service (LJC-051)	D,S	7
h. (S8) (007.A1.01) Respond to High PRT Parameters	N,S	5

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

System / JPM Title	Type Code*	Safety Function
i. (P1) (010.A2.02) Transfer Pressurizer Heaters to Backup Power (LJP-029)	D,E,L	3
j. (P2) (E05.EA1.1) Reset the Turbine Driven Aux Feedwater Pump (LJP-012A)	D,E,L,R	4S
k. (P3) (068.AA1.31) Start a D/G and Restore Power to a Vital Bus following Control Room Evacuation (LJP-003A)	A,D,E,L	8

@ All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for 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	≥ 1 / ≥ 1 / ≥ 1 (control room system
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Diablo Canyon Date of Examination: 04/18/2016

Exam Level: RO ☐ SRO-I ☒ SRO-U ☐ Operating Test Number: L141

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U)

System / JPM Title	Type Code*	Safety Function
a. (S1) (004.A2.25) Makeup Control – Dilute (LJC-075)	A,M,S	1
b. (S2) (013.A2.01) Reinitiate ECCS Flow Following SI Termination	A,E,EN,L,N,S	2
c. (S3) (006.A4.02) Isolate Accumulators Following a LOCA (LJC-048)	A,D,E,L,S	3
d. (S4) (074.EA2.02) Restore Temporary Core Cooling During ICC Event	A,E,L,N,S	4P
e.		
f. (S6) (062.A4.07) Transfer Vital 4kV Buses from Auxiliary to Start-Up Power	A,L,N,S	6
g. (S7) (015.A2.02) Remove Power Range Channel N42 from Service (LJC-051)	D,S	7
h. (S8) (007.A1.01) Respond to High PRT Parameters	N,S	5

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

i. (P1) (010.A2.02) Transfer Pressurizer Heaters to Backup Power (LJP-029)	D,E,L	3
j. (P2) (E05.EA1.1) Reset the Turbine Driven Aux Feedwater Pump (LJP-012A)	D,E,L,R	4S
k. (P3) (068.AA1.31) Start a D/G and Restore Power to a Vital Bus following Control Room Evacuation (LJP-003A)	A,D,E,L	8

@ All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for 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	≥ 1 / ≥ 1 / ≥ 1 (control room system
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Diablo Canyon Date of Examination: 04/18/2016

Exam Level: RO ☐ SRO-I ☐ SRO-U ☒ Operating Test Number: L141

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U)

System / JPM Title	Type Code*	Safety Function
a. (S1) (004.A2.25) Makeup Control – Dilute (LJC-075)	A,M,S	1
b. (S2) (013.A2.01) Reinitiate ECCS Flow Following SI Termination	A,E,EN,L,N,S	2
c.		
d.		
e.		
f.		
g. (S7) (015.A2.02) Remove Power Range Channel N42 from Service (LJC-051)	D,S	7
h.		

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

i.		
j. (P2) (E05.EA1.1) Reset the Turbine Driven Aux Feedwater Pump (LJP-012A)	D,E,L,R	4S
k. (P3) (068.AA1.31) Start a D/G and Restore Power to a Vital Bus following Control Room Evacuation (LJP-003A)	A,D,E,L	8

@ All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for 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	≥ 1 / ≥ 1 / ≥ 1 (control room system
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Diablo Canyon (PWR) **Scenario No:** 1 **Op-Test No:** L141 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: **2% with MFW in service, aligned to Start-Up Power, MOL, 1431 ppm boron**

Turnover: **In OP L-3, performing step 6.28, raising power to 8%.**

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	N/A	R(ATC, SRO)	Raise reactor power from 2% to ≈ 8% **OP L-3,** sec 6.28**.**
2	MAL_NIS3B	SRO	Intermediate range detector NI-36 fails low @ ≈ 3.0%. (**AP-5, TS 3.3.1.F**) (Used for SRO TS Only).
3	CNV_MFW9_3	I (BOP, SRO)	Feed Reg Bypass Valve FCV-1510 oscillation; manual control required (**PK09-15**).
4	MAL_EPS4D_2	C (ALL)	4kV Vital Bus G differential trip results in loss of charging and letdown isolation (**AP-17**, **AP-27**, **TS 3.4.11.C, 3.8.1.B, 3.8.4.A**).
5	MAL_RCP1A, XMT_RCP17_3, XMT_RCP18_3	M (ALL)	RCP 1-1 #1 seal leak requiring Rx Trip and tripping of RCP 1-1. **(AP-28)**
6	MAL_RCS4C	M (ALL)	400 gpm Post-Trip SGTR on S/G 1-3 (**CT-18, 19, 20, 21**)
7	CNV_MSS24_2	C (BOP)	PORV on ruptured S/G fails open, requires backup air to close

***(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor**

Target Quantitative Attributes (Per Scenario; See Section D.5.d) (from form ES301-4)	Actual Attributes
1. Total malfunctions (5–8) (Events 2,3,4,5,6,7)	6
2. Malfunctions after EOP entry (1-2) (Event 7)	1
3. Abnormal events (2–4) (Events 2,3,4)	3
4. Major transients (1-2) (Event 5,6)	2
5. EOPs entered/requiring substantive actions (1–2) (E-3)	1
6. EOP contingencies requiring substantive actions (0–2)	0
7. Critical tasks (2–3)(See description below) (CT-18, 19, 20)	3

Critical Task	Justification	Reference
(CT-18) Isolate the ruptured steam generator from the intact steam generators prior to commencing cooldown of the RCS in step 9.c (40% stem dumps) or 10.b (10% steam dump) by completing the following: Isolate feedwater by ensuring closed: • LCV-108 (MDAFW Level Control Valve) • LCV-115 (TDAFW Level Control Valve) Isolate steamflow by ensuring closed: • FCV-43 and FCV-23 (S/G 1-3 MSIV and bypass) (VB3) • S/G 1-3 supply to TD AFW Pp (FCV-38) • FCV-157 and FCV-246 (Blowdown and Sample Isolation Outside Containment)	SG inventory increase leads to water release through the S/G PORV or safety valve(s) or to SG overfill, which would seriously compromise the SG as a fission-product barrier and complicate mitigation.	• W Margin to Overfill (CN-CRA-05-53 Rev1) • W Offsite Doses (CN-CRA-05-54) • SGTR UFSAR 15.4.3 • WCAP-17711-NP

(continued on next page)

Critical Task	Justification	Reference
(CT-19) Perform RCS cooldown at *maximum rate** to CETC target temperature specified in E-3, step 6, using steam dumps such that: • MAGENTA PATH on RCS INTEGRITY is avoided • RCS subcooled margin still exists following the cooldown. • Target temperature is reached within 20 minutes of initiation of cooldown (based on 2x NRC scenario validation time) *For 40% steam dumps, maximum rate limit is 100 psi/min (PPC value). Above this, main steam line isolation will occur. Operator should attempt highest rate possible without getting main steam line isolation (<u>not</u> critical). If 40% dumps are not available or if steam line isolation occurs, maximum rate cooldown requires 10% steam dumps on intact S/Gs to be fully opened.	Transition to contingency procedures to address inadequate subcooling or Pressurized Thermal Shock conditions results in delaying RCS depressurization and SI termination. This delay allows excess inventory in the ruptured S/G to continue to increase, with the potential of challenging SG overpressure components or causing an overfill condition to occur.	• W Margin to Overfill (CN-CRA-05-53 Rev1) • SGTR UFSAR 15.4.3 • WCAP-17711-NP
(CT-20) Depressurize the RCS to meet depressurization criteria specified in E-3, App GG prior to stopping Safety Injection pumps.	Failure to stop reactor coolant leakage into a ruptured SG by depressurizing the RCS complicates mitigation of the event and constitutes a "significant reduction of safety margin beyond that irreparably introduced by the scenario".	• W Margin to Overfill (CN-CRA-05-53 Rev1) • SGTR UFSAR 15.4.3 • WCAP-17711-NP

SCENARIO SUMMARY – NRC #1

1. Control rods are used to raise power from 2% to ≈ 8% **OP L-3, Secondary Plant Startup,** step 6.28**.** ATC operator complies with 1 step pull and wait procedural requirement while monitoring relevant controls and diverse indicators. Shift Foreman provides reactivity oversight.

2. Intermediate range detector NI-36 fails low @ ≈ 3.0%. The ATC operator identifies single intermediate range detector lowering with no other indications corroborate a power decrease. SFM directs power ascension placed on hold. **OP AP-5, Malfunction of Eagle 21 Protection or Control Channel** may be referenced. Shift Foreman addresses **TS 3.3.1.F, One Intermediate Range Neutron Flux channel inoperable.**

3. Feed Reg Bypass valve FCV-1510 begins to oscillate causing an unexpected rise in S/G 1-1 feedwater flow and level. Crew responds to **AR PK09-15, Digital Feedwater Cont System,** taking manual control of the failed valve. Shift Foreman establishes level control band for manual operation. May refer to **TS 3.7.3** for MFRV bypass valves, but LCO is not applicable.

4. 4 kV bus G trips on differential causing a loss of charging and letdown isolation. The crew responds by entering **OP AP-17, Loss of Charging**, and **OP AP-27, Loss of Vital 4kV and/or 480V Bus**. Shift Foreman establishes priorities and maintains oversight while the board operators implement abnormal procedures as assigned. Normal charging and letdown are restored and redundant/backup equipment is placed in service. Shift Foreman addresses applicable short action Tech Specs: **TS 3.4.11.C-Pressurizer Power Operated Relief Valves; 3.8.1.B-AC Sources Operating for D/G 1-2; 3.8.4.A-DC Sources**.

5. RCP 1-1 seal leak ramps in over 2 minutes resulting in high seal return flow. Crew responds to **AR PK05-01, RCP NO 11** for seal leakoff flow greater than 5.0 gpm and transitions to **OP AP-28, Section B, RCP Number 1 Seal Failure**. When seal leakoff and radial out bearing temperatures begin to rise, SFM directs Rx Trip and subsequent tripping of RCP 1-1 and closure of associated pressurizer spray valve.

6. Crew enters **EOP E-0, Reactor Trip or Safety Injection**, and transitions into **EOP E-0.1, Reactor Trip Response** to stabilize the plant. A 400 gpm SGTR*** on S/G 1-3 ramps in approximately two minutes post-trip. The crew identifies the need to safety inject when pressurizer pressure and level begin to lower rapidly. The Shift Foreman re-enters **EOP E-0, Reactor Trip or Safety Injection**, and transitions to **EOP E-3, Steam Generator Tube Rupture**, based on RM-73 and rising S/G 1-3 level, where the crew will address the following three critical tasks:

   CT-18: Isolate the ruptured steam generator prior to commencing a cooldown of the RCS.

   CT-19: Perform RCS cooldown at maximum rate such that MAGENTA PATH on RCS INTEGRITY is avoided and RCS subcooled margin still exists following the cooldown (accomplished by reaching the target temperature specified in E-3, step 6).

   CT-20: Depressurize the RCS to meet depressurization criteria specified in E-3, App GG prior to stopping Safety Injection pumps.

(continued on next page)

7. The 10% Pressure Operated Relief Valve on the ruptured steam generator inadvertently actuates (steam generator pressure is still over 100# below setpoint). The Control Operator identifies the open valve by the audible sound of steam and a red position indicator. The valve is successfully closed by cutting in backup air and taking the backup control switch to close. Isolation of the valve is part of CT-18.

The scenario is terminated once RCS depressurization is complete (E-3, ready to perform step 24).

*** **CT / TCOA note**: SGTR was evaluated against Time Critical Operator Actions (TCOAs) # 2,3, and 4 (SGTR); initial power level and supporting equipment conditions differ significantly from the conditions used in this scenario. For these reasons, the S/G TCOAs will remain critical (a critical task, per WOG), but TCOA time limits will not be applied to this scenario.

Facility: Diablo Canyon (PWR) **Scenario No:** 2 **Op-Test No:** L141 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: 50% MOL 1085 ppm boron; Bus F week: CFCU 1-1 and D/G 1-3 OOS

Turnover: Maintenance has requested Pressurizer Heater Group 1-2 energized on vital backup power.

Event No	Malf No.	Event Type*	Event Description
1	PZ01PRZ_PRH12_1RTAPWRRT, LOA_PZR31	C (ATC, SRO)	Pressurizer Heater Grp 1-2 ground and power failure during routine maintenance (**PK17-24, TS 3.4.9**)
2	H_V1_034M_1, XMT_VEN6_3, XMT_VEN7_3, XMT_VEN8_3	C(BOP, SRO)	CFCU 1-2 high stator/bearing temperature due to low CCW flow (**PK01-21, TS 3.6.6**)
3	XMT_CVC2_3	I(BOP, SRO)	PT-135 Fails High causing letdown pressure control valve to go full open (**PK04-21**).
4	MAL_CVC8A	C (ATC, SRO)	Seal Injection Filter 1-1 plugs causing reduction in charging flow to RCP seals (**PK04-22).**
5	MAL_GEN4_2	R (ATC) C (BOP, SRO)	Main Generator underfrequency at 50% causes load rejection (**AP-2**).
6	MAL_MSS3A, MAL_SYD2	M (ALL)	Steamline break outside containment upstream of flow restrictor (S/G 1-1); requires Rx trip and SI. MDAFW Pump LCV-110 fails open and must be closed locally. RCPs trip due to a loss of 12kV power, driving conditions for entry into **FR-P.1** (**CT-48, CT-CD**).
7	MAL_PPL5A, MAL_PPL5B CNV_AFW1_2 1	C (BOP)	ATWS condition; requires manual opening of 13D/E breakers from C/R (**CT-1**).

***(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor**

Target Quantitative Attributes (Per Scenario; See Section D.5.d) (from form ES301-4)	Actual Attributes
1. Total malfunctions (5–8) (Events 1,2,3,4,5,6,7)	7
2. Malfunctions after EOP entry (1-2) (Events 7)	1
3. Abnormal events (2–4) (Events 1,2,3,4,5)	5
4. Major transients (1-2) (Event 6)	1
5. EOPs entered/requiring substantive actions (1–2) (E-2)	1
6. EOP contingencies requiring substantive actions (0–2) (FR-P.1)	1
7. Critical tasks (2–3)(See description below)	3

Critical Task	Justification	Reference
(CT-1) Reactor tripped by completion of E-0, step 1.	The safeguards systems that protect the plant during accidents are designed assuming that only decay heat and pump heat are being added to the RCS. Failure to manually trip the reactor causes a challenge to the subcriticality critical safety function beyond that irreparably introduced by the postulated conditions.	• Westinghouse Owner's Group WCAP-17711-NP • Calc G.2 Rev 5 (08151-2169) • OP1.ID2, Time Critical Operator Actions Rev 8A, #34.
(CT-48) Terminate ECCS flow by the completion of FR-P.1, step 11.	Failure to terminate ECCS flow when SI termination criteria are met causes the PRZR to fill and RCS pressure to increase. Additionally, the unnecessary continuation of ECCS flow needlessly aggravates the thermal stress on the reactor vessel. This constitutes an incorrect performance that causes a significant reduction of safety margin beyond that irreparably introduced by the scenario.	• Westinghouse Owner's Group WCAP-17711-NP
(CT-CD) Stop RCS Cooldown by: • Isolating feedflow to S/G 1-1 by closing/verifying closed LCV-106 and LCV-110. • Verifying FCV-25 and FCV-41 are closed (S/G 1-1 steamline isolation) • Verifying all steam dumps closed. • Throttling Feedflow to S/Gs 1-2, 1-3, and 1-4 while maintaining the minimum heatsink requirements (435 gpm until S/G level is greater than 15% in one non-faulted S/G). prior to performing step 15.d, isolation of Accumulators, in FR-P.1	An event or series of events which leads to a relatively rapid and severe reactor vessel downcomer cooldown can result in a thermal shock to the vessel wall that may lead to a small flaw, which may already exist in the vessel wall, growing into a larger crack. The growth or extension of such a flaw may lead, in some cases (where propagation is not stopped within the wall), to a loss of vessel integrity	• Background Information for WOG Emergency Response Guideline

SCENARIO SUMMARY – NRC #2

1. Reactor Operator places Pressurizer Heater Group 1-2 (aligned to vital backup power supply) in service to support routine maintenance. Associated control board power meter spikes to high end of scale followed by annunciator alarm **PK17-24**, **480V Bus 1G Ground** on the associated bus. Operator recognizes the abnormal condition and secures heater. If breaker is not opened manually, break trips open in 15 seconds. Shift Foreman enters **TS 3.4.9 Condition B**, for one required group of pressurizer heaters inoperable (72 hour shutdown tech spec).

2. CFCU 1-2 has loss of CCW flow due to debris migration causing stator and motor bearing temperatures to rise rapidly and bring in annunciator alarm **PK01-21, Contmt Fan Clr.** Reactor operators identify low flow indications on vertical boards and rapidly rising stator/bearing temperatures using plant process computer trends. Crew secures the CFCU to prevent motor damage and contact maintenance/engineering for assistance. Shift Foreman enters **TS 3.6.6 Condition C**, one required CFCU system inoperable such that a minimum of two CFCUs remain OPERABLE (7 day shutdown tech spec).

   (*Note: Malfunction is designed to trip the CFCU if crew has not shut the fan down within 5 minutes of stator temperature reaching 380^oF*).

3. PT-135, Transmitter for Letdown Pressure Control Valve**,** Fails High causing letdown pressure control valve to go full open and letdown flow to rise. **AR PK04-21, LETDOWN PRESS / FLOW TEMP** comes into alarm for Letdown Heat Exchanger Outlet Pressure High as a result of the failed transmitter, while actual letdown pressure lowers to approximately 90 psig as a result of full open control valve response. Letdown flow increases approximately 8 gpm above normal, resulting in a charging/letdown mismatch. Procedural guidance in AR PK04-21 directs crew to take manual control of PCV-135. Crew performs diagnostic brief to determine nature of the malfunction as well as actions required to restore letdown pressure back to normal band.

4. In-service Seal Injection Filter 1-1 plugs, reducing flow to RCP seals and bringing in **AR PK04-22, RCP Seal Inj Fltr Delta-P Hi**. Reactor Operators verify CCP seal cooling is still being maintained by CCW and ATC operator throttles RCP seal injection hand control valve, HCV-142, as needed to maintain pressurizer level. Shift Foreman establishes bands for pressurizer level and confirms field operators have been dispatched to swap seal injection filters.

5. An underfrequency condition on the grid results in a full load rejection on Unit 1 from 50% power. Crew recognizes condition based on numerous power level alarms and the ensuing secondary side transient. The crew monitors primary and secondary systems response, most notably rod control, steam dumps, and digital feedwater to ensure all systems respond appropriately in automatic. Shift Foreman implements **OP AP-2, Full Load Rejection** to stabilize the plant. The ATC operator places rods in manual to maintain reactor power between 20%-30% reactor power and determines required boration to stabilize the plant. BOP operator performs secondary realignments.

6. A main steamline break develops gradually (4 minute ramp) upstream of the Main Steam Isolation Valves, outside containment. The crew identifies the need to safety inject and enter Emergency Operating Procedures based on pressurizer pressure and level lowering rapidly. Initial diagnosis may also note the absence of parameters indicative of a primary side break. Shift Foreman directs Safety Injection (SI) and entry into **EOP E-0, Reactor Trip or Safety Injection.**

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SCENARIO SUMMARY – NRC #2

7. On the SI, the reactor fails to automatically actuate (ATWS). Manual Rx Trip switches are ineffective as well. Control board operators will perform their respective immediate actions: ATC drives control rods inward and BOP manually opens control rod breakers 13D/E on VB5 (**CT-1, Reactor tripped by completion of E-0, step 1)****. The Shift Foreman maintains his oversight position to ensure immediate actions are performed properly and in a timely manner, prior to formally entering **EOP E-0, Reactor Trip or Safety Injection**.

8. A reactor operator is assigned the task of isolating S/G 1-1 per **EOP E-2, Faulted Steam Generator Isolation**, **Appendix HH,** while the Shift Foreman and remaining crew member continue on through the main procedure body in parallel.

   The loss of power to the RCPs (Startup power lost shortly after Rx trip), combined with the blowdown of S/G 1-1 eventually leads to a MAGENTA and subsequent RED path on RCS INTEGRITY. The Shift Foreman transitions to **EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition**, where the crew completes the critical task of SI termination (**CT-48, Terminate ECCS flow by the completion of FR-P.1, step 11**)** and stopping the RCS cooldown (**CT-CD, Stop the RCS cooldown prior to performing step 15, isolation of Accumulators, in FR-P.1**)**.

The scenario is terminated once Safety Injection is terminated in FR-P.1, Response to Imminent Pressurized Thermal Shock Condition (completion of step 11), provided all Critical Tasks are complete. If steps to stop the cooldown have not been completed, the scenario should be continued to the bounding step 15, in FR-P.1.

*** **TCOA note**: Steam break was evaluated against Time Critical Operator Actions (TCOAs) # 18 & 19 (MSLB IC & OC); the break sizes, ramp times, initial power levels, and other conditions differ significantly from the conditions used in this scenario.

*** **TCOA note**: The ATWS was evaluated against Time Critical Operator Actions (TCOA) # 34 and is similar to the TCOA basis event. TCOA time limits will be applied to the scenario. Operators must trip the reactor from the control room within 90 seconds.

Facility: Diablo Canyon (PWR) **Scenario No:** 3 **Op-Test No:** L141 NRC

Examiners: ____________ **Operators:** ____________

Initial Conditions: **100% MOL 960 ppm boron; AFW 1-3 and CFCU 1-5 OOS**

Turnover: Maintain 100% Power

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	N/A	R(ALL)	Grid Control Center backdown order (200 MW within 30 minutes) (**AP-25**).
2	XMT_PZR24_3	I (BOP, SRO)	PT-474, Pressurizer Pressure Transmitter, Fails Low (**AP-5**, **TS 3.3.1.E, M, 3.3.2.D, 3.4.11**)
3	XMT_ASW2, XMT_ASW1, PMP_ASW2, AS01ASW_ASP11_MTFSEIZUR	C (BOP, SRO)	ASW Pp 1-1 high bearing temperature / trip due to loss of motor oil (**PK01-03, AP-10, TS 3.7.8.A**).
4	DSC_ROD1	C (ATC, SRO)	DRPI power supply failure during ramp. (**PK03-21, TS 3.1.7.B**).
5	MAL_CVC1	C (ALL)	35 gpm letdown leak inside containment (**AP-18).**
6	ZMLSEI1, MAL_SEI1, MAL_RCS2A	M (ALL)	Seismically induced 100% DBA LBLOCA. Reactor Trip and Safety Injection auto-initiate. Crew transitions to **E-1.3** Cold Leg Recirculation when RWST reaches 33% (**CT-36**).
7	PMP_ASW2_1	C (BOP)	ASW Pp 1-2 trip during initial bus transfer; manual restart required) (**CT-9**).
8	RLY_PPL73, RLY_PPL74, RLY_PPL75, RLY_PPL76	C (BOP)	Containment spray fails to actuate (manually alignment required) (**CT-3**).
9	AF01AFW_AFP12_MTFSHEAR	C (ATC)	AFW Pp 1-2 shaft shears and TDAFW auto-start fails.
10	PMP_RHR2_1, PMP_RHR1_1	C (ATC)	RHR pumps fail to trip @ 33% RWST level; manual stop required.

***(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor**

Target Quantitative Attributes (Per Scenario; See Section D.5.d) (from form ES301-4)	Actual Attributes
1. Total malfunctions (5–8) (Events 2,3,4,5,6,7,8,9,10)	9
2. Malfunctions after EOP entry (1-2) (Events 7,8,9,10)	4
3. Abnormal events (2–4) (Events 3,4,5)	3
4. Major transients (1-2) (Event 6)	1
5. EOPs entered/requiring substantive actions (1–2) (E-1, E-1.3)	2
6. EOP contingencies requiring substantive actions (0–2)	0
7. Critical tasks (2–3)(See description below)	3

Critical Task	Justification	Reference
(CT-9) Manually start ASW Pump 1-2 by completion of EOP E-0, Appendix E, step 5.	ASW train is required to remove accident generated and core decay heat following a design basis LOCA. Without ASW, CFCUs cannot remove heat from the containment atmosphere. Additionally, ASW serves as the ultimate heat sink during the recirculation mode of ECCS cooling. Failure to start the minimum number of required ASW pumps places the plant in an unanalyzed condition.	• FSAR, Section 6.2. & 9.2.7. • Westinghouse Owner's Group WCAP-17711-NP • Technical Specification Basis B.3.7.8
(CT-3) Manually align at least 1 train of Containment Spray (1 pump and associated valves) by completion of EOP E-0, Appendix E, step 7.	Failure to initiate the minimum required Containment Spray equipment as a means of pressure suppression represents a severe challenge to Containment Safety Function.	• EOP FR-Z.1 Background Document • Westinghouse Owner's Group WCAP-17711-NP
(CT-36) Establish one train of cold leg recirculation within 10 minutes of the RWST reaching low level set point of 33% as verified by: <u>Train A (RHR HX 1-1 in Service)</u> • 8700A CLOSED • 8982A OPEN • 8804A OPEN • Flow Indicated on FI-970A/B OR <u>Train B (RHR HX 1-2 in Service)</u> • 8700B CLOSED • 8982B OPEN • 8804B OPEN • 8807A OR 8807B OPEN • Flow Indicated on FI-971A/B	Transfer to cold leg recirculation within the TCOA time frame is license commitment. Failing to perform RHR suction realignment within the specified time constraint can lead to inadequate RHR NPSH and degraded the emergency core cooling system performance.	• STA-061 (07938-3-21, W Letter PGE-99-546 (07711-1153) • OP1.ID2, Time Critical Operator Actions Rev 8A, #8. • Westinghouse Owner's Group WCAP-17711-NP

SCENARIO SUMMARY – NRC #3

1. Shift Manager reports a confirmed Grid Control Center backdown order due to 500 kV line fire risk. Unit 1 is directed to shed 200 MW within 30 minutes. The Shift Foreman determines an appropriate ramp rate to meet the backdown order requirement (may assign this task to reactor operator) and implements **OP AP-25, Rapid Load Reduction or Shutdown.** The ATC determines an initial boration based on the Reactivity Handbook and advises the Shift Foreman of his recommendation. The BOP enters the programmed ramp into the turbine control system. The reactivity evolutions are implemented sequentially, with the Shift Foreman providing oversight.

2. PT-474, Pressurizer Pressure Transmitter, Fails low bringing in multiple Annunciator Alarms. There is no transient associated with this failure, but the failure has significant Operational implications due to its input function as part of various Reactor Protection logic schemes. When failed low, PT-474's interlock function prevents Pressurizer PORVs PCV-455C and PCV-474 from opening on a valid high pressure signal; only PCV-456 will still function. The Shift Foreman may elect to enter any of the associated Annunciator Response alarms, but in all cases, will be directed to **OP AP-5, Malfunction of Eagle-21 Protection or Control Channel**, which provides information regarding indications, controls, and a listing of the associated Tech Specs:
   - **TS 3.3.1.E, PC-474C High Press Trip & TC 441C OT Delta T Trip (72 hrs)**
   - **TS 3.3.1.M, PC 474A Low Press Trip (72 hrs)**
   - **TS 3.3.2.D, PC 474D Low Press S.I. (72 hrs)**
   - **TS 3.4.11, PC 474B PORV Press Interlock**
     - **PCV-474 (non-class I), 3.4.11.B1 & B2 to close & remove power from associated block valve (1 hr)**
     - **PCV-455C (class I), 3.4.11.B1 & B2 to close & remove power from associated block valve (1 hr); 3.4.11.B3 to return to OPERABLE status (72 hrs)**

3. A loss of motor oil to ASW Pp 1-1 causes high motor bearing temperatures, bringing in annunciator **AR PK01-03, Aux Salt Water Pumps.** Reactor operators follow procedural guidance and identify rapidly rising upper and lower motor bearing temperatures on ASW Pp 1-1 using the plant process computer (PPC). The local field operator is dispatched to determine the cause of the alarm and the Shift Foreman directs a swap to the standby ASW pump. Shift Foreman enters **TS 3.7.8.A,** for one ASW train inoperable (72 hour shutdown tech spec).

   (*Note: ASW Pp 1-1 will experience a seized shaft malfunction if it is stilling running 8 minutes after the high temperature alarm actuates ~ 350^{o}F*).

4. The normal power supply to DRPI trips opens at 1000 MW (close to end of target ramp). Crew responds per **AR PK03-21, DRPI FAILURE/ROD BOTTOM**, placing rods in manual. The Shift Foreman directs the ramp to be placed on hold and field operators are contacted to place DRPI on backup power. Shift Foreman provides crew with procedural guidance regarding rod motion (minor adjustments as well as actions required should a major transient occur) with DRPI unavailable and before entering **TS 3.1.7.B** for more than one DRPI per group inoperable (immediate TS action is to place rods in manual).

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SCENARIO SUMMARY – NRC #3

5. A 35 gpm letdown line leak inside containment, downstream RO 27/28/29 ramps in over 3 minutes. The ATC identifies lowering letdown flow, VCT level, and rising charging. The BOP operator checks containment parameters to aid diagnostic efforts. Diagnostic brief by the crew identifies letdown line inside containment as likely leak source (pressurizer pressure stable, structure sumps rising, RM-12 in alarm). Crew enters **OP AP-18, Letdown Line Failure** to address the leak. Normal letdown is isolated and excess letdown is placed in service.

   (*Note: If leak is not identified as being on the letdown line during initial diagnosis, crew will enter OP AP-1, and be directed to OP AP-18*).

6. Large seismic results in 100% DBA LBLOCA. Reactor Trip and Safety Injection occur immediately and the crew enters **E-0, Reactor Trip or Safety Injection**. The crew performs their immediate actions and checks for actuation of emergency safeguards equipment, diagnosing conditions consistent with a large break LOCA (high containment pressure, loss of pressurizer pressure and level, loss of subcooling, high containment sump levels). The crew identifies RCP trip criteria have been met, with Shift Foreman concurrence, trip all four RCPs (**TCOA**).*** Shift Foreman directs the BOP Operator to complete **Appendix E, ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS**, and continues on in E-0.

7. Reactor Operator notes that ASW Pp 1-2 is not running (tripped off during bus transfer) and manually restart the pump (**CT-9, Manually start ASW Pump 1-2 to provide at least the minimum required number of ASW pumps in an operating safeguards train by the completion of E-0, Appendix E, step 5**). **

   (*Note: This action may be performed after individuals have verified immediate actions or as part of Appendix E, ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS) .*

8. Reactor Operator identifies Containment Spray has failed to actuate as required based on high high containment pressure and an active Safety Injection signal. Action is taken to manually start containment spray pumps and align spray valves. (**CT-3, Manually align at least 1 train of Containment Spray (1 pump and associated valves) by completion of EOP E-0, Appendix E, step 7**).**

   (*Note: This action may be performed after individuals have verified immediate actions or as part of Appendix E, ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS).*

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9. Reactor Operator identifies low S/G level and no flow condition from running AFW pump (sheared shaft) along with TDAFW failed auto start. TDAFW pump is manually started to restore S/G level.

   Crew continues through E-0 diagnostic steps, and transitions to **E-1, Loss of Reactor or Secondary Coolant**. Functional restoration status trees are checked and crew identifies transition criteria for **FR-P.1, Response to Imminent Pressurized Thermal Shock**. Conditions will be met for exiting the procedure at the first step.

   The Shift Foreman performs a procedure transition brief to review plant conditions, priorities, and review the expected procedural flow path prior to continuing on into E-1. The task of monitoring for **E-1.3, Transfer to Cold Leg Recirculation** transition criteria is assigned to one of the Reactor Operators as a Fold Out Page (FOP) item during the brief.

10. When RWST level will reach 33%, both RHR pumps fail to trip, but can be manually stopped in the Control Room. The crew to transition immediately to **E-1.3, Transfer to Cold Leg Recirculation**, performing the TCOA/CT*** of Cold Leg Recirculation Alignment (**CT-36, Transfer to Cold Leg Recirculation and establish at least 1 train of ECCS flow within 10 minutes of the RWST level reaching 33%**). The Shift Foreman will assign one operator (usually the BOP) Appendix EE to perform RHR Hx alignment, while he and the remaining operator complete the TCOA alignment steps of E-1.3. The loss of one train of ASW early in the scenario will limit the crew to a single RHR Hx, requiring the crew to clearly communicate and choreograph the final valve alignment process.

The scenario is terminated once a single train of Cold Leg Recirculation Alignment is complete per EOP E-1.3, Transfer to Cold Leg Recirculation.

*** **TCOA note**: This DBA LBLOCA was evaluated against TCOA #8, and is similar to the TCOA bases event, so TCOA time limits will be applied to the scenario (operators have 10 min to align to cold leg recirculation, as timed from the RWST reaching 33% [alarm comes in] and finishing the alignment). Phase B, RCP Trip Criteria in this scenario was evaluated against TCOA #67 and determined to apply. Operators have 5 minutes to trip all four RCPs from the initial Phase B actuation signal.

Facility: Diablo Canyon (PWR) **Scenario No:** 4 **Op-Test No:** L141 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: 100% MOL 960 ppm boron; AFW 1-3 and D/G 1-3 OOS

Turnover: Perform remainder of STP P-CCP-11, continuing from step 12.9.11

Event No	Malf No.	Event Type*	Event Description
1	H_V2_164M_1	C (ALL)	STP P-CCP-11 surveillance test unsat due to high amp reading; pump swap required. **TS 3.5.2.A**
2	VLV_PZR6_2	C (BOP, SRO)	Pressurizer PORV PCV-474 seat leak-by to PRT. (**PK05-23, TS 3.4.11.A**).
3	MAL_RCS4H	C (ALL)	10 gpm SGTL ramped in over 1 minute; plant shutdown required (**OP O-4**, **AP-3, AP-25**, **TS 3.4.13.B**).
4	VLV_SGB1_1, VLV_SGB2_1, VLV_SGB3_1, VLV_SGB4_1, VLV_SGB1_9, VLV_SGB10_1, VLV_SGB11_1, VLV_SGB12_1, VLV_SGB13_1, VLV_SGB14_1,	C(BOP, SRO)	SG Blowdown High Rad auto actuations fail (**PK11-17**).
5	LOA_CND1	M (ALL)	Condenser vacuum requiring turbine trip/Rx trip (**AP-7**).
6	VLV_PZR4_2	C (BOP)	Pressurizer PORV PCV-455C fails slightly open on trip requiring manual isolation by associated block valve (**CT-10**).
7	MAL_AFW1 PMP_AFW1	C (BOP)	Turbine Driven Aux Feedwater Pump trips on overspeed. Motor Driven Aux Feedwater Pump 1-2 fails to autostart, manual start available.
8	PMP_AFW_2	M (ALL)	AFW Pp 1-2 overcurrent trip at less than the minimum required S/G level causing Loss of Heat Sink extreme challenge and entry into **FR-H.1** (**CT-43**).

***(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor**

Target Quantitative Attributes (Per Scenario; See Section D.5.d) (from form ES301-4)	Actual Attributes
• Total malfunctions (5–8) (Events 1,2,3,4,5,6,7,8)	8
• Malfunctions after EOP entry (1-2) (Events 6,7,8)	3
• Abnormal events (2–4) (Events 2,3,4)	3
• Major transients (1-2) (Events 5 and 8)	2
• EOPs entered/requiring substantive actions (1–2) (E-0.1)	1
• EOP contingencies requiring substantive actions (0–2) (FR-H.1)	1
• Critical tasks (2–3)(See Scenario Summary)	2

Critical Task	Justification	Reference
(CT-10) Close the block MOV upstream of the stuck open PORV such that the PRT remains intact for the duration of the scenario.	The open PORV and block valve constitute the degradation of a fission product barrier. Closing the block valve is essential to safety since failure to do so results in the unnecessary continuation of the degraded condition.	• Westinghouse Owner's Group WCAP-17711-NP
(CT-43) Establish feedwater flow from the Condensate System to at least one S/G within 50 minutes of AFW Pump 1-2 trip (2x NRC scenario validation time of 25 minutes) as indicated by WR S/G level rising and/or Core Exit Thermocouple temperatures lowering. Associated Performance Indicators are as follows: • Verifies Main Feedwater Isolation Valves OPEN • Throttles open MFRVs (feed reg valves) or MFRV bypasses • Depressurizes one S/G using the 10% steam dumps, at maximum rate (100% open), down to condensate injection pressure (< 490 psig)	Primary to secondary heat transfer deteriorates If S/G dryout is allowed to occur. The resultant rise in RCS temperature and pressure can lead to RCS barrier loss when pressure rises above the Pressurizer PORV setpoint, causing a loss of inventory and eventual fuel over-heating and damage.	• Westinghouse Owner's Group WCAP-17711-NP

SCENARIO SUMMARY – NRC #4

1. Crew takes the watch with **STP P-CCP-11, Routine Surveillance Test of Centrifugal Charging Pump 1-1** already in progress. Pump amps are found to be slightly above the normal operating range, and the crew follows the STP procedural guidance to shut down the pump. The crew notes CCP 1-3 was in-service just prior to the test, and performs a pump swap following the guidance of OP B-1A:V. The field operator assigned to check for bearing oil flow reports no oil was visible in the sight glass while the pump was running. Shift Foreman enters **TS 3.5.2.A,** for one ECCS train inoperable (72 hours to determine no common cause failure and 14 days to restore the train to OPERABLE status).

2. Pressurizer Pressure Operated Relief Valve PCV-474 begins leaking by, bringing in annunciator **AR PK05-23, PZR SAFETY OR RELIEF LINE TEMP.** The crew performs diagnostics, confirming the existence of seat leakage based on PZR PORV tailpipe temperature and rising Pressurizer Relief Tank pressure and temperature. Annunciator guidance is followed to identify and isolate PCV-474 by closing its associated block valve, 8000A. Shift Foreman addresses **TS 3.4.11.A**, for one or more PORVs inoperable solely due to excessive seat leakage. (1 hour to close the associated block valve).

3. Steam Generator 1-4 develops a 10 gpm tube leak over a one minute period. Rising counts on various radiation monitors alert the crew to both the nature and location of the leak. The Crew estimates the leak rate and enters **OP AP-3, Steam Generator Tube Failure**. The leak is also evaluated per **OP O-4, Primary to Secondary Steam Generator Tube Leak Detection**, which directs the crew to reduce power by 50% in the next hour and be in Mode 3 within two hours. Shift Foreman determines **TS 3.4.13.B**, RCS Operational Leakage applies and enters **OP AP-25, Rapid Load Reduction or ShutdowN** for the ramp off-line.

4. **AR PK11-17, SG BLOWDOWN HI RAD** comes into alarm during the ramp as a result of high radiation on RM- 23. Isolation relays fail to actuate and the crew takes manual action to close the affected valves on the steam generator blowdown sample header and swap steam generator blow down discharge over to the Equipment Drain Receiver.

5. The vacuum breaker at the main condenser develops a leak at approximately 1150 MW, causing condenser vacuum to steadily degrade. The crew enters **OP AP-7, Degraded Condenser,** and may attempt to adjust the ramp rate, but will be unsuccessful in avoiding an automatic turbine trip/Rx trip. The Shift Foreman will monitor the turbine automatic trip set point and the may elect to manually actuate once it is clear that a trip is unavoidable.

6. The crew enters **E-0, Reactor Trip or Safety Injection** and performs their immediate actions. Board operators also identify PCV-455C in mid-position. The valve will not close and must be isolated using the associated block valve 8000B (**Critical Task CT-10, close the block MOV upstream of the stuck open PORV such that the PRT remains intact for the duration of the scenario**).**

7. Operators identify low S/G level with no AFW pumps running while preparing to transition out of E-0. The turbine driven Auxiliary Feedwater pump is tripped on overspeed and cannot be restarted. The remaining Motor Driven Aux Feedwater Pump (MDAFW 1-2) failed to auto start and is started manually.

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8. Crew transitions from **E-0, Reactor Trip or Safety Injection** to **E-0.1, Reactor Trip Response** to stabilize the plant. When S/G 1 & 2 reach approximately 13.5%, Motor Driven Aux Feedwater 1-2 trips on overcurrent, leading to Loss Of Heat Sink condition. **EOP FR-H.1, Response to Loss of Secondary Heat Sink** is used to establish secondary feedwater from the condensate system. (**Critical Task CT-43: Establish feedwater flow into at least one S/G within 50 minutes of AFW Pump 1-2 trip (2x NRC scenario validation time of 25 minutes)).****

The scenario is terminated in FR-H.1, once condensate flow to the steam generators has been established.