Facility: Diablo Canyon Date of Exam: March 6, 2020

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

3. Generic Knowledge and Abilities Categories	RO 1	RO 2	RO 3	RO 4	RO Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
					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 outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the “Tier Totals” in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)
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 outline. Systems or evolutions that do not apply at the facility should be deleted with justification. 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’ 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 a category 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.

G* Generic K/As

* These systems/evolutions must be included as part of the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan. They are not required to be included when using earlier revisions of the K/A catalog.

** These systems/evolutions may be eliminated from the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan.

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	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7) Reactor Trip, Stabilization, Recovery / 1				X			Ability to operate and monitor the following as they apply to a reactor trip: (CFR 41.7 / 45.5 / 45.6) EA1.03 RCS pressure and temperature	4.2	40
000008 (APE 8) Pressurizer Vapor Space Accident / 3						X	2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6)	4.5	42
000009 (EPE 9) Small Break LOCA / 3									
000011 (EPE 11) Large Break LOCA / 3	X						Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: (CFR 41.8 / 41.10 / 45.3) EK1.01 Natural circulation and cooling, including reflux boiling	4.1	55
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4				X			Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): (CFR 41.7 / 45.5 / 45.6) AA1.03 Reactor trip alarms, switches, and indicators	3.7*	52
000022 (APE 22) Loss of Reactor Coolant Makeup / 2			X				Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: (CFR 41.5, 41.10 / 45.6 / 45.13) AK3.06 RCP thermal barrier cooling	3.2	45
000025 (APE 25) Loss of Residual Heat Removal System / 4		X					Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: (CFR 41.7 / 45.7) AK2.01 RHR heat exchangers	2.9	44
000026 (APE 26) Loss of Component Cooling Water / 8									
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3				X			Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: (CFR 41.7 / 45.5 / 45.6) AA1.01 PZR heaters, sprays, and PORVs	4.0	46
000029 (EPE 29) Anticipated Transient Without Scram / 1									
000038 (EPE 38) Steam Generator Tube Rupture / 3					X		Ability to determine or interpret the following as they apply to a SGTR: (CFR 43.5 / 45.13) EA2.12 Status of MSIV activating system	3.9*	53
000040 (APE 40) Steam Line Rupture—Excessive Heat Transfer / 4					X		Ability to determine and interpret the following as they apply to the Steam Line Rupture: (CFR: 43.5 / 45.13) AA2.01 Occurrence and location of a steam line rupture from pressure and flow indications	4.2	41
000054 (APE 54) Loss of Main Feedwater /4									
000055 (EPE 55) Station Blackout / 6			X				Knowledge of the reasons for the following responses as the apply to the Station Blackout: (CFR 41.5 / 41.10 / 45.6 / 45.13) EK3.02 Actions contained in EOP for loss of offsite and onsite power	4.3	39
000056 (APE 56) Loss of Offsite Power / 6						X	2.4.46 Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.2	54

000057 (APE 57) Loss of Vital AC Instrument Bus / 6			X				Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: (CFR 41.5,41.10 / 45.6 / 45.13) AK3.01 Actions contained in EOP for loss of vital ac electrical instrument bus	4.1	51
000058 (APE 58) Loss of DC Power / 6	X						Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: (CFR 41.8 / 41.10 / 45.3) AK1.01 Battery charger equipment and instrumentation	2.8	43
000062 (APE 62) Loss of Nuclear Service Water / 4			X				Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: (CFR 41.4, 41.8 / 45.7) AK3.03 Guidance actions contained in EOP for Loss of nuclear service water	4.0	49
000065 (APE 65) Loss of Instrument Air / 8				X			Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: (CFR 41.7 / 45.5 / 45.6) AA1.03 Restoration of systems served by instrument air when pressure is regained	2.9	56
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6									
(W E04) LOCA Outside Containment / 3		X					Knowledge of the interrelations between the (LOCA Outside Containment) and the following: (CFR: 41.7 / 45.7) EK2.2 Facility*s heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	3.8	50
(W E11) Loss of Emergency Coolant Recirculation / 4						X	2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)	4.0	48
(W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4					X		Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) (CFR: 43.5 / 45.13) EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	3.7	47
K/A Category Totals:	2	2	4	4	3	3	Group Point Total:		18

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	**K1**	**K2**	**K3**	**A1**	**A2**	**G***	**K/A Topic(s)**	**IR**	**#**
000001 (APE 1) Continuous Rod Withdrawal / 1					X		Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: (CFR: 43.5 / 45.13) AA2.05 Uncontrolled rod withdrawal, from available indications	4.4	62
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7			X				Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: (CFR 41.5,41.10 / 45.6 / 45.13) AK3.01 Termination of startup following loss of intermediate range instrumentation	3.2	59
000036 (APE 36) Fuel-Handling Incidents / 8									
000037 (APE 37) Steam Generator Tube Leak / 3						X	G2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	3.8	63
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9						X	2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.1	57
000060 (APE 60) Accidental Gaseous Radwaste Release / 9		X					Knowledge of the interrelations between the Accidental Gaseous Radwaste Release and the following: (CFR 41.7 / 45.7) AK2.02 Auxiliary building ventilation system	2.7	65
000061 (APE 61) Area Radiation Monitoring System Alarms / 7						X	AA2.01 - Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: - ARM panel displays	3.5	60
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74) Inadequate Core Cooling / 4	X						Knowledge of the operational implications of the following concepts as they apply to the Inadequate Core Cooling: (CFR 41.8 / 41.10 / 45.3) EK1.03 - Processes for removing decay heat from the core	4.5	64

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				X			Ability to operate and / or monitor the following as they apply to the (Steam Generator Overpressure) (CFR: 41.7 / 45.5 / 45.6) EA1.2 Operating behavior characteristics of the facility	3.0	61
(W E15) Containment Flooding / 5									
(W E16) High Containment Radiation /9	X						Knowledge of the operational implications of the following concepts as they apply to the (High Containment Radiation) (CFR: 41.8 / 41.10, 45.3) EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the (High Containment Radiation)	3.0	58
(BW A01) Plant Runback / 1									
(BW A02 & A03) Loss of NNI-X/Y/7									
(BW A04) Turbine Trip / 4									
(BW A05) Emergency Diesel Actuation / 6									
(BW A07) Flooding / 8									
(BW E03) Inadequate Subcooling Margin / 4									
(BW E08; W E03) LOCA Cooldown—Depressurization / 4									
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4									
(BW E13 & E14) EOP Rules and Enclosures									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4									
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals:	2	1	2	1	1	2	Group Point Total:		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 (SF4P RCP) Reactor Coolant Pump			X									Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: (CFR: 41.7 / 45.6) K3.02 S/G	3.5	1
004 (SF1; SF2 CVCS) Chemical and Volume Control						X						Knowledge of the effect of a loss or malfunction on the following CVCS components: (CFR: 41.7 / 45.7) K6.29 Methods of pressure control of solid plant (PZR relief and water inventory)	3.8	15
005 (SF4P RHR) Residual Heat Removal							X					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: (CFR: 41.5 / 45.5) A1.03 Closed cooling water flow rate and temperature	2.5	16
006 (SF2; SF3 ECCS) Emergency Core Cooling		X										Knowledge of bus power supplies to the following: (CFR: 41.7) K2.04 ESFAS-operated valves	3.6	22
007 (SF5 PRTS) Pressurizer Relief/Quench Tank								X				Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.06 Bubble formation in PZR	2.6	28
008 (SF8 CCW) Component Cooling Water									X			Ability to monitor automatic operation of the CCWS, including: (CFR: 41.7 / 45.5) A3.02 Operation of the CCW pumps, including interlocks and the CCW booster pump	3.2	7
010 (SF3 PZR PCS) Pressurizer Pressure Control					X							Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: (CFR: 41.5 / 45.7) K5.01 Determination of condition of fluid in PZR, using steam tables	3.5	25
012 (SF7 RPS) Reactor Protection					X							Knowledge of the operational implications of the following concepts as the apply to the RPS: (CFR: 41.5 / 45.7) K5.02 Power density	3.3*	24
013 (SF2 ESFAS) Engineered Safety Features Actuation			X									Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: (CFR: 41.7 / 45.6) K3.03 Containment	4.3	23
022 (SF5 CCS) Containment Cooling	X											Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.01 SWS/cooling system	3.5	21
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray				X								Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) K4.04 Reduction of temperature and pressure in containment after a LOCA by condensing steam, to reduce radiological hazard, and protect equipment from corrosion damage (spray)	3.7	3

039 (SF4S MSS) Main and Reheat Steam			X									Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: (CFR: 41.7 / 45.6) K3.03 AFW pumps	3.2*	12
059 (SF4S MFW) Main Feedwater								X				Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.07 Tripping of MFW pump turbine	3.0*	4
061 (SF4S AFW) Auxiliary/Emergency Feedwater						X						Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: (CFR: 41.7 / 45.7) K6.02 Pumps	2.6	6
062 (SF6 ED AC) AC Electrical Distribution							X					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: (CFR: 41.5 / 45.5) A1.01 Significance of D/G load limits	3.4	27
063 (SF6 ED DC) DC Electrical Distribution	X											Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.02 AC electrical system	2.7	10
064 (SF6 EDG) Emergency Diesel Generator									X			Ability to monitor automatic operation of the ED/G system, including: (CFR: 41.7 / 45.5) A3.01 Automatic start of compressor and ED/G	4.1	14
073 (SF7 PRM) Process Radiation Monitoring											X	2.2.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)	3.6	19
076 (SF4S SW) Service Water	X											Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.15 FPS	2.5	26
078 (SF8 IAS) Instrument Air										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.01 Pressure gauges	3.1	20
103 (SF5 CNT) Containment								X				Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2 03 Phase A and B isolation	3.5*	17
053 (SF1; SF4P ICS*) Integrated Control														
006 (SF2; SF3 ECCS) Emergency Core Cooling											X	2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)	3.9	9
008 (SF8 CCW) Component Cooling Water							X					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: (CFR: 41.5 / 45.5) A1.02 CCW temperature	2.9	18

010 (SF3 PZR PCS) Pressurizer Pressure Control									X			Ability to monitor automatic operation of the PZR PCS, including: (CFR: 41.7 / 45.5) A3.02 PZR Pressure (3.6)	3.6	5
026 (SF5 CSS) Containment Spray		X										Knowledge of bus power supplies to the following: (CFR: 41.7) K2.01 Containment spray pumps	3.4*	11
076 (SF4S SW) Service Water										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.04 Emergency heat loads	3.5*	8
078 (SF8 IAS) Instrument Air				X								Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) K4.03 Securing of SAS upon loss of cooling water	3.1*	2
103 (SF5 CNT) Containment				X								Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) K4.06 Containment isolation system	3.1	13
K/A Category Point Totals:	2	2	3	3	2	2	3	3	3	3	2	Group Point Total:		28

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 (SF1 CRDS) Control Rod Drive														
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control														
014 (SF1 RPI) Rod Position Indication					X							Knowledge of the operational implications of the following concepts as they apply to the RPIS: (CFR: 41.5 / 45.7) K5.01 Reasons for differences between RPIS and step counter	2.7	37
015 (SF7 NI) Nuclear Instrumentation				K								K4.03 Knowledge of NIS design feature(s) and/or interlock(s) provide for the following: Reading of source range/intermediate range/power range outside control room	3.9	29
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor			X									Knowledge of the effect that a loss or malfunction of the ITM system will have on the following: (CFR: 41.7 / 45.6) K3.01 Natural circulation indications	3.5*	38
027 (SF5 CIRS) Containment Iodine Removal										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.03 CIRS fans	3.3*	34
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control											X	2.1.27 Ability to use plant computers to evaluate system or component status. (CFR: 41.10 / 45.12): Knowledge of system purpose and/or function	3.9	35
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling							X					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel A1.02 Pool Cooling System operating the controls including: (CFR: 41.5 / 45.5) A1.02 Radiation monitoring systems	2.8	31
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator														
041 (SF4S SDS) Steam Dump/Turbine Bypass Control								X				Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.02 Steam valve stuck open	3.6	32
045 (SF 4S MTG) Main Turbine Generator	X											Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.18 RPS	3.6	36
055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste									X			Ability to monitor automatic operation of the Liquid Radwaste System including: (CFR: 41.7 A3.02 Automatic isolation	3.6	33

071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air														
086 Fire Protection						X						Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the: (CFR: 41.7 / 45.7) K6.04 Fire, smoke, and heat detectors	2.6	30
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:	1	0	1	1	1	1	1	1	1	1	1	Group Point Total:		10

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.	G2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen)..	3.4	73		
	2.1.	2.1.37 Knowledge of procedures, guidelines, or limitations associated with reactivity management. (CFR: 41.1 / 43.6 / 45.6)	4.3	74		
	2.1.	2.1.44 Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation. (CFR: 41.10 / 43.7 / 45.12)	3.9	66		
	Subtotal		3			
2. Equipment Control	2.2.	2.2.2 Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels. (CFR: 41.6 / 41.7 / 45.2)	4.6	68		
	2.2.	2.2.40 Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)	3.4	67		
	2.2.	2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)	4.2	70		
	Subtotal		3			
3. Radiation Control	2.3.	2.3.11 Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)	3.8	75		
	2.3.	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. (CFR: 41.12 / 45.9 / 45.10)	3.2	71		
	Subtotal		2			
4. Emergency Procedures/Plan	2.4.	2.4.17 Knowledge of EOP terms and definitions. (CFR: 41.10 / 45.13)	3.9	69		
	2.4.	2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)	3.8	72		
	Subtotal		2			
Tier 3 Point Total				10		

Tier / Group	Randomly Selected K/A	Reason for Rejection
T1/G2	APE 061 G2.4.8 (#63)	Oversampled and unable to write a question to KA and not have overlap with other similar (rad montor KA's). Also no instances of EOP /AOP usage for Area Rad monitor alarms. Kept G2.4.8 and selected new system, APE 037
T1/G2	EPE 074 EK1.07 (#64)	Covered with question 25 – 010 K5.01. Selected EK1.03 (IR 4.3)
T2/G1	010 A3.02 (#05)	Unable to write satisfactory question to meet KA. Selected only other available A3. A3.02 (IR 3.6)
T2/G1	076 K1.05 (#26)	The relationship of SWS to D/G does not exist at the station. Selected K1.15 (IR 2.5).
T2/G2	029 K4.02 (#29)	No testable tie between negative pressure and containment – no interlocks/design features. DCPP is not a negative containment pressure plant. Selected the only other available KA, K4.03 (IR 3.2)
T2/G2	034 A1.02 (#31)	No testable RO tie to Refueling Canal. Because there is no other A1 with IR greater than 2.5, shifted to SF8 system 033 A1.02 (IR2.8)
T2/G2	028 G2.1.19 (#35)	No tie between computer and HRPS. Selected G2.1.27 (IR 3.9)
T3/G1	2.1.28 (#73)	Unable to write a Tier 3 question to this KA. Selected G2.1.26 (IR 3.4)
T2/G2	045 K1.06 (#36)	Unable to write satisfactory question. KA is very specific and trivial knowledge of a seldomly performed surveillance. From the remaining K1, randomly selected K1.18,RPS, (IR 3.6)
T1G2	APE 024 AK3.02 (#31)	Replaced due to overlap with SRO question 93. Replaced with APE 061 AA2.01 (IR 3.5)
T2/G2	029 K4.02 (#29)	Replaced due to overlap with SRO question. Replaced with 015 K4.03 (IR 3.9)
T2G1	004 K6.29	Replaced due to overlap with Operating test. Replaced with 004 K6.26 (IR 3.8)

Facility: Diablo Canyon Date of Exam: March 6, 2020

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1				N/A					N/A			18	3	3	6
	2												9	2	2	4
	Tier Totals												27	5	5	10
2. Plant Systems	1												28	3	2	5
	2												10	1	2	3
	Tier Totals												38	4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 / 2	3 / 4	7
														2 / 2	1 / 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 outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)

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 outline. Systems or evolutions that do not apply at the facility should be deleted with justification. 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' 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 a category 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.

G* Generic K/As

* These systems/evolutions must be included as part of the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan. They are not required to be included when using earlier revisions of the K/A catalog.

** These systems/evolutions may be eliminated from the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan.

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1									
000008 (APE 8) Pressurizer Vapor Space Accident / 3									
000009 (EPE 9) Small Break LOCA / 3					X		Ability to determine or interpret the following as they apply to a small break LOCA: (CFR 43.5 / 45.13) EA2.34 Conditions for throttling or stopping HPI	4.2	88
000011 (EPE 11) Large Break LOCA / 3									
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4									
000022 (APE 22) Loss of Reactor Coolant Makeup / 2									
000025 (APE 25) Loss of Residual Heat Removal System / 4									
000026 (APE 26) Loss of Component Cooling Water / 8						X	2.4.18 Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)	4.0	89
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3									
000029 (EPE 29) Anticipated Transient Without Scram / 1						X	2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	4.2	87
000038 (EPE 38) Steam Generator Tube Rupture / 3									
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4									
000054 (APE 54; CE E06) Loss of Main Feedwater /4					X		Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): (CFR: 43.5 / 45.13) AA2.08 Steam flow-feed trend recorder	3.3*	86
000055 (EPE 55) Station Blackout / 6									
000056 (APE 56) Loss of Offsite Power / 6									
000057 (APE 57) Loss of Vital AC Instrument Bus / 6									
000058 (APE 58) Loss of DC Power / 6									
000062 (APE 62) Loss of Nuclear Service Water / 4									
000065 (APE 65) Loss of Instrument Air / 8									
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6						X	2.4.30 - Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	4.1	85

(W E04) LOCA Outside Containment / 3					X		Ability to determine and interpret the following as they apply to the (LOCA Outside Containment) (CFR: 43.5 / 45.13) EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.	4.2	84
(W E11) Loss of Emergency Coolant Recirculation / 4									
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4									
K/A Category Totals:					3	3	Group Point Total:		6

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	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1									
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1						X	2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	4.7	93
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7									
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8						X	2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8)	4.6	91
000037 (APE 37) Steam Generator Tube Leak / 3									
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9									
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4									
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 E08; W E03) LOCA Cooldown—Depressurization / 4									
(W E10) Natural Circulation with Steam Void in Vessel with/without RVLIS/4					X		Ability to determine and interpret the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS) (CFR: 43.5 / 45.13) EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	3.9	90
(BW E13 & E14) EOP Rules and Enclosures									

(W E08) RCS Overcooling—Pressurized Thermal Shock / 4					X		EA2. Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock) (CFR: 43.5 / 45.13) EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	4.2	92
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
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	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump														
004 (SF1; SF2 CVCS) Chemical and Volume Control								X				Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5/ 43/5 / 45/3 / 45/5) A2.27 Improper RWST boron concentration	4.2	76
005 (SF4P RHR) Residual Heat Removal														
006 (SF2; SF3 ECCS) Emergency Core Cooling														
007 (SF5 PRTS) Pressurizer Relief/Quench Tank														
008 (SF8 CCW) Component Cooling Water														
010 (SF3 PZR PCS) Pressurizer Pressure Control														
012 (SF7 RPS) Reactor Protection														
013 (SF2 ESFAS) Engineered Safety Features Actuation								X				Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.01 LOCA	4.8	80
022 (SF5 CCS) Containment Cooling														
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray														
039 (SF4S MSS) Main and Reheat Steam											X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)	4.7	77
059 (SF4S MFW) Main Feedwater														
061 (SF4S AFW) Auxiliary/Emergency Feedwater								X				Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.07 Air or MOV failure	3.5	78
062 (SF6 ED AC) AC Electrical Distribution														

063 (SF6 ED DC) DC Electrical Distribution											X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)	4.2	79
064 (SF6 EDG) Emergency Diesel Generator														
073 (SF7 PRM) Process Radiation Monitoring														
076 (SF4S SW) Service Water														
078 (SF8 IAS) Instrument Air														
103 (SF5 CNT) Containment														
053 (SF1; SF4P ICS*) Integrated Control														
K/A Category Point Totals:								3			2	Group Point Total:		5

ES-401	PWR Examination Outline Plant Systems—Tier 2/Group 2 (RO/SRO)													Form ES-401-2
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive														
002 (SF2; SF4P RCS) Reactor Coolant											X	2.4.4. Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.6 / 45.4)	34.7	83
011 (SF2 PZR LCS) Pressurizer Level Control														
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation														
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor														
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge											X	2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	4.0	81
033 (SF8 SFPCS) Spent Fuel Pool Cooling														
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator								X				Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5) A2.03 Pressure/level transmitter failure	3.6	82
041 (SF4S SDS) Steam Dump/Turbine Bypass Control														
045 (SF 4S MTG) Main Turbine Generator														
055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air														
086 Fire Protection														
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:								1			2	Group Point Total:		3

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.	2.1.34 Knowledge of primary and secondary plant chemistry limits. (CFR: 41.10 / 43.5 / 45.12)			3.5	94
	2.1.	2.1.41 Knowledge of the refueling process.			3.7	97
	Subtotal				2	
2. Equipment Control	2.2.	2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)			4.6	98
	2.2.	2.2.43 Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)			3.3	95
	Subtotal				2	
3. Radiation Control	2.3.	2.3.13 Knowledge of radiological safety 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, aligning filters, etc. (CFR: 41.12 / 43.4 / 45.9 / 45.10)			3.8	99
	Subtotal				2	
4. Emergency Procedures/Plan	2.4.	2.4.29 Knowledge of the emergency plan. (CFR: 41.10 / 43.5 / 45.11)			4.4	96
	2.4.	2.4.40 Knowledge of SRO responsibilities in emergency plan implementation. (CFR: 41.10 / 43.5 / 45.11)			4.5	100
	Subtotal				2	
Tier 3 Point Total						7

Tier / Group	Randomly Selected K/A	Reason for Rejection
T2/G2	079 G2.1.32 (#81)	Station Air – cannot write SRO level question to KA. Selected System 029 – Containment Purge, KA unchanged.
T1/G1	APE 077 G2.4.21 (#85)	Unable to tie APE to generic KA dealing with critical safety functions. Shifted to 2.4.30 (IR 4.1)
T3/G1	G2.1.43 (#97)	Unable to write a suitable SRO level question. Replaced with 2.1.41 (IR 3.7)
T2/G2	002 G2.4.3 (83)	Unable to write a suitable SRO level question. Replaced with 002 2.4.4 (IR 4.7

Facility: Diablo Canyon   Date of Examination: 02/24/2020

Examination Level: RO ☒ SRO ☐   Operating Test Number: L181

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations (NRCL181-A1)	M, R	**Determine H2 Recombiner Settings** 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (4.3) (STP Exam 09-2017)
Conduct of Operations (NRCL181-A2)	M, R	**Estimate Decay Heat and Heat Removal Rate** 2.1.25 Ability to interpret reference materials such as graphs, curves, tables, etc. (3.9) (Bank: LJC-014)
Equipment Control (NRCL181-A3)	N, R	**Perform STP I-1A** 2.2.37 Ability to determine Operability and/or availability of safety related equipment. (3.6)
Radiation Control (NRCL181-A4)	N, R	**Perform RM-19 Channel Check** 2.3.5 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (2.9)

NOTE: All items (five total) are required for SROs. RO applicants require only four items unless they are retaking only the administrative topics (which would require all five items).

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

Facility: Diablo Canyon	Date of Examination: 02/24/2020
Examination Level: RO ☐ SRO ☒	Operating Test Number: L181

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations (NRCL181-A5)	M, R	**Apply Overtime Limit Restrictions** 2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc (3.9) (NRCADM061-COO-SRO1)
Conduct of Operations (NRCL181-A6)	M, R	**Evaluate Fire Zone Operability** 2.1.25 Ability to interpret reference materials such as graphs, curves, tables, etc. (4.2) (Bank: LJC-014)
Equipment Control (NRCL181-A7)	N, R	**Determine 230 kV Operability** 2.2.37 Ability to determine Operability and/or availability of safety related equipment. (4.6)
Radiation Control (NRCL181-A8)	M, R	**Authorize Emergency Exposure** 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (3.7) (NRCL161-A8)
Emergency Plan (NRCL181-A9)	N, R	**Review Emergency Notification for Steam Generator Tube Rupture** 2.4.40 Knowledge of SRO responsibilities in emergency plan implementation. (4.5)

NOTE: All items (five total) are required for SROs. RO applicants require only four items unless they are retaking only the administrative topics (which would require all five items).

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

Facility: Diablo Canyon	Date of Examination: 02/24/2020
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test Number: L181

Control Room Systems:* 8 for RO, 7 for SRO-I, and 2 or 3 for SRO-U

System/JPM Title	Type Code*	Safety Function
a. (S1) (001.A2.11) Dropped Rods During Rod Misalignment Verification (Modified LJC-066)	A,M,S	1
b. (S2) (013.A2.01) Resp to Changing Plant Params During Rx Trip Resp	A,N,EN,L,S	2
c. (S3) (E04.EA1.1) Isolate LOCA Outside Containment (Bank LJC-118)	D,L,S	3
d. (S4P) (011.EA1.11) Transfer to Cold Leg Recirc (Bank LJC-27A)	A,D,L,S	4P
e. (S4S) (059.A2.07) Perform OP AP-15 Immediate Actions for MFP Trip (Bank LJC-247)	A,D,S	4S
f. (S5) (E14.E1.2) Manually Initiate Containment Spray (Bank LJC-010)	D,L,S	5
g. (S6) (064.A4.06) Crosstie Vital Bus G to H (LJC-032)	D,L,S	6
h. (S8) (067.AA2.17) Fire in 480V Bus G Switchgear Room	A,N,S	8

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

System/JPM Title	Type Code*	Safety Function
i. (P1) (010.A2.02) Transfer Pzr Heater Grp 23 to Backup Pwr (LJP-029A)	D	3
j. (P2) (062.A2.11) Transfer the TSC to Vital Power(LJP-058A)	A,D,E,L	6
k. (P3) (G2.1.30) Clear Component Cooling Water Header "A"	E,L,N,R	8

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

* Type Codes	Criteria for R /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: 02/24/2020
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test Number: L181

Control Room Systems:* 8 for RO, 7 for SRO-I, and 2 or 3 for SRO-U

System/JPM Title	Type Code*	Safety Function
a. (S1) (001.A2.11) Dropped Rods During Rod Misalignment Verification (Modified LJC-066)	A,M,S	1
b. (S2) (013.A2.01) Resp to Changing Plant Params During Rx Trip Resp	A,N,EN,L,S	2
c. (S3) (E04.EA1.1) Isolate LOCA Outside Containment (Bank LJC-118)	D,L,S	3
d. (S4P) (011.EA1.11) Transfer to Cold Leg Recirc (Bank LJC-27A)	A,D,L,S	4P
e. (S4S) (059.A2.07) Perform OP AP-15 Immediate Actions for MFP Trip (Bank LJC-247)	A,D,S	4S
f. (S5) (E14.E1.2) Manually Initiate Containment Spray (Bank LJC-010)	D,L,S	5
g.		
h. (S8) (067.AA2.17) Fire in 480V Bus G Switchgear Room	A,N,S	8

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

i. (P1) (010.A2.02) Transfer Pzr Heater Grp 23 to Backup Pwr (LJP-029A)	D	3
j. (P2) (062.A2.11) Transfer the TSC to Vital Power(LJP-058A)	A,D,E,L	6
k. (P3) (G2.1.30) Clear Component Cooling Water Header "A"	E,L,N,R	8

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

* Type Codes	Criteria for R /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: 02/24/2020
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒	Operating Test Number: L181

Control Room Systems:* 8 for RO, 7 for SRO-I, and 2 or 3 for SRO-U

System/JPM Title	Type Code*	Safety Function
a. (S1) (001.A2.11) Dropped Rods During Rod Misalignment Verification (Modified LJC-066)	A,M,S	1
b. (S2) (013.A2.01) Resp to Changing Plant Params During Rx Trip Resp	A,N,EN,L,S	2
c.		
d.		
e.		
f.		
g.		
h.		

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

System/JPM Title	Type Code*	Safety Function
i. (P1) (010.A2.02) Transfer Pzr Heater Grp 23 to Backup Pwr (LJP-029A)	D	3
j. (P2) (062.A2.11) Transfer the TSC to Vital Power(LJP-058A)	A,D,E,L	6
k. (P3) (G2.1.30) Clear Component Cooling Water Header "A"	E,L,N,R	8

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

* Type Codes	Criteria for R /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:** L181 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: **2% with MFW in service, aligned to Start-Up Power. MOL with CFCU 1-1 OOS.**

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	VLV_CVC22_2 .5 delay=0 ramp=15	**I (ALL)**	Regen Hx Isolation Valve, LCV-459, fails to mid-position (**OP AP-18**).
3	H_V1_034M_1, XMT_VEN6_3, XMT_VEN7_3, XMT_VEN8_3	**TS, C (BOP, SRO)**	CFCU 1-2 high stator/bearing temperature due to low CCW flow (**AR PK01-21, TS 3.6.6.C**).
4	RLY_PPL63_2 OPEN RLY_PPL59_2 OPEN	**TS, I (ALL)**	SSPS relay actuation causes inadvertent start of TDAFW pump and blowdown sample isolation valves to close (**AR PK04-03, OP D-1:III, OP1.DC10**; **TS 3.7.5.B**).
5	MAL_MSS4 1.57E+07 ramp=30	**M (ALL)**	MSLB outside containment.
6	VLV_MSS7_2, VLV_MSS8_2, VLV_MSS9_2, VLV_MSS10_2 1	**C (ALL)**	All MSIVs fail open; No manual close for FCV-42.
7	MAL_PPL3B BOTH	**C (BOP)**	Safety Injection, Train B fails to actuate.

***(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) (Events 6,7)	2
3. Abnormal events (1–4) (Events 2,3,4)	3
4. Major transients (1-2) (Event 5)	1
5. EOPs entered/requiring substantive actions (1–2) (E-2, E-1.1)	2
6. EOP contingencies requiring substantive actions (0–2)	0
7. Critical tasks (2–3)(See description below)	3

Critical Task	Justification	Reference
(S1CT-1) Shutdown TD AFW pump prior to any Steam Generator Overfill (S/G wide range greater than 100%) by either: • Closing LCV-106,107, 108, 109 to the individual S/Gs OR • Closing steam supply valves FCV-37 and FCV-38 to leads 2 and 3 respectively OR • Directing FCV-95 closed in the field	Carryover into the steam lines can result in damage to downstream piping and valves, placing the secondary heat sink at risk. High steam generator level can also result in reactivity excursions due to excessive cooldown of the primary system.	• Tech Spec 3.3.2 Basis Documentation
(S1CT-2) Stop uncontrolled RCS cooldown before a severe challenge to Integrity Safety Function develops (magenta path on F-0.4 RCS Integrity) as follows: • Close Main Steam Isolation Valves FCV-41, FCV-43, FCV-44. • Dispatch Operator to close FCV-42 (S/G 1-2 steamline isolation). • Isolate feed flow to S/G 1-2 by closing/verifying closed LCV-107 and LCV-111. *(Note: LCV-107 is critical only when TDAFW pump is running or capable of an autostart).* • Isolate steam flow from S/G 1-2 by closing/verifying closed FCV-37. • Maintains the minimum heatsink requirements (435 gpm until S/G NR level is greater than 15% in one non-faulted S/G) by controlling flow to S/Gs 1-1, 1-3, and 1-4.	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
(S1CT-3) Terminate SI prior to rupture of PRT by closing 8801A/B OR 8803A/B.	Failure to terminate ECCS flow when SI termination criteria are met results in overfill of the Pressurizer and the eventual rupture of the PRT. This constitutes the avoidable degradation of the RCS as a fission product barrier.	• Westinghouse Owner's Group WCAP-17711-NP

Per NUREG-1021, Appendix D, if an operator or 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.

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. Regen Hx Isolation Valve, LCV-459, drifts to mid-position causing letdown orifice valve 8149C to close. Shift Foreman enters **OP AP-18, Letdown Line Failure**. Excess Letdown is established per **OP B-1A:IV CVCS - Excess Letdown - Place In Service and Remove From Service.**

3. CFCU 1-2 has a 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. The crew secures the CFCU to prevent motor damage and contacts 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).

4. SSPS relay actuation results in Turbine Driven AFW (TDAFW) Pump Steam Supply Isolation Valve, FCV-95, failing open and isolation of half of the blowdown sample valves inside and outside containment. S/G levels rise and RCS temperature lowers. FCV-95 cannot be closed and the crew must isolate the TDAFW Pump by closing the LCVs to the individual S/Gs or by closing steam supply valves FCV-37 and FCV-38 from leads 2 and 3 respectively, or by directing FCV-95 manually closed in the field **(S1CT-1) Shutdown TD AFW pump prior to Steam Generator Overfill**. Shift Foreman implements **TS 3.7.5.B, AFW System** for one AFW train inoperable (72 hrs).

5. A main steamline break develops downstream of the Main Steam Isolation Valves, outside containment. S/G pressure drops rapidly resulting in an automatic Reactor Trip and Safety Injection. The crew enters **EOP E-0, Reactor Trip or Safety Injection.**

6. Train B of Safety Injection fails to actuate, requiring the crew to perform numerous manual alignments and pump starts as part of **Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status.**

7. All four main steam isolation valves fail open. Steam leads 1, 3, and 4 may be closed from the control room, but lead 2 (FCV-42) requires field action. The crew transitions to **EOP E-2, Faulted Steam Generator Isolation** to isolate S/G 1-2 and dispatches an operator to locally close FCV-42 as part of the critical task to stop the uncontrolled cooldown **(S1CT-2) Stop uncontrolled cooldown before a severe challenge (magenta path) develops on F-0.4 RCS Integrity.**

8. The crew transitions to **EOP E-1.1, SI Termination** where they complete the final critical task of the scenario (**S1CT-3) Terminate SI prior to rupture of PRT**.

The scenario is terminated once the final critical task is complete.

Facility: Diablo Canyon (PWR) **Scenario No:** 2 **Op-Test No:** L181 NRC

Examiners: **Operators:**

Initial Conditions: **75% Power, MOL with AFW 1-2 cleared for a bearing oil leak**

Turnover: **At 75% power for SCCW HX Clearance**

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	H5ESP_INIT_455G 0 ramp=20 RCCIPCSDI_H5DHC455GSPUPTFFREEZE TRUE	**I (ATC, SRO)**	Pressurizer Spray Valve PCV-455A setpoint failure causing RCS pressure to slowly lower (**AR PK05-16, OP AP-13**).
2	XMT_PZR24_3 ramp=1	**TS, I (BOP, SRO)**	PT-474, Pressurizer Pressure Transmitter, Fails Low (**OP AP-5**, **TS 3.3.1.E, M, 3.3.2.D, 3.4.11.B**).
3	XMT_CVC2_3 ramp=75	**I (BOP, SRO)**	PT-135 Fails High causing letdown pressure control valve to go full open (**AR PK04-21**).
4	MAL_RCS4H 30.0	**TS, C (ALL)**	30 gpm SGTL on loop 4; plant shutdown required (**OP AP-3, OP AP-25, TS 3.4.13.B**).
5	MAL_RCS4H 400.0 ramp=60	**M (ALL)**	Tube leak grows to 400 gpm rupture during ramp offline.
6	MAL_EPS5A_2 DIFFERENTIAL cd='H_V5_194B_1' delay=10	**C (BOP, SRO)**	12 kV Bus D feeder breaker trips on differential on transfer to startup power.
7	MAL_EPS4D_2 DIFFERENTIAL cd='h_v4_221r_1'	**C (ALL)**	4kV Bus G differential trip on transfer to startup power.

***(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 6,7)	2
3. Abnormal events (1–4) (Events 1,2,3,4)	4
4. Major transients (1-2) (Event 5)	1
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)	3

Critical Task	Justification	Reference
(S2CT-1) Isolate the ruptured steam generator from the intact steam generators prior to commencing cooldown of the RCS in step 10.b (10% steam dump) by completing the following: Isolate feedwater by ensuring closed: LCV-109 (TDAFW Level Control Valve) LCV-113 (MDAFW Level Control Valve) Isolate steam flow by closing FCV-44 (MSIV)	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
(S2CT-2) Perform RCS cooldown at maximum rate to CETC target temperature specified in E-3, step 6, using steam dumps such that RCS subcooled margin still exists following the cooldown. *Maximum rate cooldown requires 10% steam dumps on intact S/Gs to be at least 90% open.*	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
(S2CT-3) Depressurize the RCS to meet depressurization criteria specified in E-3, App GG prior to stopping any Safety Injection pump.	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
Per NUREG-1021, Appendix D, if an operator or 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.		

SCENARIO SUMMARY – NRC #2

1. Pressurizer Spray Valve controller failure causes PCV-455A to ramp open and RCS pressure begins to lower slowly. **PK05-16, PZR PRESSURE HI/LO** alarms when RCS pressure reaches 2210 psig. The crew follows AR PK05-16 guidance to take manual control and close the spray valve. The crew may follow up with the actions of **OP AP-13, Malfunction of Reactor Pressure Control System** or **OP AP-5, Malfunction of Eagle-21 Protection or Control Channel**, to restore pressure to normal using manual control. Alternately, the crew may diagnose the failure prior to the annunciator response activating and enter OP AP-13 directly which also directs taking manual control and closing the spray valve.

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. 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. Steam Generator 1-4 develops a 30 gpm tube leak as indicated by rising counts on various radiation monitors. The crew enters **OP AP-3, Steam Generator Tube Failure**. Shift Foreman determines **TS 3.4.13.B, RCS Operational Leakage** applies and enters **OP AP-25, Rapid Load Reduction or Shutdow** for the ramp off-line.

5. During the ramp the tube leak develops into a 400 gpm rupture. The crew determines the leak is substantial in size based on a rapid drop in pressurizer level. The Shift Foreman directs a reactor trip and safety injection and the crew enters **EOP E-0, Reactor Trip or Safety Injection.**

(continued)

SCENARIO SUMMARY – NRC #2

6. On the transfer to start up power, 4 kV bus G experiences a differential fault. 12kV bus D also trips on differential causing a loss of vacuum as well as tripping of RCPs 1-2 and 1-4.

7. The crew transitions to **EOP E-3, Steam Generator Tube Rupture,** based on RM-74 and rising S/G 1-4 level, where they address the following critical tasks:

    - **(S2CT-1) Isolate the ruptured steam generator from the intact steam generators prior to commencing cooldown.**
    - **(S2CT-2) Perform RCS cooldown at maximum rate to CETC target temperature.**
    - **(S2CT-3) Depressurize the RCS to meet depressurization criteria specified in Appendix GG.**

The scenario is terminated once the crew has completed critical task S2CT-3.

Facility: Diablo Canyon (PWR) **Scenario No:** 3 **Op-Test No:** L181 NRC

Examiners: ______________________ **Operators:** ______________________

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Initial Conditions: **100% Power, MOL with AFW 1-2 cleared**

Turnover: **At 100% power with AFW 1-2 cleared**

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	PMP_CVC3_2 OVERLOAD_DEV_FAIL	**TS, C (ALL)**	CCP 1-3 OC Trip (**OP AP-17, ECG 8.1.A**).
2	AS01ASW_ASP11_MTFSEIZUR 1	**TS, C (BOP, SRO)**	ASW Pp 1-1 Seizes; Pp 1-2 can not be started (**OP AP-10, TS 3.0.3**).
3	MAL_CWS3A 80 MAL_CWS3B 75 MAL_CWS1A 0.15 ramp=120 MAL_CWS1B 0.15 ramp=150	**C (ALL)**	High DP on Intake Screens; ramp required (**AR PK13-01, OP AP-7, OP AP-25**).
4	PMP_CWS1_2 OVERLOAD_DEV_FAIL cd='smss lt 1140' PMP_CWS2_2 OVERLOAD_DEV_FAIL cd='smss lt 1080'	**M (ALL)**	Both Circ Water pumps trip off during ramp, requiring Reactor Trip (**OP AP-7**).
5	MAL_EPS4C_2 DIFFERENTIAL cd='fnispr lt 5' delay=30	**C (ALL)**	Vital 4kV Bus F differential trip.
6	VLV_PZR4_2 0.3 cd='jpplp4' del ay=60	**C (BOP)**	Pressurizer PORV PCV-455C fails slightly open on trip requiring manual isolation by associated block valve
7	MAL_AFW1 1 cd='fnispr lt 5'	**C (ALL)**	Turbine driven AFW pump overspeed trip.

***(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 5,6,7)	3
3. Abnormal events (1–4) (Events 1,2,3)	3
4. Major transients (1-2) (Event 4)	1
5. EOPs entered/requiring substantive actions (1–2) (E-0.1, FR-H.1)	2
6. EOP contingencies requiring substantive actions (0–2) (FR-H.1)	1
7. Critical tasks (2–3)(See description below)	2

Critical Task	Justification	Reference
(S3CT-1) Close the motor operated block valve upstream of the stuck open PORV prior to rupture of the PRT.	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
(S3CT-2) Establish a secondary heat sink as indicated by: • WR level rising • Core Exit Thermocouple temperatures lowering Prior to reaching bleed and feed criteria which is defined as wide range S/G level in any three S/Gs less than 18% [26%] AND narrow range S/G level in all four S/Gs less than 15% [25%] narrow range.	A loss of all feedwater transient is characterized by a depletion of secondary inventory and eventual degradation of secondary heat transfer capability. As secondary heat transfer capability degrades, core decay heat generation will increase RCS temperature and pressure causing loss of RCS inventory similar in nature to a small break loss of coolant accident. Failure to restore a secondary heat sink when it is possible to do so constitutes "a significant reduction of safety margin beyond that irreparably introduced by the scenario."	• FR-H.1 Background Document (HFRH1BG), Rev. 3.
Per NUREG-1021, Appendix D, if an operator or 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.		

SCENARIO SUMMARY – NRC #3

1. Charging Pump CCP 1-3 trips on over current. The crew responds by entering **OP AP-17, Loss of Charging** to restore normal charging and letdown. Shift Foreman enters **ECG 8.1.A – Charging Pump No. 3 Inoperable** (establish fire watch; restore to operable status within 7 days).

2. ASW Pump 1-1 trips due to a seized shaft. Standby ASW Pump 1-2 fails to autostart and cannot be started manually. The Shift Foreman implements **OP AP-10, Loss of Auxiliary Salt Water** and cross-ties to the Unit 2 ASW system via the ASW cross-tie valve FCV-601. Shift Foreman enters **T.S. 3.0.3** for two trains of ASW inoperable on Unit 1.

3. Screen differential pressure begins to rise quickly, bringing in **AR PK13-01, Bar Racks/Screens**. Following annunciator guidance, the crew enters **OP AP-7, Degraded Condenser, Section C: Traveling Screen Problem** and begins to reduce load to 50% or less per **OP AP-25, Rapid Load Reduction**.

4. Both Circ Water pumps trip off during ramp, requiring the crew to manually trip the Reactor per OP AP-7. The crew enters **EOP E-0, Reactor Trip or Safety Injection** and performs their immediate actions.

5. On the trip, vital 4 kV bus F trips on differential. DRPI loses power, but crew is able to determine the reactor has tripped based on diverse indications (lowering reactor power and reactor trip breakers open). MDAFW Pump 1-3 is also lost due to the bus failure.

6. Board operators also identify PCV-455C in mid-position. The valve will not close and must be isolated using the associated block valve 8000B **(S3CT-1) Close the motor operated block valve upstream of the stuck open PORV prior to rupture of the PRT.**

7. The TDAFW pump trips on overspeed. Steam Generator levels are initially high enough to provide an adequate secondary side heat sink and the crew transitions to **EOP E-0.1, Reactor Trip Response** to stabilize the plant. Steam Generator levels slowly lower below the minimum required level of 15% narrow range and the crew transitions to **EOP FR-H.1, Response to Loss of Secondary Heat Sink**. With the condenser unavailable, Condensate is used to restore a secondary side heat sink **(S3CT-2) Establish a secondary heat sink.**

The scenario is terminated once Critical Task S3CT-2 is complete

Facility: Diablo Canyon (PWR) **Scenario No:** 4 **Op-Test No:** L181 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: 75% Power, MOL with D/G 1-2 OOS

Turnover: At 75% power for SCCW HX Clearance

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	AB01ABV_E2_MTVIGAIN 65 cd='H_V4_176R_1 and H_V4_180L_1' delay=5	**TS, C (BOP, SRO)**	Overcurrent trip on E-2 during Aux Building fan swaps (OP H-1:II, **AR PK 15-17, TS 3.7.12.B**).
2	XMT_CVC19_3 0.0 ramp=120	**I (ATC, SRO)**	LT-112 Fails Low (auto make-up) (**OP AP-19, AP-5**).
3	MAL_RCS3B .07	**TS, C (ALL)**	70 gpm RCS leak on Loop 2 (**OP AP-1, TS 3.4.13.A**).
4	MAL_SEI1 0.21 ramp=10 MAL_GEN4_3 TRIP delay=10 cd='jmlsei1' LOA_SYD6 (SYD7, SYD8, SYD16) OPEN delay=15 cd='jmlsei1' LOA_SYD16 Energized cd='jmlsei1' delay=15	**C (BOP, SRO)**	Seismic event causing Full Load Rejection (**OP AP-2, AP-25**).
5	DR04CND_HDP11_MTFSHEAR 1 PMP_CND1_1 AS_IS dellA PMP_CND1_1 2 delay=0 cd='V3_223S_3'	**C (ATC, SRO)**	Digital Feedwater controller failure requiring reactor trip (**OP AP-2**).
6	MAL_SYD2 0 cd='fnispr lt 5' delay=2 MAL_EPS4E_2 DIFFERENTIAL cd='h_v4_217r_1' MAL_DEG1C_2 NO_RESET cd='H_V4_224R_1'	**M (ALL)**	Loss of all A/C power.
7	VLV_AFW7_1 1 DellA VLV_AFW7_1 2 cd='V3_219S_3'	**C (BOP, SRO)**	TDAFW Pump fails to autostart – manual start 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 1,2,3,4,5,6,7)	7
2. Malfunctions after EOP entry (1-2) (Event 7)	1
3. Abnormal events (1–4) (Events 1,2,3,4,5)	5
4. Major transients (1-2) (Event 6)	1
5. EOPs entered/requiring substantive actions (1–2) (ECA-0.0, ECA-0.2)	2
6. EOP contingencies requiring substantive actions (0–2) (ECA-0.0, ECA-0.2)	2
7. Critical tasks (2–3)(See description below)	2

Critical Task	Justification	Reference
(S4CT-1) Energize at least one vital AC bus prior to implementation of FLEX strategies (ECA-0.0, step 10 RNO) associated with entry into Extended Loss of AC Power Event (ELAP) conditions	Failure to restore vital AC power from an available source when available represents an unnecessary continuation of a degraded electrical condition and unnecessarily complicates the mitigation strategy	• WCAP-17711-NP, CT-24 • ECA-0.0 Background Document (HECA00BG), Rev. 3.
(S4CT-2) Establish flow from at least one safety injection pump prior to transition out of ECA-0.2.	Failure to manually start at least one high/intermediate head injection pump under the postulated conditions constitutes misoperation or incorrect crew performance in which the crew does not prevent degraded core cooling capacity.	• WCAP-17711-NP, CT-7

Per NUREG-1021, Appendix D, if an operator or 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.

SCENARIO SUMMARY – NRC #4

1. The crew performs a normal weekly fan swap for the Aux Building Ventilation System per **OP H-1:II, Auxiliary Building Safeguards Ventilation (ABVS) – Normal Operation, Section 6.1.** Exhaust Fan E-2 trips shortly after the swap bringing in **AR PK15-17, Aux & FHB Vent Pwr Failure.** Exhaust Fan E-1 restarts automatically and a field operator is dispatched to investigate. Shift Foreman enters **TS 3.7.12.B, Auxiliary Building Ventilation System (ABVS)** for one ABVS train inoperable (7 days).
2. Volume Control Tank (VCT) level channel LT-112 fails low, causing a continuous (and erroneous) makeup signal. The crew diagnoses the level channel failure by comparing other VCT parameters, and by using **OP AP-19, Malfunction of the Reactor Makeup Control System**. The makeup system is secured, and makeup is accomplished (if needed) using manual mode (or enabling the auto mode for short periods). Crew may elect to use **OP AP-5, Malfunction of Eagle 21 Protection or Control Channel** to take manual control of **Makeup Control System.**
3. A 70 gpm RCS leak develops, requiring entry in **OP AP-1, Excessive Reactor Coolant System Leakage**. The crew adjusts charging flow and eventually starts a second charging pump to stabilized pressurizer pressure. VCT level can be maintained at the current leak rate, however, and the crew determines a plant shutdown is required. Shift Foreman enters **TS 3.4.13.A, RCS Operational Leakage** (4 hrs).
4. A significant seismic event results in a full load rejection on Unit 1. The crew recognizes the condition based on numerous power level alarms and the ensuing secondary side transient. The crew monitors primary and secondary system responses, 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.
5. During the ramp down, Digital Feedwater fails to manual and the Shift Foreman directs a Reactor trip.
6. Startup power is lost on the trip followed by a bus differential fault on vital 4kV bus H. Diesel Generator 1-3 trips and cannot be reset. The crew transitions to **EOP ECA-0.0, Loss of All Vital AC Power.**
7. The turbine driven AFW pump fails to autostart requiring the crew to manually start the pump and dispatch field operators to throttle flow.
8. Grid Control Center informs the crew that 230kV start up power is not available, but 500 kV is available. The crew performs actions to isolate RCP seal cooling.
9. Power is restored to vital 4kV buses F and G following the guidance of **ECA-0.0, Appendix DD (Backfeed from 500kV Power**) **(S4CT-1) Energize at least one vital AC bus from prior to implementation of FLEX strategies**.
10. The crew transitions to **EOP ECA-0.2, Loss of All AC Power with SI Required** where they manually start safeguards equipment. The crew performs the final scenario critical task **(S4CT-2) Establish flow from at least one safety injection pump.**

The scenario is terminated once injection flow is established.

Facility: Diablo Canyon (PWR) **Scenario No:** 5 **Op-Test No:** L181 NRC

Examiners: **Operators:**

Initial Conditions: 100% Power, MOL with PT-403 OOS

Turnover: At 100% power with PT-403 OOS

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	None	N (ATC, BOP)	Swap to CCP 1-1 from CCP 1-3 per **OP B-1A:V, section 6.1**
2	CC01CCW_CCP11_MTFSHEAR 1	TS, C (BOP, SRO)	CCW Pp 11 Shaft Shear (**AR PK01-11; TS 3.7.7.A**).
3	MAL_PPL7J 1	TS, I (ALL)	Eagle 21 DFP-1 Halt in Rack 10 (**OP AP-5; TS 3.3.1.E,M; 3.3.2.D, L; 3.4.11**).
4	PK1421_0829 1	C (ALL)	Loss of Main Transformer Cooling (**AR PK14-21, AP-25**)
5	MAL_SEI1 0.31 delay=0 ramp=15 MAL_PPL5A; PPL5B BOTH	M (ALL)	Large seismic with no automatic or manual reactor trip (ATWS).
6	MAL_RCS3C 10.0 cd='jmlsei1' delay=10 ramp=60	M (ALL)	SBLOCA following seismic; ramps in over 60 seconds.
7	MAL_PPL1A FAILURE_TO_INIT MAL_PPL1B FAILURE_TO_INIT	C (BOP)	Phase A – Train A and B fail to actuate requiring manual alignment.
8	MAL_SYD2 0 cd='jpplsia' delay=15 PMP_SIS2_2 OVERLOAD_DEV_FAIL cd='h_v4_218r_1' delay=3 PMP_CVC2_1 AS_IS BKR_EPS15 AS_IS BKR_EPS9_1 OVERCURRENT cd='H_V4_225R_1' MAL_AFW1 TRIP cd='h_v3_109m_1 gt 3000'	C (ALL)	Combination of electrical and mechanical failures result in no high or intermediate injection along with degraded secondary side heat removal capabilities requiring manual starts of available charging pump safety injection pump.

***(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)	7
2. Malfunctions after EOP entry (1-2) (Events 7,8)	2
3. Abnormal events (1–4) (Events 2,3,4)	3
4. Major transients (1-2) (Event 5,6)	2
5. EOPs entered/requiring substantive actions (1–2) (E-1)	1
6. EOP contingencies requiring substantive actions (0–2)	0
7. Critical tasks (2–3)(See description below)	3

Critical Task	Justification	Reference
(S5CT-1) Trip the Reactor by manually de-energizing 480V Buses 13D and 13E within 90 seconds of AR PK04-11, Reactor Trip Initiate coming into alarm.	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 12, #34.
(S5CT-2) Manually close containment isolation valves such that at least one valve is closed on each Phase A containment penetration before transitioning out of EOP E-0.	Failure to perform the critical task leads to an unnecessary release of fission products to the auxiliary building, increasing the potential for release to the environment and reducing accessibility to vital equipment within the auxiliary building	• WCAP-17711-NP, CT-11
(S5CT-3) Start CCP 1-2 and SIP 1-2, so as to avoid a severe (Magenta) challenge to the Core Cooling critical status function.	Failure to manually start available ECCS pumps under postulated conditions constitutes misoperation or incorrect crew performance in which the crew does not prevent "degraded emergency core cooling system (ECCS) capacity.	• WCAP-17711-NP, CT-39 • HFRC1BG Rev 3
Per NUREG-1021, Appendix D, if an operator or 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.		

SCENARIO SUMMARY – NRC #5

1. Crew performs normal charging pump swap per **OP B-1A:V, CVCS – Transfer Charging Pumps, Section 6.1.**
2. **AR PK01-11, CCW Pp 1-1 Recirc** comes into alarm for FCV-606, CCW Pump 1-1 Recirc Valve, open. Crew identifies low pump amps on VB-1 and dispatches Nuclear Operator to investigate. Field reports no audible flow sound in spite of indications motor is running. CCW Pump 1-3 is started manually and CCW Pump 1-1 shutdown. **TS 3.7.7.A, Vital Component Cooling Water (CCW) System**, is entered for one loop of CCW inoperable (72 hrs).
3. Eagle 21 experiences a Digital Filter Processor (DFP) halt on rack 10. Associated indicators PI-456, LI-460A, FI-415, FI-425, FI-435, FI-445 (VB2), and PR-445, LR-459 (CC2) fail "as-is" as well as control channels for PORV 456 (PT-456) and Pressurizer Level Control (LT-460). Crew responds per OP AP-5, Malfunction of Eagle 21 Protection or Control Channel. Shift Foreman reviews Tech Specs, entering:
   - **TS 3.3.2.D, PC 456D Low Press SI** (72 hrs).
   - **TS 3.3.1.E, PC 456A High Press Trip** (72 hrs).
   - **TS 3.3.1.M, PC 456C Low Press Trip** (72 hrs).
   - **TS 3.3.1.M, LC 460A High Level Trip** (72 hrs).
   - **TS 3.3.1.M, FC-415(425,435,445) RCS Loop 1 (2,3,4) Flow** (72 hrs).
   - **TS 3.3.2.L, PC-456 B, P-11** (1 hr).
   - **TS 3.4.11.B1, B2, & B3 PC-456E, to close & remove power from associated block valve** (1 hr) **and restore to operable** (72 hrs).
4. Crew responds to **AR PK14-21, MAIN TRANSF**. A nuclear operator is dispatched to investigate local alarms and reports back that NO cooling fans or oil pumps are running on the Main Bank C Transformer. Shift Foreman enters **OP AP-25, Rapid Load Reduction or Shutdown** and directs a 50 MW/min power reduction while Maintenance and field Operators attempt to restore transformer cooling.
5. A large earthquake (0.31 g) occurs during the ramp, but the reactor fails to trip automatically. The crew performs the immediate actions of **EOP E-0, Reactor Trip or Safety Injection** and successfully trips the reactor by opening the breakers for 480 V buses 13D and 13E to de-energize the control rod drive mechanism (CRDM) allowing control rods to fully drop into the core **(S5CT-1) Trip the Reactor by manually de-energizing 480V Buses 13D and 13E.**
6. A SBLOCA occurs as a result of the earthquake, but both trains of Phase A fail to actuate. The crew performs manual alignment of Phase A containment isolation valves per **Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status** **(S5CT-2) Manually close containment isolation valves such that at least one valve is closed on each Phase A containment penetration.**
7. Startup power is lost shortly after Safety Injection initiates and a combination of electrical and mechanical failures result in the loss of both ECCS charging pumps and safety injection pumps, with CCP 1-2 and SIP 1-2 capable of being started manually. Secondary heat removal is affected as well. The turbine driven AFW pump trips on overspeed and AFW pump 1-3 has no power due to a loss of 4kV bus F. The crew performs the critical task of starting the available ECCS pumps **(S5CT-3) Start CCP 1-2 and SIP 1-2 so as to avoid a severe (Magenta) challenge to the Core Cooling critical status function.**

(continued)

SCENARIO SUMMARY – NRC #5

8. The crew proceeds through E-0, transitioning to **E-1, Loss of Reactor or Secondary Coolant,** where they check for subsequent failures and determine the optimal procedure flow path for long term recovery.

The scenario is terminated once the crew begins evaluation of Plant Status in E-1.