Facility: **Diablo Canyon** Date of Exam: **2021-01**

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	3	3	6
	2	1	2	2				2	1			1	9	2	2	4
	Tier Totals	4	5	5				5	4			4	27	5	5	10
2. Plant Systems	1	3	3	2	3	2	2	3	3	2	3	2	28	2	3	5
	2	1	0	1	1	1	1	1	1	1	1	1	10	1 1	1	3
	Tier Totals	4	3	3	4	3	3	4	4	3	4	3	38	4	4	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	2	2	1	2	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 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 / SF1			R				EK3.01 Knowledge of the reasons for the following responses as they apply to a reactor trip: Actions contained in EOP for reactor trip (41.5 /41.10)	4.0	11
000008 (APE 8) Pressurizer Vapor Space Accident / SF3		R					AK2.02 Knowledge of the interrelations between Pressurizer vapor space accident and the following: sensors and detectors (41. 7)	2.7*	12
000009 (EPE 9) Small Break LOCA / SF3				R			EA1.02 Ability to operate and monitor as they apply to a SBLOCA: RB sump level (41.7)	3.8	13
000011 (EPE 11) Large Break LOCA / SF3	R						EK1.01 Knowledge of the operational implications of the following concepts as they apply to Large break LOCA: Natural circulation and cooling, including reflux boiling (41.8 / 41.10)	4.1	14
000015 (APE 15) Reactor Coolant Pump Malfunctions / SF4					R		AA2.09 Ability to determine and interpret the following as they apply to RCP malfunctions: When to secure RCPs on high stator temperatures (41.10)	3.4	15
000022 (APE 22) Loss of Reactor Coolant Makeup / SF2			R				AK3.01 Knowledge of the interrelations between the Loss of reactor coolant makeup and the following adjustment of the RCP seal back pressure regulator valve to obtain normal flow (41.5 / 41.10)	2.7	16
000025 (APE 25) Loss of Residual Heat Removal System / SF4		R					AK2.02 Knowledge of the interrelations between the loss of RHRS and the following: LPI or Decay Heat Removal/RHR Pumps (41.7)	3.2	17
000026 (APE 26) Loss of Component Cooling Water / SF8						S	G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (43.2)	4.2	76
000027 (APE 27) Pressurizer Pressure Control System Malfunction / SF3				R			AA1.04 Ability to operate and monitor as they apply to a PZR Pressure Control System Malfunction: Pressure recovery using emergency only heaters (41.7)	3.9*	18
000029 (EPE 29) Anticipated Transient Without Scram / SF1						R	G2.1.8 Ability to coordinate personnel activities outside the control room (41.10)	3.4	19
000038 (EPE 38) Steam Generator Tube Rupture / SF3						S	G2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc (43.5)	4.6	77
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / SF4					S		W E12 EA2.2 Uncontrolled Depressurization of all Steam Generators: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments (43.5)	3.9	78
000054 (APE 54; CE E06) Loss of Main Feedwater /4			R				AK3.05 Knowledge of the reason for the following responses as they apply to loss of main feedwater: HPI/PORV cycling upon total feedwater loss (41.5)	4.6	20
000055 (EPE 55) Station Blackout /SF 6	R						EK1.02 Knowledge of the operational implications of the following concepts as they apply to SBO: Natural circulation cooling (41.8 / 41.10 / 45.3)	4.1	21
000056 (APE 56) Loss of Offsite Power / SF6						S	G2.2.37 Ability to determine operability and/or availability of safety-related equipment (43.2)	4.6	79

000057 (APE 57) Loss of Vital AC Instrument Bus / SF6					R		AA2.19 Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus and the following: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus (41.10 / 43.5 / 45.13)	4.0	22
000058 (APE 58) Loss of DC Power / SF6	R						AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation (41.8 / 41.10 / 45.3)	2.8	23
000058 (APE 58) Loss of DC Power / SF6						S	AA2.03 Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems (CFR: 43.5 / 45.13)	3.9	81
000062 (APE 62) Loss of Nuclear Service Water / SF4				R			AA1.01 Ability to operate and/or monitor the following as they apply to Loss of Nuclear Service Water (SWS): Nuclear service water temperature indications (41.7 / 43.5 / 45.6)	3.1	24
000065 (APE 65) Loss of Instrument Air / SF8					R		AA2.08 Ability to determine and interpret the following as they apply to the loss of instrument air: Failure modes of air operated equipment (43.5 / 45.13)	3.3	25
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / SF6					S		AA2.09 Ability to determine and interpret the following as they apply to Generator Voltage and Grid Disturbances: Operational status of the emergency diesel generators (43.5)	4.3	80
(W E04) LOCA Outside Containment / SF3		R					EK2.2 Knowledge of the interrelations (between the LOCA outside containment) and the following: 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 (41.7 / 45.7)	3.8	26
(W E11) Loss of Emergency Coolant Recirculation / SF4						R	G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrumentation interpretation (41.5)	4.4	27
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / SF4						R	G2.4.1 Knowledge of EOP entry conditions and immediate action steps (41.10)	4.6fc	28
K/A Category Totals:	3	3	3	3	3/**3**	3/**3**	Group Point Total:		18/**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							Not sampled		
000003 (APE 3) Dropped Control Rod / 1		R					~~AK2.03~~ **AK2.05 - Knowledge of the interrelations between the Dropped Control Rod and the following: control rod drive power supplies and logic circuits (41.7) **** corrected typo**	2.5	57
000005 (APE 5) Inoperable/Stuck Control Rod / 1							Not sampled		
000024 (APE 24) Emergency Boration / 1						R	G2.1.30 Ability to locate and operate components, including local controls (41.7 / 45.7)	4.4	58
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2	R						AK1.01 Knowledge of the operational implications of the following concepts as they apply to PZR Level control malfunction: PZR reference leak abnormalities (41.8 / 41.10)	2.8*	59
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7							Not sampled		
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7							Not sampled		
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8							Not sampled		
000037 (APE 37) Steam Generator Tube Leak / 3						S	G2.4.8 Knowledge of how Abnormal Operating Procedures are used in conjunction with EOPs. (43.5)	4.5	82
000051 (APE 51) Loss of Condenser Vacuum / 4					R		AA2.02 Ability to determine and interpret the following as they apply to loss of condenser vacuum: Conditions requiring reactor and/or turbine trip (41.7)	3.9	60
000060 (APE 60) Accidental Gaseous Radwaste Release / 9							Not sampled		
000061 (APE 61) Area Radiation Monitoring System Alarms / 7		R					AK2.01 Knowledge of the interrelations between the Area Radiation Monitoring System alarms and the following: detectors at each ARM location (41.7)	2.5*	61
000067 (APE 67) Plant Fire On Site / 8						S	G2.4.41 Knowledge of the emergency action level thresholds and classifications (41.10 / 43.5 / 45.11)	4.6	83
000068 (APE 68; BW A06) Control Room Evacuation / 8							Not sampled		
000069 (APE 69; W E14) Loss of Containment Integrity / 5					S		AA2.01 Ability to determine and interpret the following as they apply to the loss of containment integrity: Loss of containment Integrity (43.5)	4.3	84
000074 (EPE 74; W E06 & E07) Inadequate or Saturated Core Cooling / 4				R			EA1.3 Ability to operate and monitor the following as they apply to Inadequate Core Cooling: Desired operating results during abnormal and emergency operations (41.7)	3.9	62
000076 (APE 76) High Reactor Coolant Activity / 9			R				EK3.06 Knowledge of the reasons for the following responses as they apply to High Reactor Coolant Activity: actions contained in the EOP for high reactor coolant activity (41.5 / 41.10)	3.2	63

000078 (APE 78*) RCS Leak / 3							Not sampled until rev3 of KA cat.		
(W E01 & E02) Rediagnosis & SI Termination / 3			R				EK3.2 Knowledge of the reasons for the following responses as they apply to SI termination: Normal, abnormal and emergency operating procedures associated with (SI Termination) (41.5 / 41.10 / 45.6 / 45.13)	3.0	64
(W E13) Steam Generator Overpressure / 4				R			EA1.3 Ability to operate and/or monitor the following as they apply to Steam Generator Overpressure: Desired operating results during abnormal and emergency conditions (41.7 / 45.5 / 45.6)	3.1	65
(W E15) Containment Flooding / 5					S		EA2.1 Ability to determine and interpret the following as they apply to the Containment Flooding: Facility conditions and selection of appropriate procedures during abnormal and emergency operations (43.5)	3.2	85
(W E16) High Containment Radiation /9							Not sampled		
(BW A01) Plant Runback / 1							N/A for this design		
(BW A02 & A03) Loss of NNI-X/Y/7							N/A for this design		
(BW A04) Turbine Trip / 4							N/A for this design		
(BW A05) Emergency Diesel Actuation / 6							N/A for this design		
(BW A07) Flooding / 8							N/A for this design		
(BW E03) Inadequate Subcooling Margin / 4							N/A for this design		
(BW E08; W E03) LOCA Cooldown—Depressurization / 4							Not sampled		
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4							Not sampled		
(BW E13 & E14) EOP Rules and Enclosures							N/A for this design		
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4							Not sampled		
(CE A16) Excess RCS Leakage / 2							N/A for this design		
(CE E09) Functional Recovery							N/A for this design		
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4							N/A for this design		
K/A Category Point Totals:	1	2	2	2	½	½	Group Point Total:		9/**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 RCPS) Reactor Coolant Pump System					R							K5.01 Knowledge of the operational implications of the following concepts as they apply to RCPS: relationship between RCPS flow rate and the nuclear power core operating parameters (quadrant power tilt, imbalance, DNB rate, local power density, difference in loop T-hot pressure (41.5)	3.3	29
004 (SF1; SF2 CVCS) Chemical and Volume Control		R										K2.01 Knowledge of bus power supplies to the following: Boric acid makeup pumps (41.7)	2.9	30
005 (SF4P RHR) Residual Heat Removal					R							K5.09 Knowledge of the operational implications of the following concepts as they apply to the RHRS: dilution and boration considerations (41.5)	3.2	31
005 (SF4P RHR) Residual Heat Removal								S				A2.02 Ability to a) predict the impacts of the following malfunctions or operations on the RHRS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown (41.5 / 43.5 / 45.3 / 45.13)	3.7	86
006 (SF2; SF3 ECCS) Emergency Core Cooling								R				A2.11 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: Rupture of ECCS header (41.5)	4.0	32
006 (SF2; SF3 ECCS) Emergency Core Cooling											S	G2.2.40 Ability to apply Technical Specifications for a system (43.2)	4.7	87
007 (SF5 PRTS) Pressurizer Relief/Quench Tank									R			A3.01 Ability to monitor automatic operation of the PRTS, including: components which discharge to the PRT (41.7 / 45.5)	2.7*	33
008 (SF8 CCW) Component Cooling Water											R	G2.1.28 Knowledge of the purpose and function of major system components and controls (41.7)	4.1	34
008 (SF8 CCW) Component Cooling Water								S				A2.03 Ability to a) predict the impacts of the following malfunctions or operations on the CCW, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/Low CCW Temperature (43.5)	3.2	88
010 (SF3 PZR PCS) Pressurizer Pressure Control				R								~~K4.02 Knowledge of the PZR PCS design feature(s) and/or interlocks which provide for the following: Prevention of uncovering heaters (41.7)~~ K4.03 Overpressure Control	~~3.0~~ 3.8	35
012 (SF7 RPS) Reactor Protection	R											K1.05 Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following: ESFAS (41.2-41.9 / 45.7 -45.8)	3.8*	36
						R						K6.07 Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Trip logic circuits (41.7)	3.1	37

013 (SF2 ESFAS) Engineered Safety Features Actuation	R											K1.18 Knowledge of the physical connections and/or cause-effect relationships between the ESFAS and the following: Premature reset of ESF actuation	3.7	38
									R			A3.02 Ability to monitor automatic operation of the ESFAS, including the following: operation of actuated equipment (41.7)	4.1	39
022 (SF5 CCS) Containment Cooling				R								K4.03 Knowledge of the design feature(s) and/or interlocks which provide for the following: Automatic containment isolation	3.6	40
025 (SF5 ICE) Ice Condenser												Not applicable for this design		
026 (SF5 CSS) Containment Spray	R											K1.01 Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following: ECCS	4.2	41
										R		A4.05 Ability to manually operate and/or monitor in the control room: containment spray reset switches (41.7)	3.5	42
039 (SF4S MRSS) Main and Reheat Steam							R					A1.05 Ability to predict and/or monitor changes in parameters to prevent exceeding design limits associated with operating the MRSS including: RCS T-ave (41.5)	3.2	43
059 (SF4S MFW) Main Feedwater			R									K3.03 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: S/Gs (41.7 / 45.6)	3.5	44
061 (SF4S AFW) Auxiliary/Emergency Feedwater							R					A1.05 Ability to predict and/or monitor changes in parameters to prevent exceeding design limits associated with operating the AFW controls including: AFW flow/amps (41.5)	3.6	45
061 (SF4S AFW) Auxiliary/Emergency Feedwater											S	G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (43.2)	4.2	89
062 (SF6 ED AC) AC Electrical Distribution										R		A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard) (41.7)	3.3	46
062 (SF6 ED AC) AC Electrical Distribution											S	G2.2.22 Knowledge of limiting conditions for operations and safety limits (43.2)	4.7	90
063 (SF6 ED DC) DC Electrical Distribution		R										K2.01 Knowledge of bus power supplies to the following: Major DC loads	2.9	47
								R				**A2.01 ~~A2.04~~ Ability to a) predict the impacts of the following malfunctions or operations on the DC Electrical system, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds (41.7) ****corrected typo (A2.01 not A2.04)**	2.5	48
064 (SF6 EDG) Emergency Diesel Generator				R								K4.02 Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Trips for ED/G while operating (normal or emergency) (41.7)	3.9	49
						R						K6.08 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks (41.7)	3.2	50

073 (SF7 PRM) Process Radiation Monitoring							R					A1.01 Ability to predict and/or monitor changes in parameters to prevent exceeding design limits associated with operating the PRM system controls including: Radiation levels (41.5)	3.2	51
076 (SF4S SW) Service Water		R								R		K2.01 Knowledge of bus power supplies to the following: Service water pumps (41.7) A4 02 Ability to manually operate and/or monitor in the control room: SWS valves (41.7)	2.7* 2.6	52 53
078 (SF8 IAS) Instrument Air			R									K3.02 Knowledge of the effect that a loss or malfunction of the IAS system will have on the following: Systems having pneumatic valves and controls (41.7)	3.4	54
103 (SF5 CNT) Containment								R			R	A2 04 Ability to a) predict the impacts of the following malfunctions or operations on the Containment, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Containment evacuation (including recognition of the alarm) G2.4.46 Ability to verify that alarms are consistent with plant conditions (41.10)	3.5 4.2	55 56
053 (SF1; SF4P ICS*) Integrated Control												N/A for this plant design		
K/A Category Point Totals:	3	3	2	3	2	2	3	3/ **2**	2	3	2/**3**	Group Point Total:		28/**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 (SF1 CRDS) Control Rod Drive							R					A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRD system controls including: Reactor Power (41.5)	4.1	1
002 (SF2; SF4P RCS) Reactor Coolant										R		A4.07 Ability to manually operate and/or monitor in the control room: Flow path linking the RWST through the RHR system to the RCS hot legs for gravity refilling of the refueling cavity (41.7)	2.8	2
011 (SF2 PZR LCS) Pressurizer Level Control						R						K6.05 Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: Function of PZR level gauges as post-accident monitors (41.7)	3.1	3
014 (SF1 RPI) Rod Position Indication				R								K4.06 Knowledge of RPIS design feature(s) and interlock(s) which provide for the following: Individual and group misalignment (41.5)	3.4	4
015 (SF7 NI) Nuclear Instrumentation											R	G2.1.20 Ability to interpret and execute procedure steps (41.10)	4.6	5
016 (SF7 NNI) Nonnuclear Instrumentation	R											K1.03 Knowledge of the physical connections and/or cause-effect relationship between the NNIS and the following systems: SDS	3.2*	6
017 (SF7 ITM) In-Core Temperature Monitor								S				A2.02 Ability to a) predict the impacts of the following malfunctions or operations on the ITM, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Core Damage (43.5)	4.1	91
027 (SF5 CIRS) Containment Iodine Removal												Not sampled		
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control					R							K5.01 Knowledge of the operational implications of the following concepts as they apply to the HRPS: Explosive hydrogen concentration (41.7)	3.4	7
029 (SF8 CPS) Containment Purge												Not sampled		
033 (SF8 SFPCS) Spent Fuel Pool Cooling			R									K3.03 Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Spent fuel temperature (41.7)	3.0	8
034 (SF8 FHS) Fuel-Handling Equipment				S								K4.01 Knowledge of FHS design feature(s) and interlock(s) which provide for the following: Fuel protection from binding and dropping (41.7)	3.4	92
035 (SF 4P SG) Steam Generator									R			A3.02 Ability to monitor automatic operation of the S/G including: MAD valves	3.7	9
041 (SF4S SDS) Steam Dump/Turbine Bypass Control												Not sampled		
045 (SF 4S MTG) Main Turbine Generator												Not sampled		
055 (SF4S CARS) Condenser Air Removal												Not sampled		
056 (SF4S CDS) Condensate												Not sampled		
068 (SF9 LRS) Liquid Radwaste								R				A2.04 Ability to a) predict the impacts of the following malfunctions or operations on the LRS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation (41.5)	3.3	10

071 (SF9 WGS) Waste Gas Disposal								S				A2.09 Ability to a) predict the impacts of the following malfunctions or operations on the WGS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Stuck open relief valve (43.5)	3.5*	93
072 (SF7 ARM) Area Radiation Monitoring												Not sampled		
075 (SF8 CW) Circulating Water												Not sampled		
079 (SF8 SAS**) Station Air												Not sampled		
086 Fire Protection												Not sampled		
050 (SF 9 CRV*) Control Room Ventilation												Not allowed to be sampled until rev3 of KA cat		
K/A Category Point Totals:	1	0	1	1/ **1**	1	1	1	1/**1**	1	1	1/**1**	Group Point Total:		10/**3**

Facility: **Diablo Canyon** Date of Exam: **2021-01**

Category	K/A #	Topic	RO IR	RO #	SRO-only IR	SRO-only #
1. Conduct of Operations	2.1.3	Knowledge of Shift or short-term relief turnover practices (41.10)	3.7	66		
	2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as ***medical requirements***, etc.	3.3	67		
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management (41.1 / 43.6 / 45.6)	4.3	68		
	2.1.34	Knowledge of primary and secondary plant chemistry limits (43.5)			3.5	94
	2.1.42	Knowledge of new and spent fuel movement procedures (43.7)			3.4	95
	Subtotal			3		**2**
2. Equipment Control	2.2.3	Knowledge of the design, procedural, and operational differences between units (41.5 thru 10)	3.8	69		
	2.2.12	Knowledge of surveillance procedures (41.10)	3.7	70		
	2.2.43	Knowledge of the process used to track inoperable alarms (41.10)	3.0	71		
	2.2.13	Knowledge of tagging and clearance procedures (43.5)			4.3	96
	2.2.20	Knowledge of the process for managing troubleshooting activities (43.5)			3.8	97
	Subtotal			3		**2**
3. Radiation Control	2.3.**4**	Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)	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. (41.12)	3.2	73		
	2.3.6	Ability to approve release permits (43.4)			3.8	98
	Subtotal			2		**1**
4. Emergency Procedures/Plan	2.4.3	Ability to identify post-accident instrumentation (41.6 / 45.4)	3.7	74		
	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions (41.10)	3.7	75		
	2.4.13	Knowledge of crew roles and responsibilities during EOP usage (43.5)			4.6	99
	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (43.5)			4.4	100
	Subtotal			2		**2**
Tier 3 Point Total				10		**7**

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO - T2/G1	010 4.02K	KA tested subject very similar to what is tested in question 18. Selected KA 4.03 (IR 3.8)
RO - T3/G3	2.3.5	Unable to write question to license level. Randomly selected 2.3.4, (IR 3.2)
SRO T1/G1	W E04 EA2.1	This is a major event in a sim scenario. Due to the limited scope of the procedure (3 steps), unable to avoid conflict with Op test with any AA2 KA's for this topic. Selected KA APE 058 AA2.03 (IR 3.9)
RO T2/G1	064 K4.10	Unable to write system based question on selected KA. Randomly selected 064 K4.02 (IR 3.9)

Facility: Diablo Canyon	Date of Examination: 01/25/2021
Examination Level: RO ☒ SRO ☐	Operating Test Number: L191

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations (NRCL191-A1)	D, R	**Determine Affected Inputs Due To Malfunction Of Eagle 21 Protection Or Control Channel** 2.1.7 Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation. (4.4) (From NRCL162-A1)
Conduct of Operations (NRCL191-A2)	N, R	**Determine DFOST Inventory** 2.1.25 Ability to interpret reference material such as graphs, curves, tables, etc. (3.9) (New)
Equipment Control (NRCL191-A3)	M, R	**Determine Clearance Points and Tagging Requirements** 2.2.13 Knowledge of tagging and clearance procedures. (4.1) (modified from NRCL141-A3)
Radiation Control (NRCL191-A4)	N, R	**Perform Manual Calculation of Primary to Secondary Leakage** 2.3.11 Ability to control radiation release (3.8) (New)

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: 01/25/2021
Examination Level: RO ☐ SRO ☒	Operating Test Number: L191

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations (NRCL191-A5)	M, R	**Review AP-5 Bistable Trip Authorization Form** 2.1.7 Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation. (4.7) (Modified from NRCL162-A5)
Conduct of Operations (NRCL191-A6)	N, R	**Review STP M-10A, DFOST Inventory** 2.1.25 Ability to interpret reference material such as graphs, curves, tables, etc. (4.2) (New)
Equipment Control (NRCL191-A7)	N, R	**Review Roving Fire Watch Check List** 2.2.40 Ability to apply Technical Specifications for a system. (4.7) (New)
Radiation Control (NRCL191-A8)	N, R	**Perform Manual Calculation of Primary to Secondary Leakage** 2.3.11 Ability to control radiation release (4.3) (New)
Emergency Plan (NRCL191-A9)	D, R	**Perform an Emergency Classification** 2.4.41 Knowledge of the emergency action level thresholds and classifications. (4.6) (Bank: LJE-143)

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:	01/25/2021
Exam Level:	RO ☒ SRO-I ☐ SRO-U ☐	Operating Test Number:	L191

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. (S2) (013.A2.06) Respond to Spurious Safety Injection	A,N,EN,L,S	2
b. (S3) (006.A4.02) Isolate Accumulators Following a LOCA (Bank LJC-048)	A,D,L,S	3
c. (S4P) (011.EA1.11) Hot Leg Recirc (Bank LJC-028)	A,D,L,S	4P
d. (S4S) (045.A4.01) Perform Load Trim to Match Tave/Tref (NRCL162-S7)	D,S	4S
e. (S5) (022.A4.01) Place CFCU Drain Collection In Service (NRCLJC051-504)	A,M,S	5
f. (S6) (062.A4.07) Transfer Vital Buses from S/U to Aux (NRCL141-S6)	A,D,L,S	6
g. (S7) (073.A4.02) Restart RE-13 Pump Following High Pressure Alarm	N,S	7
h. (S8) (029.A1.02) Containment Vent Alignment Check of RM-44A (NRCL061CLJC-S7)	D,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) (040.AA1.03) Close MSIV and Bypass Locally (Bank LJP-212)	D,E,EN,L	4S
j. (P2) (004.A2.06) Isolate Dilution Flow Paths (Bank LJP-062)	D,E,L,R	1
k. (P3) (010.A2.01) Secure Pressurizer Heaters	E,N	3

* 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: 01/25/2021

Exam Level: RO ☐ SRO-I ☒ SRO-U ☐ Operating Test Number: L191

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. (S2) (013.A2.06) Respond to Spurious Safety Injection	A,N,EN,L,S	2
b. (S3) (006.A4.02) Isolate Accumulators Following a LOCA (Bank LJC-048)	A,D,L,S	3
c. (S4P) (011.EA1.11) Hot Leg Recirc (Bank LJC-028)	A,D,L,S	4P
d.		
e. (S5) (022.A4.01) Place CFCU Drain Collection In Service (NRCLJC051-504)	A,M,S	5
f. (S6) (062.A4.07) Transfer Vital Buses from S/U to Aux (NRCL141-S6)	A,D,L,S	6
g. (S7) (073.A4.02) Restart RE-13 Pump Following High Pressure Alarm	N,S	7
h. (S8) (029.A1.02) Containment Vent Alignment Check of RM-44A (NRCL061CLJC-S7)	D,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) (040.AA1.03) Close MSIV and Bypass Locally (Bank LJP-212)	D,E,EN,L	4S
j. (P2) (004.A2.06) Isolate Dilution Flow Paths (Bank LJP-062)	D,E,L,R	1
k. (P3) (010.A2.01) Secure Pressurizer Heaters	E,N	3

* 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:	01/25/2021
Exam Level:	RO ☐ SRO-I ☐ SRO-U ☒	Operating Test Number:	L191

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. (S2) (013.A2.06) Respond to Spurious Safety Injection	A,N,EN,L,S	2
b.		
c.		
d.		
e. (S5) (022.A4.01) Place CFCU Drain Collection In Service (NRCLJC051-504)	A,M,S	5
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) (040.AA1.03) Close MSIV and Bypass Locally (Bank LJP-212)	D,E,EN,L	4S
j. (P2) (004.A2.06) Isolate Dilution Flow Paths (Bank LJP-062)	D,E,L,R	1
k. (P3) (010.A2.01) Secure Pressurizer Heaters	E,N	3

* 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	

Scenario 2 was designated the spare and is not included in the Adams upload

Scenario 3 was dropped from the exam content before exam validation due to class size dropping prior to validation week and was not needed.

Facility: Diablo Canyon (PWR) **Scenario No:** 1 **Op-Test No:** L191 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: **2% Power BOL with MFWP 1-1 in service, aligned to Start-Up Power.**

Turnover: **In OP L-3, performing step 6.29, 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 6.29**.
2	VLV_CVC22_2 .5 ramp=15	**I (BOP, SRO)**	TE-130 fails high; valve fails open (**PK04-21 or AP-5**)
3	BST_MFW1_1 1 PMP_AFW2_2 OVERLOAD_DEV_FAIL	**TS, C (ALL)**	MFW Pump 1-1 trip w/no AFW autostart; MDAFW Pump 1-3 OC trip (**PK09-12, AP-15; TS 3.7.5.B**)
4	MAL_RCS3B .3	**TS, C (ALL)**	300 gpm RCS Leak (**AP-1**; **TS 3.4.13.A**) *Note: If crew trips reactor before this event, TS for event 4 becomes 3.4.13.B*
5	MAL_RCS1C 100%_DBA	**M (ALL)**	100% DBA LBLOCA
6	PMP_RHR1_1 AS_IS PMP_RHR1_1 AS_IS	**C(BOP)**	Low Head Injection Failure
7	RLY_PPL73_1, RLY_PPL74_1, RLY_PPL75_1, RLY_PPL76_1 TRUE	**C (BOP)**	Containment Spray Actuation Failure
8	MAL_CNM3 100	**C (ALL)**	Sump Blockage

***(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 6,7,8)	3
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-1, E-1.3, ECA-1.3)	3
6. EOP contingencies requiring substantive actions (0–2) (ECA-1.3)	1
7. Critical tasks (2–3)(See description below)	3

<table>
<tr><th>Critical Task</th><th>Justification</th><th>Reference</th></tr>
<tr><td>(S1CT-1) Manually start at least one RHR pump by the completion of E-0, Appendix E.</td><td>Acceptable results based on FSAR analysis of a large break LOCA are predicated on the assumption of minimum ECCS pumped injection. The minimum rate varies with RCS pressure. In the case of a LBLOCA, low head injection flow is assumed as part of the analysis. Failure to start the available low head injection pumps constitutes a violation of the facility license condition.</td><td>Westinghouse Owner's Group WCAP-17711-NP
Backgrnd HE0BG_R3</td></tr>
<tr><td>(S1CT-2) Manually align at least 1 train of Containment Spray (1 pump and associated discharge valve) by the completion of E-0, Appendix E.
<table><tr><th>Train A</th><th>Train B</th></tr><tr><td>• CSP 1-1</td><td>• CSP 1-2</td></tr><tr><td>• 9001A</td><td>• 9001B</td></tr></table></td><td>Failure to initiate the minimum required Containment Spray equipment as a means of pressure suppression represents a severe challenge to Containment Safety Function.</td><td>EOP FR-Z.1 Background Document
Westinghouse Owner's Group WCAP-17711-NP</td></tr>
<tr><td>(S1CT-3) Stop all running ECCS pumps with suction aligned to the containment recirc sump per continuous action requirements of ECA-1.3, step 1.
<table><tr><td>• CCP 1-1</td><td>• CCP 1-2</td></tr><tr><td>• SIP 1-1</td><td>• SIP 1-2</td></tr><tr><td>• RHR 1-1</td><td>• RHR 1-2</td></tr></table>Note: Core damage is expected to occur within approximately 30 minutes of loss of pump suction (CETs above 700°F). Operations Management would consider failure to shutdown the cavitating ECCS pumps within 10 minutes a critical gap as defined by TQ2.DC3, Licensed Operator Continuing Training Program.</td><td>High When any pump loses its suction source, operators must stop the pump to prevent potential pump damage. High-head pumps, such as high-head SI pumps and SI/charging pumps may experience permanent damage within a short time of loss of suction. Operators should immediately stop these pumps if they observe symptoms of loss of suction. Lower-head pumps, such as low-head SI pumps and spray pumps can operate with cavitation for longer times, but not indefinitely without permanent damage.</td><td>Background Information for Westinghouse Owners Group Sump Blockage Guideline, Rev 0
DCPP Sump Blockage Guidelines & Background Document</td></tr>
<tr><td colspan="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.</td></tr>
</table>

SCENARIO SUMMARY – NRC #1

1. Control rods are used to raise power from 2% to ≈ 8% **OP L-3, Secondary Plant Startup,** step 6.29**.** ATC operator complies with 1 step pull and wait procedural requirement while monitoring relevant controls and diverse indicators. Shift Foreman provides reactivity oversight.

2. Letdown heat exchanger temperature element TE-130 fails high, causing TCV-130 to open fully and actual letdown temperature to lower. **PK04-21, LETDOWN PRESS / FLO TEMP** comes into alarm, directing the crew to take manual control of letdown temperature (TCV-130) and restore and maintain temperature within normal range for the duration of the scenario. The Shift Foreman may elect to follow the guidance of **OP AP-5, Malfunction of Eagle 21 Protection or Control Channel** as an alternate method for addressing the failure.

3. MFP 1-1 trips on high vibration and both MDAFW pumps fail to auto start. The crew enters **AR PK09-12, Main Feedwater Pump Trip**, and **OP AP-15, Loss of Feedwater Flow, Section B: Single Operating MFP Trips.** TDAFW Pump 1-1 and both motor driven pumps are manually started, however MDAFW Pump 1-3 trips on overcurrent. The crew follows the guidance to trip the turbine, and insert rods to reduce power to back down to 2%. Shift Foreman enters **Tech Spec 3.7.5.B, AFW System,** for one AFW train inoperable.

4. A 300 gpm RCS leak develops on the loop 3 cold leg, requiring entry in **OP AP-1, Excessive Reactor Coolant System Leakage**. The leak ramps in slowly, allowing the crew time to manually adjust flow, start second charging pump, and eventually isolate letdown in an attempt to maintain pressurizer level. VCT level cannot be maintained at the current leak rate, and the Shift Foreman directs initiation of a reactor trip and safety injection. **TS 3.4.13.A, RCS Operational Leakage,** applies.

   Note: If event executed post-trip, TS becomes 3.4.13 Condition B for Pressure boundary LEAKAGE existing

5. The crew enters **E-0, Reactor Trip or Safety Injection** and begins performing their immediate actions, when a large break LOCA (100% DBA) occurs. The crew 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, and with Shift Foreman concurrence, all four RCPs are tripped. Shift Foreman directs the BOP Operator to complete **Appendix E, ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS**, and continues on in E-0.

6. During the performance of E-0, Appendix E, the BOP Operator identifies both RHR pumps have failed to start and Containment Spray has failed to actuate as required. Action is taken to manually start RHR pumps, containment spray pumps and align spray valves. **(S1CT-1), Manually start at least one RHR pump by the completion of E-0, Appendix E, and (S1CT-2), Manually align at least 1 train of Containment Spray (1 pump and associated discharge valve) by the completion of E-0, Appendix E).**

(continued)

SCENARIO SUMMARY – NRC #1

7. The Shift Foreman 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. When RWST level reaches 33%, the crew transitions immediately to **E-1.3, Transfer to Cold Leg Recirculation**, and performs the required alignment steps. ECCS recirculation flow is lost due to sump blockage. The crew transitions to **ECA-1.3, Sump Blockage** either directly, or by way of **ECA-1.1, Loss of Emergency Coolant Recirculation**, where they secure all running ECCS pumps **(S1CT-3) Stop all running ECCS pumps with suction aligned to the containment recirc sump per continuous action requirements of ECA-1.3, step 1.**

The scenario is terminated once Critical Task S1CT-3 is complete

Facility: Diablo Canyon (PWR) **Scenario No:** 4 **Op-Test No:** L191 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: 75% Power, EOL with CCP 1-2 OOS

Turnover: Unit is operating at 75% due to a SCCW Hx Clearance

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	XMT_CVC4_3 0 delay=0 ramp=60	TS, I (ATC, SRO)	FT-128 Fails low causing high charging flow (**OP AP-5; OP AP-17**, T.S. 3.3.4.A)
2	VLV_CVC16_2 0.15 ramp=5	TS, C (ALL)	Letdown Hx Inlet Valve, CVCS- 8152 Fails 90% Closed (**PK04-21, AP-18;** TS 3.6.3.A)
3	MAL_GEN4_3 TRIP LOA_SYD6, LOA_SYD7, LOA_SYD8, CLOSED delay=5 LOA_SYD16 Energized delay=5	C (ALL)	Full Load Rejection (**AP-2**)
4	CNV_MFW6_2 0.6 delay=0 ramp=300 dellA CNV_MFW6_2 2 delay=0 cd= FTDFCV540_MAN CNV_MFW6_2 1 delay=0 ramp=30	C (BOP, SRO)	Feedreg Valve FCV-540 control problem, eventually failing full open; automatic turbine trip disabled (**PK09-15**)
5	MAL_MFW5D 2e+007 cd='fnispr lt 5' delay=2 ramp=10	M (ALL)	(Major) Feedline break inside containment
6	MAL_SYD2 0 cd='H_V4_169B_1' delay=5	C (ALL)	Loss of 230 kV on transfer to Startup
7	MAL_DEG1A_1, MAL_DEG1B_1, and MAL_DEG1C_1 STOP	C (ALL)	Shutdown relay on all 3 D/Gs; C/R start available
8	VLV_MFW4_2 1 dellA VLV_MFW4_2 2 cd='V3_193S_1'	C (BOP)	FCV-441 fails open; Isolate feedflow as part of Critical Task

***(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 1,2,3,4,5,6,7,8)	8
2. Malfunctions after EOP entry (1-2) (Events 6,7,8)	3
3. Abnormal events (1–4) (Events 1,2,3,4)	4
4. Major transients (1-2) (Events 5)	1
5. EOPs entered/requiring substantive actions (1–2) (E-2, ECA-0.0)	2
6. EOP contingencies requiring substantive actions (ECA-0.0)	1
7. Critical tasks (2–3)(See description below)	3

Critical Task	Justification	Reference
(S4CT-1) Manually trip the turbine OR reactor before S/G 1-4 reaches 92% narrow range.	Steam Generator Level above the High High setpoint (P-14) normally generates a turbine trip signal to protect against high feedwater flow and carryover into the steam lines when one out of four S/Gs has reached a narrow range level greater than 92%. Carryover into the steam lines can result in damage to downstream piping and valves, placing the secondary heat sink at risk.	• Generic Letter 81-28 • WOG Background HFRH3BG_32
(S4CT-2) Energize at least one vital AC bus from Control Room 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-3) Manually isolate feedline break by closing the following S/G 1-4 components: • Feedwater Iso Valve FCV-441 • MDAFW Level Control Valve LCV-113 • TDAFW Level Control Valve LCV-109 before a severe challenge develops to the containment safety function based on containment wide range level greater than 94 feet. *Note: Bus restoration timing and methodology may result in field isolation of valves and is acceptable.*	Failure to isolate feedflow into containment leads to an unnecessary and avoidable severe challenge to the containment integrity safety function as a result of flooding.	• WOG Background HFRZ2BG_R2
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. FT-128 (charging flow) fails low, causing actual charging flow to rise. The crew responds per **OP AP-5, Malfunction of Eagle 21 Protection or Control Channel**. FCV-128 and HC-459D are taken to manual, and charging flow is monitored using alternate indications (RCP seals, Pzr level, VCT level, etc) for the remainder of the scenario. **OP AP-17, Loss of Charging**, Section B (Charging System Equipment Malfunctions), may also be used to respond to the failure. **TS 3.3.4.A, Remote Shutdown Systems,** is implemented.

2. Letdown Hx Inlet Valve, CVCS-8152 fails 90% closed causing letdown to divert to the Pressurizer Relief Tank. **AR PK04-21, LETDOWN PRESS / FLOW TEMP** comes into alarm, directing the crew to isolate Normal Letdown and place Excess Letdown in service per **OP B-1A:IV, CVCS – Excess Letdown – Place In Service and Remove From Service**. Alternately, the crew may elect to enter **OP AP-18, Letdown Line Failure**, which provides equivalent guidance. Shift Foreman enters **TS 3.6.3.A – Containment Isolation Valves,** for one containment isolation valve inoperable.

3. Full load rejection on Unit 1. The crew recognizes the condition based on numerous power level alarms and the ensuing secondary side transient and monitors the plant for appropriate automatic system responses. Shift Foreman implements **OP AP-2, Full Load Rejection** to stabilize the plant at approximately 30% power. The earlier failure of FT-128 will require board operators to manually control charging using diverse indications. Secondary realignments are performed as time permits.

4. Loop 4 Feedwater Reg Valve, FCV-540, begins drifting open but will initially respond to manual control when the Control Operator matches valve position to demand. A subsequent failure causes manual valve control to fail as well, with the valve drifting to full open. Shift Foreman directs a manual reactor trip **(S4CT-1) Manually trip the turbine OR reactor before S/G 1-4 reaches 92% narrow range.**

5. On the trip, the feedline header to S/G 1-4 ruptures, causing S/G 1-4 to depressurize into containment.

6. Startup power is lost on the bus transfer and all three D/Gs fail to start as a result of a shutdown relay common mode failure. The crew transitions to **EOP ECA-0.0, Loss of All Vital AC Power,** taking action to close RCS sample lines while attempting to restore vital 4kV power.

7. The crew is able to restore power to all three vital buses following ECA-0.0 guidance to reset diesel generator shutdown relay for each of the D/Gs **(S4CT-2) Energize at least one vital AC bus from Control Room prior to implementation of FLEX strategies.**

8. Crew enters **EOP E-0, Reactor Trip or Safety Injection** and performs their immediate actions. E-0 diagnostic steps direct the crew to **E-2, Faulted Steam Generator Isolation** to perform the final critical task of isolating the feedline break. **(S4CT-3) Manually isolate feedline break before a severe challenge develops to the containment safety function as the result of containment wide range level exceeding 94 feet.**

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

Facility: Diablo Canyon (PWR) **Scenario No:** 5 **Op-Test No:** L191 NRC

Examiners: ______________________ **Operators:** ______________________

______________________ ______________________

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Initial Conditions: **50% Power, MOL with CCP 1-2 OOS**

Turnover: **Place 120 gpm letdown in service per OP B-1A:XII, Section 6.3**

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	CNV_CVC5_2 0.65 cd='NOT H_V2_212G_1' delIA CNV_CVC5_2 2 cd='V2_212S_1 AND H_V2_212R_1'	**C (ALL)**	Letdown high pressure condition during 120 gpm alignment evolution.
2	DSC_ROD1 BREAKER_OPEN	**TS, C (ATC, SRO)**	DRPI power supply failure. (**PK03-21, TS 3.1.7.A and 3.1.7.B**).
3	XMT_CCW25_3 374 ramp=300 H_V1_086M_1 68 ramp=300	**TS, C (BOP, SRO**	CCW Pp 1-2 High Stator Temp (**PK01-09, AP-11; TS 3.7.7.A**)
4	RC08RC_8948ATVLEAK 0.5 delay=2 RC08RC_8818ATVLEAK 0.06 delay=2 ramp=480 RC09RC_8742ATVLEAK 0.1 delay=2 MAL_RHR2A 0.05 delay=2 ramp=60	**C (ALL, SRO)**	Intersystem LOCA **(PK02-17, PK05-21, PK02-16, AP-1)**
5	MAL_RCP4A 1	**M (ALL)**	RCP 1-1 Locked Rotor causing Rx Trip initiate signal
6	MAL_PPL5A, MAL_PPL5B BOTH V5_239S_1 0 V5_245S_1 0	**C (ALL)**	ATWS – Attempts to trip from C/R unsuccessful

***(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)	6
2. Malfunctions after EOP entry (1-2) (Events 6)	1
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) (FR-S.1, ECA-1.2)	2
6. EOP contingencies requiring substantive actions (FR-S.1, ECA-1.2)	2
7. Critical tasks (2–3)(See description below)	2

Critical Task	Justification	Reference
(S5CT-1) Insert negative reactivity into the core by driving rods per EOP FR-S.1 such that power is reduced to less than 5% by the completion of step 19.	Failure to insert negative reactivity as procedurally directed constitutes a failure to provide appropriate reactivity control and represents an unnecessary and avoidable challenge to the criticality safety function.	• WCAP-17711-NP, CT-52 • FR-S.1 Background Document, Rev. 3.
(S5CT-2) Manually isolate LOCA outside containment by closing 8809A such that a positive RCS pressure trend is established prior to transition out of EOP ECA-1.2, LOCA Outside Containment.	The intersystem LOCA results in a failed fission product barrier, permitting RCS fluid into the Aux Building through an unintended path. Failure to isolate the leak allows the RCS to remain in a degraded state when it is within the capability of the crew to correct the problem and restore the barrier.	• WCAP-17711-NP, CT-32 • WOG Background HECA12BG_R3
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. The crew places 120 gpm letdown in service for RCS cleanup per OP B-1A:XII, Section 6.3. Letdown pressure rises above 400 psig when the crew attempts alignment of the 45 gpm orifice valve. The crew determines 120 gpm letdown cannot be established and follows procedural guidance to restore letdown alignment and charging back to the original 87 gpm charging alignment.

2. The normal power supply to DRPI trips opens and crew responds per **AR PK03-21, DRPI FAILURE/ROD BOTTOM**. Shift Foreman directs the ATC to place rods in manual and field operators are contacted to place DRPI on backup power. Shift Foreman enters **TS 3.1.7.A** for on DRPI per group inoperable in one or more groups (indirectly verify rod position by using core power distribution measurement information) and **TS 3.1.7.B** for more than one DRPI per group inoperable (immediate TS action is to place rods in manual).

3. Component Cooling Water pump CCW 1-2 alarms on high stator temperatures. The Shift Foreman enters **AR PK01-09, CCW Pumps** and dispatches field operators who report strong acrid smell in the area and motor casing hot to the touch**.** CCW pump 1-3 is placed in service and 1-2 pump is shutdown. **T.S. 3.7.7.A, Vital Component Cooling Water (CCW) System**, is entered for one loop of CCW inoperable.

4. **PK02-17, RHR PUMPS** alarms, followed shortly by **PK02-16, RHR SYSTEM** and **PK05-21, PZR LEVEL HI/LO**. The crew recognizes the conditions as indicative of an intersystem LOCA between the RCS and RHR. **OP AP-1, Excessive Reactor Coolant System Leakage** is entered in an attempt to maintain pressurizer level. Flow is maximized using two charging pumps. Letdown isolation triggers the next event.

5. Locked rotor on RCP 1-1 results in a reactor trip initiate signal due to low RCS loop flow, but the reactor fails to trip. **PK04-11, REACTOR TRIP INITIATE** illuminates, with no corresponding **PK04-14, REACTOR TRIP ACTUATED**, alerting the crew to an ATWS condition. Additional PKs come into alarm and are used by the crew to identify that nature of the failure. RCP 1-1 trips on overcurrent after approximately a minute if the crew has not already tripped the pump.

6. The crew identifies the ATWS condition and the Shift Foreman enters **EOP E-0, Reactor Trip or Safety Injection,** and transitioning immediately to **EOP FR-S.1, Response to Nuclear Power Generation / ATWS.** Attempts to trip the reactor from the Control Room are unsuccessful. The crew performs the critical task of adding negative reactivity by manually driving rods **(S5CT-1) Insert negative reactivity into the core by driving rods per EOP FR-S.1 such that power is reduced to less than 5% by the completion of step 19.** The crew continues working through FR-S.1 until field operators open the reactor trip breakers.

7. RCS pressure and pressurizer level continue to lower resulting in a need to safety inject the plant due to the ongoing intersystem LOCA. Safety injection may be directed by the Shift Foreman or will eventually occur automatically based on low pressure.

8. The crew re-enters E-0, performing the diagnostic steps, noting RCS pressure continuing to slowly lower with no indications of a leak inside containment. High head injection is lost shortly after instrument air is restored to containment. The crew is directed to **EOP ECA-1.2, Intersystem LOCA.** to perform the critical task of closing 8809A **(S5CT-2) Manually isolate LOCA outside containment by closing 8809A such that a positive RCS pressure trend is established prior to transition out of EOP ECA E-1.2, LOCA Outside Containment.**

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

Facility: Diablo Canyon (PWR) **Scenario No:** 6 **Op-Test No:** L191 NRC

Examiners: ______________ **Operators:** ______________

Initial Conditions: 100% Power, MOL with CCP 1-2 OOS

Turnover: Maintain 100% Power

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	PMP_ASW2_1 AS_IS dellA PMP_ASW2_1 2 cd='V1_243S_3' PMP_ASW1_2 OVERLOAD_DEV_FAIL	**TS, C (BOP, SRO)**	ASW 1-1 OC Trip, standby req man start (**AP-10; TS 3.7.8.A**)
2	MAL_NIS6B 200 delay=0 ramp=420	**TS, C (ALL)**	NI-42 slow failure HIGH (**AP-5; TS 3.3.1.D,E,S,T, ECG 37.2.A, 37.3.A**)
3	MAL_SEI1 0.12 ramp=10 PLP_AUX6 19 cd='jmlsei1' delay=15 ramp=1200	**C (ALL)**	Seismic event results in RWST crack and inadequate sump level. TS calls for ramp (**PK06-20, AP-25; TS 3.5.4.B**)
4	MAL_RCS3C 6.0 cd='v2_233r_1' delay=10 ramp=3	**M (ALL)**	SBLOCA during ramp offline
5	MAL_PPL1A FAILURE_TO_INIT MAL_PPL1B FAILURE_TO_INIT	**C (BOP)**	Failure of Phase A Isolation valves, inside and outside containment on Safety Injection

***(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)	5
2. Malfunctions after EOP entry (1-2) (Event 5)	1
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-1, E-1.3, ECA-1.1)	3
6. EOP contingencies requiring substantive actions (ECA-1.1)	1
7. Critical tasks (2–3)(See description below)	2

<table>
<tr><th>Critical Task</th><th>Justification</th><th>Reference</th></tr>
<tr><td>(S6CT-1), Close containment isolation Phase A valves:
<table>
<tr><td>** FCV-254, 256, 257, 260, 501</td><td>Rad Waste Isolation (I.C.)</td></tr>
<tr><td>** FCV-253, 255, 258, 500</td><td>Rad Waste Isolation (O.C.)</td></tr>
<tr><td>** 9356 A/B</td><td>RCS Sample</td></tr>
<tr><td>** 9355 A/B</td><td>Pzr Liquid Sample</td></tr>
<tr><td>** 8112, 8100</td><td>Seal Water Return</td></tr>
<tr><td>8045</td><td>PRT N2 Supply</td></tr>
<tr><td>8029</td><td>PRT Primary Water Supply</td></tr>
<tr><td>8880</td><td>Accum N2 Supply</td></tr>
<tr><td>FCV-584</td><td>Cont Instrument Air</td></tr>
<tr><td>FCV 633</td><td>Cont Fire Water Isolation</td></tr>
</table>
by the completion of E-0, Appendix E.

** (Note: closing either the inside or outside isolation valve meets the critical task requirement)</td>
<td>Failure to initiate Safety Injection under the postulated conditions of this scenario places the plant outside the assumptions of the FSAR accident analysis and in violation of the facility licensing condition.</td>
<td>• Westinghouse Owner's Group WCAP-17711-NP</td></tr>
<tr><td>(S6CT-2) Stop the following ECCS pumps
<table><tr><td>CCP 1-1</td><td>SIP 1-1</td><td>SIP 1-2</td></tr></table>
before insufficient RWST level results in ECCS pump cavitation as indicated by rapid swings in pump amperage.</td>
<td>Damage to the RWST in this scenario results in a continuous loss of level and eventual inability to meet the minimum NPSH requirements for the running ECCS pumps. Failure to stop the pumps before cavitation occurs can lead to pump damage sufficient to render the pumps unavailable for use once an alternate make-up supply is aligned to the RCS.</td>
<td>• WCAP-17711-NP, CT-28
• WOG Background HECA11BG_R2</td></tr>
<tr><td colspan="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.</td></tr>
</table>

SCENARIO SUMMARY – NRC #6

1. ASW Pp 1-1 trips on overcurrent in annunciator **PK01-03, Aux Salt Water Pumps.** ASW Pp 1-2 fails to auto start. Shift Foreman enters **OP AP-10, Loss of Auxiliary Salt Water** and directs a swap to the standby ASW pump. **TS 3.7.8.A, Auxiliary Saltwater System,** is entered for one ASW train inoperable (72 hour shutdown tech spec).

2. Power Range Nuclear Instrument NI-42 slowly fails high causing inward rod motion. Crew diagnoses failure, and once motion is deemed unwarranted, takes rods to manual. Failure is addressed per **OP AP-5, Malfunction of Eagle 21 Protection or Control Channel**, which removes the failed channel from service and directs the Shift Foreman to address **Tech Specs 3.3.1.D,E,S,T Reactor Trip System Instrumentation; ECG 37.2.A Axial Flux Difference (AFD) monitoring, and ECG 37.3.A Quadrant Power Tilt Ratio Alarms (QPR).**

3. A 0.12 g seismic event results in a rupture of RWST, causing level to lower rapidly. The crew identifies RWST level lowering by monitoring level indications on VB-2 or by evaluating **AR PK06-20, PPC Select** which identifies RWST level is below the alarm setpoint. Field operators report a crack in the RWST extending down to ground level. The Shift Foreman enters **TS 3.5.4.B – Refueling Water Storage Tank (RWST)** for borated water volume less than the required minimum of 455,300 gallons (~94%). Shift Manager will direct the Shift Foreman to ramp at 6 MW/min due to overall degraded conditions of the Unit. Shift Foreman enters **OP AP-25, Rapid Load Reduction or Shutdown** to ramp the unit offline.

4. A SBLOCA occurs on cold leg 3. 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**, where they perform their immediate actions.

5. Both trains of Phase A fail to actuate resulting in all Phase A Containment Isolation valves (inside and outside containment on a single process line) to remain open. Shift Foreman directs the BOP Operator to complete **Appendix E, ESF AUTO ACTIONS, SECONDARY AND AUXILIARIES STATUS**, which addresses the critical task of manually closing the failed open containment isolation valves **(S6CT-1), Close containment isolation Phase A valves by the completion of E-0, Appendix E.**

6. The Shift Foreman will continue on, working through E-0 diagnostic steps. The crew briefly enters **E-1, Loss of Reactor or Secondary Coolant** prior to transitioning to **E-1.3, Transfer to Cold Leg Recirculation** when RWST level reaches 33%, which happens quickly due to the leaking RWST. With the Containment Recirc Sump Level less than 92%, the crew is forced into **EOP ECA-1.1, Loss of Emergency Coolant Recirculation.** The fissure location causes the RWST to continue to drain, requiring the crew to perform the second critical task **(S6CT-2), Stop ECCS pumps aligned to the RWST before insufficient level results in ECCS pump cavitation**.

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

Facility: Diablo Canyon (PWR) **Scenario No:** 7 **Op-Test No:** L191 NRC

Examiners: ______ **Operators:** ______

Initial Conditions: **75% Power, MOL with CCP 1-2 OOS**

Turnover: **At 75% power from SCCW HX Clearance. Directed to return to 100% power following guidance of ramp plan provided with crew turnover.**

Event No	Malf No.	Event Type*	Event Description (See Summary for Narrative Detail)
1	N/A – Normal Event	**N (ATC)**	Prepare for return to 100% power following ramp plan and **B-1A:VII, Attachment 1**.
2	CNV_CVC1_1 1 delay=20 cd='NOT V2_234G_1' VLV_CVC14_1 1 delay=20 cd='NOT V2_234G_1' H_V2_273R_1 1 cd='NOT V2_234G_1' delIA CNV_CVC1_1 2 cd='V2_273S_1' delIA H_V2_273R_1 2 delay=1 cd='v2_273s_1' delIA VLV_CVC14_1 2 cd='V2_234S_1'	**C (ALL)**	Makeup Controller failure causing continuous dilution (**AP-19**)
3	EECKSELECT2382371XPWR 0	**C (SRO, BOP)**	Load Tap Changer Auto Control Failure (**PK20-04, TS 3.8.1.A**)
4	VLV_PZR6_2 0.25 ramp=5	**C (BOP, SRO)**	PCV-455C slowly drifts open following small seismic (**PK05-20, AP-13; TS 3.4.11.B**)
5	MAL_CWS2C .2 ramp=20	**C (ALL)**	Condenser In-leakage (**PK12-05, AP-20 & 25**)
6	MAL_MSS4 3.6E+07	**M (ALL)**	MSLB downstream of MSIVs
7	VLV_MSS7_1, VLV_MSS8_1, VLV_MSS9_1, VLV_MSS10_1	**C (ALL)**	All MSIVs fail open

***(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 7)	1
3. Abnormal events (1–4) (Events 2,3,4,5)	4
4. Major transients (1-2) (Events 6)	1
5. EOPs entered/requiring substantive actions (1–2) (E-2, ECA-2.1, FR-P.1)	3
6. EOP contingencies requiring substantive actions (ECA-2.1, FR-P.1)	2
7. Critical tasks (2–3)(See description below)	3

Critical Task	Justification	Reference
(S7CT-1) Close the motor operated block valve 8000B upstream of the stuck open PORV PCV-455C prior to reaching reactor trip setpoint of RCS pressure less than 1950 psig.	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
(S7CT-2): Minimize the uncontrolled cooldown by throttling AFW flow to approximately 25 gpm per lead prior to SI termination by performing the following actions: • Isolate TDAFW pump: ○ Close turbine steam supply valves FCV-37 and 38 OR ○ Close S/G feed flow valves LCV-106, 107, 108, and 109 from TDAFW pump (may be done on step 6 of E-0) • Throttle S/G feed flow from MDAFW pumps to approx 25 gpm per lead (LCV-110, 111, 115, 113).	Events which lead 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. Feedflow from the TDAFW pump may be stopped by isolating the motive force (steam supply) or feed path (LCV flow valves).	• Background Information for WOG Emergency Response Guideline
(S7CT-3): Terminate ECCS flow before PRT rupture by closing 8801A/B and/or 8803A/B.	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
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 #7

1. Crew will prepare to ramp back to 100% power following completion of repairs to SCCW HX. Ramp plan provided as part of Turnover prior to taking the watch. Heaters and sprays are already in service as part of the initial setup. The crew will begin on with performance of a 100 gallon dilution per **OP B-1A:VII, Makeup Control System Operation, Attachment 1 – Dilution Checklist.**

2. Makeup controller fails to terminate dilution when the batched target is reached. Crew takes manual action to stop the dilution following the guidance of **OP AP-19, Malfunction of Reactor Makeup Control System.**

3. Startup Transformer 1-1 Load Tap Changer control power supply fails. Crew responds per **AR PK20-04, SU TRANSF 11, 12, OR 21 LOCAL ANNUN,** and manually controls transformer voltage. Shift Foreman enters **TS 3.8.1.A, AC Sources – Operating** for one required offsite circuit inoperable (perform SR 3.8.1.1 for required operable circuit in one hour and once per 8 hours thereafter; restore required offsite circuit to operable status within 72 hrs.

4. Pressurizer Pressure Control Valve PCV-455C drifts open bringing **PK05-20 Pressurizer Relief/Safety Valves Open** and **PK05-23, Pressurizer Safety or Relief Line Temperature** into alarm. Valve must be isolated using the associated 8000B block valve **(S7CT-1) Close the motor operated block valve upstream of the stuck open PORV**. Crew may also consult guidance of **OP AP-13, Malfunction of Reactor Pressure Control System.** Shift Forman enters **TS 3.4.11.B Pressurizer Power Operated Relief Valves (PORVs)** – for one PORV inoperable for reasons other than excessive seat leakage.

5. A saltwater leak develops in the SW quadrant of the condenser, bringing in **PK12-05, COND PPS DISCH HDR CATION CONDT'Y HI.** The crew determines cation conductivity is elevated and the Shift Foreman enters **OP AP-20, Condenser Tube Leak**, which calls for a 100 MW/min ramp offline along with a shutdown of the affected circulator. The crew immediately implements **OP AP-25, Rapid Load Reduction or Shutdown** to commence the ramp**.** Feedwater cation conductivity rises above the threshold indicating condensate demin breakthrough and the Shift Foreman directs a reactor trip.

6. On the trip, a main steamline break develops downstream of the Main Steam Isolation Valves, outside containment and the crew enters **EOP E-0, Reactor Trip or Safety Injection.**

7. All four main steam isolation valves fail open. The crew transitions to **EOP E-2, Faulted Steam Generator Isolation**, and then to **EOP ECA-2.1, Uncontrolled Depressurization of All Steam Generators** where the crew performs the critical task of minimizing the uncontrolled cooldown **(S7CT-2) Minimize the uncontrolled cooldown by throttling AFW flow to approximately 25 gpm per lead.** It is possible that conditions will degrade leading to a MAGENTA and subsequent RED path on RCS INTEGRITY, and a transition is made to contingency procedure **EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition,** which also contains guidance to minimize the uncontrolled cooldown.

8. With temperature stabilizing, pressure begins to rise quickly and the crew performs the final critical task of terminating safety injection flow **(S7CT-3): Terminate ECCS flow before PRT rupture by closing 8801A/B and/or 8803A/B.**

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