Facility: Callaway Date of Exam: 2/23/22

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			6
	2	2	1	1				1	2			2	9			4
	Tier Totals	5	4	4				4	5			5	27			10
2. Plant Systems	1	2	2	2	2	2	3	3	3	3	3	3	28			5
	2	1	0	1	1	1	1	1	1	1	1	1	10			3
	Tier Totals	3	2	3	3	3	4	4	4	4	4	4	38			8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
					3		2		2		3					

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		X					Knowledge of the interrelations between a reactor trip and the following: (CFR 41.7 / 45.7) EK2.03 Reactor trip status panel	3.5	11
000008 (APE 8) Pressurizer Vapor Space Accident / 3					X		Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: (CFR: 43.5 / 45.13) AA2.10 High-pressure injection valves and controllers	3.6	18
000009 (EPE 9) Small Break LOCA / 3									
000011 (EPE 11) Large Break LOCA / 3				X			Ability to operate and monitor the following as they apply to a Large Break LOCA: (CFR 41.7 / 45.5 / 45.6) EA1.06 D/Gs	4.2	12
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4		X					Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: (CFR 41.7 / 45.7) AK2.08 CCWS	2.6	4
000022 (APE 22) Loss of Reactor Coolant Makeup / 2					X		Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: (CFR 43.5/ 45.13) AA2.03 Failures of flow control valve or controller	3.1	13
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) AK2.03 Service water or closed cooling water pumps	2.7	17
000026 (APE 26) Loss of Component Cooling Water / 8						X	2.4.1 Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10 / 43.5 / 45.13)	4.6	8
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3	X						Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: (CFR 41.8 / 41.10 / 45.3) AK1.02 Expansion of liquids as temperature increases	2.8	6
000029 (EPE 29) Anticipated Transient Without Scram / 1									
000038 (EPE 38) Steam Generator Tube Rupture / 3									
000040 (APE 40; ~~BW E05; CE E05; W E12~~) Steam Line Rupture—~~Excessive Heat Transfer Uncontrolled Depressurization of all Steam Generators~~ / 4	X						Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: (CFR 41.8 / 41.10 / 45.3) AK1.06 High-energy steam line break considerations	3.7	7
000054 (APE 54; ~~CE E06~~) Loss of Main Feedwater /4				X			Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): (CFR 41.7 / 45.5 / 45.6) AA1.04 HPI, under total feedwater loss conditions	4.4	10
000055 (EPE 55) Station Blackout / 6						X	2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)	4.2	2
000056 (APE 56) Loss of Offsite Power / 6						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	14

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	5
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	9
000062 (APE 62) Loss of Nuclear Service Water / 4									
000065 (APE 65) Loss of Instrument Air / 8			X				Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: (CFR 41.5,41.10 / 45.6 / 45.13) AK3.03 Knowing effects on plant operation of isolating certain equipment from instrument air	2.9	16
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6			X				Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8) AK3.01 Reactor and turbine trip criteria	3.9	1
(W E04) LOCA Outside Containment / 3				X			Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment) (CFR: 41.7 / 45.5 / 45.6) EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	4.0	15
(W E11) Loss of Emergency Coolant Recirculation / 4					X		Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation) (CFR: 43.5 / 45.13) EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.4	3
(~~BW E04~~; W E05) ~~Inadequate Heat Transfer—~~Loss of Secondary Heat Sink / 4									
K/A Category Totals:	3	3	3	3	3	3	Group Point Total:		18

ES-401 PWR Examination Outline Form ES-401-2 Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)									
E/APE # / Name / Safety Function	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	X						Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: (CFR 41.8 / 41.10 / 45.3) AK1.07 Effect of dropped rod on insertion limits and SDM	3.1	26
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2	X						Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions: (CFR 41.8 / 41.10 / 45.3) AK1.01 PZR reference leak abnormalities	2.8	27
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7					X		Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: (CFR: 43.5 / 45.13) AA2.05 Nature of abnormality, from rapid survey of control room data	3.0	21
000036 (APE 36; ~~BW/A08~~) Fuel-Handling Incidents / 8						X	2.1.27 Knowledge of system purpose and/or function. (CFR: 41.7)	3.9	20
000037 (APE 37) Steam Generator Tube Leak / 3									
000051 (APE 51) Loss of Condenser Vacuum / 4			X				Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: (CFR 41.5,41.10 / 45.6 / 45.13) AK3.01 Loss of steam dump capability upon loss of condenser vacuum	2.8	25
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 -~~ High Containment Pressure / 5				X			Ability to operate and / or monitor the following as they apply to the (High Containment Pressure) (CFR: 41.7 / 45.5 / 45.6) EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.7	23
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling – Degraded Core Cooling – Saturated 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						X	2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (CFR: 41.10 / 45.12)	4.6	19
(W E15) Containment Flooding / 5									
(W E16) High Containment Radiation /9		X					Knowledge of the interrelations between the (High Containment Radiation) 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	2.6	22
~~(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 Operations - Natural Circulation with Steam Void in Vessel with/without RVLIS /4									
~~(BW E13 & E14) EOP Rules and Enclosures~~									
(~~CE A11**~~; W E08) ~~RCS Overcooling—~~Pressurized Thermal Shock / 4					X		Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock) (CFR: 43.5 / 45.13) EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments	3.5	24
~~(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	1	1	2	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) K3.04 RPS	3.9	28
004 (SF1; SF2 CVCS) Chemical and Volume Control		X										Knowledge of bus power supplies to the following: (CFR: 41.7) K2.06 Control instrumentation	2.6	43
005 (SF4P RHR) Residual Heat Removal										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.03 RHR temperature, PZR heaters and flow, and nitrogen	2.8	49
006 (SF2; SF3 ECCS) Emergency Core Cooling						X						Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: (CFR: 41.7 / 45.7) K6.05 HPI/LPI cooling water	3.0	48
007 (SF5 PRTS) Pressurizer Relief/Quench Tank									X			Ability to monitor automatic operations of the PRTS including: (CFR: 41.7 / 45.5) A3.01 Components which discharge to the PRT	2.7	44
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) A1.01 CCW flow rate	2.8	52
010 (SF3 PZR PCS) Pressurizer Pressure Control	X											Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.02 ESFAS	3.9	41
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) K5.01 DNB	3.3	47
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) K3.02 RCS	4.3	33
022 (SF5 CCS) Containment Cooling								X				Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; 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) A2.01 Fan motor over-current	2.5	55
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray								X				Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; 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) A2.04 Failure of spray pump	3.9	46
039 (SF4S MSS) Main and Reheat Steam											X	2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	3.8	35

059 (SF4S MFW) Main Feedwater									X			Ability to monitor automatic operation of the MFW, including: (CFR: 41.7 / 45.5) A3.06 Feedwater Isolation	3.2	39
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	36
062 (SF6 ED AC) AC Electrical Distribution				X								Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) K4.10 Uninterruptable ac power sources	3.1	45
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	54
064 (SF6 EDG) Emergency Diesel Generator				X								Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) K4.08 ED/G fuel isolation valves	2.9	34
073 (SF7 PRM) Process Radiation Monitoring											X	2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	4.4	30
076 (SF4S SW) Service Water							X					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: (CFR: 41.5 / 45.5) A1.02 Reactor and turbine building closed cooling water temperatures	2.6	53
078 (SF8 IAS) Instrument Air		X										Knowledge of bus power supplies to the following: (CFR: 41.7) K2.01 Instrument air compressor	2.7	42
103 (SF5 CNT) Containment										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.01 Flow control, pressure control, and temperature control valves, including pneumatic valve controller	3.2	31
053 (SF1; SF4P ICS*) Integrated Control														
006 (SF2; SF3 ECCS) Emergency Core Cooling											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)	4.5	50
010 (SF3 PZR PCS) Pressurizer Pressure Control						X						Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: (CFR: 41.7 / 45.7) K6.04 PRT	4.5	37
022 (SF5 CCS) Containment Cooling									X			Ability to monitor automatic operation of the CCS, including: (CFR: 41.7 / 45.5) A3.01 Initiation of safeguards mode of operation	4.1	40
039 (SF4S MSS) Main and Reheat Steam					X							Knowledge of the operational implications of the following concepts as the apply to the MRSS: (CFR: 441.5 / 45.7) K5.08 Effect of steam removal on reactivity	3.6	51

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) A2.12 Failure of feedwater regulating valves	3.1	32
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) A4.02 SWS valves	2.6	38
103 (SF5 CNT) Containment							X					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: (CFR: 41.5 / 45.5) A1.01 Containment pressure, temperature, and humidity	3.7	29
K/A Category Point Totals:	2	2	2	2	2	3	3	3	3	3	3	Group Point Total:		28

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

Plant Systems—Tier 2/Group 2 (RO/SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive					X							Knowledge of the following operational implications as they apply to the CRDS: (CFR: 41.5/45.7) K5.97 Relationship of T-Ave. to T-Ref	3.3	65
002 (SF2; SF4P RCS) Reactor Coolant						X						Knowledge of the effect or a loss or malfunction on the following RCS components: (CFR: 41.7 / 45.7) K6.02 RCP	3.6	63
011 (SF2 PZR LCS) Pressurizer Level Control											X	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	60
014 (SF1 RPI) Rod Position Indication			X									Knowledge of the effect that a loss or malfunction of the RPIS will have on the following: (CFR: 41.7 / 45.6) K3.02 Plant Computer	2.5	58
015 (SF7 NI) Nuclear Instrumentation														
016 (SF7 NNI) Nonnuclear Instrumentation	X											Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.09 ESFAS	3.7	57
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				Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge 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) A2.01 Maintenance or other activity taking place inside containment	2.9	56
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 Pool Cooling System operating the controls including: (CFR: 41.5 / 45.5) A1.01 Spent fuel pool water level	2.7	62
034 (SF8 FHS) Fuel-Handling Equipment				X								Knowledge of design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) K4.02 Fuel movement	2.5	59
035 (SF 4P SG) Steam Generator														
041 (SF4S SDS) Steam Dump/Turbine Bypass Control										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.06 Atmospheric relief valve controllers	2.9	61
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									X			Ability to monitor automatic operation of the Fire Protective System including: (CRF 41.7 / 45.5) A3.03 Actuation of fire detectors	2.9	64
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: Callaway		Date of Exam: 2/23/22				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.2	Knowledge of operator responsibilities during all modes of plant operation. (CFR: 41.10 / 45.13)	4.1	69		
	2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc (CFR: 41.10 / 43.2)	3.3	66		
	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	74		
	Subtotal			3		
2. Equipment Control	2.2.40	Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)	3.4	67		
	2.2.17	Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. (CFR: 41.10 / 43.5 / 45.13)	2.6	73		
	Subtotal			2		
3. Radiation Control	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)	3.4	72		
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)	2.9	68		
	Subtotal			2		
4. Emergency Procedures/Plan	2.4.16	Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines. (CFR: 41.10 / 43.5 / 45.13)	3.5	71		
	2.4.26	Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage. (CFR: 41.10 / 43.5 / 45.12)	3.1	70		
	2.4.29	Knowledge of the emergency plan. (CFR: 41.10 / 43.5 / 45.11)	3.1	75		
	Subtotal			3		
Tier 3 Point Total				10		

Facility: Callaway Date of Exam: 2/23/22

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				N/A					N/A			9	2	2	4
	Tier Totals				N/A					N/A			27	5	5	10
2. Plant Systems	1												28	3	2	5
	2												10	2	1	3
	Tier Totals												38	5	3	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1: 2, 2: 2	3: 1, 4: 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 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; ~~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) EA2.06 Whether PZR water inventory loss is imminent	4.3	80
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						X	2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)	4.5	76
000026 (APE 26) Loss of Component Cooling Water / 8									
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3									
000029 (EPE 29) Anticipated Transient Without Scram / 1					X		Ability to determine or interpret the following as they apply to a ATWS: (CFR 43.5 / 45.13) EA2.06 Main turbine trip switch position indication	3.9	78
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) EA2.16 Actions to be taken if S/G goes solid and water enters steam line	4.6	77
000040 (APE 40; ~~BW E05; CE E05~~; W E12) Steam Line Rupture~~—Excessive Heat Transfer~~ Uncontrolled Depressurization of all Steam Generators / 4									
000054 (APE 54; ~~CE E06~~) Loss of Main Feedwater /4									
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						X	2.4.18 Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)	4.0	79
000065 (APE 65) Loss of Instrument Air / 8									
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6									
(W E04) LOCA Outside Containment / 3									
(W E11) Loss of Emergency Coolant Recirculation / 4									
(~~BW E04~~; W E05) ~~Inadequate Heat Transfer—~~Loss of Secondary Heat Sink / 4						X	2.1.20 Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)	4.6	81

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						X	2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)	4.3	83
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									
000036 (APE 36; ~~BW/A08~~) Fuel-Handling Incidents / 8									
000037 (APE 37) Steam Generator Tube Leak / 3					X		Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: (CFR: 43.5 / 45.13) AA2.13 Which S/G is leaking	4.3	85
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						X	2.4.41 Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11)	4.6	82
000069 (APE 69; W E14) Loss of Containment Integrity - High Containment Pressure / 5									
000074 (EPE 74; ~~W E06 & E07~~) Inadequate Core Cooling – ~~Degraded Core Cooling – Saturated Core Cooling~~ / 4					X		Ability to determine or interpret the following as they apply to Inadequate Core Cooling: (CFR 43.5 / 45.13) EA2.02 Availability of main or auxiliary feedwater	4.6	84
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									

(~~BW E09~~; ~~CE A13**~~; W E09 & E10) Natural Circulation Operations - Natural Circulation with Steam Void in Vessel with/without RVLIS /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	2	Group Point Total:			4

ES-401											PWR Examination Outline Plant Systems—Tier 2/Group 1 (RO/SRO)		Form ES-401-2
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	2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13)	4.2	90
004 (SF1; SF2 CVCS) Chemical and Volume Control														
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								X				Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.01 Loss of CCW pump	3.6	89
010 (SF3 PZR PCS) Pressurizer Pressure Control														
012 (SF7 RPS) Reactor Protection														
013 (SF2 ESFAS) Engineered Safety Features Actuation														
022 (SF5 CCS) Containment Cooling														
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray											X	2.2.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)	4.5	88
039 (SF4S MSS) Main and Reheat Steam														
059 (SF4S MFW) Main Feedwater														
061 (SF4S AFW) Auxiliary/Emergency Feedwater														
062 (SF6 ED AC) AC Electrical Distribution								X				Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.10 Effects of switching power supplies on instruments and controls	3.3	86
063 (SF6 ED DC) DC Electrical Distribution														
064 (SF6 EDG) Emergency Diesel Generator														
073 (SF7 PRM) Process Radiation Monitoring														

076 (SF4S SW) Service Water														
078 (SF8 IAS) Instrument Air								X				Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; 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) A2.01 Air dryer and filter malfunctions	2.9	87
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 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														
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														
033 (SF8 SFPCS) Spent Fuel Pool Cooling														
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator														
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								X				Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate 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) A2.04 Loss of condensate pumps	2.8	93
068 (SF9 LRS) Liquid Radwaste								X				Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste 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) A2.02 Lack of tank recirculation prior to release	2.8	91
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring											X	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)	4.4	92
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:								2			1	Group Point Total:	3

Facility: Callaway Date of Exam: 2/23/22

Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	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	96
	2.1.15	Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc. (CFR: 41.10 / 45.12)			3.4	94
	Subtotal					2
2. Equipment Control	2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc. (CFR: 41.10 / 43.5 / 45.13)			3.9	97
	2.2.21	2.2.21 Knowledge of pre- and post-maintenance operability requirements. (CFR: 41.10 / 43.2)			4.1	98
	Subtotal					2
3. Radiation Control	2.3.11	Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)			4.3	100
	Subtotal					1
4. Emergency Procedures/Plan	2.4.13	Knowledge of crew roles and responsibilities during EOP usage. (CFR: 41.10 / 45.12)			4.6	95
	2.4.40	Knowledge of SRO responsibilities in emergency plan implementation. (CFR: 41.10 / 43.5 / 45.11)			4.5	99
	Subtotal					2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
		None (NRC Generated Outline)

Facility: Callaway	Date of Examination: 2/14/2022
Examination Level: RO ☒ SRO ☐	Operating Test Number: 2022-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations A1	R, N	G2.1.40 (2.8) Knowledge of refueling administrative requirements JPM: Determine RWST Gravity Feed to RCS requirements
Equipment Control A2	R, N	G2.2.41 (3.5) Ability to obtain and interpret station electrical and mechanical drawings. JPM: Determine WPA/Tagout requirements for 'B' CCP, PBG05B.
Radiation Control A3	M, R	G2.3.12 (3.2) 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. JPM: Determine Estimated Dose and Make Shielding Recommendation
Emergency Plan / Procedures A4	N, R	G2.4.6 (3.7) Knowledge of EOP mitigation strategies. JPM: Calculate Maximum Reactor Vessel Venting Time per EOP Addendum 33

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)

RO Administrative JPMs:

A1 This is a NEW JPM. The applicant will have to determine RWST Gravity Feed to RCS requirements for 2 situations per EDP-ZZ-01129 Attachment 12, RWST Gravity Feed to RCS. Situation #1 - With a RWST level of 66% and a required vent area equal to the RPV head and 2 PZR Safeties, what is the EARLIEST time, in days after reactor shutdown, the RWST Gravity Feed to RCS path can be credited? Situation #2 - 1.3 days after reactor shutdown with an available vent area equal to the RPV head and 1 PZR Safety, what is the LOWEST RWST Level (in %) that the RWST Gravity Feed to RCS path can be credited?

A2 This is a NEW JPM. The applicant will have to determine a hold off tag is required and tagged out the 'B' CCP Pump, PBG05B, with a hold off tag(s) on the NB0201 Breaker (racked out / disengaged position), 2 Suction Valves (closed), and 3 Discharge Valves (closed). One of the three possible charging vent or drain valves will be required along with one CCW drain path.

A3 This is a MODIFIED, BANK JPM. The parent JPM (Admin3-RO-O-001, Rev date of 4/11/2017) was last used on an ILT NRC Exam administered at Callaway in 2017. Upon completion of this JPM, the applicant will have calculated total estimated dose for the work without installing shielding to be 30 mrem and with shielding to be 23.75 mrem. The applicant recommends shielding be requested.
Note: this JPM was modified by changing both calculations (by revised estimate work time and dose rates) and by changing the recommendation in favor of shielding.

A4 This is a NEW JPM. The applicant will have to calculate containment air volume at STP, maximum hydrogen volume that can be vented, and determine a hydrogen flow rate per figure 1 to determine the maximum RCS venting time of 15.75 minutes (acceptable range of 15.67 to 15.93 minutes).

Facility: Callaway

Date of Examination: 2/14/2022

Examination Level: RO ☐ SRO ☒

Operating Test Number: 2022-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations A5	N, R	G2.1.25 (4.2) Ability to interpret reference material, such as graphs, curves, tables, etc. JPM: Review ESW Train A Valve stroke surveillance, OPS-EF-V001A, and determine required actions.
Conduct of Operations A6	N, R	G2.1.2 (4.1) Knowledge of operator responsibilities during all modes of plant operations. JPM: Determine Fire Protection Equipment Operability and Compensatory Measures.
Equipment Control A7	R, P	G2.2.17 (3.8) Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. JPM: Review Work Week Schedule and determine Technical Specifications and risk mitigation strategies.
Radiation Control A8	N, R	G2.3.11 (3.8) Ability to approve release permits. JPM: Review CA0855 Liquid Release Worksheet and TRM limits for upcoming liquid release.
Emergency Plan A9	N, R	G2.4.44 (4.4) Knowledge of emergency plan protective action recommendations. JPM: Complete CA 2843, PAR Flowchart.

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)

SRO Administrative JPMs:

A5 This is a NEW JPM. The applicant will be given a completed surveillance sheet of stroke time data for 9 ESW 'A' Train valves. The applicant determined 3 valves are operable (EFHV0045/0051/0059), 4 valves should be only retested (EFHV0031/0049/0047/0065), and 2 valves are inoperable and will require retesting (EFHV0033/0037).

A6 This is a NEW JPM. The applicant will be given a set of plant data from which the applicant will determine that the Halon Fire Protection equipment protecting ESF Switchgear NB01 is inoperable. Additionally, the applicant will determine that Detection Zones #314 and #315 are affected, room #3301 is affected, along with the fact that this will impact maintenance rule subsystem availability. Finally, the applicant will determine the compensatory action of "establish a continuous fire watch in Rooms 3301 and 3302" is required.

A7 This is a Bank JPM that was used on the 2019 ILT NRC Exam. The Bank JPM # is Admin2-SRO-SO-001, Review Work Week Schedule to determine Technical Specifications and risk mitigation strategies. The applicant will be required to review a work week schedule with 5 planned activities. Out of these activities, 1 will require Technical Specification 3.8.1 Condition B entry. Furthermore, the candidate will determine that the planned work activities cannot occur due to parallel work on the Security Diesel and the "A" EDG day tank.

A8 This is a NEW JPM. The applicant will be given a set of liquid release data and a completed CA0855, Liquid / Gaseous Release Worksheet, to review for accuracy. The applicant will determine there are 3 errors on the CA0855 per the enclosed KEY. Additionally, the applicant will determine that if the liquid release would occur, a violation of TRM 16.11.1.2 would occur, specifically the dose to any organ would exceed the 5 mrem per calendar quarter limit.

A9 This is a NEW TIME CRITICAL JPM. At the completion of this JPM, the applicant determined that the Affect Sectors are L, M, and N. These affected sectors should be Sheltered in Place 5 miles downwind. All sectors within a 2 mile radius of the plant should be Sheltered in Place. Additionally, the applicant completed CA2843 sections: map outline, method, evacuate, reason for type of PAR, and impediments considered correctly (per the included KEY) in less than or equal to (≤) 15 minutes of the start of the JPM.
Note: while completing a PAR was a part of the 2020 ILT Exam, the 2020 JPM and this A9 JPM are different in multiple ways including: impediments, not rapidly progress, different affects zones, shelter order instead of evacuate, and a different

radius (5 miles vs 10 miles). Therefore, these JPMs are significantly different and any knowledge of the 2020 Exam would not provide an unfair advantage for the SRO applicant.

Facility: Callaway	Date of Examination: 2/14/2022
Exam Level: RO ☒ SRO-I ☒ SRO-U ☐	Operating Test Number: 2022-1

Control Room Systems:* 8 for RO, 7 for SRO-I

System/JPM Title	Type Code*	Safety Function
S1. 001 Control Rod Drive System / Make Control Bank A Incapable of Withdrawal – RO Applicants ONLY	L, S, N	1
S2. 004 Chemical and Volume Control System / Establish Charging and RCP Seal injection in Mode 5 – All Applicants	A, L, S, N	2
S3. 006 Emergency Core Cooling System / Raise Safety Injection Accumulator Level – All Applicants	P, S, EN	3
S4. 039 Main and Reheat Steam System / Failure of D MSIV to isolate normally and from SA075A/B, EOP Addendum 13 to isolate main steam valves downstream – All Applicants	A, S, N	4S
S5. 026 Containment Spray System / Perform 'B' Containment Spray Pump Inservice Test then respond to Containment Flooding due to component Failure – All Applicants	A, S, EN, M	5
S6. 064 Emergency Diesel Generator / Remove 'B' EDG from 4160 ESF bus NB02 – All Applicants	D, S, EN	6
S7. 015 Nuclear Instrumentation / Perform OSP-SE-00005, Boron Dilution Mitigation System then respond to a SR NI failure and restore from CCP swapover to the RWST (BDMS) – All Applicants	A, L, S, N	7
S8. 029 Containment Purge System / Raise Containment Pressure per OTN-GT-00001, Addendum 1, and then respond to an Containment Isolation Signal with failures – All Applicants	A, S, N	8

In-Plant Systems:* 3 for RO, 3 for SRO-I		
P1. 062 AC Electrical distribution / Swapping NN01 power supply from normal to swing inverter then bypass power source – All Applicants	A, N	6
P2. 008 Component Cooling Water System / Bypass and isolate CCW to the Seal Water Heat Exchanger per Off Normal Procedure – All Applicants	N, R, E	8
P3. 002 Reactor Coolant System / Swap CVCS Seal Injection Filters – All Applicants	P, R	2

* 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	Actual for R / SRO-I
(A)lternate path	4–6 / 4–6	6 / 6
(C)ontrol room		
(D)irect from bank	≤ 9 / ≤ 8	3* / 3*
(E)mergency or abnormal in-plant	≥ 1/ ≥ 1	1 / 1
(EN)gineered safety feature	≥ 1 / ≥ 1	3 / 3
(L)ow-Power/Shutdown	≥ 1 / ≥ 1	3 / 2
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2	8 / 7
(P)revious 2 exams	≤ 3 / ≤ 3	2 / 2
(R)CA	≥ 1 / ≥ 1	2 / 2
(S)imulator		

Note 1: The JPMs from the 2019 exam were randomly selected by placing 11 slips of paper labeled “A” through “K” in a container. No JPMs from the 2020 NRC exam were available for random selection as those JPMs will be used as a part of 2022 Audit Exam.

(*) JPMs S3 and P3 were on a previous exam (P) but as these JPMs were not modified, they were included on the count on Direct from bank (D) also.

Simulator JPMs

S1 This is a New JPM. The plant is in Mode 5 and the applicant will have to make Control Bank A rods incapable of withdrawal to support troubleshooting. Upon completion of the JPM, the applicant will have made Control Bank A rods incapable of withdrawal by placing switches Mech1,2,5 & 6 (Rods H6,H10, F8, K8 respectively) in the downward (disconnect) position.

S2 This is a New Alternate Path JPM. The plant is in Mode 5 with no charging or letdown in service. The applicant will be directed to establish charging with the NCP but will have determine that the Normal charging pump flowpath is not available. The applicant will then establish charging and seal injection flow with the 'A' CCP subsystem. The applicant will have started the 'A' CCP using BG HIS-1A, opened both BG FK-121 and BG HC-182 ('A' CCP FCV and Charging header Backpressure control valve respectively) and balanced CCP system flow to achieve RCP Seal Injection flow of 8-13 gpm per pump and a charging header flow of ~75 gpm (70 to 80 gpm is acceptable).

S3 This is a Bank JPM that was used on the 2019 ILT NRC Exam. The applicant will have started the 'B' SI pump, raised SI Accumulator 'A' level to between 35% and 55%, restored the Safety Injection System lineup.

S4 This is a New Alternate Path JPM. The applicant will have isolated D SG after a failure of MSIV(s) to close normally and from panels SA075A/B. Specifically, the applicant will have to close BM HIS-4A and AB HIS-10 per E-3, SGTR. However, due to the multiple failures associated with closing the D MSIV, the applicant will have to expand the isolation boundary and disarm Condenser Steam Dump logic and close 8 Steam Supply valves per EOP Addendum 13, MS Header Isolation – Control Room Actions.

S5 This is a Modified Bank JPM. The bank JPM has not been used on the last 4 ILT NRC exams. The applicant will have performed the B Containment Spray Pump inservice test, by starting PEN01B with EN HIS-9, calculated a pump differential pressure of 173.8 psid, and then stopped the RWST from draining into the CTMT Recirc Sump (due to a malfunction) by closing at least one suction valve (ENHV0007 and/or BNHV0003) before Annunciator 60D alarms or RWST level reaches 75% (whichever occurs first).

S6 This is a Bank JPM. The bank JPM (NE-RO-S-004) has not been used on the last 4 ILT NRC exams. The applicant will have lowered 'B' EDG load to 0.2 to 0.4 KW using KJ HIS-107A and then placed NE HIS-26 in the Trip position. Lastly, NE HS-6 will be placed to reset. If the applicant lowered EDG load too quickly / too much and caused a reverse power EDG trip, (or was the cause of any EDG automatic trip), the JPM should be considered UNSAT.

S7 This is a New Alternate Path JPM. The applicant the applicant will have responded to a Source Range flux doubling on SR N31 (channel being tested) and a swapover from the VCT to the RWST. The applicant will pressed RESET pushbutton on SE HS-11/12, then swapped CCP suction back to the VCT by opening BG HIS-112B&C then closing BN HIS-112D/E using the handswitches on RL001/2.

S8 This is a New Alternate Path JPM. Reactor Power is 100%The applicant will have started to raise CTMT pressure by opening GTHZ0026/0027, GTHZ0004/0005, GTHZ0041/0042. After CTMT pressure begins to rise, the applicant will respond to a Containment Purge Isolation Signal (CPIS) and close at least one of the 2 dampers: GTHZ0005 and/or GTHZ0004.

In Plant JPMs

P1 This is a New Alternate Path JPM. The applicant will be directed to swap NN01 power supply from the normal power supply (NK0111 via the normal inverter, N11) to the swing inverter (NN17) but due to a malfunction the Swing inverter cannot take the load. The applicant will have to assess plant conditions and determine that the NN01 can be powered from the bypass power source (NG01ABR1) via a different section of the procedure. Upon completion of this JPM, the applicant will have powered NN01 from the bypass power source (NG01AABR1) by pressing NN11S202, Bypass to Load Pushbutton, and then bypassed the static transfer switch by placing NN11S1, Maintenance Bypass Switch, in Bypass.

P2 This is a New JPM. The applicant will have bypassed and isolated CCW to the Seal water Heat Exchanger by manually opening BG8400 and manually closing 4 isolation valves: BG-8393A&B, BGV0206, and EGV0085 per OTO-EG-00001, CCW System Malfunction, Attachment A.

P3 This is a Bank JPM that was used on the 2019 ILT NRC Exam. The applicant will be directed to swap RCP Seal Injection filters per a normal procedure. The JPM will be complete when the 'B' CVCS seal water injection filter will have been placed in service and 'A' placed in standby.

Facility: Callaway Date of Exam: 2/14/2022 Operating Test No. 2022-1

APPLICANT	EVENT TYPE	Scenarios – **Team 1: I1, I2, R1**															
		1 #			2			3			4			TOTAL	MINIMUM(*)		
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP				
		#	#	#	**I1**	**I2**	**R1**	**I2**	**I1**	**R1**	**I2**	**R1**	**I1**		R	I	U
I1	RX				2									1	1	1	0
	NOR	1		1	1				1				1	3	1	1	1
	I/C	2,3,4,5,6	2,5,6	3,4	3,4,6,7				2,3,4,5,6				2,3,4,7	13	4	4	2
	MAJ	7	7	7	5				7				6	3	2	2	1
	TS	3,4			2,4									2	0	2	2
I2	RX					2					2			2	1	1	0
	NOR							1			1			2	1	1	1
	I/C					4,6		2,3,4,5,6			3,4,5,7			11	4	4	2
	MAJ					5		7			6			3	2	2	1
	TS							2,4			2,5			4	0	2	2
R1	RX											2		1	1	1	0
	NOR						1							1	1	1	1
	I/C						2,3,4,6,7			4,5,6		5,7		10	4	4	2
	MAJ						5			7		6		3	2	2	1
	TS														0	2	2

Facility: Callaway Date of Exam: 2/14/2022 Operating Test No. 2022-1

Scenarios – **Team 2 &(3): I3, R2, R3 & (I4, R4, R5) + Surrogate(S1)**

APPLICANT	EVENT TYPE	1 #			2			3			4			TOTAL	MINIMUM(*)		
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP				
		#	#	#	**I3 (I4)**	**R2 (R4)**	**R3 (R5)**	**I3 (I4)**	**R3 (R5)**	**R2 (R4)**	**S1**	**I3 (I4)**	**R2 (R4)**		R	I	U
I3 (I4)	RX				2							2		2	1	1	0
	NOR	1		1	1			1						2	1	1	1
	I/C	2,3,4,5,6	2,5,6	3,4	3,4,6,7			2,3,4,5,6				5,7		11	4	4	2
	MAJ	7	7	7	5			7				6		3	2	2	1
	TS	3,4			2,4			2,4						4	0	2	2
R2 (R4)	RX					2								1	1	1	0
	NOR												1	1	1	1	1
	I/C					4,6				4,5,6			2,3,4,7	9	4	4	2
	MAJ					5				7			6	3	2	2	1
	TS														0	2	2
R3 (R5)	RX													(*)	1	1	0
	NOR						1		1					2	1	1	1
	I/C						2,3,4,6,7		2,3,4,5,6					10	4	4	2
	MAJ						5		7					2	2	2	1
	TS														0	2	2

Facility: Callaway Date of Exam: 2/14/2022 Operating Test No. 2022-1

APPLICANT	EVENT TYPE	Scenarios – **Team 4: I5, R6, Surrogate(S1)**												TOTAL	MINIMUM(*)		
		1 #			2			3 @			4						
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP				
		#	#	#	**I5**	**R6**	**S1**	@	@	@	**S1**	**I5**	**R6**		R	I	U
I5	RX				2							2		2	1	1	0
	NOR	1		1	1									1	1	1	1
	I/C	2,3,4,5,6	2,5,6	3,4	3,4,6,7							5,7		6	4	4	2
	MAJ	7	7	7	5							6		2	2	2	1
	TS	3,4			2,4									2	0	2	2
R6	RX					2								1	1	1	0
	NOR												1	1	1	1	1
	I/C					4,6							2,3,4,7	6	4	4	2
	MAJ					5							6	2	2	2	1
	TS														0	2	2

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the at-the-controls (ATC) and balance-of-plant (BOP) positions. Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-I *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.

2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional I/C malfunctions on a one-for-one basis.

3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.

4. For new reactor facility licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

NOTES:

All 4 scenarios and their attributes are listed as they are labeled for ease of comparison. The Total Columns is summed for Scenario #2 through 4 (or 2 scenarios as appropriate for Team 4) while Scenario #1 and its attributes are shown as the spare. This in no way means that Callaway Energy Center desires Scenario #1 as the spare; specifically Callaway Energy Center would prefer the Chief Examiner to determine which scenario to designate as the spare based on the ES-D1's provided and on site validation. Callaway Energy Center will then update this ES-301-5 per NRC direction.

@ Team 4 will require 2 scenarios. Spare Scenario (#) is listed but Scneario #3 is not listed to avoid summing confusion. The total attributes listed are the totals present for Scenario #2 and #4. This in no way means that Callaway Energy Center desires Team 4 to take Scenarios #2 and #4 and have Scenario#1 or #3 as the spare; specifically Callaway Energy Center would prefer the Chief Examiner to determine which scenarios Team 4 will take based on the ES-D1's provided and on site validation. The number of normal and reactivity events combined with the number of malfunctions will need to be reevaluated for R6. Callaway Energy Center will then update this ES-301-5 per NRC direction.

Facility: Callaway Scenario No.1, Rev 0 Op-Test No. 2022-1

Examiners: ______________________________ Operators: ______________________________

Initial Conditions: Mode 3, BOC, NOP/NOT, Shutdown Banks Withdrawn, Equipment OOS: None

Turnover: Place mini purge in service per OTN-GT-00001, Containment Purge System, Section 5.2 to support a containment entry later in shift.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	SRO (N) BOP (N)	Place Mini-Purge in service per OTN-GT-00001, Containment Purge System, Section 5.2
2	BG / BG LT149	SRO (I) ATC (I)	BG LT 149 fails low. OTO-BG-00004, VCT Level Channel Failures
3	AB / ABPV0003	SRO (C) BOP (C)	Atmospheric Steam Dump 'C' fails open with manual control. OTO-AB-00001, Steam Dump Malfunction (Tech Spec 3.7.4)
4	NB / NB02_F	SRO (C) BOP (C)	Lockout of 4160 VAC Bus NB02. OTO-NB0002, Loss of Power to NB02. (Tech Spec 3.8.1 / 3.8.4 / 3.8.9 / 3.7.20)
5	BB / BBLT0459	SRO (I) ATC (I)	Pressurizer Level Channel LT-459 Fails Low. OTO-BG-00001, Pressurizer Level Control Malfunctions.
6	MD / multiple	SRO (C) ATC (C)	Loss of the switchyard. E-0, Reactor Trip or Safety Injection.
7	NE / multiple	SRO (M) ATC (M) BOP (M)	Failure of the 'A' EDG field to flash causing the output breaker to close. Station blackout. Restore Power from COOP power to NB01 per EOP Addendum 39. Transition to ECA recovery procedure, verify RCP Seal Isolation, then restore high pressure injection with the 'A' CCP. CT-1 Energize NB01 AC Emergency Bus using EOP Addendum 39 CT-2 Ensure RCP Seal isolation is complete prior to starting the 'A' charging pump

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

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Total malfunctions (5-8)	6
2. Malfunctions after EOP entry (1-2)	1
3. Abnormal events (2-4)	4
4. Major transients (1-2)	1
5. EOPs entered/requiring substantive actions (1-2)	1
6. EOP contingencies requiring substantive actions (0-2)	2
7. Critical tasks (2-3)	2

Scenario #1 Event Description
Callaway 2022-1 NRC ES-D-1, rev. 0

The Plant is Mode 3 at Normal Operating Pressure and Temperature (NOP / NOT). Shutdown Banks are withdrawn.

After the reactivity brief is complete, the crew will place containment mini purge in service per OTN-GT-00001, Containment Purge System, Section 5.2 to support a containment entry later in shift.

After mini purge is placed in service, BG LT 149 fails downscale. The crew will enter OTO-BG-00004, VCT Level Channel Failures and will stop the makeup to the VCT by placing the reactor makeup control switch (BG HS 25) to stop.

After the makeup to the is stopped, the 'C' Atmospheric Steam Dump fails open. The BOP operator should close the dump valve using manual control. The crew should enter OTO-AB-00001, Steam Dump Malfunction. Tech Spec 3.7.4 applies.

After the 'C' ASD is closed and Technical Specifications are addressed, NB02 will lockout due to a bus fault. The crew will enter OTO-NB-00002, Loss of Power to NB02, to mitigate the event. The crew will swap CCW service loop to the A train, verify all NB02 feeder breakers are open, secure the B EDG, and place both B Train CCW pump handswitches (EG HIS-22/24) in Pull To Lock (PTL). Technical specifications 3.8.1, 3.8.4, 3.8.9 and 3.7.20 apply.

After B Train CCW pumps handswitches are in PTL, Pressurizer level channel BB LT-459 fails low. The crew will enter OTO-BG-00001, Pressurizer Level Control Malfunction. Actions should be taken to transfer Pressurizer level control to an operable channel. The failure results in a loss of RCS letdown and charging flow should be adjusted to supply the RCP seals only. Letdown flow should be reestablished and Pressurizer level stabilized. The event is terminated when Letdown is restored and pressurizer level control is in Auto.

After the BB LT-459 failure is mitigated, the transmission operations supervisor calls and informs the crew that there is a grid disturbance occurring in the greater Saint Louis Area including Montgomery Substation. 1 minute after this, both Montgomery-Cal offsite 345 kV lines are lost. 2 minutes after this, the remaining switchyard lines are lost. The crew will enter E-0, Reactor Trip or Safety injection. The 'A' EDG will start but the field will fail to flash causing the output breaker, NB0111, to fail to close. These failures can not be mitigated manually from the control room. This will result in a station blackout and the crew will enter ECA-0.0, Loss of All AC Power.

The crew will isolated the RCS, place A Train ECCS components in PTL, direct RCP seal isolation per EOP Addendum 22, and restore power to NB01 from COOP power per EOP Addendum 39. Once AC power is restored to NB01, the crew will transition to ECA-0.1, Loss of All AC Power Recovery without SI required, or ECA-0.2, Loss of All AC Power Recovery with SI required, based on plant conditions present at ECA-0.0, step #31.

The scenario is complete when the crew has started the 'A' CCP per step #3 of ECA-0.1 or per step #6 of ECA-0.2.

Critical Tasks:

	CT-1	CT-2
Critical Tasks	Energize NB01 AC Emergency Bus using EOP Addendum 39, Alternate Emergency Power Supply, within 30 minutes from the beginning of the SBO.	Ensure RCP Seal isolation is complete prior to starting the 'A' charging pump per ECA-0.1, Step 3 or ECA-0.2 Step #6
EVENT	7	7
Safety significance	In the scenario, failure to energize at least one ac emergency bus results in the needless continuation of a situation in which the pumped ECCS capacity and the emergency power capacity are both in a completely degraded status, as are all other active safeguards requiring electrical power. Although the completely degraded status is not due to the crew's action (was not initiated by operator error), continuation in the completely degraded status is a result of the crew's failure to energize at least one ac emergency bus.	Failure to isolate RCP seal injection before starting a charging pump, under the postulated plant conditions, can result in unnecessary and avoidable degradation of the RCS fission-product barrier, specifically at the point of the RCP seals, especially if RCPs are subsequently started. Additionally, failure to perform the critical task results in "significant degradation in the mitigative capability of the plant" in that the RCPs are not available for subsequent event recovery actions. The control room crew is responsible for the coordination and execution of restore charging flow without the unnecessary degradation of the RCS barrier.
Cueing	Indication and/or annunciation that all ac emergency buses are de-energized • Bus energized lamps extinguished • Circuit Breaker Position • Bus Voltage • EDG status	Indication and/or annunciation that all ac emergency buses are de-energized • Bus energized lamps extinguished • Circuit Breaker Position • Bus Voltage • EDG status AND • Step 8 of ECA-0.0 is reached]
Performance indicator	Manipulation of controls as required to energize NB01 from COOP power: ○ PB0502 (AEPS FDR BKR PB0503 to NB0114) ○ NB HIS-67 (Nb01 AEPS SPLY BKR NB0114)	Dispatching of personnel to locally close valves and/or manipulation of controls as required to isolate RCP seal injection • Control switches for the RCP seal injection isolation valves in the closed position • Control switch indication that the RCP seal injection isolation valves are closed
Performance feedback	Indication that NB01 is energized: • NB01 Bus energized light • NB01 bus voltage	• Report from personnel that RCP seals are isolated (or that the appropriate step(s) of the procedure have been completed) • Seal injection flow rate indication of zero when the charging pump is started
Justification for the chosen performance limit	Failure to perform the critical task prior within 30 minutes results in the plant being in a unnecessary degraded condition which could impact RCP Seals, RCS Inventory, and Emergency Plan Implementation and Actions. 30 minutes represents twice the 15 minute time limit to identify or upgrade existing Emergency Action Levels. Specifically an alert, SA1.1, to a site area emergency, SS1.1, would be appropriate if power was not restored from offsite to 4160VAC emergency bus NB02 within 15 minutes. The facility endorses the use of twice the E Plan time limit to evaluate an applicant for a RO or SRO license.	Failure to ensure RCP Seal isolation is complete before restoring a charging pump to service could result in unnecessary "shocking" or degradation of the RCP Seal package and lead to a SBLOCA inside containment. In the process of a recovery from a station blackout, PZR Pressure control will be restored by starting a CCP in ECA-0.1, Loss of All AC Power Recovery without SI Required, Step #3, or in ECA-0.2, Loss of All AC Power Recovery with SI Required, Step #6. Therefore, RCP Seal isolation is required prior to starting a CCP to ensure the RCS fission product barrier.
PWR Owners Group Appendix	CT - 24, Energize at least one ac emergency bus	CT-27 Isolate RCP seal injection

References
OTN-GT-00001, Containment Purge System, rev 32
OTO-BG-00004, VCT Level Channel Failures, rev 20
OTO-AB-00001, Steam Dump Malfunction, rev 19
OTO-NB-00002, Loss of Power to NB02, rev 35
OTO-BG-00001, Pressurizer Level Control Malfunction, rev 25
E-0, Reactor Trip or Safety Injection, rev 27
ECA-0.0, Loss of All AC Power, rev 32
ECA-0.1, Loss of All AC Power Recovery without SI Required, rev 13
ECA-0.1, Loss of All AC Power Recovery with SI Required, rev 12
EOP Addendum 22, Local RCP Seal Isolation, rev 2
EOP Addendum 39, Alternate Emergency Power Supply, rev 15
Tech Spec 3.7.4, Atmospheric Steam Dump Valves (ASDs)
Tech Spec 3.8.1, AC Sources - Operating
Tech Spec 3.8.4, DC Sources - Operating
Tech Spec 3.8.9, Distributions Sources - Operating
ODP-ZZ-00025, EOP/OTO User's Guide

PRA Systems, Events or Operator Actions
1. Station Blackout is a 19% contribution to CDF by Accident Type
2. Alternate Electric Power Supply is the #6 System of the Top 10 Callaway Risk Important Systems.

Facility: Callaway Scenario No.2, Rev 0 Op-Test No. 2022-1

Examiners: ______________ Operators: ______________

Initial Conditions: Mode 1, BOC, 50%, Equipment OOS: 'D' CCW pump

Turnover: Once turnover is complete, swap the CCW Service Loop from B Train to A Train per OTN-EG-00001, CCW System, Section 5.7. The 'A' CCW Pump has the least run time.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	SRO (N) BOP (N)	Swap CCW Service Loop from B to A Train per OTN-EG-00001, CCW System, Section 5.7
2	SF / SFC05_DR	SRO (R) ATC (R) BOP (C)	Shutdown Bank D dropped rod. OTO-SF-00001, Rod Control Malfunction. (Tech Spec 3.1.4)
3	EG / PEG01A	SRO (C) BOP (C)	A Train CCW Pump trip will failure of other pump to autostart, OTO-EG-00001, CCW System Malfunction.
4	BB / EBB01D=35	SRO (C) ATC (C) BOP (C)	Tube Leak on Steam Generator 'D', OTO-BB-00001, Steam Generator Tube Leak. (Tech Spec 3.4.13 and 3.4.17)
5	BB / EBB01D=200	SRO (M) ATC (M) BOP (M)	Tube Leak grows into Tube Rupture in Steam Generator 'D' requiring a manual reactor and entry into E-0, Reactor Trip or Safety Injection.
6	SF / SF006 = Both Modes	SRO (C) ATC (C) BOP (C)	Reactor Failure to trip, transition from E-0 to FR-S.1 at E-0 Step #1 RNO. Local Actions to trip MG set successful, transition back to E-0. CT-1, Insert negative reactivity into the core
7	AL / ALHV0005_ MTFASIS=1	SRO (C) BOP (C)	Tube Rupture in Steam Generator 'D'. E-3, Steam Generator Tube Rupture. SG 'D' AFW FCV (AL HK-5) failure to close. CT-2, Control initial RCS cooldown CT-3, Depressurize RCS to E-3 SI termination criteria CT-4, Isolate the Ruptured SG

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

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Total malfunctions (5-8)	7
2. Malfunctions after EOP entry (1-2)	2
3. Abnormal events (2-4)	3
4. Major transients (1-2)	1
5. EOPs entered/requiring substantive actions (1-2)	2
6. EOP contingencies requiring substantive actions (0-2)	1
7. Critical tasks (2-3)	4

The Plant is Mode 1 at 50% with the 'D' CCW Pump OOS for motor overhaul.

After the reactivity brief is complete, the crew will Swap CCW Service Loop from B to A Train per OTN-EG-00001, CCW System, Section 5.7, Running Both Trains, Shifting Trains Or Service Loop From Train B To Train A.

After the service loop train swap is complete, a D shutdown bank control rod, C5, drops into the core. The crew will enter OTO-SF-00001, Rod Control Malfunctions, and mitigate the event. Tech Spec 3.1.4 applies. The crew will adjust turbine load per OTO-MA-00008, Rapid Load Reduction, to match Tavg/Tref to within 1.5°F.

After Tavg/Tref mismatch is return to band, the A CCW Pump trips and the backup (C CCW Pump) fails to autostart. The crew may start the C CCW pump per prudent operator actions or will start the C CCW pump per step #1 RNO of OTO-EG-00001, CCW System Malfunctions. The event is over once A Train CCW system flow is restored.

Once the CCW malfunction is addressed, a 35 gpm tube leak develops in Steam Generator 'D'. The crew should enter OTO-BB-00001, Steam Generator Tube Leak, and quantify the leak to be greater than 150 gpd. This failure will result in Technical Specification 3.4.13 and 3.4.17 not being met.

After technical specifications are addressed, the Tube Leak will grow into a Tube Rupture. The crew should attempt to manually trip the reactor but the reactor fails to trip. E-0 Step #1 RNO actions to deenergize PG19 and PG20 (to secure rod control MG sets) are not completely successful (PG19 remains energized) and the crew should transition to FR-S.1, Response to Nuclear Power Generation / ATWS. Operators will insert negative reactivity into the core and direct operators to the A MG Set to open the reactor trip and bypass breakers. Once the reactor is shutdown, the crew will transition back to E-0, Reactor Trip or Safety Injection, and continue until a transition to E-3, Steam Generator Tube Rupture, is made.

During the performance of E-3, while isolating auxiliary feedwater flow to the D SG, AL HK-5A (MDAFP FCV to D SG) will not close which will require securing the B MDAFP with AL HIS-22A in Pull To Lock.

The scenario is complete when the crew has has depressurized the RCS using normal spray per E-3 Step #16.

Critical Tasks:

	CT-1	CT-2
Critical Tasks	Insert negative reactivity into the core by at least one of the following methods including dispatching operators to locally Trip the Reactor at the A MG Set • Deenergize PG20 using PG HIS-18 (PG19 can not be deenergized from the control room) • Insert Control Rods • Establish emergency boration flow to the RCS Before 'D' SG NR level is >91% (yellow path on SG Level)	Establish/maintain an RCS temperature so that transition from E-3 does not occur because the RCS temperature is in either of the following conditions: • Too high to maintain minimum required subcooling. (RCS Subcooling is required to be GREATER THAN 50°F [70°F] to prevent a transition to ECA-3.1) at Step #15 RNO OR • Below the RCS temperature that causes an extreme (RED path) or a severe (ORANGE path) challenge to the subcriticality and/or the integrity CSF
EVENT	6	7
Safety significance	In the scenario, failure to insert negative reactivity by one of the methods listed previously can result in the needless continuation of an extreme or a severe challenge to the subcriticality CSF. Although the challenge was not initiated by the crew (was not initiated by operator error), continuation of the challenge is a result of the crew's failure to insert negative reactivity.	Failure to establish and maintain the correct RCS temperature during a SGTR leads to a transition from E-3 to a contingency ERG. This failure constitutes an incorrect performance that "necessitates the crew taking compensating action that would complicate the event mitigation strategy...."
Cueing	Indication of ATWS (the reactor is not tripped and that a manual reactor trip is not effective)	All of the following: • Indication and/or annunciation of SGTR in one SG ○ Increasing SG water level / Radiation • Indication and/or annunciation of reactor trip • Indication and/or annunciation of SI • Indication of ruptured SG pressure greater than minimum required pressure

Performance indicator	Manipulation of controls in the control room as required to initiate the insertion of negative reactivity into the core (at least one of the following)
	• Open supply breakers to PG19 and PG20.
	  ○ PG HIS-16 and PG HIS-18
	• Insert Control Rods at the Maximum Rate.
	• ALIGN emergency boration flow path:
	  ○ Start boric acid transfer pumps
	    ▪ BG HIS-5A and BG HIS-6A
	  ○ OPEN Emergency Borate To Charging Pump Suction valve: BG HIS-8104
		Manipulation of controls as required to establish and maintain RCS temperature
		• Steam dump valve position lamps and/or indicators indicate closed
		• SG PORV valve position lamps and/or indicators indicate closed

Performance feedback	Crew will observe the following: • Indication of a negative SUR on the intermediate range of the excore NIS • Indication of less than 5% power on the power range of the excore NIS	Indication of steam flow rate greater than zero • Indication of RCS temperature decreasing OR • Indication of RCS temperature less than target value
Justification for the chosen performance limit	With a SG Tube rupture on D SG in progress, D SG level will be rising. If the ATWS is not mitigated in a timely manner, D SG parameters of Level and Pressure may rise the point where pressure or level would be released to the environment. This would result in a direct RCS inventory release to the environment effectively bypassing one of the fission product barriers and challenge protecting the health and safety of the public. The Narrow Range level of 91% was selected as a level of this amount would represent a Steam bubble compression causing the D Sg pressure to rise and this parameter is easy to monitor as it is a Yellow CSF path entry parameter.	Terminating the RCS cooldown before reaching the target temperature prevents achieving the minimum RCS subcooling. Failure to achieve the required RCS subcooling results in a condition that forces the crew to transition to contingency ERG ECA-3.1, thereby delaying the RCS depressurization and SI termination. Such a delay allows the excessive inventory increase of the ruptured SG to continue until the SG overpressure components release water or until SG overfill occurs. Terminating the cooldown too late challenges either the subcriticality CSF or the integrity CSF. Because the crew is directed to cool down at the maximum rate, late termination of cooldown could force the RCS temperature low enough to challenge the integrity CSF. The crew must then transition to one of the integrity FRGs. The transition also delays RCS depressurization and SI termination.
PWR Owners Group Appendix	CT- 52, Insert negative reactivity into the core	CT-19, Control initial RCS cooldown

Critical Tasks:

	CT-3	CT-4
Critical Tasks	Depressurize RCS using normal PZR spray per E-3 step#16 until E-3 SI termination criteria is met without the RCS reaching saturation conditions (Subcooling = 0) or Pressurizer NR level reaching 100%.	Isolate feedwater flow into and steam flow from the 'D' SG before a transition to ECA-3.1 occurs.
EVENT	7	7
Safety significance	RCS depressurization decreases the RCS leakage into the SG, helping to mitigate the inventory increase in the ruptured SG. The RCS depressurization also helps the ECCS restore RCS inventory, which in turn allows SI termination. SI termination eliminates the remaining cause of leakage from the RCS into the SG.	Isolating the ruptured SG maintains a differential pressure between the ruptured SG and the intact SGs. The differential pressure (250 psi) ensures that minimum RCS subcooling remains after RCS depressurization.

Cueing	All of the following:
	• Indication and/or annunciation of SGTR in one SG
	• Indication and/or annunciation of reactor trip and SI
	• Indication that the RCS is cooled down to the target temperature per E-3 Step #13
		All of the following:
		• Indication and/or annunciation of SGTR in one SG
		  ○ Increasing SG water level
		  ○ Radiation
		• Indication and/or annunciation of reactor trip
		• Indication and/or annunciation of SI

Performance indicator	Manipulation of controls as required to depressurize the RCS
	• Valve position indications and controls for PZR spray valve
		Manipulation of controls as required to isolate the ruptured 'D' SG
		• Adjust ruptured SG(s) ASD controller setpoint to 1160 psig:
		  ○ AB PIC-4A (SG D)
		• Close Steamline Low Point Drain Valve:
		  ○ AB HIS-10 (SG D)
		• Close D MSIVs and Bypass valves:
		  ○ AB HIS-11 (SG D)
		• Stop Auxiliary feed flow to ruptured SG
		  ○ CLOSE AL HK-6A and place AL HIS-22A to PTL (B MDAFP)

Performance feedback	Crew will observe the following: • Indication of RCS pressure decreasing • Indication of PRZR level increasing	Crew will observe the following: • Indication of stable or increasing pressure in the ruptured SG • Indication of decreasing or zero feedwater flow rate in the ruptured SG

Justification for the chosen performance limit	The intent is to depressurize to establish and maintain the criteria that allow the crew to terminate SI. E-3 SI termination criteria is any of the following conditions satisfied (per E-3 Step #17b):
	• Both of the following:
	  • RCS pressure – LESS THAN RUPTURED SG(s) PRESSURE
	  • PZR level – GREATER THAN 9% [29%]
	OR
	• PZR level - GREATER THAN 74% [64%]
	OR
	• RCS subcooling – LESS THAN 30°F [50°F]
	Before depressurization, the crew has met most of the criteria for SI termination. The most likely criterion not met is adequate pressurizer level. The depressurization establishes pressurizer level within the range to allow termination. However, if the crew depressurizes too much, the existing subcooling can be lost, inhibiting termination. In addition, if the crew fails to realign the controls after depressurization, RCS pressure will continue to decrease, also inhibiting termination.
	If the RCS reaches saturation conditions (Subcooling = 0) or Pressurizer NR level reaches 100% (i.e PZR complete fills which would cause RCS Pressure to begin to
		When the crew cannot maintain the 250 psi differential, the ERGs require a transition to contingency ERG ECA-3.1. This transition unnecessarily delays the sequence of actions leading to RCS depressurization and SI termination.

	rise inhibiting SG Tube Rupture mitigation), the depressurization continued for too long and this Critical Task is not met.	
PWR Owners Group Appendix	CT-20, Depressurize RCS to E-3 SI termination criteria	CT-18, Isolate the Ruptured SG

References
OTN-EG-00001, CCW System, rev 62
OTO-SF-00001, Rod Control Malfunctions, rev 23
OTO-MA-00008, Rapid Load Reduction, rev 41
OTO-EG-00001, CCW System Malfunction, rev 18
OTO-BB-00001, Steam Generator Tube Leak, rev 31
E-0, Reactor Trip or Safety Injection, rev 27
FR-S.1, Response to Nuclear Power Generation / ATWS, rev 14
E-3, Steam Generator Tube Rupture, rev 25
Technical Specification 3.1.4, Rod Group Alignment Limits
Technical Specification 3.4.13, RCS Operational Leakage
Technical Specification 3.4.17, Steam Generator (SG) Tube Integrity
ODP-ZZ-00025, EOP/OTO User's Guide

PRA Systems, Events or Operator Actions:

1. SG Tube Rupture is a 2% contribution to CDF by Accident Type

Facility: Callaway Scenario No.3, Rev 0 Op-Test No. 2022-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: Mode 1, MOC, 100%. Equipment OOS: 'D' CCW Pump

Turnover: One the crew takes the watch, the crew is to perform OSP-SF-00002, Section 6.2 for control Bank A rods.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	SRO (N) ATC (N)	OSP-SF-00002, Exercise Control Bank A rods
2	AE / AELT0549 = 143.469	SRO (I) ATC (I)	AE LT 549 SG NR Level Instrument Fails downscale. OTO-AE-00002, Steam Generator Water Level Control Instrument Malfunctions (Tech Specs 3.3.1 and 3.3.2)
3	BG / PBG04 = Trip	SRO (C) ATC (C)	Trip of the Normal Charging Pump. OTO-BG-00001, Pressurizer Level Control Malfunction
4	SE / SEN0041 = 1	SRO (I) ATC (I) BOP (I)	Power Range Channel N41 fails low. OTO-SE-00001, Nuclear Instrument Malfunction (Tech Spec 3.3.1)
5	MA / MA01TVH2F1 = 1 EA / EATV0007ZMANTYP = True (1) EA / EATV0007TASTEM=0.01	SRO (C) ATC (C) BOP (C)	Main Generator Machine Gas High (>56°C) OTO-MA-00004, Generator Gas System Malfunction, and Rapid Load reduction per OTO-MA-00008, Rapid Load Reduction
6	SF / SF006 = Both Modes	SRO (C) ATC (C) BOP (C)	Reactor Failure to trip remain in E-0. Control Room Actions successful, transition to ES-0.1. CT-1, Manually trip the reactor before any SG level indicates less than 10% WR level
7	AL / PAL02_1=1 AL / PAL01A_1=1 AL / PAL01B_1=1	SRO (M) ATC (M) BOP (M)	Loss of Auxiliary Feedwater resulting in a Loss of Secondary Heat Sink, FR-H.1. NFASP successful. CT-2, Establish Secondary Heat Sink by establish SG flow from the NSAFP before any SG reaches dryout conditions (10% WR level)

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

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Total malfunctions (5-8)	7
2. Malfunctions after EOP entry (1-2)	2
3. Abnormal events (2-4)	4
4. Major transients (1-2)	1
5. EOPs entered/requiring substantive actions (1-2)	1
6. EOP contingencies requiring substantive actions (0-2)	2
7. Critical tasks (2-3)	2

The Plant is Mode 1 at 100% Reactor Power.

After the reactivity brief is complete, the crew will perform OSP-SF-00002, Control Rod Partial Movement, Section 6.2 for the Control Bank A.

After Control Bank A has been exercised, SG NR Level will fail downscale. The crew will enter OTO-AE-00002, Steam Generator Water Level Control Malfunctions, due to AE LI-549 failing low. The crew will removed the failed instrument from service and restore SG NR to the range of 45% to 55%. The CRS will determine that Technical Specification 3.3.1 Condition A and E apply due to function 14a&b. Technical Specification 3.3.2 Condition A, D, I apply function 5c, 5e and 6d.

After AE LT 549 Technical Specifications are addressed, the Normal Charging Pump, PBG04, trips. The RO should perform the Immediate Actions of OTO-BG-00001, Pressurizer Level Control Malfunction, to start a CCP and return PZR back to program level.

Once the plant has been stabilized, Power Range Nuclear Instrument Channel N41 fails low. The crew will enter OTO-SE-00001, Nuclear Instrument Malfunction, to bypass channel N41 and restore control rods to desired position. Technical Specification 3.3.1 is not met.

After Technical specifications have been addressed, a malfunction occurs which causes a lack of Main Generator H2 Cooling. As a result, the crew will enter OTO-MA-00004 and once H2 temperature has been confirmed to be greater than 56°C, the crew will begin a load reduction. Reducing load will not correct the issue and when H2 Temperature has been greater than 56°C for greater than 15 minutes, the crew should trip the reactor and the main turbine.

The crew enters E-0, Reactor Trip or Safety Injection, and performs the immediate actions. The Reactor will not trip with either of the MCB switches (SB HS-1 or SB HS-42) but opening the supply breakers to PG-19 & PG-20 will successfully deenergize the MG sets allowing the control rods to fully insert into the core. At Step #4, a SI is not required and the crew transitions to ES-0.1, Reactor Trip Response. With the transition from E-0, CSFs are now monitored by the crew and the crew should identify a RED Path on Secondary Heat Sink and then transition to FR-H.1, Response to Loss of Secondary Heat Sink.

The TDAFP pump can not be started from the control room or locally. The 'B' MDAFP trips on overcurrent shortly after its autostart after the transition from E-0. The 'A' MDAFP is showing signs of cavitation and has significantly reduced flow. As a result, the crew will place the NSAFP per EOP Addendum 38.

The scenario is complete when the crew has establish flow from the NSAFP to at least one SG.

Critical Tasks:

	CT-1	CT-2
Critical Tasks	Manually trip the reactor before any SG level indicates less than 10% WR level.	Establish Secondary Heat Sink by establish SG flow from the NSAFP before any SG reaches dryout conditions (10% WR level).
EVENT	6	7
Safety significance	Failure to manually trip the reactor causes a challenge to the subcriticality CSF beyond that irreparably introduced by the postulated conditions. Additionally, it constitutes an incorrect performance that "necessitates the crew taking compensating action that would complicate the event mitigation strategy" and demonstrates the inability of the crew to "recognize a failure or an incorrect automatic actuation of an ESF system or component."	Failure to establish the minimum required feedwater flow rate, under the postulated plant conditions, results in "adverse consequences or significant degradation in the mitigative capability of the plant." In this case, the minimum required feedwater flow rate can be established by performing the appropriate manual action.
Cueing	Indication and/or annunciation that plant parameter(s) exist that should result in automatic reactor trip but reactor does not automatically trip • Manual Reactor Trip Switch did not trip the reactor when depressed.	Indication and/or annunciation of reactor trip Indication and/or annunciation that secondary heat sink is required Indication and/or annunciation that the feedwater flow rate is less than the minimum required • Total feedwater flow rate indicates less than the minimum required • Total AFW flow rate indicates less than the minimum required • Control switch indication that the steam supply valves to the turbine-driven AFW pump are closed • AFW valve position indication that a flow path is not established to at least one SG
Performance indicator	Manipulation of control room PG 19& PG 20 feeder breakers switches (PB HIS 16 & 18) to deenergize the control rod drive MG Sets allowing the rods to fall into the core. • Power Range NI lowering • DRPI indication of rods at bottom	Manipulation of controls in the control room as required to establish the minimum required feedwater flow rate to the SGs: • Valve position indication that the steam admission valve(s) for the turbine-driven AFW pump are open
Performance feedback	Indications of reactor trip • Control rods at bottom of core • Neutron flux decreasing	Indication that at least the minimum required feedwater flow rate is being delivered to the SGs Indication of increasing SG levels
Justification for the chosen performance limit	Not tripping the reactor prior to any SG reaching dryout conditions when it is possible to do so forces an immediate extreme challenge to the subcriticality CSF, availability of the heat sink, and containment. Additionally, the incorrect performance of failing to trip the reactor necessitates the crew taking compensating action that seriously complicates the event mitigation strategy. This misoperation constitutes a "significant reduction of safety margin beyond that irreparably introduced by the scenario."	Because the secondary heat sink is required but not satisfactorily provided, the RCS heats up. If feedwater flow rate commensurate with core decay heat is not established, the heat sink CSF is eventually challenged. With continued insufficient feedwater flow, the SGs dry out, causing an RCS pressure increase that opens the pressurizer PORVs. The open PORVs create a small-break LOCA that eventually challenges the core cooling CSF. Ultimately, the fuel matrix/clad (a fission-product barrier) is challenged.
PWR Owners Group Appendix	CT-1, Manually trip the reactor	CT 45, Establish minimum required feedwater flow rate to SGs before SG dryout

"NOTE: (Per NUREG-1021, Appendix D) If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review. "

References
OSP-SF-00002, Control Rod Partial Movement, Rev 28
OTO-AE-00002, Steam Generator Water Level Control Malfunctions, Rev 13
OTO-BG-00001, Pressurizer Level Control Malfunction, Rev 25
OTO-SE-00001, Nuclear Instrument Malfunction, Rev 33
OTO-MA-00004, Generator Gas System Malfunction, Rev 11
OTO-MA-00008, Rapid Load Reduction, Rev 41
E-0, Reactor Trip or Safety Injection, Rev 27
ES-0.1, Reactor Trip Response, Rev 24
FR-H.1, Response to Loss of Secondary Heat Sink, Rev 19
EOP Addendum 38, Non Safety Auxiliary Feedwater Pump, Rev 12
Tech Spec 3.3.1, Reactor Trip System (RTS) Instrumentation
Tech Spec 3.3.2, EFSAS Instrumentation
ODP-ZZ-00025, EOP/OTO User's Guide

PRA Systems, Events or Operator Actions

1. ATWS is 16% Contribution to CDF by Accident Type
2. #6 of the Top 10 Risk Reduction Operator Actions is to "Open Reactor Trip Breakers, manually insert rods and emergency borate to achieve lon term shutdown after ATWS"
3. Auxiliary Feedwater is the #1 Callaway Risk Important System

Facility: Callaway Scenario No.4, Rev 0 Op-Test No. 2022-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: Mode 1, BOC, 75%. Equipment OOS: None. Rod Control is in Manual.

Turnover: One the crew takes the watch, the crew is to perform OSP-AC-00001, Section 6.1 for Turbine Stop Valve (MSV) testing for MSV#1.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	SRO (N) BOP (N)	Full Stroke Test of Turbine Main Stop Valve #1 per OSP-AC-00001.
2	BB / BBTE411A1	SRO (R) ATC (R) BOP (I)	RCS Loop 1 Thot RTD Fails High, OTO-BB-00004, RCS RTD Failures. (Tech Spec 3.3.1)
3	AE / AE FK-520	SRO (C) BOP (C)	'B' MFRV fails open. OTO-AE-00001, Feedwater System Malfunction.
4	CD / PCD01 CD / PCD03	SRO (C) BOP (C)	Main Seal Oil Pump trip with failure of Emergency Seal Oil Pump to autostart. OTO-MA-00002, Generator Seal Oil System Malfunction.
5	BB / BB001_C=35	SRO (C) ATC (C)	A 35 gpm RCS leak develops. OTO-BB-00003, RCS Excessive Leakage. (Tech Spec 3.4.13)
6	BB / BB001_C = 3500	SRO (M) ATC (M) BOP (M)	3500 gpm LOCA, E-0 Reactor Trip or Safety Injection.
7	NF / NF039A_1 (failure to start) BG / PBG05B_1 =1	SRO (C) ATC (C) BOP (C)	A LOCA Sequencer fails to start (prior to Step 0). B CCP Fails to Autostart. CT#1, Trip all RCPs within 5 minutes of meeting RCP trip criteria. CT#2, Manually start the 'A' SI pump, 'A' & 'B' CCPs for RCS Injection. CT#3, Manually start the 'A' or 'C' CCW Pump.

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

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Total malfunctions (5-8)	7
2. Malfunctions after EOP entry (1-2)	2
3. Abnormal events (2-4)	4
4. Major transients (1-2)	1
5. EOPs entered/requiring substantive actions (1-2)	1
6. EOP contingencies requiring substantive actions (0-2)	0
7. Critical tasks (2-3)	3

Scenario #4 Event Description
Callaway 2022-1 NRC ES-D-1, rev. 0

The Plant is Mode 1 at 75% Reactor Power. Rod Control is in Manual.

After the reactivity brief is complete, the crew will perform OSP-AC-00001, Section 6.1, and full stoke Main Stop Valve #1 (MSV#1).

After OSP-AC-00001, Section 6.1.2, is complete for MSV#1, the Loop 1 Thot RCS RTD, AB TE411 fails high. The crew will enter OTO-BB-00004, RCS RTD Failures, and remove the failed instrument from service. The CRS will determine that Technical Specification 3.3.1 Condition A and E are not met apply due to function 6 & 7.

After technical specifications are addressed, the 'B' MFRV fails opens. The crew will enter OTO-AE-00001, Feedwater System Malfunctions, and place the affected MFW Reg Valve in Manual then restore level to between 45 to 55%.

After the crew has stabilized SG levels and returned them to the control band, the Main Seal Oil Pump, PCD01, trips and the Emergency Seal Oil Pump (ESOP), PCD03, fails to autostart. The crew will enter OTO-MA-00002, Generator Seal Oil System Malfunction, and place the ESOP in service.

After the ESOP is in service, a small RCS Leak (35 gpm) ramps in over 2 minutes. The crew will enter OTO-BB-00003, RCS Excessive Leakage. The CRS will determine that Technical Specification 3.4.13 Condition A is not met.

After Technical Specifications have been addressed, a 3500 gpm LOCA occurs requiring a Reactor trip and Safety Injection. The A LOCA sequencers fails prior to start (NF039A fails autostart) and the B CCP Fails to autostart on its sequencer requiring manual action to start and establish high head RCS Injection flow. Additionally the 'A' or 'C' CCW pump must be started to provide cooling to 'A' train ECCS components. Finally, once RCP trip criteria is met, the crew must trip the RCPs within 5 minutes of meeting this criteria.

The scenario is complete when the crew has transitioned to ES-1.2, Post LOCA Cooldown and Depressurization.

Critical Tasks:

	CT-1	CT-2
Critical Tasks	Trip all RCPs within 5 minutes of meeting RCP trip criteria.	Manually start the 'A' SI pump. 'A' & 'B' CCPs for RCS Injection before completion of E-0 Attachment A
EVENT	7	7
Safety significance	Failure to trip the RCPs under the postulated plant conditions leads to core uncovery and to fuel cladding temperatures in excess of 2200°F, which is the limit specified in the ECCS acceptance criteria. Thus, failure to perform the task represents misoperation or incorrect crew performance in which the crew has failed to prevent “degradation of...{the fuel cladding} ...barrier to fission product release” and which leads to “violation of the facility license condition.”	The acceptable results obtained in the FSAR analysis of a small-break LOCA are predicated on the assumption of minimum ECCS pumped injection. The analysis assumes that a minimum pumped ECCS flow rate, which varies with RCS pressure, is injected into the core. The flow rate values assumed for minimum pumped injection are based on operation of one each of the following ECCS pumps: Charging/SI pump (HP plants only), high-head SI pump, and low-head SI pump. Operation of this minimum required complement of ECCS injection pumps is consistent with the FSAR assumption that only minimum safeguards are actuated. Because compliance with the assumptions of the FSAR is part of the facility license condition, failure to perform the critical task (under the postulated plant conditions) constitutes a violation of the license condition.
Cueing	Indications of a SBLOCA AND Indication and/or annunciation of safety injection AND Indication and/or annunciation that at least one CCP/SI pump is running AND Indication that the RCP trip criteria are met	Indication and/or annunciation that Charging/SI pump injection is required • SI actuation • RCS pressure below the shutoff head of the Charging/SI pump Indication and/or annunciation that no Charging/SI pump is injecting into the core • Control switch indication that the circuit breakers or contactors for both Charging/SI pumps are open • All Charging/SI pump discharge pressure indicators read zero • All flow rate indicators for Charging/SI pump injection read zero
Performance indicator	Manipulation of controls as required to trip all RCPs • RCP breaker position lights indicate breaker open	Starting the 'A' SI pump using EM HIS-4 Starting the 'B' CCP using BG HIS-2A Starting the 'A' CCP using BG HIS-1A
Performance feedback	Indication that all RCPs are stopped • RCP breaker position lights • RCP flow decreasing • RCP motor amps decreasing	With the 'B' train of ECCS have lost its cooling water pump and the size of the LOCA, the 'A' CCP will not be able to restore RCS water level while RCS pressure remains greater than the 'A' RHR pump head. As RCS Pressures lowers below the SI pump shutoff head, the 'A' SI pump will be required and once the 'A' SI pump is started, Indication and/or annunciation that the ‘A’ SI is injecting and Flow rate indication of injection from the ‘A’ SI pump
Justification for the chosen performance limit	In a letter to the NRC titled “Justification of the Manual RCP Trip for Small Break LOCA Events” (OG-117, March 1984) (also known as the Sheppard letter), the WOG provided the required assurance based on the results of the analyses performed in conjunction with WCAP-9584. The WOG showed that for all Westinghouse plants, more than two minutes were available between onset of the trip criteria and depletion of RCS inventory to the critical inventory. In fact, additional analyses sponsored by the WOG in connection with OG-117 conservatively showed that manual RCP trip could be delayed for five minutes beyond the onset of the RCP trip criteria without incurring any adverse consequence.	“before completion of Attachment A of E-0” is in accordance with the PWR Owners Group Emergency Response Guidelines. It allows enough time for the crew to take the correct action while at the same time preventing avoidable adverse consequences.
PWR Owners Group Appendix	CT-16, Manually Trip RCPs	CT-6, Establish flow from at least one Charging/SI pump

	CT-3	
Critical Tasks	Manually start the 'A' or 'C' CCW Pump before completion of E-0 Attachment A	
EVENT	7	
Safety significance	Failure to manually start at least the minimum number of CCW pumps required to provide adequate component cooling for the operating safeguards train(s)represents a failure by the crew to “demonstrate the following abilities: • Effectively direct or manipulate engineered safety feature (ESF) controls that would prevent a significant reduction of safety margin beyond that irreparably introduced by the scenario • Recognize a failure or an incorrect automatic actuation of an ESF system or component” Additionally, under the postulated plant conditions, failure to manually start at least the minimum required number of CCW pumps (when it is possible to do so) is a “violation of the facility license condition.” Operation of the ECCS injection pumps without CCW could lead to pump failure or damage, which would constitute misoperation or incorrect crew performance in which the crew does not prevent “degraded emergency core cooling system (ECCS) … capacity.”	
Cueing	Indications of a Reactor trip AND Indications of a Safety Injection AND Indications NO 'A' Train CCW pumps running AND Indications that at least one of the 'A' CCW pumps can be started (No overload aka power available, etc)	
Performance indicator	Manipulation of controls as required to start at least the minimum number of CCW pumps required to provide adequate component cooling for the operating safeguards train(s) Control switch indication that the appropriate circuit breaker(s) or contactor(s) are closed	
Performance feedback	Indication and/or annunciation that at least the minimum number of CCW pumps required to provide adequate component cooling for the operating safeguards train(s) is running • CCW low pressure condition clear; indication of pressure • CCW low flow condition clear; indication of flow	
Justification for the chosen performance limit	“before completion of Attachment A of E-0” is in accordance with the PWR Owners Group Emergency Response Guidelines. It allows enough time for the crew to take the correct action while at the same time preventing avoidable adverse consequences.	
PWR Owners Group Appendix	CT-8, Manually start CCW pump	

References
OSP-AC-00001, Turbine Stop Valve Trip Actuating Device Test, Rev 17
OTO-BB-00004, RCS RTD Channel Failures, Rev 21
OTO-AE-00001, Feedwater System Malfunction, Rev 39
OTO-MA-00002, Generator Seal Oil system Malfunction, Rev 12
OTA-RK-00026, Addendum 130E, Generator Auxiliary Trouble, Rev 3
OTO-BB-00003, RCS Excessive Leakage, Rev 27
E-0, Reactor Trip or Safety Injection, Rev 27
E-1, Loss of Reactor or Secondary Coolant, Rev 23
ES-1.2, Post LOCA Cooldown and Depressurization, Rev 21
Tech Spec 3.3.1, Reactor Trip System (RTS) Instrumentation
Tech Spec 3.4.13, RCS Operational Leakage
ODP-ZZ-00025, EOP/OTO User's Guide

PRA Systems, Events or Operator Actions

1. Medium Size LOCA is a 19% Contribution to CDF