September 27, 2018

Kelly D. Clayton
NRC Chief Examiner
U. S. Nuclear Regulatory Commission
Region IV
1600 East Lamar Blvd.
Arlington, TX 76011-4511

Callaway Plant, Unit 1
Facility Operating License No. NPF-30
Docket No. STN 50-483

Dear Mr. Clayton

Subject: Submittal of Initial Operator License Training Outline Materials

Enclosed are the examination materials supporting the initial license exam scheduled for 03/04/2019 through 03/13/2019, at Callaway Station.

This submittal includes the Senior Reactor Operator and Reactor Operator Written Examination Outlines, Job Performance Measure Outline, and Integrated Plant Operation Scenario Guide Outline.

These examination materials have been developed in accordance with NUREG-1021, "Operator Licensing Examination Standards," Rev 11.

In accordance with NUREG-1021, Rev 11, Section ES-201, please ensure that these materials are withheld from public disclosure until after the examinations are complete.

Should you have any questions concerning the examination materials, please contact Phil Swan at 573-544-8102 or Mark Otten (573)544-8071

Respectfully,

Mark Covey

Mark Covey
Manager Operations - Support
Callaway Station

Enclosures: mailed to Kelly D. Clayton, Chief Examiner, NRC Region IV)

- Examination Outline Quality Checklist (Form ES-201-2)
- Examination Security Agreements (Form ES-201-3)
- 4 Scenario Outlines (Form ES-D-1s)
- Transient and Event Checklist (Form ES-301-5)
- RO and SRO Administrative Topics Outlines (Form ES-301-1)
- RO, SRO-I, SRO-U Control Room / In-Plant Systems Outline (Form ES-301-2)
- PWR Examination Outlines (ES-401-2 and ES-401-3)
- Record of Rejected K/As (Form ES-401-4)

Cc: (without enclosures)
Chief, NRC Operator Licensing Branch
NRC Senior Resident Inspector – Callaway Station

Bcc: (without enclosures)
Vice President Nuclear Operations – Callaway Station
Regulatory Affairs Manager - Callaway Station
Operations Director – Callaway Station
Training Director – Callaway Station

Facility: **Callaway** Date of Exam: **2019-03**

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

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

Note:
1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points, and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the outline. Systems or evolutions that do not apply at the facility should be deleted with justification. Operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible. Sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' IRs for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. If fuel-handling equipment is sampled in a category other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2. (Note 1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

G* Generic K/As

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

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

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/**SRO**)	Form ES-401-2	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / SF1	1						EK1.05 Knowledge of the operational implications of the following concepts as they apply to reactor trip: Decay Power as a function of time (41.8 / 41.10 / 45.3)	3.3	3
000008 (APE 8) Pressurizer Vapor Space Accident / SF3					**1**		**AA2.26 Ability to determine or interpret the following as they apply to PZR vapor space accident: Probable PZR steam space leage paths other than PORV or code safety (43.5 / 45.13)**	**3.4**	**SRO 76**
000009 (EPE 9) Small Break LOCA / SF3				1			EA1.15 Ability to operate and monitor as they apply to a SBLOCA: PORV and PORV block valve (41.7 / 45.5 / 45.6)	3.9	4
000011 (EPE 11) Large Break LOCA / SF3		1					EK2.02 Knowledge of the interrelations between the LBLOCA and the following: Pumps (41.7 / 45.7)	2.6*	2
000015 (APE 15) Reactor Coolant Pump Malfunctions / SF4					1		AA2.09 Ability to determine and interpret the following as they apply to RCP malfunctions: When to secure RCPs on high stator temperatures (41.10 / 43.5 / 45.13)	3.4	5
000022 (APE 22) Loss of Reactor Coolant Makeup / SF2			1				AK3.04 Knowledge of the reasons for the following responses as they apply to the loss of reactor coolant makeup: Isolating letdown (41.5 / 41.10 / 45.6 / 45.13)	3.2	6
000025 (APE 25) Loss of Residual Heat Removal System / SF4						1	G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (41.5 / 41.7 / 43.2)	3.2	7
000026 (APE 26) Loss of Component Cooling Water / SF8				1			AA1.02 Ability to operate and/or monitor the following as they apply to loss of CCW: Loads on the CCWS in the control room (41.7 / 45.5 / 45.6)	3.2	8
000027 (APE 27) Pressurizer Pressure Control System Malfunction / SF3					1		AA2.17 Ability to determine and interpret the following as they apply to the PZR Pressure control malfunction and the following: Allowable RCS temperature difference versus reactor power (41.10 / 43.5 / 45.13)	3.1	9
000029 (EPE 29) Anticipated Transient Without Scram / SF1						1	G2.1.8 Ability to coordinate personnel activities outside the control room (41.10 / 45.5 / 45.12-13)	3.4	10
000038 (EPE 38) Steam Generator Tube Rupture / SF3						**1**	**G2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc (41.7 / 43.5 / 45.12)**	**4.6**	**SRO 77**
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / SF4			1				AK3.03 Knowledge of the reason for the following responses as they apply Steam line rupture: Steam line non-return valves (41.5 / 41.10 / 45.6 / 45.13)	3.2*	11
000054 (APE 54; CE E06) Loss of Main Feedwater /4							Not sampled		
000055 (EPE 55) Station Blackout /SF 6	1						EK1.02 Knowledge of the operational implications of the following concepts as they apply to SBO: Natural circulation cooling (41.8 / 41.10 / 45.3)	4.1	12
000055 (EPE 55) Station Blackout / SF6					**1**		**EA2.04 Ability to determine or interpret the following as they apply to a Station blackout: Instruments and controls operable with only dc battery power available (43.5 / 45.13)**	**4.1**	**SRO 78**

000056 (APE 56) Loss of Offsite Power / SF6						**1**	**G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operation and safety limits (41.5 / 41.7 / 43.2)**	**4.2**	**SRO 79**
000057 (APE 57) Loss of Vital AC Instrument Bus / SF6					1		AA2.19 Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus and the following: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus (41.10 / 43.5 / 45.13)	4.0	13
000058 (APE 58) Loss of DC Power / SF6	1						AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation (41.8 / 41.10 / 45.3)	2.8	14
000062 (APE 62) Loss of Nuclear Service Water / SF4				1			AA1.01 Ability to operate and/or monitor the following as they apply to Loss of Nuclear Service Water (SWS): Nuclear service water temperature indications (41.7 / 43.5 / 45.6)	3.1	15
000065 (APE 65) Loss of Instrument Air / SF8					**1**		**AA2.08 Ability to determine and interpret the following as they apply to the loss of instrument air: Failure modes of air operated equipment (43.5 / 45.13)**	**3.3**	**SRO 80**
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / SF6		1					AK2.02 Knowledge of the interrelations between generator voltage and electric Grid Disturbances and the following: Breakers, relays (41.4 / 41.5 / 41.7 / 41.10 / 45.8)	3.1	16
(W E04) LOCA Outside Containment / SF3		1					EK2.2 Knowledge of the interrelations (between the LOCA outside containment) and the following: Facility*s heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility (41.7 / 45.7)	3.8	17
(W E11) Loss of Emergency Coolant Recirculation / SF4						1	G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (41.5 / 43.5 / 45.12 / 45.13)	4.4	18
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / SF4			1				EK3.1 Knowledge of the reasons for the following responses as they apply to Loss of Secondary heat Sink: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics (41.5 / 41.10 / 45.6 / 45.13)	3.4	19
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / SF4						**1**	**G2.4.6 Knowledge of EOP mitigation strategies (41.10 / 43.5 / 45.13)**	**4.7**	**SRO 81**
K/A Category Totals:	3	3	3	3	3/**3**	3/**3**	Group Point Total:		18/**6**

ES-401 PWR Examination Outline Form ES-401-2 Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1									
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						**1**	**G2.4.8 Knowledge of how Abnormal operating procedures are used in conjunction with EOPs. (41.10 / 43.5 / 45.13)**	**4.5**	**SRO 82**
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7						1	G2.1.30 Ability to locate and operate components, including local controls (41.7 / 45.7)	4.4	20
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8									
000037 (APE 37) Steam Generator Tube Leak / 3	1						AK1.01 Knowledge of the operational implications of the following concepts as they apply to Steam generator tube leak: Use of steam tables (41.8 / 41.10 / 45.3)	2.9*	21
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9					**1**		**AA2.02 Ability to determine and interpret the following as they apply to the accidental liquid radwaste release: The permit for liquid radioactive-waste release (43.5 / 45.13)**	**3.9**	**SRO 83**
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7		1					AK2.01 knowledge of the interrelations between the Area radiation Monitoring System Alarms and the following: Detectors at each ARM system location (41.8 / 41.10 / 45.3)	2.5*	22
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8					1		AA2.03 Ability to determine and interpret the following as they apply to control room evacuation: T-hot, T-cold, and in-core temperatures (43.5 / 45.13)	4.0	23
000069 (APE 69; W E14) Loss of Containment Integrity / 5						**1**	**G2.4.41 Knowledge of the emergency action level thresholds and classifications (41.10 / 43.5 / 45.11)**	**4.6**	**SRO 84**
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4				1			EA1.29 Ability to operate and monitor the following as they apply to Inadequate Core Cooling: Quench tank temperature, pressure, and level instrumentation (41.7 / 45.5 / 45.6)	3.4	24
000076 (APE 76) High Reactor Coolant Activity / 9									
000078 (APE 78*) RCS Leak / 3									

(W E01 & E02) Rediagnosis & SI Termination / 3			1				EK3.2 Knowledge of the reasons for the following responses as they apply to SI termination: Normal, abnormal and emergency operating procedures associated with (SI Termination) (41.5 / 41.10 / 45.6 / 45.13)	3.0	25
(W E13) Steam Generator Overpressure / 4				1			EA1.3 Ability to operate and/or monitor the following as they apply to Steam Generator Overpressure: Desired operating results during abnormal and emergency conditions (41.7 / 45.5 / 45.6)	3.1	26
(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/4			1				EK3.1 Knowledge of the reasons for the following responses as they apply to natural Circulation with Steam Void in Vessel with/without RVLIS: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics (41.5 / 41.10 / 45.6 / 45.13)	3.4	27
(BW E13 & E14) EOP Rules and Enclosures						1	G2.2.37 Ability to determine operability and/or availability of safety related equipment (41.7 / 43.5 / 45.12)	3.6	28
(CE A11; W E08) RCS Overcooling—Pressurized Thermal Shock / 4**					**1**		**EA2.1 Ability to determine and interpret the following as they apply to the (Pressurized thermal Shock): Facility conditions and selection of appropriate procedures during abnormal and emergency conditions (43.5 / 45.13)**	**4.2**	**SRO 85**
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals:	1	1	2	2	1/**2**	2/**2**	Group Point Total:		9/**4**

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump							1					A1.02 Ability to predict and monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCP controls including: RCP pump and motor bearing temperatures (41.5 / 45.5)	2.9	29
											1	G2.4.6 Knowledge of EOP mitigation strategies (41.10 / 43.5 / 45.13)	3.7	30
004 (SF1; SF2 CVCS) Chemical and Volume Control						1						K6.37 Knowledge of the effect of a loss or malfunction on the following CVCS components: Boron loading of demineralizer resin (41.7 / 45.7)	2.9	31
005 (SF4P RHR) Residual Heat Removal		1										K2.03 Knowledge of bus power supplies to the following: RCS pressure boundary motor-operated valves (41.7)	2.7*	1
005 (SF4P RHR) Residual Heat Removal								**1**				**A2.02 Ability to a) predict the impacts of the following malfunctions or operations on the RHRS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown (41.5 / 43.5 / 45.3 / 45.13)**	**3.7**	**SRO 86**
006 (SF2; SF3 ECCS) Emergency Core Cooling					1							K5.07 Knowledge of the operational implications of the following concepts as they apply to ECCS: Expected temperature levels in various locations of the RCS due to various plant conditions (41.5 / 45.7)	2.7	32
											1	G2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies (41.10 / 43.5 / 45.13)	3.8	33
007 (SF5 PRTS) Pressurizer Relief/Quench Tank									1			A3.01 Ability to monitor automatic operation of the PRTS, including: components which discharge to the PRT (41.7 / 45.5)	2.7*	34
008 (SF8 CCW) Component Cooling Water				1								K4.01 Knowledge of the design feature(s) and/or interlocks which provide for the following: Automatic start of standby pump (41.7)	3.1	35
010 (SF3 PZR PCS) Pressurizer Pressure Control								1				A2.02 Ability to a) predict the impacts of the following malfunctions or operations on the PZR PCS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures (41.5 / 43.5 / 45.3 / 45.13)	3.9	36
010 (SF3 PZR PCS) Pressurizer Pressure Control											**1**	**G2.2.42 Ability to recognize system parameters that are entry conditions for Technical Specifications (41.7 / 41.10 / 43.2 / 43.3 / 45.3)**	**4.6**	**SRO 87**

012 (SF7 RPS) Reactor Protection	1								1			K1.05 Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following: ESFAS (41.2-41.9 / 45.7 -45.8) A3.03 Ability to monitor automatic operation of the RPS, including the following: power supply (41.7 / 45.5)	3.8* 3.4	37 38
012 (SF7 RPS) Reactor Protection								**1**				**A2.02 Ability to a) predict the impacts of the following malfunctions or operations on the RPS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument power (41.5 / 43.5 / 45.3 / 45.5)**	**3.9**	**SRO 88**
013 (SF2 ESFAS) Engineered Safety Features Actuation										1		A4.02 Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels (41.7 / 45.5 to 45.8)	4.3	39
013 (SF2 ESFAS) Engineered Safety Features Actuation											**1**	**G2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations (41.10, 43.2 / 45.13)**	**4.2**	**SRO 89**
022 (SF5 CCS) Containment Cooling				1								K4.03 Knowledge of the design feature(s) and/or interlocks which provide for the following: Automatic containment isolation	3.6	40
022 (SF5 CCS) Containment Cooling											**1**	**G2.2.22 Knowledge of limiting conditions for operations and safety limits (41.5 / 43.2 / 45.2)**	**4.7**	**SRO 90**
025 (SF5 ICE) Ice Condenser												Not applicable for this design		
026 (SF5 CSS) Containment Spray	1	1										K1.01 Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following: ECCS K2.01 Knowledge of bus power supplies to the following: Containment spray pumps	4.2 3.4	41 42
039 (SF4S MSS) Main and Reheat Steam					1							K5.01 Knowledge of the operational implications of the following concepts as they apply to MRSS: Definition and causes of steam/water hammer (41.5 / 45.7)	2.9	43
059 (SF4S MFW) Main Feedwater			1									K3.03 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: S/Gs (41.7 / 45.6)	3.5	44
061 (SF4S AFW) Auxiliary/Emergency Feedwater								1				A2.04 Ability to a) predict the impacts of the following malfunctions or operations on the AFW system, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: pump failure or improper operation	3.4	45
062 (SF6 ED AC) AC Electrical Distribution										1		A4.03 Ability to manually operate and/or monitor in the control room: Synchroscope, including an understanding of running and incoming voltages	2.8	46

063 (SF6 ED DC) DC Electrical Distribution		1					1					K2.01 Knowledge of bus power supplies to the following: Major DC loads A1.01 Ability to predict and monitor changes in parameters (to prevent exceeding design limits) associated with operating the DC Electrical Distribution system including: Battery capacity as it is affected by discharge rate	2.9 2.5	47 48
064 (SF6 EDG) Emergency Diesel Generator			1			1						K6.07 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers K3.02 Knowledge of the effect that a loss or malfunction of the EDG will have on the following: ESFAS controlled or actuated systems	2.7 4.2	49 50
073 (SF7 PRM) Process Radiation Monitoring											1	G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation	4.3	51
076 (SF4S SW) Service Water			1									K3.07 Knowledge of the effect that a loss or malfunction of the SW system will have on the following: ESF loads	3.7	52
078 (SF8 IAS) Instrument Air				1								K4.01 Knowledge of the design feature(s) and/or interlocks which provide for the following: Manual/automatic transfers of control	2.7	53
103 (SF5 CNT) Containment								1		1		A2 04 Ability to a) predict the impacts of the following malfunctions or operations on the Containment, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Containment evacuation (including recognition of the alarm) A4 04 Ability to manually operate and/or monitor in the control room: Phase A and phase B resets	3.5 3.5	54 55
053 (SF1; SF4P ICS*) Integrated Control												N/A for this plant design		
K/A Category Point Totals:	2	3	3	3	2	2	2	3/2	2	3	3/3	Group Point Total:		28/5

ES-401												PWR Examination Outline Plant Systems—Tier 2/Group 2 (RO/SRO)		Form ES-401-2
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive								1				A2.03 Ability to a) predict the impacts of the following malfunctions or operations on the CRDS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect of stuck rod or Misaligned rod	3.5	56
002 (SF2; SF4P RCS) Reactor Coolant										1		A4.07 Ability to manually operate and/or monitor in the control room: Flow path linking the RWST through the RHR system to the RCS hot legs for gravity refilling of the refueling cavity	2.9	57
011 (SF2 PZR LCS) Pressurizer Level Control	1											K1.01 Knowledge of the physical connections and/or cause-effect relationships between the PZR LCS and the following systems CVCS	3.6	58
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation								**1**				**A2.02 Ability to a) predict the impacts of the following malfunctions or operations on the NIS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty or erratic operation of detectors or compensating components (41.5 / 43.5 / 45.3 / 45.5)**	**3.5***	**SRO 91**
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor							1					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including: Core exit temperature	3.7	59
027 (SF5 CIRS) Containment Iodine Removal					1							K5.01 Knowledge of the operational implications of the following concepts as they apply to the CIRS: Purpose of charcoal filters	3.1	60
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling			1									K3.03 Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Spent fuel temperature	3.0	61
034 (SF8 FHS) Fuel-Handling Equipment								**1**				**A2.01 Ability to a) predict the impacts of the following malfunctions or operations on the FHE, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped fuel element (41.5 / 43.5 / 43.7 / 45.3 / 45.13)**	**4.4**	**SRO 92**
035 (SF 4P SG) Steam Generator									1			A3.01 Ability to monitor automatic operation of the S/G including: S/G water	4.0	62
041 (SF4S SDS) Steam Dump/Turbine Bypass Control						1						K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS	2.7	63
045 (SF 4S MTG) Main Turbine Generator														

055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate											1	G2.1.20 Ability to interpret and execute procedure steps	4.6	64
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water		1										K2.03 Knowledge of bus power supplies to the following: Emergency/essential SWS pumps	2.6	65
079 (SF8 SAS**) Station Air														
086 Fire Protection														
050 (SF 9 CRV*) Control Room Ventilation											**1**	**G2.2.40 Ability to apply Technical Specifications for a system (41.10 / 43.2 / 43.5 / 45.3)**	**4.7**	**SRO 93**
K/A Category Point Totals:	1	1	1	0	1	1	1	1/**2**	1	1	1/**1**	Group Point Total:		10/**3**

Facility: **Callaway** Date of Exam: **2019-03**

Category	K/A #	Topic	RO IR	RO #	SRO-only IR	SRO-only #
1. Conduct of Operations	2.1.1	Knowledge of Conduct of operations requirements (41.10 / 45.13)	3.8	66		
	2.1.20	Ability to interpret and execute procedure steps (41.10 / 43.5 / 45.12)	4.6	67		
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management (41.1 / 43.6 / 45.6)	4.3	68		
	2.1.5	**Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc (41.10 / 43.5 / 45.12)**			**3.9**	**94**
	2.1.40	**Knowledge of refueling administrative requirements (41.10 / 43.5 / 43.7 / 45.13)**			**3.9**	**95**
	Subtotal			3		**2**
2. Equipment Control	2.2.6	Knowledge of the process for making changes to procedures (41.10 / 43.3 / 45.13)	3.0	69		
	2.2.13	Knowledge of tagging and clearance procedures (41.10 / 45.13)	4.1	70		
	2.2.12	**Knowledge of surveillance procedures (41.10 / 43.5 / 43.7 / 45.13)**			**4.1**	**96**
	2.2.19	**Knowledge of maintenance work order requirements (41.10 / 43.5 / 45.13)**			**3.4**	**97**
	Subtotal			2		**2**
3. Radiation Control	2.3.5	Ability to use rad monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (41.11 / 41.12 / 43.4 / 45.9)	2.9	71		
	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions (41.12 / 45.10)	3.5	72		
	2.3.11	**Ability to control radiation releases (41.11 / 43.4 / 45.10)**			**4.3**	**98**
	Subtotal			2		**1**
4. Emergency Procedures/Plan	2.4.3	Ability to identify post-accident instrumentation (41.6 / 45.4)	3.7	73		
	2.4.13	Knowledge of crew roles and responsibilities during EOP usage (41.10 / 45.12)	4.0	74		
	2.4.27	Knowledge of “fire in the plant” procedures (41.10 / 43.5 / 45.13)	3.4	75		
	2.4.25	**Knowledge of fire protection procedures (41.10 / 43.5 / 45.13)**			**3.7**	**99**
	2.4.22	**Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (41.7 / 43.5 / 45.12)**			**4.4**	**100**
	Subtotal			3		**2**
Tier 3 Point Total				10		**7**

Tier / Group	Randomly Selected K/A	Reason for Rejection

Facility: Callaway	Date of Examination: 3/4/19
Examination Level: RO ☒ SRO ☐	Operating Test Number: 2019-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations RO Admin 1	R,M	2.1.41 (2.8) Knowledge of the refueling process. JPM: Calculate Maximum RHR pump run time while filling the Refueling pool.
Conduct of Operations RO Admin 2	R,M	2.1.37 (4.3) Knowledge of procedures, guidelines, or limitations ssociated with reactivity management. JPM: Manual Makeup Setting Calculations
Equipment Control RO Admin 3	R,D	2.2.13 (4.1) Knowledge of tagging and clearance procedures JPM: Determine WPA/Tagout requirements for the 'A' CLCW pump (PEB01A)
Radiation Control RO Admin 4	R,D,P[1]	2.3.7 (3.5) Ability to comply with radiation work permit requirements during normal or abnormal conditions. JPM: Determine entry requirements for HRA in the RCA.
Emergency Plan		

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)

Note 1. The JPMs from the 2016 exam were randomly selected by placing 4 slips of paper labeled “A” through “D” in a container. No JPMs from the 2017

NRC exam were available for random selection as those JPMs will be used as a part of 2019 Audit Exam.

RO Administrative JPMs:

RO Admin #1 This is a Modified JPM. The Bank JPM, id # URO-ADM-01-A006J, has not been used on ILT NRC Exam since at least 2013. The JPM was modified in that the RHR pump flowrate and initial RWST volume were changed. This JPM requires the RO candidate to lookup 2 RWST volume values from references and then calculate a maximum RHR pump run time without exceeding a FSAR limit.

RO Admin #2 This is a Modified JPM. The Bank JPM, id #URO-ADM-02_A011J has not been used on ILT NRC Exam since at least 2013. The JPM was modified in that the RCS and BAST concentrations are different, the makeup flow rate is different, and the desired VCT level change is different. This JPM requires the RO candidate to calculate 4 controller settings based on the procedure and figure 7-2 of the curve book.

RO Admin #3 This is a Bank JPM. The Bank JPM, id # URO-ADM-01-A003J, has not been used on ILT NRC Exam since at least 2013. This JPM requires the RO candidate to determine WPA/Tagout requirements for the 'A' CLCW pump (PEB01A). The candidate will determine that Holdoff tags are required on the MCC Breaker (off/open), Suction Valve (closed), and Discharge Valve (closed) and that the tagging sequence is not critical.

RO Admin #4 This is a Bank JPM last used on the 2016 Exam. This JPM requires the RO candidate to review given conditions and determine dose received for a task, required authorization for that dose, and posting requirements for the area where the task will be performed; in accordance with APA-ZZ-01004, Radiological Work standards, and HDP-ZZ-01500, Radiological Postings.

Facility: Callaway	Date of Examination: 3/4/19
Examination Level: RO ☐ SRO ☒	Operating Test Number: 2019-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations A1	R,D,P[1]	2.1.37 (4.6) Knowledge of procedures, guidelines, or limitations associated with reactivity management. JPM: Review a QPTR Calculation
Conduct of Operations A2	R, N	2.1.18 (3.8) Ability to make accurate, clear, and concise logs, records, status boards, and reports. JPM: Determine Reportability
Equipment Control A3	R, N	2.2.40 (4.7) Ability to apply Technical Specifications for a system. JPM: Determine Technical Specifications for CRACS and CREVS
Radiation Control A4	R, N	2.3.6 (3.6) Ability to approve release permits. JPM: Review CA0855 for accuracy and determine ODCM LCO limits will be exceeded
Emergency Plan A5	R, D	2.4.41 (4.6) Knowledge of the Emergency Action Level (EAL) thresholds and classification JPM: Classify Event and Complete Notification Form – Sentry Not Available

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)

Note 1. The JPMs from the 2016 exam were randomly selected by placing 5 slips of paper labeled “A” through “E” in a container. No JPMs from the 2017 NRC exam were available for random selection as those JPMs will be used as a part of 2019 Audit Exam.

SRO Administrative JPMs:

SRO Admin #1 This is a BANK JPM. This JPM was used on the 2016 Exam. The SRO candidate will be required to review a QPTR calculation and determine that an error occurred in the calculation and determine the QTPR is not within the limits of TS 3.2.4 and that required actions A.1, A.2, A.3, A.4, A.5 and A.6 must be performed.

SRO Admin #2 This is a NEW JPM. The SRO candidate will be required to determine the reportability for a deviation and implementation of 10CFR50.54 X&Y during a severe weather event. A 1 hour report is required to the NRC Operations Center.

SRO Admin #3 This is a NEW JPM. The SRO candidate will be required to determine that both the Train A CRACS and CREVS systems are inoperable (due to a ductwork failure) for reasons other than the CRE or CBE boundary and declare T.S. LCO 3.7.10 and 3.7.11 not met with a 7 day and 30 day completion time respectively.

SRO Admin #4 This is a NEW JPM. The SRO candidate will be required to find multiple errors on CA0855, Liquid / Gaseous Release Worksheet, concerning a 'B' Waste Gas Decay Tank Batch release. Additionally, the SRO candidate will determine that an ODCM Gaseous Effluent LCO will not be met if the Batch release is completed.

SRO Admin #5 This is a BANK JPM. This JPM has not been used on a previous ILT NRC Exam. This JPM is Time Critical and the candidate will have 15 minutes to classify an event based on the conditions given and then an additional 15 minutes (from the time of declaration) to complete EIP-ZZ-00102 Attachment 4, Notification Form – Sentry NOT Available. Sentry not available is a change from the original bank JPM and will allow the JPM to be given in a classroom setting

Facility: Callaway	Date of Examination: 3/4/19
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test Number: 2019-1

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

System/JPM Title	Type Code*	Safety Function
S1. 001 Rod Control / Low power repositions control banks with a shutdown bank rod drop / Reactor shutdown required.	M, S, L, A	1
S2. 004 CVCS (BG) / Swap From the NCP to 'B' CCP	D, S, A, P[1]	2
S3. 006 Emergency Core Cooling System / Transfer to Hot Leg Recirc per ES-1.4	D, S, EN	3
S4. 003 Reactor Coolant Pump System / Respond to a loss of cooling to the RCPs	D, S, L, A	4P
S5. 022 Containment Cooling System / Start the 'A' Containment Cooler	D, S	5
S6. 015 Nuclear Instrument System / Perform a PR NI Gain Adjustment	M, S	7
S7. 029 Containment Purge System / Place CTMT Mini purge in Service	N, S	8
S8. 062 AC Electrical Distribution / Cross Connect Safety Related Load Centers	D, S,	6

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

System/JPM Title	Type Code*	Safety Function
P1. 035 Main and Reheat Steam System (AB) / Isolate a Failed Open Atmospheric Steam Dump	A, D, E, R, P[1]	4S
P2. 064 Emergency Diesel Generators / Locally Start EDGs per EOP Addendum 21	A, D, E	6
P3. 086 Electric Fire Pump / Electric Fire Pump (PKC1001A) Test per OSP-KC-00001	N	8

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

* Type Codes	Criteria for R /SRO-I/SRO-U

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

Note 1. The JPMs from the 2016 exam were randomly selected by placing 11 slips of paper labeled “A” through “K” in a container. No JPMs from the 2017 NRC exam were available for random selection as those JPMs will be used as a part of 2019 Audit Exam.

Simulator JPMs

S1 This is an ALTERNATE PATH, MODIFIED bank JPM. The reactor is critical at 10E-8 amps and the candidate is directed to raise power to 1% per OTG-ZZ-00003, Plant Startup Hot Zero Power to 30% Power – IPTE. While moving Control Bank 'D' rods, a shutdown bank rod, N-7, drops fully into the core. The candidate will enter OTO-SF-00001 and Attachment A directs the reactor to be shutdown if less than 5% power. The candidate must begin the reactor shutdown. The Bank JPM, id#URO-SSF-03-C120J, is not an alternate path nor does it include the dropped rod and requirement to shutdown the reactor.

S2 This is an ALTERNATE PATH, bank JPM id#BG-RO-S-004(A). This JPM was used on the 2016 Exam. The candidate will perform the actions of OTN-BG-00001, Addendum 1 to shift from the NCP to the B CCP. After the B CCP is started and during the transition from the NCP flow controller to the B CCP flow controller, the B CCP will Trip, requiring the candidate to restore charging flow. Upon completion of this JPM, the candidate will have restored charging flow to normal.

S3 This is a bank JPM id#EOP-RO-S-007. This JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to transfer to Hot leg Recirculation per ES-1.4. The JPM will be complete when the candidate has successfully completed ES-1.4 step #2.

S4 This is an ALTERNATE PATH, bank JPM id#URO-SBB-04-C166J(A)(TC). This JPM is TIME CRITICAL and has not been used on ILT NRC Exam since at least 2013. The candidate will be required to respond to a loss cooling water flow and restore flow to/from RCPs within 10 minutes per OTO-BB-00002, RCP Off Normal.

S5 This is a bank JPM id#GN-RO-S-002. This JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to place the 'A' Containment Cooler running in FAST Speed per OTN-GN-00001, Containment Cooling and CRDM Cooling.

S6 This is a MODIFIED bank JPM. The bank JPM id #SE-RO-S-003(A). The JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to perform a PR NI gain adjustment per OSP-SE-00004 Attachment 1. This JPM was modified from an Alternate Path JPM by removing a malfunction which required additional actions to reset PR NI trips. This JPM is now a normal evolution/task (i.e not alternate path).

S7 This is a NEW JPM. The candidate will place CTMT Mini Purge in service per section 5.2 of OTN-GT-00001, Containment Purge System.

S8 This JPM is for the RO candidates ONLY. This is a Bank JPM id#URO-SNG-01-C083J. This JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to cross connected Safety Related Load Centers NG01 and NG03 with NG03 feeder breaker supplying both load centers and the NG01 feeder breaker opened.

In Plant JPMs

P1 This is an ALTERNATE PATH BANK JPM. This JPM was used on the 2016 Exam and the Bank ID is AB-NLO-P-001(A). The candidate will be assigned the task of locally closing Atmospheric Steam Dumps, AB PV-1 AND AB PV-4. Upon completion of this JPM, the candidate will have closed AB PV-1 and isolated AB PV-4. AB PV-1 was closed by isolating Air/N2 from the valve. AB PV-4 was isolated by closing the manual isolation valve, ABV0007.

P2 This is a BANK JPM. This JPM (or same task) was on the 2013 NRC exam. The candidate will be informed that the plant is in a Station Blackout and direct to locally start the "A" EDG, NE01, per EOP Addendum 21, Local Start of Emergency DGs. The JPM will be complete when the 'A' EDG, Emergency Diesel Generator NE01 is running with Master Transfer Switch KJ-HS-9 in AUTO.

P3 This is a NEW JPM. The candidate will perform a test run of the Electric Fire Pump, PKC1001A, per OSP-KC-00001 section 6.2. The JPM will be complete when the pump has been test run for at least 15 minutes and the appropriate data recorded on OSP-KC-00001 Attachment 2.

Facility: Callaway	Date of Examination: 3/4/19
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test Number: 2019-1

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

System/JPM Title	Type Code*	Safety Function
S1. 001 Rod Control / Low power repositions control banks with a shutdown bank rod drop / Reactor shutdown required.	M, S, L, A	1
S2. 004 CVCS (BG) / Swap From the NCP to 'B' CCP	D, S, A, P[1]	2
S3. 006 Emergency Core Cooling System / Transfer to Hot Leg Recirc per ES-1.4	D, S, EN	3
S4. 003 Reactor Coolant Pump System / Respond to a loss of cooling to the RCPs	D, S, L, A	4P
S5. 022 Containment Cooling System / Start the 'A' Containment Cooler	D, S	5
S6. 015 Nuclear Instrument System / Perform a PR NI Gain Adjustment	M, S	7
S7. 029 Containment Purge System / Place CTMT Mini purge in Service	N, S	8

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

System/JPM Title	Type Code*	Safety Function
P1. 035 Main and Reheat Steam System (AB) / Isolate a Failed Open Atmospheric Steam Dump	A, D, E, R, P[1]	4S
P2. 064 Emergency Diesel Generators / Locally Start EDGs per EOP Addendum 21	A, D, E	6
P3. 086 Electric Fire Pump / Electric Fire Pump (PKC1001A) Test per OSP-KC-00001	N	8

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

* Type Codes	Criteria for R /SRO-I/SRO-U

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

Note 1. The JPMs from the 2016 exam were randomly selected by placing 11 slips of paper labeled “A” through “K” in a container. No JPMs from the 2017 NRC exam were available for random selection as those JPMs will be used as a part of 2019 Audit Exam.

Simulator JPMs

S1 This is an ALTERNATE PATH, MODIFIED bank JPM. The reactor is critical at 10E-8 amps and the candidate is directed to raise power to 1% per OTG-ZZ-00003, Plant Startup Hot Zero Power to 30% Power – IPTE. While moving Control Bank 'D' rods, a shutdown bank rod, N-7, drops fully into the core. The candidate will enter OTO-SF-00001 and Attachment A directs the reactor to be shutdown if less than 5% power. The candidate must begin the reactor shutdown. The Bank JPM, id#URO-SSF-03-C120J, is not an alternate path nor does it include the dropped rod and requirement to shutdown the reactor.

S2 This is an ALTERNATE PATH, bank JPM id#BG-RO-S-004(A). This JPM was used on the 2016 Exam. The candidate will perform the actions of OTN-BG-00001, Addendum 1 to shift from the NCP to the B CCP. After the B CCP is started and during the transition from the NCP flow controller to the B CCP flow controller, the B CCP will Trip, requiring the candidate to restore charging flow. Upon completion of this JPM, the candidate will have restored charging flow to normal.

S3 This is a bank JPM id#EOP-RO-S-007. This JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to transfer to Hot leg Recirculation per ES-1.4. The JPM will be complete when the candidate has successfully completed ES-1.4 step #2.

S4 This is an ALTERNATE PATH, bank JPM id#URO-SBB-04-C166J(A)(TC). This JPM is TIME CRITICAL and has not been used on ILT NRC Exam since at least 2013. The candidate will be required to respond to a loss cooling water flow and restore flow to/from RCPs within 10 minutes per OTO-BB-00002, RCP Off Normal.

S5 This is a bank JPM id#GN-RO-S-002. This JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to place the 'A' Containment Cooler running in FAST Speed per OTN-GN-00001, Containment Cooling and CRDM Cooling.

S6 This is a MODIFIED bank JPM. The bank JPM id #SE-RO-S-003(A). The JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to perform a PR NI gain adjustment per OSP-SE-00004 Attachment 1. This JPM was modified from an Alternate Path JPM by removing a malfunction which required additional actions to reset PR NI trips. This JPM is now a normal evolution/task (i.e not alternate path).

S7 This is a NEW JPM. The candidate will place CTMT Mini Purge in service per section 5.2 of OTN-GT-00001, Containment Purge System.

In Plant JPMs

P1 This is an ALTERNATE PATH BANK JPM. This JPM was used on the 2016 Exam and the Bank ID is AB-NLO-P-001(A). The candidate will be assigned the task of locally closing Atmospheric Steam Dumps, AB PV-1 AND AB PV-4. Upon completion of this JPM, the candidate will have closed AB PV-1 and isolated AB PV-4. AB PV-1 was closed by isolating Air/N2 from the valve. AB PV-4 was isolated by closing the manual isolation valve, ABV0007.

P2 This is a BANK JPM. This JPM (or same task) was on the 2013 NRC exam. The candidate will be informed that the plant is in a Station Blackout and direct to locally start the "A" EDG, NE01, per EOP Addendum 21, Local Start of Emergency DGs. The JPM will be complete when the 'A' EDG, Emergency Diesel Generator NE01 is running with Master Transfer Switch KJ-HS-9 in AUTO.

P3 This is a NEW JPM. The candidate will perform a test run of the Electric Fire Pump, PKC1001A, per OSP-KC-00001 section 6.2. The JPM will be complete when the pump has been test run for at least 15 minutes and the appropriate data recorded on OSP-KC-00001 Attachment 2.

Facility: Callaway		Date of Examination: 3/4/19	
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒		Operating Test Number: 2019-1	

Control Room Systems:* 8 for RO, 7 for SRO-I, and 2 or 3 for SRO-U		
System/JPM Title	Type Code*	Safety Function
S1. 001 Rod Control / Low power repositions control banks with a shutdown bank rod drop / Reactor shutdown required.	M, S, L, A	1
S3. 006 Emergency Core Cooling System / Transfer to Hot Leg Recirc per ES-1.4	D, S, EN	3
In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U		
P1. 035 Main and Reheat Steam System (AB) / Isolate a Failed Open Atmospheric Steam Dump	A, D, E, R, P[1]	4S
P2. 064 Emergency Diesel Generators / Locally Start EDGs per EOP Addendum 21	A, D, E	6
P3. 086 Electric Fire Pump / Electric Fire Pump (PKC1001A) Test per OSP-KC-00001	N	8

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

* Type Codes	Criteria for R /SRO-I/SRO-U

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

Note 1. The JPMs from the 2016 exam were randomly selected by placing 11 slips of paper labeled “A” through “K” in a container. No JPMs from the 2017 NRC exam were available for random selection as those JPMs will be used as a part of 2019 Audit Exam.

Simulator JPMs

S1 This is an ALTERNATE PATH, MODIFIED bank JPM. The reactor is critical at 10E-8 amps and the candidate is directed to raise power to 1% per OTG-ZZ-00003, Plant Startup Hot Zero Power to 30% Power – IPTE. While moving Control Bank 'D' rods, a shutdown bank rod, N-7, drops fully into the core. The candidate will enter OTO-SF-00001 and Attachment A directs the reactor to be shutdown if less than 5% power. The candidate must begin the reactor shutdown. The Bank JPM, id#URO-SSF-03-C120J, is not an alternate path nor does it include the dropped rod and requirement to shutdown the reactor.

S3 This is a bank JPM id#EOP-RO-S-007. This JPM has not been used on ILT NRC Exam since at least 2013. The candidate will be required to transfer to Hot leg Recirculation per ES-1.4. The JPM will be complete when the candidate has successfully completed ES-1.4 step #2.

In Plant JPMs

P1 This is an ALTERNATE PATH BANK JPM. This JPM was used on the 2016 Exam and the Bank ID is AB-NLO-P-001(A). The candidate will be assigned the task of locally closing Atmospheric Steam Dumps, AB PV-1 AND AB PV-4. Upon completion of this JPM, the candidate will have closed AB PV-1 and isolated AB PV-4. AB PV-1 was closed by isolating Air/N2 from the valve. AB PV-4 was isolated by closing the manual isolation valve, ABV0007.

P2 This is a BANK JPM. This JPM (or same task) was on the 2013 NRC exam. The candidate will be informed that the plant is in a Station Blackout and direct to locally start the "A" EDG, NE01, per EOP Addendum 21, Local Start of Emergency DGs. The JPM will be complete when the 'A' EDG, Emergency Diesel Generator NE01 is running with Master Transfer Switch KJ-HS-9 in AUTO.

P3 This is a NEW JPM. The candidate will perform a test run of the Electric Fire Pump, PKC1001A, per OSP-KC-00001 section 6.2. The JPM will be complete when the pump has been test run for at least 15 minutes and the appropriate data recorded on OSP-KC-00001 Attachment 2.

Facility: **Callaway** Date of Exam: **Week of 3/4/19** Operating Test No.: **2019-1**

APPLICANT	EVENT TYPE	Scenarios: **Team 1: S1, S2, S3 and Team 2: S4, S5, S6**												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		R	I	U
		S1 S4	S2 S5	S3 S6	S2 S5	S3 S6	S1 S4	S3 S6	S1 S4	S2 S5	#	#	#	#			
SRO-I S1 S4	RX	3							2		2	2		2	1	1	0
	NOR													0*	1	1	1
	I/C	1,2,4					2,4,7		1,3,4		1,3,4,6	3,4,	1,2,3,4,6	9*	4	4	2
	MAJ	5,6					5		5		5	5	5	4	2	2	1
	TS	1,2									1,2			2	0	2	2
SRO-I S2 S5	RX		3		1									2	1	1	0
	NOR													0*	1	1	1
	I/C		1,2,4		2,3,4,6,7					2,4,				10*	4	4	2
	MAJ		5,6		5					5				4	2	2	1
	TS				2,4									2	0	2	2
SRO-I S3 S6	RX					1		2						2	1	1	0
	NOR													0*	1	1	1
	I/C			3,4		3,4,6		1,3,4						8*	4	4	2
	MAJ			5,6		5		5						4	2	2	1
	TS							1,3						2	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.

All 4 scenarios and their attributes are listed as they are labeled for ease of comparison. The Total Columns is summed for Scenario #1 through 3 while Scenario #4 and its attributes are shown as the spare. This in no way means that Callaway Energy Center desires Scenario #4 as the spare; specifically Callaway Energy Center would like the Chief Examiner to determine which scenario to designate as the spare based on the ES-D1's provided and their above attributes. Callaway Energy Center will then update this ES-301-5 per NRC direction.

Facility: **Callaway** Date of Exam: **Week of 3/4/19** Operating Test No.: **2019-1**

APPLICANT	EVENT TYPE	Scenarios: **Team 3 S7, S8, U1, R1**												TOTAL	MINIMUM(*)		
		1			2			3			4						
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO U1	ATC S7	BOP S8	SRO S7	ATC3 S8	BOP R1	SRO S8	ATC R1	BOP S7	SRO #	ATC #	BOP #	#	R	I	U
SRO-I S7	RX		3		1						2	2		2	1	1	0
	NOR													0*	1	1	1
	I/C		1,2,4		2,3,4,6,7					2,4	1,3,4,6	3,4,	1,2,3,4,6	10*	4	4	2
	MAJ		5,6		5					5	5	5	5	4	2	2	1
	TS				2,4						1,2			2	0	2	2
SRO-I S8	RX					1		2						2	1	1	0
	NOR													0*	1	1	1
	I/C			3,4		3,4,6		1,3,4						8*	4	4	2
	MAJ			5,6		5		5						4	2	2	1
	TS							1,3						2	0	2	2
SRO-U U1	RX	3												1	1	1	0
	NOR													0*	1	1	1
	I/C	1,2,4												3*	4	4	2
	MAJ	5,6												2	2	2	1
	TS	1,2												2	0	2	2
RO R1	RX								2					1	1	1	0
	NOR													0*	1	1	1
	I/C						2,4,7,		1,3,4					6*	4	4	2
	MAJ						5		5					2	2	2	1
	TS													0	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.

All 4 scenarios and their attributes are listed as they are labeled for ease of comparison. The Total Columns is summed for Scenario #1 through 3 while Scenario #4 and its attributes are shown as the spare. This in no way means that Callaway Energy Center desires Scenario #4 as the spare; specifically Callaway Energy Center would like the Chief Examiner to determine which scenario to designate as the spare based on the ES-D1's provided and their above attributes. Callaway Energy Center will then update this ES-301-5 per NRC direction.

Facility: Callaway	Scenario No.:2 , Rev 0	Op-Test No.: 2019-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: Mode 3 with shutdown banks withdrawn

Turnover: No equipment out of service. The crew is directed to dilute the RCS to desired ECP boron concentration.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	SRO (R) RO (R)	Dilute RCS to ECP boron concentration OTN-BG-00001, Reactor Makeup Control and Boron Thermal Regeneration system
2	AB / ABPV0001A_1	SRO (C) BOP (C)	Atmospheric Steam Dump 'A' fails open with manual control OTO-AB-00001, Steam Dump Malfunction (Tech Spec 3.7.4)
3	BB / CRCPV2_1	SRO (C) RO (C)	“A” RCP High Vibration OTO-BB-00002, RCP Off Normal
4	BB / EBB01D	SRO (C) RO (C) BOP (C)	'D' SG Tube Leak OTO-BB-00001, Steam Generator Tube Leak (Tech Spec 3.4.13)
5	BB / EBB01D	SRO (M) RO (M) BOP (M)	'D' SG Tube Rupture / RX Trip E-0, Reactor Trip or Safety Injection and then E-3 Steam Generator Tube Rupture
6	MD / MDCB1 MD / MDLC1 MD / MDMT7 MD / MDMT8 MD / MMDESFB	SRO (C) RO (C)	Loss of Offsite Power – post trip
7	SA / SAS9XX_4 SA / SAS9XX_8	SRO (C) BOP (C)	Failure of 'D' MSIV to close

* (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)	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)	0
7. Critical tasks (2-3)	3

The Plant is stable in Mode 3 with shutdown banks withdrawn. The crew has been directed to dilute the RCS to the desired Estimated Critical Position boron concentration per step 5.3.10 a of OTG-ZZ-00001, Plant Heatup Cold Shutdown to Hot Standby, using OTN-BG-00002, Reactor Makeup Control and Boron Thermal Regeneration System. The crew will begin a RCS dilution that will last 25 minutes.

Once the dilution has been initiated, the 'A' 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. This failure will result in Technical Specification 3.7.4 not being met.

After the ASD is addressed including its Technical Specifications, a mechanical failure causes ‘A’ RCP vibrations to rise rapidly above the immediate trip setpoint. This will drive the crew to enter OTO-BB-00002, RCP Off Normal. The crew will recognize the need to immediately trip the 'A' RCP.

‘D’ SG will develop a tube leak and the crew will respond IAW OTO-BB-00001, SG Tube Leak. This failure will result in Technical Specification 3.4.13 not being met.

After Technical Specification have been addressed, the tube leak will increase into a rupture. The crew will trip the reactor due to the increased leak and respond IAW E-0, Reactor Trip or Safety Injection. The crew will transition to E-3, Steam Generator Tube Rupture and isolate the ruptured ‘D’ SG.

During the performance of E-0, offsite power will be lost. Additionally, during the performance of E-3, 'D' MSIV will fail to close with its HS. These post trip failures will cause the crew to depressurize using the PZR PORVS instead of normal PZR Spray and fast close all other MSIVs and cooldown using intact SG ASDs vice the preferred path of the condenser.

This scenario is complete when Step#17 of E-3 is complete. The scenario may be terminated earlier at the option of the Lead Evaluator, provided all critical tasks have been completed.

Critical Tasks:

Critical Tasks	Isolate feedwater flow into and steam flow from the ruptured SG before a transition to ECA-3.1 occurs.	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 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	5	5
Safety significance	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.	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	All of the following: • Indication and/or annunciation of SGTR in one SG o Increasing SG water level o Radiation • Indication and/or annunciation of reactor trip • Indication and/or annunciation of SI	Indication and/or annunciation of SGTR in one SG • Increasing SG water level • Radiation AND Indication and/or annunciation of reactor trip AND Indication and/or annunciation of SI AND Indication of ruptured SG pressure greater than minimum required pressure
Performance indicator	Manipulation of controls as required to isolate the ruptured SG • Adjust ruptured SG 'D' ASD controller setpoint to 1160 psig • Fast Close all remaining MSIVs and Bypass valves. (SG 'A', 'B', and 'C')	Manipulation of controls as required to establish and maintain RCS temperature 1. Steam dump valve position lamps and/or indicators indicate closed 2. SG PORV valve position lamps and/or indicators indicate closed
Performance feedback	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	Indication of steam flow rate greater than zero • Indication of RCS temperature decreasing OR • Indication of RCS temperature less than target temperature
Justification for the chosen performance limit	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.	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 18, Isolate the Ruptured SG.	CT 19, Control initial RCS cooldown

Critical Tasks	Depressurize RCS to achieve SI termination criteria of >9% [29%] pressurizer level and >30F [50F] RCS subcooling so that a transition to ECA-3.1 does not occur.	
EVENT	5	
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.	
Cueing	Indication and/or annunciation of SGTR in one SG AND Indication and/or annunciation of reactor trip and SI AND Indication that the RCS is cooled down to the target temperature	
Performance indicator	Manipulation of controls as required to depressurize the RCS 3. Valve position lamps show PRZR PORV open	
Performance feedback	Indication of RCS pressure decreasing Indication of PRZR level increasing	
Justification for the chosen performance limit	The intent is to depressurize to establish and maintain the criteria that allow the crew to terminate SI. 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.	
PWR Owners Group Appendix	CT 20, Depressurize RCS to E-3 SI termination criteria	

References
OTG-ZZ-00001, Plant Heatup Cold Shutdown to Hot Standby
OTN-BG-00002, Reactor Makeup Control and Boron Thermal Regeneration System
OTO-BB-00001, Steam Generator Tube Leak
OTO-BB-00002, RCP Off Normal
OTO-AD-00001, Steam Dump Malfunction
E-0, Reactor Trip or Safety Injection
E-3, Steam Generator Tube Rupture
Technical Specification 3.7.4 Atmospheric Steam Dump Valves
Technical Specification 3.4.13 RCS Operational Leakage
ODP-ZZ-00025, EOP/OTO User's Guide

PRA Systems, Events or Operator Actions

1. SG Tube rupture (2% Contribution to CDF)

Facility: Callaway Scenario No.:3 , Rev 0 Op-Test No.: 2019-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: 100%

Turnover: 'A' MDAFP is out of service.

Event No.	Malf. No.	Event Type*	Event Description
1	SF / SFB08_DR	SRO (C) RO (C)	Dropped rod OTO-SF-00001, Rod Control Malfunctions (Tech Spec 3.1.4)
2	AC / FCV0049MANTYP AC / FCV0049TASTEM	SRO (R) RO (R) BOP (C)	Main Turbine Control Valve #3 fails closed OTO-MA-00001, Turbine Load Reject
3	TVHM1705 RCCFUELFAIL	SRO (C) RO (C)	RCS High Activity OTO-BB-00005 / Place 120 gpm Letdown in-service (Tech Spec 3.4.16)
4	AE / AEPT0508A, B, C FC / FCXY0001A_2 FC / FCXY0003A_1	SRO (I) RO (I) BOP (I)	DFWCS failure – loss of MFW with a manual reactor trip required (failure of the automatic reactor trip), E-0 Reactor Trip or Safety Injection
5	AB / AB002_D	SRO (M) RO (M) BOP (M)	'D' SG Fault, E-2 Faulted SG Isolation with a failure of the automatic SL isolation

* (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)	1
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)	0
7. Critical tasks (2-3)	2

Scenario #3 Event Description
Callaway 2019-1 NRC ES-D-1, rev. 0

The Plant is stable at 100% with the 'A' MDAFP out of service.

After the reactivity brief is complete, a dropped rod occurs as indicated by DRPI and Control Rod Alarms. The crew will establish conditions for rod recovery per OTO-SF-00001, identify Technical Specifications and begin restoration. Technical Specification 3.1.4 is not met.

After Tech Specs for the dropped rod have been addressed, Control Valve #3 will fail closed and crew will take actions per OTO-MA-00001. The crew will stabilize the plant and adjust Tavg and turbine controls in response.

Once the plant is stable, a report from Chemistry indicates high activity in the RCS. The Crew will enter OTO-BB-00005, RCS High Activity and establish 120 gpm letdown flow IAW OTN-BG-00001 Addendum 04, Operation of CVCS Letdown. Technical Specification 3.4.16 is not met.

After the plant is stabilized, a failure of the DFWCS occurs resulting in both main feed pump speed lowering and therefore flow lowering to the point that a manual reactor trip is required (auto trip will not work). The crew will enter E-0, Reactor Trip or Safety Injection, and perform the immediate actions.

After E-0 immediate actions are completed, the 'D' SG develops a fault which can be seen by the crew as RCS pressure and temperature lower. A transition should be made to E-2, Faulted Steam Generator Isolation, at E-0 step#14. When performing E-2, the crew should determine that the automatic steamline isolation failed to occur and manually initiate it.

Once the faulted Steam Generator is isolated and, after the secondary side blows dry, SI termination criteria should be met. The crew will transition to ES-1.1, SI Termination from E-2.

The scenario is complete when the Boron Injection Header is isolated in ES-1.1, SI Termination.

Critical Tasks:

Critical Tasks	Manually trip the reactor before any SG level indicates less than 10% WR	Isolate the faulted 'D' SG before transition out of E-2 MSIV D Closed AFW Isolated From SG D
EVENT	4	5
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 isolate a faulted SG that can be isolated causes challenges to CSFs beyond those irreparably introduced by the postulated conditions. Failure to isolate a faulted SG can result in challenges to the following CSFs: • Integrity • Subcriticality • Containment (if the break is inside containment)
Cueing	Indication and/or annunciation that plant parameter(s) exist that should result in automatic reactor trip but reactor does not automatically trip • SG lev low low RX trip annunciator (85A)	Both of the following: • Steam pressure and flow rate indications that make it possible to identify 'D' SG as faulted AND • Valve position and flow rate indication that AFW continues to be delivered to the faulted 'D' SG
Performance indicator	Manipulation of control room reactor trip switches as required to trip the reactor • Reactor trip and bypass breakers indicate open	ISOLATE AFW flow to faulted SG(s): • CLOSE associated MD AFP Flow Control Valve(s): o AL HK-5A (SG D) • CLOSE associated TD AFP Flow Control Valve(s): o AL HK-6A (SG D) • CLOSE Steamline Low Point Drain valve from faulted SG(s): o AB HIS-10 (SG D) • FAST CLOSE all MSIVs and Bypass valves: o AB HS-79 o AB HS-80
Performance feedback	Indications of reactor trip • Control rods at bottom of core • Neutron flux decreasing	Crew will observe the following: • Any depressurization of intact SGs stops • AFW flow rate indication to faulted SG of zero
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."	"before transition out of E-2" 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-1, Manually trip the reactor	CT-17 Isolate faulted SG

References
OTO-SF-00001, Rod Control Malfunctions
OTO-MA-00001, Turbine Load Reject
OTO-BB-00005, RCS High Activity
OTN-BG-00001 Addendum 04, Operation of CVCS Letdown
E-0, Reactor Trip or Safety Injection
E-2, Faulted Steam Generator Isolation
ES-1.1, SI Termination
Technical Specification 3.1.4, Rod Group Alignment Limits
Technical Specification 3.4.16, RCS Specific Activity
ODP-ZZ-00025, EOP/OTO User's Guide

PRA Systems, Events or Operator Actions

1. Secondary Line Breaks (10% contribution to CDF)
2. Loss of MFW (1% contribution to CDF)

Facility: Callaway Scenario No.:4 , Rev 0 Op-Test No.: 2019-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: 50%

Turnover: 'A' MDAFP is out of service.

Event No.	Malf. No.	Event Type*	Event Description
1	AE / AELT0519	SRO (I) BOP (I)	SG Level Instrument Fails to 75% OTO-AE-00002, Steam Generator Water Level Control Instrument Malfunctions (Tech Specs 3.3.1 and 3.3.2)
2	SE / SEN0043	SRO (R) RO (R) BOP (I)	Power Range Channel N43 fails high. OTO-SE-00001, Nuclear Instrument Malfunction (Tech Spec 3.3.1)
3	NB / XNB02_4 EF / PEF01B	SRO (C) BOP (C) RO (C)	Loss of ESF transformer XNB02 causing a Loss of NB02/ EDG “B” starts, ESW Pump “B” trips. OTO-NB-00002, Loss of Power to NB02
4	MD / MDCB1 MD / MDLC1 MD / MT7 MD / MDMT8 MD / MDESFB #	SRO (C) BOP (C) RO (C)	Loss of offsite power / RX Trip / E-0, Reactor Trip or Safety Injection
5	NB / NB01_F	SRO (M) RO (M) BOP (M)	NB01 Bus Lockout / Loss of All AC Power / ECA-0.0 Loss of All AC Power / Restore Power with AEPS
6	AL / PAL02_3	SRO (C) BOP (C)	TDAFW fails to Auto Start

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

Note: Grid voltage swings accomplished by adjusting GRID_VOLTAGE (Grid Voltage Multiplier) under external parameters in browser

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)	2

The Plant is stable at 50% with the 'A' MDAFP out of service.

After the reactivity brief is complete, ‘A’ SG controlling level channel slowly fails to 75%. The crew will respond using OTO-AE-00002, Steam Generator Water Level Control Instrument Malfunctions, to control SG level. Technical Specification 3.3.1 and 3.3.2 are not met.

After Tech Specs are addressed, Power Range Nuclear Instrument Channel N43 fails high causing an automatic rod insertion. The crew should respond to the rod insertion by placing rods in manual. The crew will enter OTO-SE-00001, Nuclear Instrument Malfunction, to bypass channel N43 and restore control rods to desired position. Technical Specification 3.3.1 is not met.

After Tech Specs are addressed, a fault on ESF Transformer XNB02 occurs, resulting in a loss of power to Bus NB02. “B” EDG starts, but Essential Service Water Pump “B” trips 3 minutes following pump start, forcing the crew to trip the affected Diesel and enter OTO-NB-00002, Loss of Power to NB02.

5 minutes after the ESW Pump "B" trip, offsite power will begin to fluctuate and a Loss of offsite power will occur resulting Rx Trip. 2 minutes after the reactor trip, NB01 lockouts due to a bus fault and the crew will transition to ECA-0.0, Loss of All AC Power.

The crew will manually start the TDAFW and restore power to NB02 using EOP Addendum 39, Alternate Emergency Power Supply.

The scenario can be terminated after power has been restored to NB02.

Critical Tasks:

Critical Tasks	Establish greater than 270,000 lbm/hr AFW flow rate to the SGs prior to SG dryout occurring.	Energize NB02 AC Emergency Bus prior to RCP seal degradation
EVENT	6	5
Safety significance	Failure to establish minimum AFW flow in this scenario is a violation of the basic objective of ECA-0.0 and of the assumptions of the analyses upon which ECA-0.0 is based. Without AFW flow, the SGs could not support any significant plant cooldown. Thus, the crew would lose the ability to delay the adverse consequences of core uncovery.	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.
Cueing	Indication of ATWS (the reactor is not tripped and that a manual reactor trip is not effective) with no AFW flow indication present	Indication and/or annunciation that all ac emergency buses are de-energized • Bus energized lamps extinguished • Circuit Breaker Position • Bus Voltage • EDG status
Performance indicator	Manipulation of the: • TDAFW Steam Supply valve(s): o AB HIS-5A(SG B) o AB HIS-6A(SG C) • TDAFP Mechanical Trip/Throttle valve: o FC HIS-312A	Manipulation of controls as required to energize at least one ac emergency bus from the AEPS: • Using PBXY0001 Close AEPS FDR BKR PB0502 to NB0214 o PB0502 • Close NB02 AEPS Supply BKR NB0214 o NB HIS-68
Performance feedback	Crew will observe the following: • Greater than 270,000 lbm/hr AFW flow to the SGs.	Indication that NB02 is energized: • NB02 Bus energized light • NB02 bus voltage
Justification for the chosen performance limit	Without AFW flow, decay heat would open the SG safety valves and would rapidly deplete the SG inventory, leading to a loss of secondary heat sink, or SG dryout. Decay heat would then increase RCS temperature and pressure until the pressurizer PORVs open, imposing a larger LOCA than RCP seal leakage.	Failure to perform the critical task also results in needless degradation of any barrier to fission product release, specifically of the RCS barrier at the point of the RCP seals. Failure to perform the critical task means that RCS inventory lost through the RCP seals cannot be replaced. It also means that the RCP seals remain without cooling and gradually deteriorate. As the seals deteriorate the rate of RCS inventory loss increases.
PWR Owners Group Appendix	CT - 23, Establish AFW flow during SBO	CT - 24, Energize at least one ac emergency bus

References
OTO-AE-00002, Steam Generator Water Level Control Instrument Malfunctions
OTO-SE-00001, Nuclear Instrument Malfunction
OTO-NB-00002, Loss of Power to NB02.
EOP Addendum 39, Alternate Emergency Power Supply
E-0, Reactor Trip or Safety Injection
ECA-0.0, Loss of All AC Power
Technical Specification 3.3.1, Reactor Trip System Instrumentation
Technical Specification 3.3.2, EFSAS Instrumentation
ODP-ZZ-00025, EOP/OTO User's Guide

PRA Systems, Events or Operator Actions

1. SBO (Loss of All AC) is a 19% contribution to CDF

PRA Systems, Events or Operator Actions

2. Alternate Electric Power Supply (PA) is #6 of the Top #10 Callaway Risk Important Systems