Facility: **Wolf Creek** **Date of Exam**: **December 2019**

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	2				1	2			1	9			4
	Tier Totals	5	4	5				4	5			4	27			10
2. Plant Systems	1	3	2	3	3	2	2	3	3	2	2	3	28			5
	2	1	0	1	1	1	1	1	1	1	1	1	10			3
	Tier Totals	4	2	4	4	3	3	4	4	3	3	4	38			8

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

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points, and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the outline. Systems or evolutions that do not apply at the facility should be deleted with justification. Operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible. Sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

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

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' IRs for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. If fuel-handling equipment is sampled in a category other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2. (Note 1 does not apply). Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

G* Generic K/As

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

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

ES-401	PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO)	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				✓			EA1.07 Ability to operate and monitor the following as they apply to a reactor trip: MT/G trip; verification that the MT/G has been tripped	**4.3**	**39**
000008 (APE 8) Pressurizer Vapor Space Accident / 3	✓						AK1.02 Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure	**3.1**	**40**
000009 (EPE 9) Small Break LOCA / 3			✓				EK3.18 Knowledge of the reasons for the following responses as the apply to the small break LOCA: Monitoring containment radiation levels	**3.9**	**41**
000011 (EPE 11) Large Break LOCA / 3		✓					EK2.02 Knowledge of the interrelations between the and the following Large Break LOCA: Pumps	**2.6**	**42**
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4			✓				AK3.03 Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction	**3.7**	**43**
000022 (APE 22) Loss of Reactor Coolant Makeup / 2									
000025 (APE 25) Loss of Residual Heat Removal System / 4				✓			AA1.09 Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: LPI pump switches, ammeter, discharge pressure gauge, flow meter, and indicators	**3.2**	**44**
000026 (APE 26) Loss of Component Cooling Water / 8					✓		AA2.02 Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The cause of possible CCW loss	**2.9**	**45**
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3				✓			AA1.03 Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure control when on a steam bubble	**3.6**	**46**
000029 (EPE 29) Anticipated Transient Without Scram / 1						✓	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.	**4.2**	**47**
000038 (EPE 38) Steam Generator Tube Rupture / 3									
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4					✓		EA2.1 Ability to determine and interpret the following as they apply to the (Uncontrolled Depressurization of all Steam Generators): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	**3.2**	**48**
000054 (APE 54; CE E06) Loss of Main Feedwater /4						✓	2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	**4.6**	**49**
000055 (EPE 55) Station Blackout / 6						✓	2.1.20 Ability to interpret and execute procedure steps.	**4.6**	**50**
000056 (APE 56) Loss of Offsite Power / 6	✓						AK1.03 Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of subcooling: use of steam tables to determine it	**3.1***	**51**
000057 (APE 57) Loss of Vital AC Instrument Bus / 6									

000058 (APE 58) Loss of DC Power / 6	✓						AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation	**2.8**	**52**
000062 (APE 62) Loss of Nuclear Service Water / 4			✓				AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the nuclear service water coolers	**3.2**	**53**
000065 (APE 65) Loss of Instrument Air / 8									
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6		✓					AK2.07 Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / Generator Control	**3.6**	**54**
(W E04) LOCA Outside Containment / 3					✓		EA2.1 Ability to determine and interpret the following as they apply to the LOCA Outside Containment, Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	**3.4**	**55**
(W E11) Loss of Emergency Coolant Recirculation / 4		✓					EK2.2 Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) 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.	**3.9**	**56**
(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)									
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			✓				AK3.01 Knowledge of the reasons for the following responses as they apply to Emergency Boration: When emergency boration is required	**4.1**	**57**
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2					✓		AA2.03 Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Charging subsystem flow indicator and controller	**2.8**	**58**
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7					✓		AA2.02 Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Expected change in source range count rate when rods are moved	**3.6**	**59**
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	✓						AK1.02 Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Leak rate vs. pressure drop	**3.5**	**60**
000051 (APE 51) Loss of Condenser Vacuum / 4			✓				AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum	**2.8**	**61**
000059 (APE 59) Accidental Liquid Radwaste Release / 9									
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74; **W E06** & E07) Inadequate Core Cooling / 4	✓						EK1.3 Knowledge of the operational implications of the following concepts as they apply to the (Degraded Core Cooling): Annunciators and conditions indicating signals, and remedial actions associated with the (Degraded Core Cooling).	**3.7**	**62**
000076 (APE 76) High Reactor Coolant Activity / 9									
~~000078 (APE 78*) RCS Leak / 3~~									

(W E01 & **E02**) Rediagnosis & SI Termination / 3				✓			EA1.3 Ability to operate and / or monitor the following as they apply to the (SI Termination) Desired operating results during abnormal and emergency situations.	**3.8**	**63**
(W E13) Steam Generator Overpressure / 4						✓	2.4.6 Knowledge of EOP mitigation strategies.	**3.7**	**64**
(W E15) Containment Flooding / 5		✓					EK2.1 Knowledge of the interrelations between the (Containment Flooding) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	**2.8**	**65**
(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									
~~(BW E13 & E14) EOP Rules and Enclosures~~									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4									
~~(CE A16) Excess RCS Leakage / 2~~									
~~(CE E09) Functional Recovery~~									
~~(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4~~									
K/A Category Point Totals:	2	1	2	1	2	1	Group Point Total:		9

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump										✓		A4.03 Ability to manually operate and/or monitor in the control room: RCP lube oil and lift pump motor controls	**2.8**	**4**
004 (SF1; SF2 CVCS) Chemical and Volume Control					✓	✓						K5.20 Knowledge of the operational implications of the following concepts as they apply to the CVCS: Reactivity effects of xenon, boration and dilution — — K6.26 Knowledge of the effect of a loss or malfunction on the following CVCS components: Methods of pressure control of solid plant (PZR relief and water inventory)	**3.6** **—** **3.8**	**5** **—** **6**
005 (SF4P RHR) Residual Heat Removal						✓						K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger	**2.5**	**7**
006 (SF2; SF3 ECCS) Emergency Core Cooling							✓					A1.18 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: PZR level and pressure	**4.0**	**1**
007 (SF5 PRTS) Pressurizer Relief/Quench Tank								✓				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Stuck-open PORV or code safety	**3.9**	**8**
008 (SF8 CCW) Component Cooling Water				✓						✓		K4.09 Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: The "standby" feature for the CCW pumps A4.07 Ability to manually operate and/or monitor in the control room: Control of minimum level in the CCWS surge tank	**2.7** **—** **2.9**	**3** **—** **9**
010 (SF3 PZR PCS) Pressurizer Pressure Control									✓		✓	A3.02 Ability to monitor automatic operation of the PZR PCS, including: PZR Pressure — — 2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	**3.6** **3.6**	**2** **10**
012 (SF7 RPS) Reactor Protection	✓							✓				K1.05 Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems: ESFAS — — A2.06 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: Failure of RPS signal to trip the reactor.	**3.8** **4.4**	**11** **12**

013 (SF2 ESFAS) Engineered Safety Features Actuation		✓							✓			K2.01 Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control — — A3.02 Ability to monitor automatic operation of the ESFAS including: Operation of actuated equipment	**3.6** **—** **4.1**	**13** **—** **14**
022 (SF5 CCS) Containment Cooling		✓										K2.01 Knowledge of power supplies to the following: Containment cooling fans	**3.0**	**15**
~~025 (SF5 ICE) Ice Condenser~~												NOT APPLICABLE		
026 (SF5 CSS) Containment Spray							✓				✓	A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment Pressure — — 2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	**3.9** **—** **4.2**	**16** **—** **17**
039 (SF4S MSS) Main and Reheat Steam				✓								K4.08 Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Interlocks on MSIV and bypass valves	**3.3**	**18**
059 (SF4S MFW) Main Feedwater	✓		✓									K1.02 Knowledge of the physical connections and/or cause/effect relationships between the MFW and the following systems: AFW system — — K3.03 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: S/GS	**3.4** **—** **3.5**	**19** **—** **20**
061 (SF4S AFW) Auxiliary/Emergency Feedwater								✓				A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc power	**3.1**	**21**
062 (SF6 ED AC) AC Electrical Distribution											✓	2.4.31 Knowledge of annunciator alarms, indications, or response procedures.	**4.2**	**22**
063 (SF6 ED DC) DC Electrical Distribution				✓								K4.02 Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following: Breaker Interlocks, permissives, bypasses and cross-ties.	**2.9***	**23**
064 (SF6 EDG) Emergency Diesel Generator			✓									K3.02 Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: ESFAS controlled or actuated systems	**4.2**	**24**
073 (SF7 PRM) Process Radiation Monitoring					✓							K5.03 Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Relationship between radiation intensity and exposure limits	**2.9***	**25**
076 (SF4S SW) Service Water							✓					A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures	**2.6***	**26**

078 (SF8 IAS) Instrument Air	✓											K1.04 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor	**2.6**	**27**
103 (SF5 CNT) Containment			✓									K3.02 Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under normal operations	**3.8**	**28**
~~053 (SF1; SF4P ICS*) Integrated Control~~														
K/A Category Point Totals:	**3**	**2**	**3**	**3**	**2**	**2**	**3**	**3**	**2**	**2**	**3**	Group Point Total:		28

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

Plant Systems—Tier 2/Group 2 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive					✓							K5.97 Knowledge of the following operational implications as they apply to the CRDS: Relationship of T-avg to T-ref.	**3.3**	**29**
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control														
014 (SF1 RPI) Rod Position Indication							✓					A1.04 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: Axial and radial power distribution	**3.5**	**30**
015 (SF7 NI) Nuclear Instrumentation			✓									K3.01 Knowledge of the effect that a loss or malfunction of the NIS will have on the following: RPS	**3.9**	**31**
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor						✓						K6.01, Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors.	**2.7**	**32**
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge	✓											K1.02 Knowledge of the physical connections and/or cause effect relationships between the Containment Purge System and the following systems: Containment radiation monitor	**3.3**	**33**
033 (SF8 SFPCS) Spent Fuel Pool Cooling									✓			A3.02 Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Spent fuel leak or rupture.	**2.9**	**34**
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator										✓		A4.05 Ability to manually operate and/or monitor in the control room: Level Control to enhance natural circulation	**3.8**	**35**
041 (SF4S SDS) Steam Dump/Turbine Bypass Control														
045 (SF 4S MTG) Main Turbine Generator											✓	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	**3.9**	**36**
055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate								✓				A2.04 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: Loss of condensate pumps	**2.6**	**37**
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				✓								K4.03 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Detection and location of fires	**3.1**	**38**
~~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: Wolf Creek **Date of Exam:** December 2019

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.	3.8	66		
	2.1.19	Ability to use plant computers to evaluate system or component status.	3.9	67		
	2.1.41	Knowledge of the refueling process.	2.8	68		
	Subtotal			3		
2. Equipment Control	2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.	2.6	69		
	2.2.22	Knowledge of limiting conditions for operations and safety limits.	4.0	70		
	Subtotal			2		
3. Radiation Control	2.3.11	Ability to control radiation releases.	3.8	71		
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4	72		
	Subtotal			2		
4. Emergency Procedures/Plan	2.4.11	Knowledge of abnormal condition procedures.	4.0	73		
	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	3.5	74		
	2.4.19	Knowledge of EOP layout, symbols, and icons	3.4	75		
	Subtotal			3		
Tier 3 Point Total				10		

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

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

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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									
000011 (EPE 11) Large Break LOCA / 3									
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4									
000022 (APE 22) Loss of Reactor Coolant Makeup / 2					✓		AA2.01 Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exists	3.8	84
000025 (APE 25) Loss of Residual Heat Removal System / 4									
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									
000038 (EPE 38) Steam Generator Tube Rupture / 3						✓	2.4.21 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.	4.6	85
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4									
000054 (APE 54; CE E06) Loss of Main Feedwater /4						✓	2.2.37 Ability to determine operability and/or availability of safety related equipment.	4.6	86
000055 (EPE 55) Station Blackout / 6									
000056 (APE 56) Loss of Offsite Power / 6									
000057 (APE 57) Loss of Vital AC Instrument Bus / 6					✓		AA2.05 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: S/G pressure and level meters	3.8	87
000058 (APE 58) Loss of DC Power / 6									
000062 (APE 62) Loss of Nuclear Service Water / 4									
000065 (APE 65) Loss of Instrument Air / 8						✓	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.	4.4	88
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					✓		EA2.2 Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.	4.3	89
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 (SRO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1					✓		AA2.03 Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place	4.8	90
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1						✓	2.4.21 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.	4.6	91
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									
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9									
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4									
000076 (APE 76) High Reactor Coolant Activity / 9					✓		AA2.02 Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS	3.4	92
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						✓	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.4	93
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4									
~~(BW E13 & E14) EOP Rules and Enclosures~~									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4									
~~(CE A16) Excess RCS Leakage / 2~~									
~~(CE E09) Functional Recovery~~									
~~(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4~~									
K/A Category Point Totals:					2	2	Group Point Total:		4

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump														
004 (SF1; SF2 CVCS) Chemical and Volume Control														
005 (SF4P RHR) Residual Heat Removal														
006 (SF2; SF3 ECCS) Emergency Core Cooling								✓				A2.12 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions requiring actuation of ECCS	4.8	76
007 (SF5 PRTS) Pressurizer Relief/Quench Tank														
008 (SF8 CCW) Component Cooling Water														
010 (SF3 PZR PCS) Pressurizer Pressure Control														
012 (SF7 RPS) Reactor Protection														
013 (SF2 ESFAS) Engineered Safety Features Actuation														
022 (SF5 CCS) Containment Cooling											✓	2.1.20 Ability to interpret and execute procedure steps.	4.6	77
~~025 (SF5 ICE) Ice Condenser~~														
026 (SF5 CSS) Containment Spray														
039 (SF4S MSS) Main and Reheat Steam														
059 (SF4S MFW) Main Feedwater														
061 (SF4S AFW) Auxiliary/Emergency Feedwater											✓	2.4.1 Knowledge of EOP entry conditions and immediate action steps.	4.8	78
062 (SF6 ED AC) AC Electrical Distribution								✓				A2.04 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: Effect on plant of de-energizing a bus	3.4	79
063 (SF6 ED DC) DC Electrical Distribution														

064 (SF6 EDG) Emergency Diesel Generator								✓				A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Parallel operation of ED/Gs	3.1	80
073 (SF7 PRM) Process Radiation Monitoring														
076 (SF4S SW) Service Water														
078 (SF8 IAS) Instrument Air														
103 (SF5 CNT) Containment														
~~053 (SF1; SF4P ICS*) Integrated Control~~														
K/A Category Point Totals:								3			2	Group Point Total:		5

ES-401 PWR Examination Outline 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								✓				A2.10 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: Failure of PZR level instrument - high	3.6	81
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											✓	2.4.6 Knowledge of EOP mitigation strategies.	4.7	82
035 (SF 4P SG) Steam Generator														
041 (SF4S SDS) Steam Dump/Turbine Bypass Control								✓				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the SAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam valve stuck open	3.9	83
045 (SF 4S MTG) Main Turbine Generator														
055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air														
086 Fire Protection														
~~050 (SF 9 CRV*) Control Room Ventilation~~														

K/A Category Point Totals:								2			1	Group Point Total:		3

Facility: Wolf Creek		Date of Exam: December 2019				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.3	Knowledge of shift or short-term relief turnover practices.			3.9	94
	2.1.15	Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.			3.4	95
	Subtotal					2
2. Equipment Control	2.2.12	Knowledge of surveillance procedures.			4.1	96
	Subtotal					1
3. Radiation Control	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions.			3.6	97
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.			3.1	98
	Subtotal					2
4. Emergency Procedures/Plan	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.			4.4	99
	2.4.44	Knowledge of Emergency Plan Protective Action Recommendations.			4.4	100
	Subtotal					2
Tier 3 Point Total						7

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO T2/G1 063 A3.01	010 A3.02	**063 - DC Electrical Distribution** Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights Replaced with 010 A3.02 due to Audit Exam Overlap and fairly narrow K/A. The previous Chief agreed to change in the interest of balance of coverage.
RO T2/G1 064 A2.03	012 A2.06	**064 – Emergency Diesel Generators** Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Parallel operation of ED/Gs. Replaced with 012 A2.06 due to overlap with SRO Only section and oversampling of the 064 K/A. 064 A2.03 is on SRO Only Section so the 064 topic was sampled 3 times before all Tier 2 Group 1 topics were sampled twice. The previous Chief agreed to change in the interest of balance of coverage.
RO T2/G1 026 A1.02	026 A1.01	**026 – Containment Spray** Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment temperature Replaced with 026 A1.01 as there is no Containment Temperature implication for Containment Spray System Operation at Wolf Creek. Containment Spray system is operated based on Containment Pressure, which is covered by K/A 026 A1.01. Will add K/A 026 A1.02 to Wolf Creek K/A Suppression list. The previous Chief agreed to this change.
RO T2/G1 064 K3.01	064 K3.02	**064 – Emergency Diesel Generators** Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: Systems controlled by automatic loader Replaced with 064 K3.02 due to audit exam overlap. The previous Chief agreed to change in the interest of balance of coverage.
RO T2/G2 001 K5.96	001 K5.97	**001 – Control Rod Drive System** Knowledge of the following operational implications as they apply to the CRDS: Sign changes (plus or minus) in reactivity, obtained when positive reactivities are added to negative reactivities. Replaced with 001 K5.97. The previous Chief agreed to change to aid in creation of a better Operationally valid question at the appropriate discriminatory level of difficulty as Tavg/Tref mismatch is the criteria which controls operation of automatic rod control.

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO T2/G2 034 K6.02	017 K6.01	**034 – Fuel-Handling Equipment** Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System: Radiation monitoring system Replaced with 017 K6.01 based on overlap with audit exam and the SRO Only section. 034 2.4.31 is on the SRO Only section so the 034 topic is sampled twice before all Tier 2/Group 2 K/As are sampled at least once. The previous Chief agreed to change in the interest of balance of coverage.
RO T2/G2 041 A3.02	033 A3.02	**041 – Steam Dump / Turbine Bypass Control** Ability to monitor automatic operation of the SDS, including: RCS pressure, RCS temperature, and reactor power Replaced with 033 A3.02 due to oversampling with SRO Only section. 041 A2.02 is on SRO Only section so the 041 topic is sampled twice before all Tier 2 / Group 2 K/As are sampled at least once. The previous Chief agreed to change in the interest of balance of coverage.
RO T2/G2 028 2.1.25	045 2.1.25	**028 – Hydrogen Recombiner and Purge Control** Ability to interpret reference materials, such as graphs, curves, tables, etc. Replaced with 014 2.1.25 due to inapplicability of the K/A. Hydrogen Recombiners are Retired-in-place at Wolf Creek and there are no graphs, curves, tables to interpret for Hydrogen Purge Control. The previous Chief agreed to change to the Turbine Topic since there are applicable associated operational curves.
RO T1/G1 029 2.2.36	029 2.2.44	**029 – Anticipated Transient Without Scram** Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. Replaced with 029 2.2.44 due to inability to write a question with the given topic and generic K/A combination. There is no maintenance activity that would result in an ATWS scenario and the overall mitigating strategy of EMG FR-S1, ATWS is to purposely remove power from the Rod Drive Motor Generator sets to cause the rods to insert on a loss of power.
RO T1/G1 077 AK2.04	077 AK2.07	**APE 077 – Generator Voltage and Electric Grid Disturbances** Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Controllers, positioners Replaced with APE 077 AK2.07 due to inapplicability of the K/A. There are no controllers, or positioners at Wolf Creek with upgraded Ovation Turbine Control System. Will add K/A 077 AK2.04 to Wolf Creek K/A Suppression list. The previous Chief agreed to this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO T1/G1 WE05 EA2.2	WE04 EA2.1	**WE05 - Inadequate Heat Transfer – Loss of Secondary Heat Sink** Ability to determine and interpret the following as they apply to the Loss of Secondary Heat Sink: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. Replaced with WE04 EA2.1 due to oversampling with SRO Only section. WE05 EA2.2 is on SRO Only section so WE05 topic is sampled twice before all Tier 1 / Group 1 topics are sampled once. The previous Chief agreed to change in the interest of balance of coverage.
RO T3/G2 2.2.39	G 2.2.22	**Generic 2.2.39 -** Knowledge of less than or equal to one-hour Technical Specification action statements for systems. Replaced with 2.2.22 due to inability to write a generic question for the given system extension K/A. This K/A cannot stand alone as a Generic Tier 3 Topic without being an extension of a Tier 2 System.
SRO T2/G1 012 2.1.20	022 2.1.20	**012 – Reactor Protection System** – Ability to interpret and execute procedure steps. Replaced with 022 2.1.20 based on oversampling. 012 Topic was already sampled twice on the RO Section.
SRO T1/G2 APE 003 2.4.8	APE 005 2.4.21	**APE 003 – Dropped Control Rod –** Knowledge of how abnormal operating procedures are used in conjunction with EOPs. Replaced with APE 005 2.4.21 due to inability to write a question at the SRO level for Dropped rod and using AOPs in conjunction with EOPs. The concept of flux distribution changes that might result from a dropped rod was also covered by RO question #30.
SRO T1/G2 APE 076 AA2.05	APE 076 AA2.02	**APE 076 – High Reactor Coolant Activity** Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: CVCS letdown flow rate indication Replaced with APE 076 AA2.02 as the letdown flowrate at Wolf Creek is based on orifice flow lineup, either 75 gpm or 120 gpm, and is independent of Reactor Coolant Activity. Raising flow to 120 gpm, as directed by Chemistry, to maximize flow through Ion Exchanger displays system level knowledge at the RO Level. The previous Chief agreed to change in the interest of asking an Operationally valid discriminatory question based on High Reactor Coolant Activity.
SRO T3/G4 G 2.4.21	G 2.4.22	**Generic 2.4.21** 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. Replaced with 2.4.22 due to overlap with #85. Both questions covered K/A 2.4.21. The previous Chief agreed to change in the interest of balance of coverage.

Tier / Group	Randomly Selected K/A	Reason for Rejection
SRO T3 G 2.4.37	G 2.4.44	**Generic 2.4.37** - Knowledge of the lines of authority during implementation of the emergency plan. Replaced with 2.4.44 due to overlap with audit exam and narrow focus of K/A. The previous Chief agreed to change in the interest of balance of coverage.

Facility: **Wolf Creek**	Date of Examination: **Dec 2019**
Examination Level: RO ☒ SRO ☐	Operating Test Number: ____________

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R,N	**A1 - 2.1.25 [3.9**] Determine dilution volume to stabilize power 1 hour after power reduction.
Conduct of Operations	R,D	**A2 - 2.1.20 [4.6]** Determine Final Accumulator Pressure per OFN EJ-015.
Equipment Control	R,N	**A3 - 2.2.13 [4.1]** Develop a Clearance Order for 'B' Containment Cooler.
Radiation Control	R,D	**A4 – 2.3.13 [3.2]** Determine maximum allowed dose per EPP 06-013 and calculate stay time.

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:

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

Facility: Wolf Creek		Date of Examination: Dec 2019	
Examination Level: RO ☐ SRO ☒		Operating Test Number:	

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, N	**A5 - 2.1.37 [4.6]** Given Data and completed 1/M plot during a reactor startup, review and determine any required follow-up actions.
Conduct of Operations	R, M	**A6 - 2.1.25 [4.2]** Given a completed STS SF-002, review and determine any related Technical Specification required actions.
Equipment Control	R, N	**A7 – 2.2.13 [4.3]** Given a prepared Clearance Order for 'B' Containment Cooler (SGN01B), review for approval and identify any errors.
Radiation Control	R, M	**A8 – 2.3.6 [3.8]** Given a prepared LRW Radioactive Release permit, review for approval and identify any errors.
Emergency Plan	R, N	**A9 – 2.4.41 [4.6]** Given plant conditions, classify the event and determine Protective Action Recommendation.

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

Facility: **Wolf Creek**	Date of Examination: **Dec 2019**
Exam Level: *RO* ☒ SRO-I ☒ **SRO-U** ☒	Operating Test Number:

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 Perform a Manual Dilution per SYS BG-200 to maintain temperature during startup.	**L, M, S**	**1**
S2 Manually align Containment Spray per EMG E-0, ATT F, Step F12 [Previous use on 2017 NRC S6]	**A, D, E, EN, P, S**	**5**
S3 Establish Hot Leg Recirculation per EMG ES-13.	A, E, EN, N, S	2
S4 Start up 'A' Train CCW and transfer Service Loop per SYS EG-201, Section 6.1	N, S	8
S5 Cycle PORV Block Valve per STS BB-201A, Section 8.1	D, L, S	3
S6 Restore AFW after LSP Actuation per ALR 00-127A	A, D, E, EN, S	4S
S7 Restore RCP Cooling per OFN BB-005	A, D, E, S	4P
S8 Change RM11 Process Rad Monitor Setpoint	*M, S*	*9*

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 Line up 'A' EDG for Autostart per SYS KJ-121, Section 6.1	**A, D, EN**	**6**
P2 Open Reactor Trip Breakers as directed by EMG FR-S1	**E, N, R**	**7**
P3 Locally close valves to Isolate RCP Seals per EMG C-0, Step 16.	**D, E, R**	**4P**

* 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 **RO** /SRO-I/SRO-U

(A)lternate path **5**	**4–6**/4–6 /2–3
(C)ontrol room	
(D)irect from bank **6**	≤ **9**/≤ 8/≤ 4
(E)mergency or abnormal in-plant **6**	≥ **1**/≥ 1/≥ 1
(EN)gineered safety feature **4**	≥ **1**/≥ 1/≥ 1 (control room system)
(L)ow-Power/Shutdown **2**	≥ **1**/≥ 1/≥ 1
(N)ew or (M)odified from bank including 1(A) **5**	≥ **2**/≥ 2/≥ 1
(P)revious 2 exams **1**	≤ **3**/≤ 3/≤ 2 (randomly selected)
(R)CA **2**	≥ **1**/≥ 1/≥ 1
(S)imulator **8**	

S1: The unit in MODE 2 at approximately 4% power. The applicant is tasked with performing a manual dilution in accordance with the Reactivity Plan. The applicant must correctly operate the Chemical and Volume Control System to add 120 gallons of water to the Volume Control Tank in accordance with SYS BG-200, REACTOR MAKEUP CONTROL SYSTEM NORMAL OPERATION, Step 6.2.

S2: A Large Break LOCA resulted in a Reactor Trip and Safety Injection Actuation. The applicant is tasked with performing EMG E-0, REACTOR TRIP OR SAFETY INJECTION, Attachment F, to verify proper automatic actuations. The applicant must recognize the ‘A’ Train of Containment Spray System failed to Auto Actuate and take proper action manually align the system for operation per step F12.

S3: The unit is aligned for Cold Leg Recirculation due to a Large Break LOCA and Safety Injection Actuation, which occurred 10 hours earlier. The applicant is tasked with performing Steps 1-8 of EMG ES-13, TRANSFER HOT LEG RECIRCULATION. During performance of this task, EJ HV-8840, RHR HOT LEG RECIRC VLV, will not open. The applicant must re-align the Residual Heat Removal System for Cold Leg Recirculation while proceeding to align the Safety Injection System for Hot Leg Recirculation.

S4: The unit operating at 100% power with Yellow Train equipment in service when corrective maintenance on the ‘A’ Centrifugal Charging Pump is complete and a post-maintenance test run is required. The applicant is tasked with starting up the ‘A’ Train of Component Cooling Water System and transferring the Service Loop to the ‘A’ train per SYS EG-201, TRANSFERRING SUPPLY OF CCW SERVICE LOOP AND CCW TRAIN SHUTDOWN, Step 6.1, to support the ‘A’ Centrifugal Charging pump run.

S5: The unit is in MODE 2 at approximately 4% power. The applicant is tasked to perform an operability test of the ‘B’ Power Operated Relief Valve Block Valve per STS BB-201A, CYCLE TEST OF PORV BLOCK VALVE.

S6: A Tornado has gone through the protected area causing a Unit Trip and damage to the Condensate Water Storage Tank. The applicant is tasked with performing ALR 00-127A, AFP SUCT PRESS LO. While performing this task, the applicant will discover the ‘A’ Train of Low Suction Pressure failed to actuate, requiring the applicant to manually align ESW to the ‘A” MD AFW Pump Suction.

S7: The unit is operating at 100% power when the crew entered OFN BB-005, RCP MALFUNCTIONS due to numerous alarms associated with the Reactor Coolant Pump thermal barriers. The applicant is tasked with performing Steps 7 & 8. The applicant will discover that one of the Component Cooling Water containment isolation valves has closed, requiring the action to bypass the valve to restore flow before Reactor Coolant Pump trip criteria is met

S8: The unit is operating at 100% power, when Chemistry issued a Gas Release Permit which requires a change to the Radwaste Effluent Radiation Monitor setpoints. The applicant is tasked to change the setpoint per given release permit, APF 07B-001-11-07, and SYS SP-121, OPERATION OF THE G.A. MONITOR SYSTEM, Step 6.3.

P1: The unit is in MODE 4, and a surveillance run of the ‘A’ Emergency Diesel Generator has just been completed. The applicant is tasked with aligning the ‘A’ Emergency Diesel Generator for automatic operation per SYS KJ-121, DIESEL GENERATOR NE01 AND NE02 LINEUP FOR AUTOMATIC OPERATION, Step 6.1. The applicant will discover that the lockout relays will require manual reset and that the Engine Driven Jacket Water Pump Air isolation must be opened to properly align the Emergency Diesel Generator for Automatic Operation.

P2: A Turbine Trip occurred from 100% power, but the Reactor failed to trip in both Automatic and Manual. The crew is performing EMG FR-S1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS. The applicant is tasked to locally open the Reactor Trip and Bypass Breakers. The applicant will locate and open the breakers.

P3: The unit tripped due to a complete loss of AC power and the crew is responding per EMG C-0, LOSS OF ALL AC POWER. The applicant is tasked with performing EMG C-0, Step 16 to isolate the Reactor Coolant Pump Seals. The applicant will enter the RCA to locate and close the five valves specified in the procedure step.

Facility: Wolf Creek Scenario No.: 1 Op-Test No.: December 2019

Examiners: ____________________ Operators: ____________________

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Initial Conditions: Unit is in MODE 2, 4% Power, BOL, Yellow Train In Service, ‘A’ MFP Running.

Turnover: The unit is operating in MODE 2 at 4% power, BOL with ‘A’ MFP in service. GEN 00-003, HOT STANDBY TO MINIMUM LOAD is in progress, on step 6.10, maintain power level while the oncoming crew briefs entry to MODE 1. Temperature is being controlled using Steam Dumps in Automatic. MCB Annunciators 103D and 103E are listed on the White Board.

Critical Tasks: **CT-1** Close either BG HIS-8160 or BG HIS-8152 to isolate CTMT. **CT-2** Manually start either ‘A’, ‘C’, or ‘D’ CCW Pump within 30 minutes after SIS. **CT-3** Realign from ECCS injection mode to cold leg recirculation before RWST level reaches 6% with a failure of EJ HV-8811B to automatically open.

Event No.	Malf. No.	Event Type*	Event Description
1		C (BOP/CRS)	Cond Vac Pump ‘A’ Trips SYS CG-120, Section 6.2.1
2		C (ATC/CRS) Tech Specs	Loss of Vital 120 VAC Instrument Bus NN03, OFN NN-021 LCO 3.8.7 COND A, LCO 3.8.9 COND C LCO 3.4.1 COND A (DNBR)
3		C (ATC/CRS) Tech Specs	Inadvertent CSAS on Red Train LCO 3.6.6 COND A, 3.3.2, Function 2.b COND A & C ALR 00-059A, OFN EN-049
4		I (All)	AB UK-33, Steam Dump Cooldown CTRL fails HIGH in Auto AP15C-003, OFN AB-041
5		M (All)	Earthquake, Large Break LOCA (18”) on Loop 4 Cold Leg EMG E-0, EMG E-1
6		C (ATC/CRS)	Valves BG HV-8160, BG HV-8152, and KA HIS-29B fail to Auto Close on CISA EMG E-0, ATT F, Step F3
7		C (BOP/CRS)	‘B’ CCW pump trips, ‘A’, ‘C’, and ‘D’ CCW Pumps fail to autostart on SIS EMG E-0, ATT F, Step F6
8		C (All)	EJ HV-8811B, CTMT SUMP TO RHR PUMP SUCTION fails to open on RWST LOLO EMG ES-12

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

Target Quantitative Attributes per Scenario (See Section D.5.d)	Actual Attributes	ES-301-5	CRS	ATC	BOP
1. Malfunctions after EOP entry (1–2)	3	Rx	0	0	0
2. Abnormal events (2–4)	4	Nor	0	0	0
3. Major transients (1–2)	1	I/C	7	5	4
4. EOPs entered/requiring substantive actions (1–2)	2	Maj	1	1	1
5. Entry into a contingency EOP with substantive actions ($\geq$ 1 per scenario set)	0	TS	2	0	0
6. Preidentified critical tasks ($\geq$ 2)	3				

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT1: Close either BG HIS-8160 or BG HIS-8152 containment isolation valves before completion of EMG E-0, Attachment F.	The non-essential containment penetrations are isolated to prevent potential release of radioactive materials from containment. With both BG HIS-8160 and BG HIS-8152 open, a release path to the environment exists. KA HIS-29B is failed open to prevent these valves from failing closed.	Red lights lit on *BG HIS-8160 *BG HIS-8152 ESFAS Status PANEL CISA Section White Lights NOT LIT. *BGHV8152 (Red) *BGHV8160(Yellow)	On Panel RL001, Depress CLOSE on: *BG HIS-8160 *BG HIS-8152	Green lights lit on *BG HIS-8160 *BG HIS-8152 ESFAS Status Panel ICSA Section White Lights LIT for Yellow Train if BG HV8160 closed. Red Train White Lights require both BG HV8152 and KA HIS28B closed.
CT2: Manually start 'A', 'C', or 'D' CCW pump to cool one Train of ECCS equipment within 30 minutes of SIS to prevent the loss of CCP or SI pumps.	Failure to maintain CCW flow to ECCS components would result in a reduction of margin of safety due to loss of all CCW flow only by improper crew response. AI 21-016 specifies TSA to trip CCPs and SIPs on a loss of CCW cooling within 30 minutes.	Green lights are lit on CCW hand switches * EG HIS-21 <u>and</u> * EG HIS-23 <u>and</u> * EG HIS-24 Amber light lit on CCW hand switch * EG HIS-22	On Panel RL-019, Manually start one Red or Yellow Train CCW Pump. Either: * EG HIS-21 <u>or</u> * EG HIS-23 <u>or</u> * EG HIS-24	Red Light on the manipulated hand switch, * EG HIS-21 <u>or</u> * EG HIS-23 <u>or</u> * EG HIS-24
CT3: Realign from ECCS injection mode to cold leg recirculation before RWST level reaches 6% with a failure of EJ HV-8811B to automatically open.	Unnecessary loss/reduction of core cooling. ECCS pumps taking suction from the RWST are required to be stopped when RWST level reaches 6% in order to prevent loss of suction flow to the pumps and potential pump damage.	RWST Level <36% Annunciator 047D On Panel RL-017, Green Light Remains Lit on * EJ HIS-8811B.	On Panel RL-017, Manipulates controls: * EJ HIS-2 to Stop * BN HIS-8812B to CLOSE * EJ HIS-8811B to OPEN * EJ HIS-2 to Run	- Green Light on EJ HIS-2 - Green light on BN HIS-8812B - Red Light on EJ HIS-8811B - Red Light on EJ HIS-2 - 'B' RHR Restoration conditions: Pressure * EJ PI-615 Flow * EJ FI-619

Note: Causing an unnecessary plant trip or ESF actuation may constitute a Critical Task failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

SCENARIO # 1 NARRATIVE

Turnover: The Unit is in MODE 2, operating at 4% power, BOL with ‘A’ MFP in service. Maintain current power level while the crew briefs for entering MODE 1. Annunciators 103D and 103E are written on the White Board.

Event 1: ‘A’ Condenser Mechanical Vacuum Pump Trips. A NPIS alarm will the first indication of pump trip as vacuum will not degrade very fast. The crew may use section 6.2 of SYS CG-120 to start a standby condenser vacuum pump. If Vacuum degrades to Ovation alarm setpoint of 4 inches, the crew may perform OFN AF-025, Attachment F. Step F6 directs using SYS CG-120 to start standby condenser vacuum pumps. Once a standby condenser vacuum pump is started and at the direction of the lead evaluator, the next will start.

Event 2: Loss of bus NN03. Annunciators 027A and 027C will actuate, indicating a loss of instrument bus power, as well as multiple annunciators that are symptoms of that power loss. Partial Trip Status PERMIS/BLOC Panel, SB-069, will also show columns of white lights for the loss of NN03 powered equipment. The CRS will direct “Select out Blue” which will prompt which will prompt ATC and BOP Operators to select alternate channels as memory actions. The ATC will select manual on the PZR Master Pressure Controller before selecting an alternate channel to prevent lifting a PORV. The BOP will manually isolate ‘C’ ARV. The crew will perform OFN NN-021 and dispatch the Turbine Building Watch to investigate the loss of power, which was due to a maintenance worker inadvertently bumping open breaker NN0301. Closing this breaker restores power to NN03. Once the crew has reenergized the bus and determined applicable technical specifications, the next event will start as directed by the Lead Examiner.

Event 3: Inadvertent CSAS. An inadvertent CSAS will actuate on ‘A’ Train and annunciator 059A will alarm. The crew will place EN HIS-3 in Pull-to-Lock as a Memory Action Step per OFN EN-049. The CSAS signal will NOT be able to be reset for the given failure, so the crew will have to evaluate LCO 3.6.3, 3.6.6, 3.6.7, 3.7.7 and 3.03 per step 16. Once Technical Specifications have been evaluated, Event 2 will start at the direction of the Lead Examiner.

Event 4: AB UK-33, Steam Dump Cooldown Controller fails HIGH in Auto. Controller Failure will be diagnosed by AB UK-33 output rising to 100% and the three Steam Dump Valves, AB UV34, AB UV-45 and AB UV41 fully opening. As a result of the steam dump valves opening, Tavg will drop, adding positive reactivity which will cause inadvertent MODE change to MODE 1 without prompt Operator Action. S/G Levels rise due to swell causing MCB Annunciators 109B-111B to actuate. The BOP should take manual control of the failed AB UK-33 controller per AP15C-003, Manual Back-up to stabilize plant conditions. Once plant conditions are stable, the Major event will start as directed by the Lead Examiner.

Event 5: Earthquake, Large Break LOCA (18”) on Loop 4 Cold Leg. The earthquake will be felt and associated annunciators will all actuate (98B, 98D, 98E). The crew will diagnose RCS pressure and PZR Level lowering, as well as degrading conditions in CTMT, and manually trip the Reactor, actuate SI and perform EMG E-0 Immediate Actions. The next three post-trip events will also be addressed by the crew.

Event 6: Three CTMT Isolation Valves fail to close on CISA. (BG HV-8160 LTDN SYS INNER CTMT ISO VLV, BG HV-8152 LTDN SYS OUTER CTMT ISO VLV, and KA HIS-29 INST AIR SPLY CTMT ISO VLV). This failure will be indicated on the ESF SYS Status Indication boards. The ATC, while performing EMG E-0, ATTACHMENT F should manually close one of the two valves Letdown valves to isolate the open path from CTMT while performing Step F3. The failure of KA HIS-29 supports the critical task as BG HIS-8160 fails closed on a loss of air to containment.

CT1: Close either BG HV-8152 or BG HV-8160 containment Phase-A isolation valves to isolate a relief path from containment prior to completion of EMG E-0, Attachment F.

Event 7: 'B' CCW pump trips, 'A', 'C', and 'D' CCW Pumps fail to autostart on SIS. The BOP, after completing Immediate Actions, should note no operating CCW Pump running to cool Red or Yellow Train Safety Loads and manually start either 'A', 'C', or 'D' CCW Pumps. The ATC also has guidance per EMG E-0, ATTACHMENT F, Step F6, to manually start one of the two pumps in each train if one is NOT running at that time.

CT2: Manually start 'A', 'C', or 'D' CCW pump to cool one Train of ECCS equipment within 30 minutes of SIS to prevent the loss of CCP or SI pumps.

Event 8: EJ HV-8811B, CTMT SUMP TO RHR PUMP SUCTION fails to open on RWST LOLO
The crew will perform actions of EMG E-1, until RWST level drops to 36% and Annunciator 47D, RWST LEV LOLO1 AUTO XFR actuates. EJ HIS-8811B fails to auto open and because of RHR pump suction interlocks, the crew will have to Stop B RHR Pump, Close BN HIS-8812B, Open EJ HIS-8811B and restart B RHR Pump to complete the lineup for Cold Leg recirc on B Train.
CT3: Realign from ECCS injection mode to cold leg recirculation before RWST level reaches 6% with a failure of EJ HV-8811B to automatically open.

The scenario is complete when the crew completed procedure EMG ES-12 and verified cold leg recirculation.

Facility: Wolf Creek Scenario No.: 2 Op-Test No.: December 2019

Examiners: ______________________ Operators: ______________________

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Initial Conditions: 100% Power, MOL, Yellow Train In Service, 'A' EDG Out service, LCO 3.8.1, COND B is entered.

Turnover: The unit is operating at 100% power, MOL Yellow Train is in Service, 'A' EDG is out of service due to repairs on the Auxiliary Lube Oil Pump. LCO 3.8.1 Condition B is entered (Actions B.1 and B.2 are current. STS NB-005 was performed 3 hours ago).

Critical Tasks: **CT-1** Manually Trip the Reactor per EMG FR-S1, Step 1, Immediate Actions **CT-2** Isolate Feed flow to Faulted 'A' S/G. **CT-3** Terminate SI prior to Rupturing PRT.

Event No.	Malf. No.	Event Type*	Event Description
1		C (ATC/CRS) Tech Specs	Breaker 4-16 to Non-Safety 4.16 KV Bus SL-41 Trips, Loss of power to 'A' SW Pump. ALR 00-011D, ALR 00-08B, TR 3.7.8, COND A
2		I (BOP/CRS)	AE FI-520, B S/G Feed Flow Channel fails LOW. OFN SB-008, ATT E.
3		C (All)	'B' HDP Trips, Downpower to 95% OFN AF-025, OFN MA-038
4		I (BOP/CRS) Tech Specs	AC PT-505, Turbine Impulse Pressure Channel fails LOW OFN SB-008, ATT D LCO 3.3.1, Function 18.f, Conditions A, T
5		C (ATC/CRS)	Letdown Orifice valve BG HIS-8149BA fails closed ALR 00-0032D
6		M (All)	'D' RCP Trips, Reactor Fails to Trip in both Auto and Manual (ATWS) EMG FR-S1
7		C (BOP/CRS)	'A' MDAFW Pump fails to Auto start EMG FR-S1, Step 3
8		M (ALL)	Three S/G Safeties on 'A' S/G fail open (Faulted S/G), EMG E-0, EMG E-2, EMG ES-03
9		I (ATC/CRS)	SI Actuates on 'A' Train ONLY. EMG E-0, Step 4

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

Target Quantitative Attributes per Scenario (See Section D.5.d)	Actual Attributes	ES-301-5	CRS	ATC	BOP
1. Malfunctions after EOP entry (1–2)	2	Rx	0	0	0
2. Abnormal events (2–4)	5	Nor	0	0	0
3. Major transients (1–2)	2	I/C	7	4	4
4. EOPs entered/requiring substantive actions (1–2)	3	Maj	2	2	2
5. Entry into a contingency EOP with substantive actions (≥ 1 per scenario set)	1	TS	2	0	0
6. Preidentified critical tasks (≥ 2)	3				

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT 1: Given an ATWS, insert maximum negative reactivity into the core by manually inserting control rods and de-energizing the control rod drive MG sets within 1 minute of the need to trip the reactor.	Failure to insert negative reactivity by one of the methods listed can result in the needless continuation of an extreme or a severe challenge to the subcriticality CSF. The safeguards systems that protect the plant during accidents are designed assuming that only decay heat and pump heat are being added to the RCS.	1) Red first out annunciator 86A lit with indications of loss of RCS flow on one loop. 2) On Panel RL-004, Red lights lit for Reactor Trip Breakers *SB ZL-1 *SB ZL-2 3) On DRPI panel, ALL Rods out. 4) Reactor Not Manually Tripped after actuating Handswitches *SB HS-1 *SB HS-42 5) Reactor Power ≥5%	On Panel RL-004 RO inserts rods in MANUAL using * SF HS-2 On Panel RL-016, BOP/3rd RO opens red handled breakers: *PG HIS-16 *PG HIS-18	1) On DRPI panel, All Rod Bottom lights lit. 2) Reactor Power <5% on PR NIs. 2) Reactor power lowering on IR NI detectors * SE NI-34B * SE NI36B 3) Negative IR SUR *SE NI-35D *SE NI-36D
CT 2: Isolate feed flow into the Faulted 'A' S/G by closing AL HK-7A and AL HK-8A, AFW REG VLV CTRLs before ANY RCS Cold Leg temperature reaches 240°F.	Failure to isolate steam from and feed to a faulted S/G causes an unnecessary and avoidable challenge to the Integrity CSF due only to improper response by the crew.	S/G pressures, flows and level indications will make it possible to identify 'A' S/G as the faulted S/G. Reports from the field help identify safety valves have lifted.	Manipulates closed the following hand switches On Panel RL-005, o AL HK-7A, SG A MD AFP AFW Reg VLV CTRL o AL HK-8A, SG A TD AFP AFW REG VLV CTRL	On panel RL-005, AL HK-7A and 8A in the left latch detent position. Indicated flow on AL FI-2A is 0 lbm/hr

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT 3: Isolate high head ECCS flow through the BIT before overfill of the RCS results in a rupturing of the pressurizer relief tank (PRT) at 91 psig.	Continued maximum injection causes RCS to go solid and PORV to open, passing excess inventory through PORVs to the PRT. Failure to terminate ECCS flow when it is possible to do so results in a rupture of the PRT, spread of radioactive coolant into Containment, and constitutes an avoidable degradation of a fission product barrier, as well as additional risk of stuck open PORV (SBLOCA).	RCS pressure and pressurizer level rise. PORVs open, flow indicated. PRT level, pressure, and temperature rise. When PRT ruptures at ~91 psig, PRT pressure drops and equalizes with Containment Pressure.	The Operator will isolate the BIT per EMG ES-03, Step 13, by Manipulation of the following handswitches on Panel RL018. *EM HIS-8803A *EM HIS-8803B *EM HIS-8801A *EM HIS-8801B	Green lights LIT and red lights extinguished for the following valves: *EM HIS-8803A *EM HIS-8803B *EM HIS-8801A *EM HIS-8801B CCP To BIT Flow indicators drop to 0 GPM. *EM FI-917A *EM FI-917B

Note: Causing an unnecessary plant trip or ESF actuation may constitute a Critical Task failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

SCENARIO # 2 NARRATIVE

Turnover: The Unit is operating at 100% power. Yellow Train is in service. ‘A’ EDG is out of service due to repairs on the Auxiliary Lube Oil pump. LCO 3.8.1 Condition B is entered (Actions B.1 and B.2 are current. STS NB-005 was performed 3 hours ago)

Event 1: Breaker 4-16 to 4.16 KV, Non-Safety, Bus SL-41Trips, Loss of power to ‘A’ SW Pump. The crew will dispatch an Operator to investigate loss of bus SL-41 per ALR 00-011D and start ‘B’ SW pump per ALR 00-08B to restore Service Water System pressure to >85 psig. Once the CRS has determined TRM 3.7.8 is applicable for loss of ‘A’ SW Pump, the next event will start as directed by the Lead Examiner.

Event 2: B S/G Feed Flow Channel indicator AE FI-520 fails LOW. The crew will respond by taking manual control of AE FK-520, B FRV to match feed and steam flows per ALR 00-109C and then address the instrument failure per OFN SB-008, ATT C. Once the Crew has restored automatic control, the next event will start as directed by the Lead Examiner.

Event 3: ‘B’ HDP Trips. Per OFN AF-025, ATTACHMENT A, Maximum unit load is at 95% for one HDP out of service. The crew will reduce load per OFN MA-038 and beginning of shift reactivity brief. Once reactor power has stabilized at the new lower power level, the next event will start as directed by the Lead Examiner.

Event 4: Turbine First Stage Pressure indictor AC PT-505 fails LOW. After control rods reach 204 steps in automatic, AC PT-505 will fail low. After confirming no load reject is in progress, the ATC operator will take rods to manual to stop inward rod motion. The crew will address the failure per OFN SB-008, ATT D. Once the CRS has determined applicable technical specifications, the next event will start at the direction of the Lead Examiner.

Event 5: Letdown Orifice valve BG HIS-8149BA fails closed: ATC will recognize loss of Letdown flow and Pressurizer Level rising. The ATC will address the failure by charging to seals only, and restoring Letdown using one of the other valves. ALR 00-032D may be initiated to control pressurizer level. The next event will start following restoration of Letdown or at the direction of the Lead Examiner

Event 6: ‘D’ RCP spuriously trips and the Reactor fails to Trip in BOTH Auto and Manual. The loss of flow causes multiple MCB alarms, including a Red First Out 86A, which indicates the reactor should have tripped due to RCS flow <89.9% on 3/3 loop flow instruments on1/4 RCS Loops while Reactor Power >48% (P8). After the crew attempts to manually trip the reactor unsuccessfully, they will perform Immediate Actions of EMG FR-S1, to open RDMG breaker power supplies for PG19 and PG20 to trip the Reactor.

CT 1: Given an ATWS, insert maximum negative reactivity into the core by manually Inserting control rods and deenergizing the control rod drive MG sets within 1 minute of the need to trip the reactor.

Event 7: ‘A’ MD AFW Pump fails to Auto Start. The BOP will manually start ‘A’ MDAFW Pump per EMG FR-S1, Step 3 RNO.

Event 8: Three S/G Safety Valves will fail open on ‘A’ S/G and SI will actuate on ‘A’ Train ONLY: As soon as the Reactor trips. Safety valves will lift on ‘A’, ‘B’ and ‘C’ S/Gs. The Safety valves will reseat on ‘B’ and ‘C’ S/Gs, while three ‘A’ S/G safety valves stick open. Steam flow noises will be heard in the control room. Once the crew closes MSIVs, the faulted ‘A’ S/G will be more evident, and they will transition to EMG E-2 to address the faulted ‘A’ S/G.

CT 2: Isolate feed flow into the Faulted ‘A’ S/G by closing AL HK-7A, SG A MD AFP AFW REG VLV CTRL and AL HK-8A SG A TD AFP AFW REG VLV CTRL before ANY RCS Cold Leg temperature reaches 240°F.

Once the crew isolates the Faulted S/G per EMG E-2, they will transition to EMG ES-03 to terminate SI.
CT 3: Isolate high head ECCS flow through the BIT before overfilling the RCS resulting in a rupture of the pressurizer relief tank (PRT) at 91 psig.

Event 9: SI fails to actuate on ‘A’ Train ONLY: SI will actuate on ‘A’ Train ONLY, The ATC will Manually Actuate SI on ‘B’ Train during performance of EMG E-0 Immediate actions before the crew continues in EMG E-0 to verify proper SI Auto actuation.

The scenario is complete when the crew has Terminated SI flow and verified ECCS Flow is NOT required per EMG ES-03, Step 18 and/or at the discretion of the Lead Examiner.

Facility: Wolf Creek Scenario No.: 3 Op-Test No.: December 2019

Examiners: ______________________ Operators: ______________________

Initial Conditions: 59% Power, MOL, Yellow Train In Service, Benton Line is out of service.

Turnover: The unit is operating at 59% power, MOL Yellow Train is in Service, Benton Line was removed from service yesterday to replace multiple damaged poles expected to return tomorrow.

Critical Tasks: **CT-1** ALL CLOSE MSIVs to isolate steam to the Turbine **CT-2** Given an open ARV on the Ruptured 'C' S/G, Isolate Feed flow and steam from Ruptured 'C' S/G prior to transition to either EMG E-2, or EMG C-31. **CT-3** Commence controlled RCS depressurization to allow for SI termination per EMG E-3 prior to overfilling the Ruptured 'C' S/G.

Event No.	**Malf. No.**	**Event Type***	**Event Description**
1		C (All)	'B' Stator Water Pump Trips, 'A' Stator Water Pump fails to Auto Start, Turbine Runback ALR 00-112C, OFN MA-001
2		I (ATC/CRS) Tech Specs	BB TI-421, Loop 2 TC Instrument channel fails LOW OFN SB-008, ATT L LCO 3.3.1, Functions 6 and 7, Conditions A, E
3		C (ATC/CRS)	BG TCV-130 fails closed in Auto ALR 00-039 B/A
4		I (BOP/CRS)	AE PT-508, Feed Header Pressure channel fails LOW OFN SB-008, ATT B
5		C (ALL) Tech Specs	'C' S/G Tube Leak, 50 gpm OFN BB-07A LCO 3.4.13, Condition B
6		M (ALL)	"C' S/G Tube leak grows to 400 gpm SGTR EMG E-0, EMG E-3
7		C (BOP/CRS)	Turbine fails to trip in both Auto and Manual, MSIV ALL CLOSE Push Button Works. EMG E-0, Step 2
8		C (BOP/CRS)	'C' ARV opens to 100% on Reactor Trip, Closes Manually AP 15C-003, EMG E-3, Step 3.b
*	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor		

Crew B tripped RX on Event 4, and lost Event 5

Target Quantitative Attributes per Scenario (See Section D.5.d)	Actual Attributes	ES-301-5	CRS	ATC	BOP
1. Malfunctions after EOP entry (1–2)	2	Rx	0	0	0
2. Abnormal events (2–4)	5	Nor	0	0	0
3. Major transients (1–2)	1	I/C	7	4	5
4. EOPs entered/requiring substantive actions (1–2)	1	Maj	1	1	1
5. Entry into a contingency EOP with substantive actions (≥ 1 per scenario set)	0	TS	2	0	0
6. Preidentified critical tasks (≥ 2)	3				

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT1 Manually ALL CLOSE main steamline isolation valves before a severe (orange-path) challenge develops to either the subcriticality (Positive IR SUR) or the integrity CSF (RCS Cold Leg Temperature <240°F)	Failure to isolate steam to the turbine given failure of auto and manual turbine trips will cause an unnecessary uncontrolled cooldown and avoidable challenges to the subcriticality and Integrity CSFs due only to lack of proper response by the crew.	Main Stop valves remain open despite reactor trip, and manual turbine trip.	On Panel RL-006 Manipulates either of the following handswitches: * AB HS-78 * AB HS-80	Green lights LIT on o AB HIS-14 o AB HIS-17 o AB HIS-20 o AB HIS-11 Indicated steam flow will drop to 0 MPPH on all four S/Gs.
CT2 Given an open ARV on ruptured S/G, Isolate feed flow into and steam flow from the ruptured 'C' S/G before making an unnecessary transition to EMG E-2 from EMG E-0, Step 16 or to EMG C-31 due to * RCS Subcooling <30°F, * PZR Level <6% or * Ruptured S/G Pressure <380 psig, by closing the following: *AB PIC-3A, ARV *AB HIS-20, MSIV *AL HK-11A MD AFW REG VLV CTRL *AL HK-12A TD AFW REG VLVL CTRL * AB V087 TDAFW Steam Supply from C S/G * AB-V082, C S/G Low Point Drain	Feedwater is isolated to prevent overfill of ruptured S/G. Steam flow out of S/G is isolated to minimized radiological release. It also maintains ruptured S/G pressure higher than non-ruptured, which prevents transition from E-3, the preferred procedure, to C-31, which will release radiation to the public.	Radiation Monitor alarms, S/G levels and S/G pressures make it possible to identify S/G 'C' as ruptured.	Manipulate controls as required to: * Close AB PIC-3A, ARV * Close AB HIS-20, MSIV * Close AL HK-11A and 12A, AFW REG VLVL CTRLs Dispatch Operator to close * AB V087 TDAFW Steam Supply * AB-V082, Low Point Drain	Green light on *AB HIS-20 0% output: *AL HK-12A *AL HK-11A *AB PIC-3A Report from Local Operator that valves are closed: *AB V-087 *AB V-082

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT3: Commence controlled RCS depressurization to allow for SI termination per EMG E-3 prior to overfilling the ruptured ‘C’ S/G (90% WR).	Depressurizing the RCS to equalize with Ruptured S/G pressure prior to overfilling the ruptured S/G minimizes radioactive release to the environment from the ruptured S/G, minimizes stress to the Main Steam Lines, and allows for a subcooled recovery vice a potential saturated recovery.	S/G Level rising in an uncontrolled manner with feed flow isolated. Radiation monitor alarms	Manipulation of Normal Spray controls as required to depressurize the RCS. *BB PK-455A, PZR PRESS MASTER CTRL	RCS Pressure reducing in a controlled manner, subcooling maintained, leak rate to ruptured S/G drops, PZR Level >6%, Ruptured S/G Level <90% WR.

Note: Causing an unnecessary plant trip or ESF actuation may constitute a Critical Task failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

SCENARIO #3 NARRATIVE

Turnover: The Unit is operating at 59% power. Yellow Train is in service. Benton Line was removed from service yesterday to replace multiple damaged poles expected to return tomorrow

Event 1: 'B' Stator Water Pump Trips and 'A' Stator Water Pump fails to Auto start. An automatic runback of the turbine will occur. The crew will address the runback per ALR 0112C and/or OFN MA-001. Once the crew has started 'A' Stator Water Pump, and stabilized plant conditions, the next event will start at the direction of the Lead Examiner.

Event 2: Loop 2 TC instrument, BB TI-421) Fails LOW. There is no automatic plant response due to the channel failure in the low direction. Multiple MCB Annunciators will actuate, including 067D, LOOP 2 T AVG LO DEV which will help the crew diagnose which instrument failed. The crew will address the instrument failure using OFN SB-008, ATT L. Once the crew has evaluated technical specifications, the next event will start at the direction of the Lead Examiner.

Event 3: BG TCV-130 fails CLOSED in Auto. Annunciators 039B, LTDN HX DISCH TEMP HI will actuate and depending on timeliness of the crew response, Annunciator 039A may also alarm indicating letdown demineralizers have been bypassed due to high temperature. The crew will perform ALR 039B and/or 039A actions. Once crew has taken manual control of BG TCV-130 with letdown heat exchanger outlet temperature lowering, the next event will commence at the direction of the Lead Examiner.

Event 4: AE PT-508, Feed water header pressure channel fails LOW. In response to rising MFP speed, rising feed water flow and rising S/G levels, the BOP should take manual control of MFP TURBS MASTER SPEED CTRL and refer to the posted figure for programmed feedwater ΔP to manually control feedwater flow as a Memory Action. The crew will address the instrument failure per OFN SB-008, ATT B. The next event will start at the direction of the Lead Examiner.

Event 5: 'C' S/G Tube Leak. Annunciator 062A will actuate for Process Radiation levels at the ALERT level. When the crew investigates which PRM is alarming they will diagnose the S/G tube leak and enter OFN BB-07A. When S/G Tube leakage exceeds 150 gpd, the CRS will enter LCO 3.4.13, COND B.

Event 6: 'C" S/G Tube Leak grows to 400 gpm SGTR. As the leak size grows, the crew will maximize charging, isolate letdown and Trip the Reactor and Actuate SI per foldout page direction. The next two post-trip events will also be addressed by the crew.

Event 7: Main Turbine fails to auto trip and will not trip using manual push buttons. While performing immediate actions, the BOP will note the turbine failed to trip and attempt to trip the turbine manually using the two pushbuttons. When that is unsuccessful, the BOP will use the ALL CLOSE push buttons to close MSIVs to isolate steam to the main turbine.

CT1: Manually ALL CLOSE main steamline isolation valves before a severe (orange-path) challenge develops to either the subcriticality (Positive IR SUR) or the integrity CSF (RCS Cold Leg Temperature <240°F

Event 8: Ruptured S/G 'C' ARV opens to 100% on Reactor Trip: The crew will identify high steam flow rate for 'C' S/G and/or open indication on 'C' ARV and the BOP will manually close the valve.

CT2: Given an open ARV on ruptured S/G, Isolate feed flow into and steam flow from the ruptured 'C' S/G before making an unnecessary transition to EMG E-2 from EMG E-0, Step 16 or to EMG C-31 due to RCS Subcooling <30°F, PZR Level <6% or Ruptured S/G Pressure <380 psig, by closing the following:
*AB PIC-3A, ARV
*AB HIS-20, MSIV
*AL HK-11A MD AFW REG VLV CTRL
*AL HK-12A TD AFW REG VLVL CTRL
*AB V087 TDAFW Steam Supply from C S/G
*AB-V082, C S/G Low Point Drain

The crew will transition to EMG E-3 to isolate 'C' S/G and cool down and depressurize the RCS to meet SI Termination criteria to minimize break flow though 'C' S/G tube rupture.

CT3: Commence controlled RCS depressurization to allow for SI termination per EMG E-3 prior to overfilling the ruptured 'C' S/G (90% WR).

The scenario is complete when the crew has depressurized the RCS per EMG E-3 Step 25 and/or at the discretion of the Lead Examiner.

Facility: Wolf Creek Scenario No.: 4 Op-Test No.: December 2019

Examiners: ______________________ Operators: ______________________

Initial Conditions: 100% Power, MOL, Red Train In Service, Letdown is at 120 gpm, ‘B’ MD AFW Pump is out of service.

Turnover: The unit is operating at 100% power, MOL, Red Train is in Service, ‘B’ MD AFW Pump was taken out of service 12 hours ago; LCO 3.7.5, Condition B is entered.

Critical Tasks: **CT-1** Manually Start ‘B’ ESW Pump prior to loaded ‘B’ EDG Tripping on high Jacket Water Temperature at 195°F. **CT-2** Commence Emergency Boration before positive IR SUR develops **CT-3** Restore Secondary Heat Sink using NS AFW Pump prior to being required to commence primary bleed and feed when 3 of 4 S/G levels degrade to <12% [28%] WR Level.

Event No.	Malf. No.	Event Type*	Event Description
1		I (ATC/CRS) Tech Specs	BB LI-459, Upper Selected PZR Level Channel fails HIGH. OFN SB-008, ATT J LCO 3.3.1, Functions 9, CONDs A, M
2		I (BOP/CRS)	AB FT-543, ‘D’ S/G Steam Flow Instrument fails LOW OFN SB-008, ATT A
3		C (ALL) Tech Specs	XNB02 Failure which results in AC Emergency Bus NB02 UV. ALR 00-022E, ALR 00-021C, OFN NB-030 LCO 3.8.1, COND A
4		C (ATC/CRS)	‘B’ ESW Pump fails to Auto Start on S/D Sequencer ALR 00-021C, Step 6
5		M (All)	Loss of Off Site Power EMG E-0, EMG ES-02
6		C (ATC/CRS)	Four Control Rods fail to fully insert. EMG E-0, Step 1, EMG ES-02, Step 12.
7		C (None)	TDAFW Pump trips and cannot be restarted
8		C (BOP/CRS)	Blowdown Containment Isolation valves fail to close (BM01/2/3/4) EMG FR-H1, Step 3a
9		C (All)	‘A’ MD AFW Pump Trips on Overcurrent EMG FR-H1, SYS AP-122

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

Target Quantitative Attributes per Scenario (See Section D.5.d)	Actual Attributes
1. Malfunctions after EOP entry (1–2)	3
2. Abnormal events (2–4)	4
3. Major transients (1–2)	1
4. EOPs entered/requiring substantive actions (1–2)	2
5. Entry into a contingency EOP with substantive actions (≥ 1 per scenario set)	1
6. Preidentified critical tasks (≥ 2)	3

ES-301-5	CRS	ATC	BOP
Rx	0	0	0
Nor	0	0	0
I/C	7	5	4
Maj	1	1	1
TS	2	0	0

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT1: Manually start 'B' ESW pump before loaded 'B' EDG trips on High Jacket Water Temperature at 195°F.	The onsite standby power system includes the Class 1E ac and dc power for equipment used to maintain a cold shutdown of the plant and to mitigate the consequences of a DBA. Not starting the ESW pumps in a timely manner could result in the loss of the EDG.	With the EDG running loaded: Green light lit on handswitch * EF HIS-56A No indicated ESW flow on * EF FI-54 No indicated ESW pressure on *EF PI-2	On Panel RL019, Manipulation of EF HIS-56A to Run Position.	Red light lit on handswitch * EF HIS-56A Indicated ESW flow on * EF FI-54 Indicated ESW pressure on *EF PI-2
CT2: Commence Emergency Boration due to more than one control rod stuck out before Positive IR SUR develops causing the crew to transition to EMG FR-S1 on an ORANGE path challenge to subcriticality CSF.	The shutdown reactivity margin must be made up through emergency boration to account for the reactivity worth of the stuck rods. Failure to emergency borate could cause an unnecessary challenge to Subcriticality CSF.	When Bus NB01 is reenergized power to DRPI panel is restored - Rods F8, B6, K10 and M4 are not on bottom.	On Panel RL001, manipulates control as necessary to start at least one BAT Pump: * BG HIS-5A <u>OR</u> * BG HIS-6A <u>AND</u> Open * BG HIS-8104	Red lights lit for operated components: BG HIS-5A BG HIS-6A BG HIS-8104 Indicated Flow >30 gpm on BG FI-121
CT3: Restore AFW Flow >270,000 lbm/hr using NSAFW Pump per EMG FR-H1 before 3 of 4 S/G levels degrade to <12% [28%] WR level.	Establishing at least 270,000 lbm/hr feedwater flow rate to the S/Gs before RCS bleed and feed is initiated to restore secondary heat sink and ensures the core will remain covered and adequately cooled. An otherwise preventable Bleed and Feed causes CTMT contamination and equipment damage due to rupturing PRT disk.	No Operating AFW Pumps Indicated flow at 0 lbm/hr on: *AL FI-2A *AL FI-3A *AL FI-4A *AL FI-1A	On panel RL005, Manipulation of AFW REG VLV CTRL *AL HK-8A *AL HK-10A *AL HK-12A *AL HK-6A	On panel RL005, Combined Indicated AFW TO SG FLOW >270,000 lbm/hr *AL FI 2A *AL FI-3A *AL FI-4A *AL FI-1A

Note: Causing an unnecessary plant trip or ESF actuation may constitute a Critical Task failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

SCENARIO # 4 NARRATIVE

Turnover: The Unit is operating at 100% power. Red Train is in service. 'B' MD AFW Pump was removed from service 12 hours ago for emergent work. LCO 3.7.5, Condition B is entered.

Event 1: BB LI-459, Upper Selected PZR Level Channel fails HIGH. Annunciator 032A will actuate for high PZR Level and charging flow will lower, causing actual PZR level to lower. The crew will address using OFN SB-008, ATTACHMENT J to remove the failed channel from service and restoring automatic control. Once the CRS has determined applicable Technical Specifications, the next event will start as directed by the Lead Examiner.

Event 2: AB FT-543, 'D' S/G Steam Flow instrument fails LOW. MCB Annunciator 111C will actuate due to feed/steam flow mismatch. The BOP will take manual control of AE FK-540, 'D' FRV to match steam and feed flows as a Memory Action Step. The crew will address the instrument failure using OFN SB-008, ATT A. Once AE FK-540 is restored to Automatic, the next event will start at the direction of the Lead Examiner.

Event 3: NB02 Bus Degraded Voltage leading to power interruption and S/D Sequencer Actuation. NB02 bus voltage drops to 3755v due to a fault on XNB02 transformer. Annunciator 022E will alarm once voltage is <3760v for 25 seconds. The crew will reference ALR 00-022E and in 94 seconds, the normal feeder breaker will trip open as designed. 'B' EDG will start and load. The crew will address the interruption of power to NB02 per ALR 00-21C and OFN NB-030, ATT B, including reducing turbine loading to maintain reactor power ≤99% due to AFAS-T Actuation. Once the crew has stabilized plant conditions, determined applicable Technical Specifications, and secured the TDAFW Pump, the major event will start at the direction of the Lead Examiner.

Event 4: 'B' ESW Pump fails to Auto Start on the S/D Sequencer. While responding to momentary loss of NB02, the ATC will note the failure of the 'B' ESW pump to auto start and manually start the pump within ~3 minutes of the EDG starting and loading to prevent the EDG from tripping on high temperature.
CT1: Manually start 'B' ESW pump before loaded 'B' EDG trips on High Jacket Water Temperature at 195°F.

Event 5: Offsite Power is Lost. The reactor will trip and the crew will perform EMG E-0 immediate actions and transition to EMG ES-02. The next four post-trip events will also be addressed by the Crew.

Event 6: Four Control Rods fail to fully insert. The ATC, while performing EMG E-0 Immediate Actions will note the four control rods not fully inserted and manually trip the Reactor per Step 1 RNO using SB HS-1. EMG ES-02, Step 12 directs the crew to Emergency Borate per OFN BG-009 for this condition.
CT 2: Commence Emergency Boration due to more than one control rod stuck out before Positive IR SUR develops causing the crew to transition to EMG FR-S1 on an ORANGE path challenge to subcriticality CSF.

Event 7: TD AFW Pump trips and cannot be restarted. The BOP will identify the TD AFW Pump tripped. Any attempts to start manually will be unsuccessful.

Event 8: SGBSIS fails to actuate in Auto. Steam Generator Blowdown Containment Isolation Valves fail to close on S/G LoLo level immediately following the reactor trip. The crew may or may not notice the failure since there is no SIS. The crew will exit EMG E-0 without performing Attachment F, which would have prompted the crew to verify SGBSIS actuation. Both EMG ES-02, Step 1 RNO and EMG FR-H1, Step 3a directs the crew to manually close the four valves that failed to Auto Close. While not specifically a critical task, failure to manually close these valves will contribute to S/G dry-out conditions, requiring the crew to bleed and feed when WR S/G levels degrade to <12% [28%].

Event 9: ‘A’ MD AFW Pump trips on overcurrent: After the crew has commenced Emergency Boration as required per EMG ES-02, Step 12, and/or at the direction of the Lead Examiner, the ‘A’ MDAFW Pump will trip on overcurrent causing the crew to transition to EMG FR-H1.

The crew will be successful restoring aux feed water flow using the NS AFW Pump per SYS AP-122, NON-SAFETY AUX FEED PUMP OPERATION.
CT3: Restore AFW Flow >270,000 lbm/hr using NSAFW Pump per EMG FR-H1 before 3 of 4 S/G levels degrade to <12% [28%] WR level.

The scenario is complete when the crew has restored the Secondary Heat Sink per EMG FR-H1, Step 8 and/or at the discretion of the Lead Examiner.

Facility: Wolf Creek Scenario No.: 5 Op-Test No.: December 2019

Examiners: ______________________ Operators: ______________________

Initial Conditions: 100% Power, MOL, Yellow Train in Service, Letdown is at 120 gpm, 'B' Safety Injection Pump has been tagged out for emergent maintenance.

Turnover: The unit is operating at 100% power, MOL Yellow Train is in Service, Letdown is at 75 gpm. 'B' SI Pump OOS for emergent maintenance, LCO 3.5.2, COND A is entered

Critical Tasks: **CT1** Establish High Head Injection by opening BIT Inlet valve(s) before RVLIS Level drops to 66% **CT2** Trip RCPs within 5 minutes of RCS pressure going below 1400 psig **CT3** Establish Alternate High Head Injection prior to CET temperatures exceeding 712°F.

Event No.	Malf. No.	Event Type*	Event Description
1		C (BOP/CRS)	TB Closed Cooling Water Pump 'B' Trips ALR 00-105A
2		C (ATC/CRS)	BG PK-131, LTDN HX OUTLET PRESS CTRL Fails HIGH in AUTO, Manual Available ALR 00-039E
3		C (ALL) Tech Spec	AC Emergency Bus NB01 Bus Lockout ALR 00-18A, OFN NB-030 LCO 3.8.9, COND B LCO 3.8.7, COND A LCO 3.8.1, Cond A,B,E
4		I (ATC/CRS) Tech Spec	PR NI 42 fails LOW OFN SB-008, ATT R LCO 3.3.1, Functions 2,3,~~5~~,6,18.b, 18.c, 18.d and 18.e 7 CONDs A, D, E, S, T, TRM 3.3.17, COND D.
5		C (ALL)	AD HIS-8, Condensate Pump 'A' Discharge valve fails closed. OFN AF-025, OFN MA-038
6		M (ALL)	Small Break LOCA in CTMT on 'C' Cold Leg EMG E-0, EMG E-1
7		C (BOP/CRS)	BIT Inlet valves, EM HIS-8803A/B, fail to AUTO OPEN on SI EMG E-0
8		C (ALL)	'B' CCP trips and BG FK-121 fails closed EMG FR-C2

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

Target Quantitative Attributes per Scenario (See Section D.5.d)	Actual Attributes	ES-301-5	CRS	ATC	BOP
1. Malfunctions after EOP entry (1–2)	2	Rx	0	0	0
2. Abnormal events (2–4)	5	Nor	0	0	0
3. Major transients (1–2)	1	I/C	7	5	5
4. EOPs entered/requiring substantive actions (1–2)	2	Maj	1	1	1
5. Entry into a contingency EOP with substantive actions (≥ 1 per scenario set)	1	TS	2	0	0
6. Preidentified critical tasks (≥ 2)	3				

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT1: Given a failure of the BIT inlet valves to open and no available Safety Injection Pumps, establish high head injection flow to the RCS by opening EM HIS-8803B before RVLIS Forced Flow Range drops to 66% w/ 4 RCPs running AND prior to tripping RCPs	With RCPs running and RVLIS <66%, the core is significantly uncovered and a degraded core cooling exists, challenging the fuel cladding fission product barrier only due failure of the crew to take the proper action.	Green light lit on * EM HIS-8803B No indication for * EM HIS-8803A, fails as is (CLOSED) No indicated High Head ECCS Flow: * EM FI-922 * EM FI-918 * EM FI-917A * EM FI-917B	On Panel RL-018, Open *EM HIS-8803B	Red Lights lit on the manipulated hand switch, Green light out. Charging flow through Bit: * EM FI-917A * EM FI-917B
CT2 Trip all RCPs within 5 minutes of RCS pressure going below 1400 psig per EMG E-0 foldout page step 1 AND after having established High Head Injection	During the initial stages of a SBLOCA, if selected parameter setpoints are reached, the RCPs should be tripped to avoid more serious impacts later due to core uncovery and loss of inventory caused by continued RCP Operation.	* RCS Pressure <1400 psig. (BB PI-455A/456/457/458) <u>AND</u> * CCP Flow or SI Pumps running with Indicated flow: (EM FI-917A/B) (EM FI-918/922) <u>AND</u> Operator Controlled Cooldown <u>NOT</u> in progress.	On Panel RL-021, take handswitches to the STOP position: *BB HIS-37 *BB HIS-38 *BB HIS-39 *BB HIS-40	Green Lights Lit on the manipulated handswitches. Indicated RCP Amps all drop to 0 on: *BB 11-1 *BB 11-2 *BB 11-3 *BB 11-4
CT3: Establish Alternate High Head Injection per EMG FR-C2 prior to CETC temperatures rising to 712°F and a transition to the RED path condition, EMG FR-C1.	The most effective method to restore adequate core cooling is to raise RCS inventory via safety injection. The NCP is the only pump remaining that can accomplish this function. This prevents a lack of decay heat removal and a RED path condition to be entered	Red Train CCP and SI Pumps without power due to NB01 Lockout. B SI Pumps out of service for maintenance at beginning of scenario. B CCP trips. No indicated flow on: EM FI-917A/B EM FI-922/918	On Panel RL001, Manipulates controls as necessary to Open *BG HC-182, *BG HIS-8105 *BG HIS-8147 *BG FK-462	100% open indication on BG HC-182 and BG FK-462. Red Lights Lit and green lights out on BG HIS-8105 and BG HIS-8146. On Panel RL002, Flow indicated on BG FI-121 CHG HDR FLOW

Note: Causing an unnecessary plant trip or ESF actuation may constitute a Critical Task failure. Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.

SCENARIO 5 NARRATIVE

Turnover: The Unit is operating at 100% power. Yellow Train is in service with letdown flow at 120 gpm, 'B' Safety Injection Pump has been tagged out for emergent maintenance. LCO 3.5.2, COND A is entered.

Event 1: Trip of 'B' TB CLCW Pump. Main Control Board Annunciators ALR 105A and 133A will both actuate. The crew should perform ALR 00-105A to restore cooling by starting 'A' CLCW pump using EB HIS-1. Once cooling is restored, the Turbine Building Watch will be dispatched to locally clear the 133A Isophase Bus Trouble Alarm. Once cooling is restored and at the direction of the Lead Examiner the next event will start.

Event 2: Letdown Outlet Pressure Controller BG PK-131 Fails HIGH in Auto. The output on Controller BG PK-131 will fail to 100% in auto, causing letdown HX high flow and Annunciator 039E to actuate. Once the ATC has taken action to manually restore proper letdown flow, the next event will start at the direction of the Lead Examiner.

Event 3: NB01 Bus Lockout: The crew will respond to a bus lockout condition per ALR 00-018A, which requires prompt action to lower turbine loading to maintain power <100% due to AFAS-T Actuation and to Start 'B' ESW pump. After plant conditions stabilize, the crew will perform OFN NB-030, ATTACHMENT A to address other equipment affected by loss of power to bus NB01. Once actions are complete and the CRS has determined technical specification implications and or at the discretion of the Lead Examiner, the next event will start.

Event 4: PRNI 42 Fails LOW. MCB Annunciators 78A and 83C will actuate. The crew will address the instrument failure using OFN SB-008, ATT R. After evaluating Technical Specifications and at the direction of the Lead Examiner, the Major event will start.

Event 5: Condensate Pump 'A' Discharge Valve (AD HIS-8) fails closed. The crew will respond using OFN AF-025 to determine maximum power with only two condensate pumps is 90% (1102 MWE) and commence rapid downpower per OFN MA-038 IAW pre-shift reactivity brief. Once plant conditions have stabilized, and at the direction of the Lead Examiner, the next event will start.

Event 6: Small Break LOCA inside CTMT. RCS leak develops on Loop 3 Cold Leg that grows to ~2.0" break over 30 seconds, crew will diagnose, Manually Trip the Reactor and Actuate Safety Injection

Event 7: BIT Inlet valves, EM HIS-8803A/B, fail to AUTO OPEN on SI. This malfunction, combined with 'B' SI pump being out of service and a bus lockout on NB01, supports the critical task to establish high head injection prior to Core Cooling conditions degrading to Orange Path CSF and before tripping RCPs. The BOP operator may identify the valve failure while monitoring foldout page actions for the RCPs. The ATC performing EMG E-0, ATTACHMENT F will also be procedurally directed to establish the correct lineup at step F13.
CT1: Given a failure of the BIT inlet valves to open and no available Safety Injection Pumps, establish high head injection flow to the RCS by opening EM HIS-8803B before RVLIS Forced Flow Range drops to 66% w/ 4 RCPs running AND prior to Tripping RCPs.

CT2: Trip all RCPs within 5 minutes of RCS pressure going below 1400 psig per EMG E-0 foldout page step 1 AND after having established High Head Injection.

Event 8: 'B' CCP Trips and BG FK-121 fails closed on SI. These failures, combined with initial conditions and NB01 bus lockout will cause crew to transition to EMG FR-C2 on an ORANGE PATH core cooling CSF where they will establish alternate high head injection using the NCP per ATTACHMENT A.

CT3: Establish Alternate High Head Injection per EMG FR-C2 prior to CET temperatures rising to 712°F and a transition to the RED path condition, EMG FR-C1.

The scenario is complete when the crew has transitioned to EMG FR-C2 and completed alignment of Alternate High Head Injection and/or at the discretion of the lead examiner