Facility: Date of Exam:

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

3. Generic Knowledge and Abilities Categories	1	2	3	4	Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	2	3	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 Form ES-401-2 Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1					X		EA2.01 Ability to determine or interpret the following as they apply to a reactor trip: Decreasing power level, from available indications. (CFR 41.7 / 45.5 / 45.6)	4.1	1
000008 (APE 8) Pressurizer Vapor Space Accident / 3						X	2.1.28 Knowledge of the purpose and function of major system components and controls. (CFR 41.7)	4.1	2
000009 (EPE 9) Small Break LOCA / 3	X						EK 1.02 Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Use of steam tables (CFR 41.8 / 41.10 / 45.3)	3.5	3
000011 (EPE 11) Large Break LOCA / 3			X				EK3.10 Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: PTS limits on RCS pressure and temperature. (CFR 41.5 / 41.10 / 45.6 / 45.13)	3.7	4
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4			X				AK3.03 Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction. (CFR 41.5, 41.10 / 45.6 / 45.13)	3.7	5
000022 (APE 22) Loss of Reactor Coolant Makeup / 2 SRO				X			AA1.08 Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Makeup: VCT level. (CFR 41.7 / 45.5 / 45.6)	3.4	6
000025 (APE 25) Loss of Residual Heat Removal System / 4					X		AA2.07 Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Pump cavitation. (CFR 43.5 / 45.13)	3.4	7
000026 (APE 26) Loss of Component Cooling Water / 8						X	2.1.19 Ability to use plant computers to evaluate system or component status (CFR 41.10 / 45.12)	3.9	8
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3	X						AK1.02 Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases. (CFR 41.8 / 41.10 / 45.3)	2.8	9
000029 (EPE 29) Anticipated Transient Without Scram / 1 SRO									
000038 (EPE 38) Steam Generator Tube Rupture / 3 SRO									
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4 SRO									
000054 (APE 54; CE E06) Loss of Main Feedwater /4 SRO									

000055 (EPE 55) Station Blackout / 6	X						EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling. (CFR 41.8 / 41.10 / 45.3)	4.1	10
000056 (APE 56) Loss of Offsite Power / 6			X				AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer. (CFR 41.5, 41.10 / 45.6 / 45.13)	3.5	11
000057 (APE 57) Loss of Vital AC Instrument Bus / 6				X			AA1.06 Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Manual control of components for which automatic control is lost. (CFR 41.7 / 45.5. 45.6)	3.5	12
000058 (APE 58) Loss of DC Power / 6					X		AA2.01 Ability to determine and interpret the following as they apply to the Loss of DC Power: That a loss of dc power has occurred; verification that substitute power sources have come online. (CFR 43.5 / 45.13)	3.7	13
000062 (APE 62) Loss of Nuclear Service Water / 4						X	2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR 41.7 / 41.10 / 43.2 / 43.3 / 45.3)	3.9	14
000065 (APE 65) Loss of Instrument Air / 8				X			AA1.03 Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Restoration of systems served by instrument air when pressure is regained. (CFR 41.7 / 45.5 / 45.6)	2.9	15
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6		X					AK2.04 Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Controllers, positioners. (CFR 41.4, 41.5, 41.7, 41.10 / 45.8)	3.0	16
(W E04) LOCA Outside Containment / 3		X					EK2.2 Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. (CFR 41.7 / 45.7)	3.8	17
(W E11) Loss of Emergency Coolant Recirculation / 4		X					EK2.1 Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR 41.7 / 45.7)	3.6	18
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4 SRO									
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									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2		X					AK2.02 Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Sensors and detectors. (CFR 41.7 / 45.7)	2.6	19
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7 SRO									
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			X				AK 3.01 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. (CFR 41.5, 41.10 / 45.6 / 45.13)	2.8	20
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				X			AA1.01 Ability to operate and / or monitor the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Automatic actuation. (CFR 41.7 / 45.5. / 45.6)	3.6	21
000067 (APE 67) Plant Fire On Site / 8					X		AA2.15 Ability to determine and interpret the following as they apply to the Plant Fire on Site: Requirements for establishing a fire watch. (CFR 43.5 / 45.13)	2.9	22
000068 (APE 68; BW A06) Control Room Evacuation / 8						X	2.4.1 Knowledge of EOP entry conditions and immediate actions steps. (CFR 41.10 / 43.5 / 45.13)	4.6	23
000069 (APE 69; W E14) Loss of Containment Integrity / 5	X						AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of Containment Integrity: Effect of pressure on leak rate. (CFR 41.8 / 41.10 / 45.3)	2.6	24
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4		X					EK2.04 Knowledge of the interrelations between the and the following Inadequate Core Cooling: HPI pumps. (CFR 41.7 / 45.7)	3.9	25
000076 (APE 76) High Reactor Coolant Activity / 9 SRO									
000078 (APE 78*) RCS Leak / 3							N/A		

(W ~~E01 &~~ E02) ~~Rediagnosis &~~ SI Termination / 3			X				EK3.3 Knowledge of the reasons for the following responses as they apply to the (SI Termination): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations. (CFR 41.5 / 41.10, 45.6, 45.13)	3.9	26
(W E13) Steam Generator Overpressure / 4									
(W E15) Containment Flooding / 5				X			EA1.3 Ability to operate and / or monitor the following as they apply to the (Containment Flooding): Desired operating results during abnormal and emergency situations.	2.8	27
(W E16) High Containment Radiation /9 SRO									
(BW A01) Plant Runback / 1							N/A		
(BW A02 & A03) Loss of NNI-X/Y/7							N/A		
(BW A04) Turbine Trip / 4							N/A		
(BW A05) Emergency Diesel Actuation / 6							N/A		
(BW A07) Flooding / 8							N/A		
(BW E03) Inadequate Subcooling Margin / 4							N/A		
(BW E08; W E03) LOCA Cooldown—Depressurization / 4									
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4 SRO									
(BW E13 & E14) EOP Rules and Enclosures							N/A		
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4									
(CE A16) Excess RCS Leakage / 2							N/A		
(CE E09) Functional Recovery							N/A		
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4							N/A		
K/A Category Point Totals:	1	2	2	2	1	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								X				A2.01 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: Problems with RCP seals, especially rates of seal leak-off. (CFR 41.5 / 43.5 / 45.3 / 45.13)	3.5	28
004 (SF1; SF2 CVCS) Chemical and Volume Control									X			A3.02 Ability to monitor automatic operation of the CVCS, including: Letdown isolation. (CFR 41.7 / 45.5.)	3.6	29
005 (SF4P RHR) Residual Heat Removal										X		A4.01 Ability to manually operate and/or monitor in the control room: Controls and indication for RHR pumps. (CFR 41.7 / 45.5. to 45.8)	3.6	30
006 (SF2; SF3 ECCS) Emergency Core Cooling											X	2.1.19 Ability to use plant computers to evaluate system or component status. (CFR 41.10 / 45.12)	3.9	31
007 (SF5 PRTS) Pressurizer Relief/Quench Tank	X											K1.03 Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: RCS (CFR 41.2 to 41.9 / 45.7 to 45.8)	3.0	32
008 (SF8 CCW) Component Cooling Water		X										K2.02 Knowledge of bus power supplies to the following: CCW pump, including emergency backup. (CFR 41.2 to 41.9 / 45.7 to 45.9)	3.0	33
010 (SF3 PZR PCS) Pressurizer Pressure Control **SRO**			X									Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RPS. (CFR 41.7 / 45.6)	4.0	34
012 (SF7 RPS) Reactor Protection				X								K4.06 Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: Automatic or manual enable/disable of RPS trips. (CFR 41.7)	3.2	35
012 (SF7 RPS) Reactor Protection					X							K5.01 Knowledge of the operational implications of the following concepts as they apply to the RPS: DNB. (CFR 41.5 / 45.7)	3.3	36
013 (SF2 ESFAS) Engineered Safety Features Actuation **SRO**						X						K6.01 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors. (CFR 41.7 / 45.5 to 45.8)	2.7	37
022 (SF5 CCS) Containment Cooling							X					A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure (CFR 41.5 / 45.5)	3.6	38

022 (SF5 CCS) Containment Cooling								X				A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Major leak in CCS (CFR 41.5 / 43.5 / 45.3 / 45.13)	3.1	39
025 (SF5 ICE) Ice Condenser												N/A		
026 (SF5 CSS) Containment Spray									X			A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Failure of spray pump. (CFR 41.5 / 43.5 / 45.3 / 45.13)	3.9	40
026 (SF5 CSS) Containment Spray										X		A3.01 Ability to monitor automatic operation of the CSS, including: Pump starts and correct MOV positioning (CFR 41.7 / 45.5)	4.3	41
039 (SF4S MSS) Main and Reheat Steam											X	2.1.20 Ability to interpret and execute procedure steps. (CFR 41.10 / 43.5 / 45.12)	4.6	42
059 (SF4S MFW) Main Feedwater	X											K1.04 Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: S/GS water level control system	3.4	43
059 (SF4S MFW) Main Feedwater			X									K3.02 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: AFW system. (CFR 41.7 / 45.6)	3.6	44
061 (SF4S AFW) Auxiliary/Emergency Feedwater			X									K3.01 Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS (CFR 41.7 / 45.6)	4.4	45
061 (SF4S AFW) Auxiliary/Emergency Feedwater				X								K4.04 Knowledge of AFW design feature(s) and/or interlocks(s) which provide for the following: Prevention of AFW runout by limiting AFW flow. (CFR 41.7)	3.1	46
062 (SF6 ED AC) AC Electrical Distribution				X								K4.10 Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Uninterruptable ac power sources. (CFR 41.7)	3.1	47
063 (SF6 ED DC) DC Electrical Distribution		X										K2.01 Knowledge of bus power supplies to the following: Major DC loads (CFR 41.7)	2.9	48

063 (SF6 ED DC) DC Electrical Distribution							X					A1.01 Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate. (CFR 41.5 / 45.5)	2.5	49
064 (SF6 EDG) Emergency Diesel Generator								X				A2.09 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: Synchronization of the ED/G with other electric power supplies. (CFR 41.5 / 43.5 / 45.3 / 45.13)	3.1	50
064 (SF6 EDG) Emergency Diesel Generator									X			A3.02 Ability to monitor automatic operation of the ED/G system, including minimum time for load pickup. (CFR 41.7 / 45.5)	3.4	51
073 (SF7 PRM) Process Radiation Monitoring **SRO**										X		A4.02 Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel. (CFR 41.7 / 45.5 to 45.8)	3.7	52
076 (SF4S SW) Service Water											X	2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR 41.10 / 43.5 / 45.3 / 45.12)	4.1	53
078 (SF8 IAS) Instrument Air **SRO**	X											K1.04 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor (CFR 41.2 to 41.9 / 45.7 to 45.8)	2.6	54
103 (SF5 CNT) Containment **SRO**							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity. (CFR 41.5 / 45.5)	3.7	55
053 (SF1; SF4P ICS*) Integrated Control												N/A		
K/A Category Point Totals:	3	2	3	3	1	1	3	3	3	3	3	Group Point Total:		28

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive										X		A4.03 Ability to manually operate and/or monitor in the control room: CRDS mode control. (CFR 41.7 / 45.5 to 45.8)	4.0	56
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control														
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation						X						K6.04 Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Bistables and logic circuits (CFR 41.7 / 45.7)	3.1	57
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor	X											K1.02 Knowledge of the physical connections and/or cause-effect relationships between the ITM system and the following systems: RCS (CFR 41.2 to 41.9 / 45.7 to 45.8)	3.3	58
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control		X										K2.01 Knowledge of bus power supplies to the following: Hydrogen recombiners (CFR 41.7)	2.5	59
029 (SF8 CPS) Containment Purge			X									K3.01 Knowledge of the effect that a loss of malfunction of the Containment Purge System will have on the following: Containment parameters. (CFR 41.7 / 45.6)	2.9	60
033 (SF8 SFPCS) Spent Fuel Pool Cooling SRO														
034 (SF8 FHS) Fuel-Handling Equipment				X								K4.03 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Overload protection. (CFR 41.7)	2.6	61
035 (SF 4P SG) Steam Generator							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the S/GS controls including: S/G wide and narrow range level during startup, shutdown, and normal operations (CFR 41.5 / 45.5)	3.6	62
041 (SF4S SDS) Steam Dump/Turbine Bypass Control									X			A3.02 Ability to monitor automatic operation of the SDS, including RCS pressure, RCS temperature, and reactor power (CFR 41.7 / 45.5)	3.3	63
045 (SF 4S MTG) Main Turbine Generator					X							K5.23 Relationship between rod control and RCS boron concentration during T/G load increases (CFR 41.5 / 45.7)	2.7	64
055 (SF4S CARS) Condenser Air Removal SRO														

056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal SRO														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air								X				A2.01 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: Cross-connection with IAS. (CFR 41.5 / 43.5 / 45.3 / 45.13)	2.8	65
086 Fire Protection														
050 (SF 9 CRV*) Control Room Ventilation												N/A		
K/A Category Point Totals:	1	1	1	1	1	1	1	1	1	1	0	Group Point Total:		10

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.21	Ability to verify the controlled procedure copy. (CFR 41.10 / 45.10 / 45.13)	3.5	66		
	2.1.31	Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup (CFR: 41.10 / 45.12)	3.8	67		
	Subtotal					
2. Equipment Control	2.2.20	Knowledge of the process for managing troubleshooting activities (CFR 41.10 / 43.5 / 45.13)	2.6	68		
	2.2.13	Knowledge of tagging and clearance procedures	4.1	69		
	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels. (CFR 41.6 / 41.7 / 45.2)	4.6	70		
	Subtotal					
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions. (CFR 41.12 / 43.4 / 45.10)	3.2	71		
	2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR 41.12 / 43.4 / 45.9 / 45.10)	3.4	72		
	Subtotal					
4. Emergency Procedures/Plan	2.4.20	Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR 41.10 / 43.5 / 45.13)	3.8	73		
	2.4.25	Knowledge of fire protection procedures. (CFR 41.10 / 43.5 / 45.13)	3.3	74		
	2.4.32	Knowledge of operator response to loss of all annunciators. (CFR 41.10 / 43.5 / 45.13)	3.6	75		75
	Subtotal					
Tier 3 Point Total				10		

Facility: Date of Exam:

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1					N/A				N/A			18	3	3	6
	2												9	2	2	4
	Tier Totals												27	5	5	10
2. Plant Systems	1												28	2	3	5
	2												10	2	1	3
	Tier Totals												38	4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
														2 2	1 2	

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

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

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

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

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

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

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

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

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

G* Generic K/As

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

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

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (SRO)		Form ES-401-2
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1									
000008 (APE 8) Pressurizer Vapor Space Accident / 3									
000009 (EPE 9) Small Break LOCA / 3									
000011 (EPE 11) Large Break LOCA / 3									
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4									
000022 (APE 22) Loss of Reactor Coolant Makeup / 2					X		AA2.01 Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Charging pump problems (CFR 43.5 / 45.13)	3.7	76
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						X	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)	4.4	77
000038 (EPE 38) Steam Generator Tube Rupture / 3						X	2.4.46 Ability to verify that the alarms are consistent with the plant conditions. (CFR 41.10 / 43.5 / 45.3 / 45.12)	4.2	78
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4						X	2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions (CFR 41.7 / 45.7 / 45.8)	4.6	79
000054 (APE 54; CE E06) Loss of Main Feedwater /4					X		AA2.06 Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): AFW adjustments needed to maintain proper T-ave, and S/G level. (CFR 43.5 / 45.13)	4.3	80
000055 (EPE 55) Station Blackout / 6									
000056 (APE 56) Loss of Offsite Power / 6									
000057 (APE 57) Loss of Vital AC Instrument Bus / 6									
000058 (APE 58) Loss of DC Power / 6									
000062 (APE 62) Loss of Nuclear Service Water / 4									
000065 (APE 65) Loss of Instrument Air / 8									
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6									
(W E04) LOCA Outside Containment / 3									

(W E11) Loss of Emergency Coolant Recirculation / 4									
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4					X		EA2.1 Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (CFR 43.5 / 45.13)	4.4	81
K/A Category Totals:					3	3	Group Point Total:		6

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (SRO)		Form ES-401-2
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1									
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7						X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (CFR 41.5 / 41.7 / 43.2)	4.2	82
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					X		AA2.03 Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: RCS radioactivity level meter. (CFR 43.5 / 45.13)	3.0	83
000078 (APE 78*) RCS Leak / 3							N/A		
(W E01 & E02) Rediagnosis & SI Termination / 3									
(W E13) Steam Generator Overpressure / 4									
(W E15) Containment Flooding / 5									
(W E16) High Containment Radiation /9						X	2.4.18 Knowledge of the specific bases for EOPs (CFR 41.10 / 43.1 / 45.13)	4.0	84
(BW A01) Plant Runback / 1							N/A		
(BW A02 & A03) Loss of NNI-X/Y/7							N/A		
(BW A04) Turbine Trip / 4							N/A		
(BW A05) Emergency Diesel Actuation / 6							N/A		
(BW A07) Flooding / 8							N/A		
(BW E03) Inadequate Subcooling Margin / 4							N/A		
(BW E08; W E03) LOCA Cooldown—Depressurization / 4									
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4					X		EA2.1 Ability to determine and interpret the following as they apply to the (Natural Circulation Operations): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (CFR 43.5 / 45.13)	3.8	85

(BW E13 & E14) EOP Rules and Enclosures							N/A		
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4									
(CE A16) Excess RCS Leakage / 2							N/A		
(CE E09) Functional Recovery							N/A		
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4							N/A		
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 (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														
007 (SF5 PRTS) Pressurizer Relief/Quench Tank														
008 (SF8 CCW) Component Cooling Water														
010 (SF3 PZR PCS) Pressurizer Pressure Control											X	2.2.12 Knowledge of surveillance procedures (CFR 41.10 / 45.13)	4.1	86
012 (SF7 RPS) Reactor Protection														
013 (SF2 ESFAS) Engineered Safety Features Actuation								X				A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Loss of instrument bus (CFR 41.5 / 43.5 / 45.3 / 45.13)	4.2	87
022 (SF5 CCS) Containment Cooling														
025 (SF5 ICE) Ice Condenser												NA		
026 (SF5 CSS) Containment Spray														
039 (SF4S MSS) Main and Reheat Steam														
059 (SF4S MFW) Main Feedwater														
061 (SF4S AFW) Auxiliary/Emergency Feedwater														
062 (SF6 ED AC) AC Electrical Distribution														
063 (SF6 ED DC) DC Electrical Distribution														
064 (SF6 EDG) Emergency Diesel Generator														
073 (SF7 PRM) Process Radiation Monitoring								X				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure (CFR 41.5 / 43.5 / 45.3 / 45.13)	3.2	88
076 (SF4S SW) Service Water														

078 (SF8 IAS) Instrument Air											X	2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entries. (CFR 41.7 / 45.7 / 45.8)	4.6	89
103 (SF5 CNT) Containment											X	2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications (CFR 41.7 / 41.10 / 43.2 / 43.3 / 45.3)	4.6	90
053 (SF1; SF4P ICS*) Integrated Control												N/A		
K/A Category Point Totals:								2			3	Group Point Total:		5

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

Plant Systems—Tier 2/Group 2 (SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive														
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control														
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation														
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor														
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling								x				A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal spent fuel pool water level or loss of water level.	3.5	91
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator														
041 (SF4S SDS) Steam Dump/Turbine Bypass Control														
045 (SF 4S MTG) Main Turbine Generator														
055 (SF4S CARS) Condenser Air Removal											X	2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR 41.5 / 43.5 / 45.12)	4.4	92
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal								X				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 Use of waste gas release monitors, radiation, gas flow rate, and totalizer. (CFR: 41.5 / 43.5 / 45.3 / 45.13	3.6	93
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												N/A		
K/A Category Point Totals:								2			1	Group Point Total:		3

Facility: Callaway Plant Date of Exam: August 31, 2020

Category	K/A #	Topic	RO IR	RO #	SRO-only IR	SRO-only #
1. Conduct of Operations	2.1.34	Knowledge of primary and secondary plant chemistry limits. (CFR: 41.10 / 43.5 / 45.12)			3.5	94
	2.1.39	Knowledge of conservative decision-making practices. (CFR: 41.10 / 43.5 / 45.12)			4.3	95
	Subtotal					
2. Equipment Control	2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc. (CFR 41.10 / 43.5 / 45.13)			3.9	96
	2.2.37	Ability to determine operability and/or availability of safety related equipment. (CFR 41.7 / 43.5 / 45.12)			4.6	97
	Subtotal					
3. Radiation Control	2.3.6	Ability to approve release permits (CFR: 41.13 / 43.4 / 45.10)			3.6	98
	Subtotal					
4. Emergency Procedures/Plan	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	99
	2.4.40	Knowledge of SRO responsibilities in emergency plan implementation. (CFR 41.10 / 43.5 / 45.11)			4.5	100
	Subtotal					
Tier 3 Point Total						7

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1 (RO)	APE 26: G2.2.22 (Q#8)	It would've been difficult to construct an RO question based on safety limits and LCO. Question was replaced with a K/A that would be more in-line with RO required level of knowledge.
1/1 (RO)	EPE 09: EK1.01 (Q#3)	The natural circulation was a repeat of another K/A. Changed to EK1.02
2/2 (RO)	015 (SF7NI) 2.2.36 (Q#57)	This K/A was replaced with one from K6 since Note 1 requirement was not met. Additionally, this would have been difficult to construct an RO level question.
3/G (RO)	2.2.25 (Q#68)	2.2.25 needed to be replaced since it is fundamentally an SRO function.
3/G (RO)	2.2.37 (Q#69)	2.2.37 needed to be replaced since it is fundamentally an SRO function.
3/G (RO)	2.2.40 (Q#70)	2.2.40 decided to replace the question with a non-T.S. related K/A.
1/2 (SRO)	000078 (APE 78*) 2.4.18	This system should have been excluded since it is not in Rev 2, Supp 1.
3/G (SRO)	2.2.20 (Q#97)	K/A was better suited for one of the RO generic equipment control questions. It was replaced with 2.2.37.
3/G (SRO)	2.4.39 (Q#100)	By definition not an SRO-only question. Replaced with 2.4.40.
1/2 (SRO)	000032 (APE 32) Loss of Source Range Nuclear Instrument 2.2.4	There are probably no differences between the STP units regarding this system. It was replaced with 2.2.25.
2/2 (RO)	045 (SF 4S MTG) Main Turbine Generator K.5.18 (Q#64)	Any reactor trip will trip the turbine. Not much of a distinction below and above 25% power. This K/A has been replaced with K.5.23
2/1 (SRO)	071 Waste Gas Disposal A2.04 (Q#93)	The A2.04 you have shown does not match the A2.04 in NUREG 1122. The question was replaced with A2.02.
1/1 (SRO)	000029 (EPE 29) Anticipated Transient Without Scram 2.2.37 (Q#77)	Original K/A not a good fit for topic of ATWS. Replaced with 2.1.23

Facility: South Texas Project | Date of Examination: XX-XX-2021

Examination Level: RO ■ SRO □ | Operating Test Number: LOT 25 NRC Exam

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M,R	A1 Determine if natural circulation cooling exists following Station Blackout G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (RO 4.6)
Conduct of Operations	D,R	A2 Performs the Independent Verification of the QPTR calculation. G2.1.20 Ability to interpret and execute procedural steps. (RO 4.6)
Equipment Control	N,R	A3 Prepare ECO for AFW. G2.2.13 Knowledge of tagging and clearance procedures. [RO 3.6]
Radiation Control	D,P,R	A4 Calculate Maximum Stay Time (Room M108C) G2.3.4 Knowledge of radiation exposure limits under normal and emergency conditions. (RO 3.2)

NOTE: RO applicants require only 4 items unless they are retaking only the administrative topics (which would require all five items, Emergency Procedures/Plan).

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

Facility: South Texas Project Date of Examination: XX-XX-2021

Examination Level: RO □ SRO ■ Operating Test Number: LOT 25 NRC Exam

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D,R	A5 Calculate SDM with a Misaligned Control Rod and Determine Applicable Technical Specifications. G2.1.37 Knowledge of procedures, guidelines or limitations associated with Reactivity Management. (SRO 4.69)
Conduct of Operations	D,R, P	A6 Review Calorimetric Heat Balance to evaluate acceptance criteria G2.1.23 Ability to perform specific system and integrated plant procedures during all modes of operation. (SRO 4.4)
Equipment Control	N,R	A7 Using technical specifications, determine operability for the applicable train based upon Lineup 11 after completion of ECO for the AFW. G2.2.40 Ability to apply Technical Specifications for a system (SRO 4.7)
Radiation Control	N,R	A8 Select individual to exceed dose limit for accident mitigation. G2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 45.10) (SRO 3.7) [NRC David You to write]
Emergency Procedures/Plan	M,R	A9 Determine EAL G2.4.41 Knowledge of the emergency action level thresholds and classifications. (SRO 4.6)

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics (which would require all five items).

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

STP LOT-25
NRC Admin JPM Description

RO

(A1) Determine if natural circulation cooling exists
Demonstrate the ability to determine if natural circulation exists using 0POP05-EO-EC01, Loss of All AC Power Recovery Without SI Required [LOT 13 Audit A2, Originally Static Simulator JPM] [NOTE: bank JPM is documented to be performed in a static simulator, in order to verify plant parameters for natural circulation. This should be modifiable for a classroom setting, by providing pictures of the various parameter trends: SG pressures, Hot Leg Temps, CETs, cold leg temps.]

(A2) Verify an Excore QPTR Calculation
Demonstrate the ability to perform and/or verify a QPTR. 0PSP10-NI-0002, Excore QPTR Determination. [LOT 22 NRC 2018 JPM A1]

(A3) Prepare ECO for the AFW
Demonstrate knowledge of tagging and clearance procedure for the AFW system. [New JPM]

(A4) Calculate Maximum Stay Time (Room 108C)
Demonstrate knowledge of radiation exposure limits under normal and emergency conditions. [LOT 23 2019 NRC JPM A4]

SRO

(A5) Calculate SDM with a Misaligned Control Rod and Determine Applicable Technical Specifications
Demonstrate the ability perform a SDM and apply appropriate TSs if required. 0PSP10-ZG-0005, Shutdown Margin Verification – Modes 1 and 2. [LOT 22 NRC 2018 JPM A5]

(A6) Review Calorimetric Heat Balance to evaluate acceptance criteria
Demonstrate the ability to perform a Calorimetric Verification and evaluate TSs. 0PEP02-CU-0001, Calorimetric Verification, and 0PSP03-NI-0001, Power Range NI Channel Calibration. [LOT 23 2019 NRC JPM A6]

(A7) Use technical specification to determine operability of AFW
Demonstrate the ability to apply Technical Specifications for a system. Use technical specifications to determine operability for the applicable train based upon Lineup 11. [NEW JPM]

(A8) Select Individual to Exceed Exposure Limits for Accident Mitigation in an Emergency
During a LOCA, choose from 8 volunteers the one individual to exceed occupational dose limits, IAW site specific procedures. [NEW JPM. See Callaway 2020 JPM A8 for reference]

(A9) Determine Emergency Action Level

Demonstrate the ability to correctly determine an Emergency Action Level for a given condition requiring entry into the STPNOC Emergency Action Plan.

Facility:	South Texas Project	Date of Examination:	July 2021
Exam Level:	*RO* SRO(I) **SRO (U)**	Operating Test No.:	LOT 25

Control Room Systems (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U, including 1 ESF)			
System / JPM Title		Type Code*	Safety Function
S1	*Perform Emergency Boration of RCS (Alt Path)*	A, D, S,	1
S2	Raise Safety Injection Accumulator Level	D, S, EN	2
S3	Respond to a Loss of Operating CCW Pump (alt path)	A, D, S, EN	8
S4	MSIV Operability Test	D, S	4S
S5	**Place H2 monitoring in service**	D, S	5
S6	**Respond to a failed SR NI**	D, L, S	7
S7	**Respond to a stuck open pressurizer spray valve. (alt path)**	A, N, S	3
S8	Perform Immediate Operator Actions following loss of offsite power. (alt path) [NRC to Write]	A, N, S	6
In-Plant Systems (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)			
P1	Locally Trip the reactor	D, E	1
P2	**Local Manual Stop of EDG 11 with 1-DG-MDA-0134 not fully latched.** (alt path)	A, N	6
P3	**Place a second Spent Fuel Pool Cooling train (B) in service.**	N, R	4P

RO: Will perform all simulator and in-plant JPMs
SRO (I): Will perform S2-S8 and all in-plant JPMs
SRO (U): S1, S3, S6 and P2-P3

@	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; 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	4-6 (5) / 4-6 (4) / 2-3 (3)
(C)ontrol room	
(D)irect from bank	≤ 9 (7) / ≤ 8 (6) / ≤ 4 (3)
(E)mergency or abnormal in-plant	≥ 1 (1) / ≥ 1 (1) / ≥ 1 (1)
(EN)gineered safety feature	≥ 1 (2) / ≥ 1 (2) / ≥ 1 (1) (control room system)
(L)ow Power / Shutdown	≥ 1 (1) / ≥ 1 (1) / ≥ 1 (1)
(N)ew or (M)odified from bank including 1(A)	≥ 2 (4) / ≥ 2 (4) / ≥ 1 (2)
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA (1)	≥ 1 (1)/ ≥ 1 (1) / ≥ 1 (1)
(S)imulator	

NRC JPM Examination
Summary Description

S1: Unit 1 failed to automatically trip when two channels of pressurizer pressure low bistables were tripped. The crew is performing actions of 0POP05-EO-FRS1, Response To Nuclear Power Generation – ATWS. The unit supervisor directs to initiate emergency boration of the RCS in accordance with the procedure. However, during the performance of the JPM, the only running charging pump will trip prior to the operator reaching Step 4b of the procedure. The applicant must correctly establish emergency boration flowpath and flowrates in accordance with procedure 0POP05-EO-FRS1, "Response To Nuclear Power Generation – ATWS" in which there are no running charging pumps. [NRC LOT-17]

S2: Unit 1 is in Mode 3. A low-level alarm for the 1A Accumulator has been received. The applicant is directed to restore level. The applicant must successfully use the HHSI Pump 1A and fill the SI accumulator to clear the alarm. Additionally, the applicant must secure filling before reaching a high level. This will be done in accordance with 0POP02-SI-0001, "Safety Injection Accumulators." This is from the licensee's bank labeled JPM Number: [NRC LOT-22 S2].

S3: Unit 1 is at 16% reactor power. Startup is in progress to 100%. The applicant is tasked by the Unit Supervisor to start CCW Pump 1C and secure CCW Pump 1A. A leak will develop when CCW Pump 1C starts. The applicant must correctly address the leak by closing the appropriate valves and isolating charging/letdown in accordance with 0POP04-CC-0001, "Component Cooling Water System Leak." This is from the licensee's bank labeled JPM Number: NRC S4. [NRC LOT-10] [May just swap back to original pump after leak develops]

S4: Unit 1 is in Mode 5. The off-going shift has completed preparations for performing operability testing "A" MSIV-7414. The applicant is directed to complete testing of MSIV-7414 in accordance with station procedure. The applicant must correctly perform timed strokes of MSIV-7414 and determine if they are within the acceptance criteria per 0PSP03-MS-0002, "Main Steam System Cold Shutdown Valve Operability Test." This is from the licensee's bank labeled JPM NO: NRC-007. [NRC LOT-10]

S5: Unit 1 has experienced a Large Break LOCA. The control room has completed standard post trip actions. The Unit Supervisor has directed to check containment H2 concentration per procedure. The applicant must correctly place the H2 monitors in service in accordance with 0POP05-EO-EO10, "Loss of Reactor or

Secondary Coolant." This is from the licensee's bank labeled JPM Number: LOT 22 Audit S1. [NRC LOT-21]

S6: Unit 1 is in Mode 3 preparing for reactor startup. Source Range Nuclear Instrument Channel NI 31 has failed low. The Unit Supervisor has directed the applicant to respond to the failure. The applicant must correctly place the Source Range Channel Level Trip to BYPASS, the High Flux Shutdown switch to BLOCK and set NI-32 for Audible Count Rate in accordance with 0POP04-NI-0001, "Nuclear Instrument Malfunction." This is from the licensee's bank labeled JPM Number: S8. [NRC LOT-22]

S7: Unit 1 is at 100% and the PRZR PRESS DEV LO B/U HTRS ON alarm actuates. The Unit Supervisor has directed the applicant to respond to the alarm. Procedures 0POP09-AN-04M8, "Annunciator Lampbox 04M8 Response Instructions" and 0POP04-RP-0001, "Loss of Automatic Pressurizer Pressure Control" will be used to determine the cause of the alarm. The applicant must identify that a normal pressurizer spray valve is open and stuck open. Actions to mitigate the pressure drop will fail resulting in manually tripping the reactor. [NEW]

S8: Unit 1 has just experienced a reactor trip due to loss of offsite power. Applicant is directed to perform immediate actions of 0POP05-EO-EO00, "Reactor Trip or Safety Injection", from memory. Applicant will have to manually close main steam to deaerater valves, and emergency start "B" EDG and close its output breaker. [NEW]

P1: Unit 1 is at 100% and experiences a loss of all feedwater. Reactor did not automatically trip when S/G LO-LO level setpoint was reached. Manual actions to trip the reactor in the control room are not successful. The unit supervisor has directed the applicant to manually trip the reactor. The applicant must correctly perform actions to manually trip the reactor in accordance with 0POP05-EO-EO00, "Reactor Trip or Safety Injection." This JPM will begin outside the control room. The applicant must ultimately operate the reactor trip breakers in the Rod Control Equipment Room. This is from the licensee's bank labeled JPM NO: P1 [NRC LOT-18].

P2: Unit 1 EDG 11 is running. Unit Supervisor has directed a local manual stop of the EDG. Applicant will use 0POP02-DG-0001, Step 8.2 to complete this action along with DG Post Run Checklist 2. Applicant will find that the Diesel Air Intake Butterfly Valve has not fully latched. Applicant must successfully reset the valve locally at the turbocharger or by depressing the engine overspeed shutdown air reset valve at the overspeed trip governor. [NEW]

P3: Unit 1 is a full power. Unit Supervisor has directed that both trains of spent fuel pool cooling be placed in service. Initial condition will have the "A" train running. Applicant must correctly step thru station procedure 0POP02-FC-0001, "Spent Fuel Pool Cooling and Cleanup System" Section 6.1 to start train B of spent fuel pool cooling. Step 6.1.1 has already been completed. [NEW]

Facility: South Texas Project **Scenario No.: 1** **Op-Test No.: LOT 25 NRC**

Examiners: ______________________

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Operators: ______________________

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Initial Conditions:
- **Reactor is at 100%.**
- **CCW Pump 1A running**

Turnover:
- **Orders have been given to lower reactor power to 98%.**

Critical Tasks:
- **CT-3: Manually initiate at least one train of containment spray**
- **CT-33: Isolate the LOCA outside containment before transitioning out of EC12**
- **CT-11: Close CIVs so that at least one valve is closed on each Phase A penetration.**

Event No.	Malf. No.	Event Type*	Event Description
1 (0 min)		R (RO, SRO)	Lower Reactor power to 98%
2 (10 min)		I (RO, SRO) TS (SRO)	Loop A T-hot instrument fails HI
3 (20 min)		C (BOP, SRO) TS (SRO)	EAB Battery Rm Exh Fan 11C shaft shears. Battery Room Exh Fan 11B fan trips when attempted to start.
4 (30 min)		C (ALL)	Instrument Air Leak. Unisolable – Reactor Trip.
5 (40 min)		M (ALL)	LBLOCA.
6 (50 min)		C (RO, SRO)	HPSI B pump seal failure following RAS.
7 (60 min)		C (BOP, SRO)	Containment Spray Pumps fail to start
8 (N/A)		C (BOP, SRO)	Phase A containment isolation valves for CVCS Letdown (OCIV MOV-0024, ICIV MOV-0023, LTDN HDR ISOL FV-0011) fail to close.

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

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)	3
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. Pre-Identified Critical tasks (≥2)	2

STP LOT-25
NRC Scenario #1 Description

Initial Conditions:

Event 1: Operators will lower reactor power to 98% per 0POP03-ZG-0008, “Power Operations.” This will have the BOP lower turbine load coordinating with the RO to perform a boration in accordance with 0POP02-CV-0001, “Makeup to the Reactor Coolant System,” Section 10.0.

Event 2: Loop A T-hot instrument fails HI. Operators will respond in accordance with 0POP04-RP-0004, “Failure Of RCS Loop RTD Protection Channel”. This will prompt multiple alarms (5M02, A-8, A-5, A-6, C-6, D-6, E-6, F-6, A-4). This will require entry into Technical Specification 3.3.1 Action 2 for functions 8 and 9.

Event 3: Respond to EAB Battery Room Fan alarm in accordance with 0POP09-AN-22M3, “Annunciator Lampbox 22M03 Response Instructions,” Section “BATT ROOM EXH FAN TRBL.” SRO should be looking at TS 3.3.3.5 and at TS Bases Table B 3.3.5-1, function 5e.

Event 4: Respond to a leak in the instrument air system per 0POP04-IA-0001, “Loss of Instrument Air.” Leak will be small enough to allow operators to step thru the procedure. However, attempts to locate and isolate the leak will be unsuccessful. Pressure will trend downward to less than 60 psig. This will require a manual reactor trip per the procedure.

Event 5: Respond to a large break LOCA (Pressurizer Low Pressure Reactor Trip) in accordance with 0POP05-EO-EO00, “Reactor Trip or Safety Injection.” Step 15 of this procedure will lead the operators to 0POP05-EO-EO10.

Event 6: While in 0POP05-EO-EO10, the B HPSI pump will have a seal failure which causes FHB radiation levels to rise greater than normal levels. Step 20 will direct operators to 0POP05-EO-EC12, “LOCA OUTSIDE CONTAINMENT”

Event 7: Containment Spray pumps will fail to automatically start. Operators will manually initiate containment spray per 0POP05-EO-EO00, Step 6.

Event 8: 0POP05-EO-EO00 Step 5 will have operators verify actions per Addendum 5. Addendum 5, Step 6b has operators check Phase A valves.

STP LOT-25
NRC Scenario #1 Description

Termination: Completion of 0POP05-EO-EC12, "LOCA OUTSIDE CONTAINMENT," Step 4.

Critical Tasks:

- **CT-3: Manually initiate at least one train of containment spray**
- **CT-11: Close at least one containment isolation valve on each Phase-A penetration**
- **CT-32: Isolate the LOCA outside containment before transitioning out of EC12**

Source: New Scenario

NOTE: Continue with additional Scenarios

Critical Tasks:

	CT-3
Critical Tasks	Manually actuate containment cooling, before completion of EO00 Addendum 5.
EVENT	7
Safety significance	Failure to manually actuate one train of containment spray under the postulated conditions demonstrates the inability of the crew to “recognize a failure or an incorrect automatic actuation of an ESF system or component.” In this case, one train of containment spray can be manually actuated from the control room. Therefore, failure to manually actuate one train of containment spray also represents a failure by the crew to demonstrate the ability to “effectively direct or manipulate engineered safety feature (ESF) controls that would prevent (degraded emergency core cooling system (ECCS) … capacity).” Additionally, under the postulated plant conditions, failure to manually actuate one train of containment spray (when it is possible to do so) results in a failure to prevent “a significant reduction of safety margin beyond that irreparably introduced by the scenario. The acceptable results obtained in the FSAR analyses of containment response to high-energy line breaks are predicated on the assumption that, at the very least, one train of safeguards actuates, including containment cooling. If one train of containment spray is not actuated, the FSAR assumptions and results are invalid. Because compliance with the assumptions of the FSAR is part of the facility license condition, failure to manually actuate at least one train of containment spray (under the postulated conditions and when it is possible to do so) constitutes a violation of the license condition.
Cueing	• Indication and/or annunciation that one train of containment spray is required • Indication and/or annunciation that one train of containment spray is not entirely actuated
Performance indicator	Manipulation of controls as required to actuate at least one train of containment spray
Performance feedback	Indication and/or annunciation that at least one train of containment spray is actuated: -Spray pump indicates running -Spray flow is indicated -Containment pressure decreasing.
Justification for the chosen performance limit	Failure to manually actuate one train of containment spray under the postulated conditions demonstrates the inability of the crew to “recognize a failure or an incorrect automatic actuation of an ESF system or component.”
PWR Owners Group Appendix	CT-3, Manually actuate containment cooling

	CT-11	CT-3
Critical Tasks	Close containment isolation valves such that at least one valve is closed on each Phase-A penetration, prior to completion of E0, Addendum 5.	
EVENT	8	
Safety significance	Actuation of Phase A containment isolation is required in order to ensure that the degree of containment integrity assumed in the accident analysis is actually established. Primary containment integrity ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leakage rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation imposed by the plant technical specifications, will limit the site boundary radiation doses to [within the dose guideline values of 10 CFR 100 during accident conditions]. The operability of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive materials to the environment will be consistent with the assumptions used in the analyses for a LOCA. In summary, the containment is a fission-product barrier. Under the plant conditions postulated, failure to close at least one containment isolation valve on each Phase A penetration results in unnecessary degradation of a fission-product barrier that is relied upon in the safety analysis for the specific accident in progress, that is, for a LOCA. For the containment barrier to possess the degree of integrity assumed in the FSAR analysis of the radiological consequences of a LOCA, at least one isolation valve on each Phase A penetration must be closed. Aside from the issue of containment integrity, failure to close containment isolation valves such that at least one valve on each critical Phase A penetration is closed represents a failure by the crew to “demonstrate the (ability to) recognize a failure or an incorrect automatic actuation of an ESF system or component.”	
Cueing	-Indication and/or annunciation that SI is actuated AND -One or more of the following: • Absence of closed valve position indication on all (both) containment isolation valves on one or more Phase A penetrations • Open valve position indication on all (both) containment isolation valves on one or more Phase A penetrations • ESF system status lamps show that all (both) containment isolation valves on one or more Phase A penetrations are not closed	
Performance indicator	Manipulation of controls as required to close at least one containment isolation valve on each Phase A penetration • Closed valve position indication for at least one containment isolation valve on each Phase A penetration	
Performance feedback	ESF system status lamps show that at least one containment isolation valve is closed on each Phase A penetration	
Justification for the chosen performance limit	Failure to ensure full containment isolation when it is possible to do so, upon completion of EO00 Addendum 5, represents an unacceptable delay in positioning ESF components which failed to reposition automatically, and which are required to satisfy the plant’s licensing basis.	
PWR Owners Group Appendix	CT-11, Close containment isolation Phase A valves	

	CT-32
Critical Tasks	Isolate the LOCA outside containment before transitioning out of EC12 Acceptance Criteria: Completion of 0POP05-EO-EC12, Step 3e (check closed 1-SI-MOV-0004B).
EVENT	6
Safety significance	Failure to isolate a LOCA outside containment (that can be isolated) degrades containment integrity beyond the level of degradation irreparably introduced by the postulated conditions. It also constitutes misoperation or incorrect crew performance that fails to prevent “degradation of any barrier to fission product release” and eventually leads to “…degraded emergency core cooling (ECCS)...capacity.” Thus, failure to perform the critical task under the postulated plant conditions leads to a “significant reduction of safety margin beyond that irreparably introduced by the scenario.” It also represents a “significant degradation in the mitigative capability of the plant.” Containment integrity is degraded because the containment fission-product barrier is bypassed via a pathway that leads from the RCS to the auxiliary building. Although the degraded status of the containment fission-product barrier is not due to the crew’s action (was not initiated by operator error), continuation in the degraded status is a result of the crew’s failure to perform the critical task. Failure to perform the critical task eventually leads to degraded ECCS capacity because the LOCA outside containment depletes the RWST inventory without causing a corresponding increase in the containment sump inventory. Thus, failure to isolate the LOCA can result in a situation in which all ECCS pumps taking suction on the RWST must be stopped because the RWST is empty and emergency coolant recirculation is unavailable.
Cueing	Indication and/or annunciation that SI is actuated and is required • RCS pressure AND Indication and/or annunciation of abnormally high radiation in the auxiliary building • Area radiation monitoring system • Process and effluent radiation monitoring system
Performance indicator	Manipulation of controls as required to isolate the LOCA; either of the following indicators, depending upon the location of the break: • All normally closed containment isolation valves indicate closed • Normally open containment isolation valve(s) upstream of the break indicate closed
Performance feedback	Indication of increasing RCS pressure.
Justification for the chosen performance limit	Completion of EC12 is the minimum acceptable time by which a LOCA bypassing containment should be secured, when it is possible to do so.
PWR Owners Group Appendix	CT-32, Isolate LOCA outside containment

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

Facility: South Texas Project **Scenario No.: 3** **Op-Test No.: LOT 25 NRC**

Examiners: ______________ **Operators:** ______________

Initial Conditions:	Critical Tasks:
• Reactor power is at 97% • CCW Pump 1A running **Turnover:** • Raise power to 99% with turbine and dilution.	• CT-1: Manually trip the reactor by dispatching operator to open RTBs • CT-52: Insert negative reactivity via emergency boration.

Event No.	Malf. No.	Event Type*	Event Description
1		R (ALL)	Raise reactor power to 99%
2		I (BOP, SRO)	SG 1A controlling feedwater flow channel fails low**
3		C (RO, SRO) TS (SRO)	30 GPM RCS leak (isolable) on the letdown line in containment.
4		I (RO, SRO) TS (SRO)	Power Range Nuclear Instrument Channel A fails HIGH
5		C(BOP, SRO) TS(SRO)	Leak on outlet piping of CCW Pump 1A (in between the isolation valves for CCW Pump 1A (surge tank LLA 66.7%)**
6		M (ALL)	Manual Reactor trip from Event 5. ATWS – local action to open RTBs required.
7		C (RO,SRO)	2 Control Rods stuck partially withdrawn > 18 steps following reactor trip, and cannot be inserted
8		C (BOP, SRO)	Main Generator output breaker fails to open.

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

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

STP LOT-25
NRC Scenario #3 Description

Initial Conditions: Reactor power is at 97%. Directions given to the oncoming crew to borate 50 gallons.

Event 1: Operators will raise power to 99% in accordance with 0POP03-ZG-0008, “Power Operations.” This will have the BOP raise turbine load coordinating with the RO to perform a dilution in accordance with 0POP02-CV-0001, “Makeup to the Reactor Coolant System,” Section 10.0.

Event 2: Operators respond to the controlling feedwater flow channel failure (LOW) for S/G 1A per 0POP04-FW-0001, “Loss of Steam Generator Water Level Control.”

Event 3: Operators will respond to RCS excessive leakage per 0POP04-RC-0003, “Excessive Leakage.” The crew will place excess letdown in service since Normal Letdown and Charging will be isolated in Addendum 3, step 4.0. The SRO will need to evaluate RCS leakage per Technical Specification 3.4.6.2, Action b). The leakage should be within limits once normal letdown is isolated per procedure.

Event 4: Operators respond to power range nuclear instrument channel A (fail HIGH) per 0POP04-NI-0001, “Nuclear Instrument Malfunction.” There should also be an assessment for entry into Technical Specification 3.3.1 (Table 3.3-1 Function 2 and 3)

Event 5: Respond to an unisolable leak from the outlet of the “A” CCW pump piping in accordance with 0POP04-CC-0001, “Component Cooling Water System Leak.” This will require entry into T.S. 3.7.3 Action a (7 day). Crew is expected to need to trip reactor at < 61.5% surge tank level.

Event 6: Continuing from previous event. Operators should enter 0POP05-EO-EO00 for manually tripping the reactor. Manual trip of the reactor from the control room will be unsuccessful. 480V LC 1K1 and 1L1 feeder breakers fail to open. Step 1 in the RNO requires entry into 0POP05-EO-FRS1, “Response to Nuclear Power Generation – ATWS.” Control room should direct plant operator to locally open the trip and bypass breakers.

Event 7: When the plant operator locally trips the RTBs, 2 control rods will fail to fully insert, > 18 steps. This is expected to occur very shortly after being directed in FRS1, before the crew has commenced emergency boration. The crew will use the CIP page to transition back to EO00 before they get to FRS1 step 4, which directs emergency boration. In this case, they will proceed thru EO00 until step 4.RNO, at which point they'll transition to ES01. ES01 Step 4 will direct emergency borating a specific volume for each rod that is not inserted. If there is a delay in opening RTBs in FRS1, the crew will emergency borate in that procedure, which is also satisfactory.

Event 8: Main Generator output breaker will fail to open. Operators respond per the RNO column in procedure 0PO05-EO-EO00.

STP LOT-??
NRC Scenario #3 Description

Termination: Completion of 0POP05-EO-ES01, “Reactor Trip Response”

Critical Tasks:

- **E6: CT-1: Manually trip the reactor from the control room or by opening feeder breakers**
- **E7: CT-52: Insert negative reactivity via emergency boration or insert rods.**

Source: New Scenario

NOTE: Continue with additional Scenarios

Critical Tasks:

	CT-1
Critical Tasks	Manually trip the reactor by dispatching an operator to open reactor trip and bypass breakers. Acceptance Criteria: prior to completing Step 1 of 0POP05-EO-EO00, Reactor Trip or Safety Injection, or Step 1 of 0POP05-EO-FRS1, RESPONSE TO NUCLEAR POWER GENERATION – ATWS, whichever comes first. Note: Step one is considered completed after performing the immediate actions AND the read through of the step.
EVENT	6
Safety significance	Failure to manually trip the reactor causes a challenge to the subcriticality CSF beyond that irreparably introduced by the postulated conditions. Additionally, it constitutes an incorrect performance that "necessitates the crew taking compensating action that would complicate the event mitigation strategy" and demonstrates the inability of the crew to "recognize a failure or an incorrect automatic actuation of an ESF system or component." The ERG Background Document for E-0 states that one function of E-0 is to verify that all required automatic protective actions occur before transitioning the crew to the appropriate ORG. The verification is important because the subsequent ORGs assume that protective systems will protect all CSFs while the ORG is implemented. The first of the E-0 verifications ensures that automatic reactor trip occurs. This highest priority verification ensures that the core heat production does not exceed the design capability of the safeguards heat removal systems, which is limited to core decay heat plus RCP heat. Any heat production above this value causes an increase in RCS temperature and pressure. If the reactor fails to automatically trip (as it does in the postulated conditions), the core fission heat production exceeds safeguards heat removal capability, leading to an RCS pressure increase. The consequences of the RCS pressure increase are further discussed in the Background Document for FR-S.1 as follows: *If the RCS is overpressurized during the transient (ASME Service Level C is used by the NRC for this purpose)…, the possibility that failure of the RCS pressure boundary may occur has been assumed by the NRC.* In addition, DNBR is a major concern. As noted in the Background Document for FR-S.1: *If the DNBR decreases too far, the possibility exists that fuel damage will occur. Once released into the coolant, this radioactivity is potentially capable of being released into containment and/or to the environment.* In either case (RCS overpressurization leading to pressure boundary failure or DNBR reduction leading to fuel damage), a fission product barrier is degraded. The RCS pressure excursion resulting from failure to manually trip the reactor (when manual trip capability is functional) can lead to lifting of PRZR PORVs and/or safety valves that would otherwise be unnecessary. Such relief/ safety valve cycling itself constitutes, in this case, an unnecessary degradation of the RCS pressure boundary and the needless introduction of a potential challenge to the integrity of that fission product barrier (should a safety / relief valve fail to close). If the reactor fails to automatically trip, a manual reactor trip by the crew is the most direct and fastest means to stop the excess heat production and to preclude the RCS pressure increase. If a manual reactor trip is not possible or not performed, other means to shut down the reactor are available, but most require additional time, which allows the RCS pressure to increase. With time being so important, the ERG directs the crew to transition immediately to FR-S.1 when automatic and manual reactor trips fail. Thus, not tripping the reactor when it is possible to do so forces an immediate extreme challenge to the subcriticality CSF. Additionally, the incorrect performance of failing to trip the reactor necessitates the crew taking compensating action that seriously complicates the event mitigation strategy. This misoperation constitutes a "significant reduction of safety margin beyond that irreparably introduced by the scenario."
Cueing	-Failure of the control rods to insert when manipulating reactor trip switch. -Failure of 480V LC 1K1 and 1L1 feeder breakers to open.
Performance indicator	Verbally direct a plant operator to open Reactor trip and bypass breakers
Performance feedback	Indications of reactor trip • Control rods at the bottom of core • Neutron flux decreasing
Justification for the chosen performance limit	Subcriticality is the highest critical safety function. Failure to take immediate action to satisfy this safety function when procedurally directed to do so constitutes a "significant reduction of safety margin beyond that irreparably introduced by the scenario."
PWR Owners Group Appendix	CT-1, Manually trip the reactor.

	CT-52
Critical Tasks	Insert negative reactivity into the core by establishing emergency boration flow to the RCS. Acceptance Criteria: Before completion of 0POP05-ES01.
EVENT	7
Safety significance	A subcritical core is verified if all rods are at the bottom according to the rod bottom lights and the rod position indicators. If these indications reveal that one rod is not inserted, no immediate action is required since the core is designed for adequate shutdown margin with one rod stuck out. However, if more than one rod fails to insert fully, the shutdown reactivity margin must be made up through emergency boration to account for the reactivity worth of the stuck rods. Failure to insert negative reactivity, under the postulated plant conditions, results in an unnecessary situation in which the reactor remains critical or returns to a critical condition. Performance of the critical task would make the reactor subcritical and provide sufficient shutdown margin to prevent (or at least minimize the power excursion associated with) any subsequent return to criticality. Failure to insert negative reactivity constitutes misoperation or incorrect crew performance which fails to prevent “incorrect reactivity control (such as failure to initiate emergency boration… or manually insert control rods).” The basic strategy of FR-S.1 is as follows: • Perform manual actions to reduce core power (such as manually driving control rods inward) and verify automatic actions such as turbine trip and AFW actuation • Initiate emergency boration of the RCS • Check for possible sources of positive reactivity and eliminate them • Verify subcriticality This strategy is based on making the reactor subcritical and providing shutdown margin as rapidly as possible. Thus, the actions to reduce core power and provide shutdown margin are prioritized according to how quickly they can be performed from the control room. Aside from the normal method of reactor trip, de-energizing the rod drive MG sets is the fastest method to shut down the reactor and provide shutdown margin (provided that it is feasible, on a plant-specific basis, to de-energize the MG sets from the control room). The second and third fastest methods are to insert control rods and to establish emergency boration flow to the RCS, respectively. It is true that local operator actions might result in reactor trip, which – if and when it occurs – would shut down the reactor faster than boration (and faster than rod insertion). However, it is anticipated that effecting the local actions will be time-consuming and that actions that can be implemented from the control room should be given precedence.
Cueing	Indication that 2 or more rods are not fully inserted
Performance indicator	Manipulation of controls in the control room as required to initiate the insertion of negative reactivity into the core via emergency boration.
Performance feedback	Indication of borated water being injected into the core at greater than or equal to 50 GPM.
Justification for the chosen performance limit	Subcriticality is the highest critical safety function. Failure to take immediate action to satisfy this safety function when procedurally directed to do so constitutes a “significant reduction of safety margin beyond that irreparably introduced by the scenario.”
PWR Owners Group Appendix	CT-52, Insert negative reactivity into the core.

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

Facility: South Texas Project **Scenario No.: 4** **Op-Test No.: LOT 25 NRC**

Examiners: ______________________ **Operators:** ______________________

Initial Conditions:
- **Mode 2 at 10E-8 AMPS Power**
- **SU SGFP #14 in service.**

Turnover:
- **Raise power to 3% (NIs)**

Critical Tasks:
- **CT-11: E6: Close Phase A CIV**
- **CT-6: E7: Start at least 1 HPSI pump**
- **CT-16: E5: Trip all RCPs**

Event No.	Malf. No.	Event Type*	Event Description
1		R (RO, SRO)	Withdraw control rods to raise power to 3%
2		I (RO, SRO) TS (SRO)	Once power is at 2%, a single rod in control bank D will slip 12 steps in. **
3		I (RO, SRO)	VCT Level transmitter LT-0112 fails high.
4		C (BOP, SRO) TS (SRO)	Inadvertent start of AFW Pump #13**
5		C (BOP, SRO)	Running Closed Loop ACW Pump trips. Standby pump fails to start automatically.
6		M (ALL)	PZR Vapor Space SBLOCA**
7		C (BOP, SRO)	Seal Return Isolation valves, MOV-0077 and MOV-0079 fail to automatically close
8		C (BOP, SRO)	ALL HPSI pumps fails to start on SI signal

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

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Malfunctions after EOP entry (1–2)	2
2. Abnormal events (2–4)	3
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. Pre-Identified Critical tasks (≥2)	3

STP LOT-25
NRC Scenario #4 Description

Initial Conditions:

Event 1: Perform power ascension per 0POP03-ZG-004, “Reactor Startup”

Event 2: Operators will respond per 0POP09-AN-05M3, “Annunciator Lampbox 5M03 (Window D-5) Response Instructions” for ROD SUPV MNTR ROD POSITION TRBL. Operators will check that one of the rods slipped 12 steps inward. The annunciator response procedure will direct operators to enter 0POP04-RS-0001, “Control Rod Malfunction.” The rod will ultimately be realigned back to its control bank.

Event 3: Operators will respond per 0POP09-AN-04M8, “Annunciator Lampbox 04M8 Response Instructions” for Window E-2, VCT LEVEL HI/LO.

Event 4: SRO will address compliance with Technical Specifications. Operators respond to stabilize the plant as allowed per Conduct of Operations Manual, Chapter 2 and manually secure the running AFW pump.

Event 5: Operators respond per procedure 0POP09-AN-09M1, Window 9M01-D-4, “ACW CLOSE LOOP PUMP TRIP” for the initial pump trip. This will subsequently direct operators to 0POP04-AC-0003, “Loss of Closed Loop Auxiliary Cooling Water” when header pressure is not greater than 67psig since the standby pump did not automatically start.

Event 6: Operators respond per 0POP05-EO-EO10, “Loss of Reactor or Secondary Coolant.”

Event 7: While performing 0POP05-EO-EO00, “Reactor Trip or Safety Injection”, Addendum 5, SI Equipment Verification, CIS Phase A will be checked per Addendum 1, and operators will note that all the valves did not automatically shut. This will require the operators to manually shut them.

Event 8: Operators should verify per 0POP05-EO-EO00, “Reactor Trip or Safety Injection” Addendum 5 that none of the HPSI pumps are running and should be manually started.

STP LOT-25
NRC Scenario #1 Description

Termination: Completion of 0POP05-EO-EO10, “Loss of Reactor or Secondary Coolant,” Step 8.

Critical Tasks:

- **CT-11: E7: Close Phase A CIV**
- **CT-6: E8: Start at least 1 HPSI pump**
- **CT-16: E6: Trip all RCPs**

Source: New

NOTE: Continue with additional Scenarios

Critical Tasks:

	CT-6
Critical Tasks	Establish flow from at least one HHSI Pump before transitioning out of EO00 during a Small Break LOCA where RCS pressure remains between 400 psig and 1680 psig, AND before tripping RCPs.
EVENT	7
Safety significance	Failure to manually start at least one HHSI pump constitutes misoperation or incorrect crew performance in which the crew does not prevent "degraded emergency core cooling system (ECCS) … capacity." In this case, at least one HHSI pump can be manually started from the control room. Therefore, failure to manually start a HHSI pump also represents a failure by the crew to "demonstrate the following abilities: • Effectively direct or manipulate engineered safety feature (ESF) controls that would prevent a significant reduction of safety margin beyond that irreparably introduced by the scenario • Recognize a failure or an incorrect automatic actuation of an ESF system or component" Additionally, under the postulated plant conditions, failure to manually start a HHSI pump (when it is possible to do so) is a "violation of the facility license condition." The acceptable results obtained in the FSAR analysis of a small-break LOCA are predicated on the assumption of minimum ECCS pumped injection. The analysis assumes that a minimum pumped ECCS flow rate, which varies with RCS pressure, is injected into the core. The flow rate values assumed for minimum pumped injection are based on operation of one each of the following ECCS pumps: high-head SI pump, and low-head SI pump. Operation of this minimum required complement of ECCS injection pumps is consistent with the FSAR assumption that only minimum safeguards are actuated. Failure to perform the critical task means that the plant is needlessly left in an unanalyzed condition. Performance of the critical task would return the plant to a condition for which analysis shows acceptable results. Because compliance with the assumptions of the FSAR is part of the facility license condition, failure to perform the critical task (under the postulated plant conditions) constitutes a violation of the license condition.
Cueing	Indication and/or annunciation that high-head SI pump injection is required: -SI actuation -RCS pressure below the shutoff head of the high-head SI pump AND Indication and/or annunciation that no HHSI pump is injecting into the core: -Control switch indication that the circuit breakers or contactors for all HHSI pumps are open -All HHSI pump discharge pressure indicators read zero -All flow rate indicators for HHSI pump injection read zero
Performance indicator	Manipulation of controls as required to establish flow from at least one HHSI pump: -Control switch indication that the circuit breaker or contractor for at least one Charging/SI pump is closed.
Performance feedback	-Indication and/or annunciation that at least one charging/SI pump is injecting -Flow rate indication of injection from at least one SI pump
Justification for the chosen performance limit	Completion of EO00 represents the minimally acceptable timeframe by which a minimally competent crew should be expected to start ECCS components which failed to start automatically, and which are required to satisfy the plant's licensing basis.
PWR Owners Group Appendix	CT-6, Establish flow from at least one SI pump

	CT-11
Critical Tasks	Close containment isolation valves such that at least one valve is closed on each Phase-A penetration, prior to completion of E0, Addendum 5.
EVENT	7
Safety significance	Actuation of Phase A containment isolation is required in order to ensure that the degree of containment integrity assumed in the accident analysis is actually established. Primary containment integrity ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leakage rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation imposed by the plant technical specifications, will limit the site boundary radiation doses to [within the dose guideline values of 10 CFR 100 during accident conditions]. The operability of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive materials to the environment will be consistent with the assumptions used in the analyses for a LOCA. In summary, the containment is a fission-product barrier. Under the plant conditions postulated, failure to close at least one containment isolation valve on each Phase A penetration results in unnecessary degradation of a fission-product barrier that is relied upon in the safety analysis for the specific accident in progress, that is, for a LOCA. For the containment barrier to possess the degree of integrity assumed in the FSAR analysis of the radiological consequences of a LOCA, at least one isolation valve on each Phase A penetration must be closed. Aside from the issue of containment integrity, failure to close containment isolation valves such that at least one valve on each critical Phase A penetration is closed represents a failure by the crew to “demonstrate the (ability to) recognize a failure or an incorrect automatic actuation of an ESF system or component.”
Cueing	-Indication and/or annunciation that SI is actuated AND -One or more of the following: • Absence of annunciation that Phase A isolation is actuated • Absence of closed valve position indication on all (both) containment isolation valves on one or more Phase A penetrations • Open valve position indication on all (both) containment isolation valves on one or more Phase A penetrations • ESF system status lamps show that all (both) containment isolation valves on one or more Phase A penetrations are not closed
Performance indicator	Manipulation of controls as required to close at least one containment isolation valve on each Phase A penetration • Closed valve position indication for at least one containment isolation valve on each Phase A penetration
Performance feedback	ESF system status lamps show that at least one containment isolation valve is closed on each Phase A penetration
Justification for the chosen performance limit	Failure to ensure full containment isolation when it is possible to do so, upon completion of EO00 Addendum 5, represents an unacceptable delay in positioning ESF components which failed to reposition automatically, and which are required to satisfy the plant’s licensing basis.
PWR Owners Group Appendix	CT-11, Close containment isolation Phase A valves

	CT-16
Critical Tasks	Manually trip RCPs Acceptance Criteria: Within 5 minutes of RCP Trip Criteria met. (At least 1 HHSI Pump running and RCS Pressure < 1430 psig)
EVENT	6
Safety significance	Failure to trip the RCPs under the postulated plant conditions leads to core uncovery and to fuel cladding temperatures in excess of 2200°F, which is the limit specified in the [ECCS acceptance criteria]. Thus, failure to perform the task represents misoperation or incorrect crew performance in which the crew has failed to prevent “degradation of...{the fuel cladding} ...barrier to fission product release” and which leads to “violation of the facility license condition.” The analysis presented in the FSAR for a SBLOCA typically assumes that the RCPs trip because of a loss of offsite power that coincides with the reactor trip. However, during a SBLOCA, offsite power might remain available and RCPs might continue to run for some period of time. Following the accident at TMI-2, the NRC expressed concern about RCP operation during a SBLOCA. In response, the WOG sponsored analyses to determine when the RCPs must be tripped if power remains available. It was determined that manually tripping the RCPs before RCS inventory is depleted to less than the “critical inventory” results in a peak cladding temperature about the same as the PCT in the FSAR analysis. Manually tripping the RCPs before depletion below the critical inventory conservatively ensures that PCT remains below 2200°F. Continued RCP operation after the RCS is depleted to the critical inventory results in more mass being lost through the break than would otherwise be the case. More mass is lost because RCP operation causes liquid to continue to flow from the break, whereas, in an RCS depleted to the critical inventory, liquid loss from the break stops when RCPs are stopped.
Cueing	• Indications of a SBLOCA AND • Indication and/or annunciation of safety injection AND • At least 1 HHSI Pump Running AND • RCS Pressure < 1430 psig
Performance indicator	Manipulation of controls as required to trip all RCPs • RCP breaker position lights indicate breaker open
Performance feedback	Indication that all RCPs are stopped • RCP breaker position lights • RCP flow decreasing • RCP motor amps decreasing
Justification for the chosen performance limit	5 minutes is the minimum time by which, if the RCPs are tripped after meeting trip criteria, fuel PCT is assured of remaining below 2200F.
PWR Owners Group Appendix	Ct-16, Manually trip RCPs

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

Facility: South Texas Project **Scenario No.: 5** **Op-Test No.: LOT 25 NRC**

Examiners: ______________________ **Operators:** ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions:

- **Unit 1 is at 85% power. EOC (IC #26)**

Turnover:

- **The crew will continue to raise power to 100%**
- **A line of thunderstorms just passed through Markham and are tracking south towards the STP site.**

Event No.	Malf. No.	Event Type*	Event Description
1		R(ALL)	Raise power to 90%
2		C (RO, SRO),	PZR level channel LT-0465 fails high
3		C (BOP,SRO) TS (SRO)	ECW Pump #13 trips
4		I (BOP, SRO)	SG 1A controlling steam flow channel FT-0512 fails high
5		I(RO, SRO) TS (SRO)	Loop A Cold Leg RTD TI-0410B Fails Low
6		M (ALL)	Main Steam Line Break SG 1B Inside Containment **(CT-17)**
7		C (BOP,SRO)	1B MSIV fails to close **(CT-12)**
8		C (BOP,SRO)	EAB HVAC fails to actuate

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

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Malfunctions after EOP entry (1–2)	2
2. Abnormal events (2–4)	4
3. Major transients (1–2)	1
4. EOPs entered/requiring substantive actions (1–2)	2
5. EOP contingencies requiring substantive actions (0–2)	1
6. Pre-Identified Critical tasks ($\geq$2)	2

STP LOT-25
NRC Scenario #5 Description

Initial Conditions: Current reactor power level is at 85%. A line of thunderstorms are tracking from the south toward the site and have effected loads on the grid.

Event 1: The crew is re-commencing raising power per step 7.51 of 0POP03-ZG-0005, Plant Startup to 90%.

Event 2: The crew acknowledges and announces annunciators:

- 0POP09-AN-04M8/A6 "PRZR LEVEL HI RX TRIP ALERT"
- 0POP09-AN-04M8/C6 "PRZR LEVEL DEV HI B/U HTRS ON"

Event 3: The crew will get annunciator 02M4 C-7, ECW Pump 1C Trip 0POP09-AN-02M4. This will be based on the trip of ECW Pump #13. Standby pump will start automatically. This will require a Technical specification call for one loop Essential Cooling water loop being OOS (3.7.4). [need to check that there will be verifiable actions, with the standby pump auto starting]

Event 4: SG 1A controlling steam flow channel FT-0512 fails low. This will a SG1A STM/FW Flow Mismatch based upon 0POP09-AN-06M3, Window E-3. The crew will respond using 0POP04-FW-0001, Loss of Steam Generator Level Control.

Event 5: Loop A Cold Leg RTD T-0410B Fails Low. The crew will respond based upon annunciators using 0POP09-AN-05M2, A-6, C-6, and D-6. SRO will direct actions consistent on 0POP04-RP-0004, Failure of RCS Loop RTD Protection Channel.

Event 6: Main Steam Line Break on S/G 1B In Containment. Crew will enter 0POP-5-EO-E000 and proceed to step 13. RNO expected since break occurred, thus crew will go to 0POP05-EO-EO20, Faulted Steam Generator Isolation, Step 1 while completing 0POP-5-EO-E000 , Addendum 5. **(Critical Task)**

Event 7: 1B MSIV fails to close. Crew performing actions with 0POP09-EO-EC-Crew performing 0POP-5-EO-E000, Addendum 5 will get to step 2b and RNO will direct to close MSIV manually using 0POP-5-EO-E000, Addendum 6. This will take the operator to Addendum 6, Failing Air to MSIVs and MSIBs **(Critical Task)**

Event 8: EAB HVAC fails to actuate. While performing actions in 0POP05-EO-EO0000, Addendum 5, operator will get to step 14 to verify ventilation actuation. RNO for step 14b will direct crew to Manually place EAB HVAC in Emergency Recirc mode.

STP LOT-25
NRC Scenario #5 Description

Termination: The scenario will terminate after the crew performs actions in 0POP05-EO-EO0000, Addendum 5, operator will get to step 14 to verify ventilation actuation. RNO for step 14b will direct crew to Manually place EAB HVAC in Emergency Recirc mode.

Critical Tasks:

- Isolate faulted SG 1B
- Adequate response to failure of MSIV 1B to close from 0POP05-EO-E000, Addendum 5

Source: New

Critical Tasks:

	CT-17
Critical Tasks	Isolate faulted SG 1B, prior to exiting EO20 [tjf We may consider failing open a FWIV or other component that will require the crew to manually isolate, to bolster this CT.]
EVENT	6
Safety significance	Failure to isolate a faulted SG that can be isolated causes challenges to CSFs beyond those irreparably introduced by the postulated conditions. Also, depending upon the plant conditions, it could constitute a failure by the crew to “demonstrate the ability to recognize a failure or an incorrect automatic actuation of an ESF system or component. ”Failure to isolate a faulted SG can result in challenges to the following CSFs: • Integrity • Subcriticality • Containment (if the break is inside containment) The plant-specific FSAR typically presents an analysis for a steam system piping failure in which only a single SG blows down completely. Typically, the transient is analyzed for the case in which offsite power is available (and RCPs are running) and for the case in which offsite power is lost simultaneously with accident initiation (and RCPs trip). The analyses for some plants show that, at about 5 minutes into the transient, reactor vessel inlet temperature has dropped from its hot zero power value by the following amounts: • About 150°F, when all RCPs are running • About 250°F, when no RCP is running The reactor vessel inlet temperature cooldown transient is worse for the case in which no RCP is running, for the following reasons. Natural circulation produces much lower RCS loop flow rates than does forced circulation. Natural circulation provides a lower mass flow rate of RCS fluid to mix with and heat the subcooled SI fluid injected into the cold leg before that fluid reaches the reactor vessel inlet. For some plants, neither of these transients (blowdown of a single SG with or without RCPs running) constitutes an ORANGE path challenge to the integrity CSF. However, if the faulted SG is not isolated, the cooldown transient for reactor vessel inlet temperature could result in an ORANGE path challenge to the integrity CSF, especially if RCPs are not running. The break may be located inside containment. In such a case, the uncontrolled depressurization of multiple SGs typically exceeds the conditions assumed in the safety analysis of the containment response to a faulted SG. The resulting containment pressure transient is a more serious challenge to the containment CSF than is the transient resulting from blowdown of only a single SG. RCP operation tends to worsen the challenge to the containment CSF because of the higher rate of heat transfer through the faulted SG to the containment atmosphere.
Cueing	Both of the following: – Steam pressure and flow rate indications that make it possible to identify a single SG as faulted AND – Valve position and flow rate indication that AFW continues to be delivered to the faulted SG -- Main steamline isolation has not actuated

Performance indicator	Manipulation of controls as required to isolate the faulted SG
	-MSIVs indicate closed
	-Indication of feedline isolation
	-Feedwater control valves indicate closed
	-Feedline isolation valves indicate closed
	-Main feed pumps indicate tripped
	
	• Verifies all FWIV’s closed.*****
	• Verifies all FWIB’s closed.*****
	• Verifies all FW Preheater bypass valves closed. *****
	• Verifies all FW Regulating and Low Power FW Regulating Valves closed.*****
	• Isolates AFW flow to ‘B’ SG ***:**
	    • Resets SI
	    • Resets ESF load sequencers
	    • Resets SG LO-LO level AFW actuations
	    • Checks SG 1D intact

	• Closes 'B' SG AFW OCIV • Verifies SG "B" PORV closed * • Verifies SG 'B' Blowdown and sample isolation valves closed *
Performance feedback	All MSIVs indicate closed Any depressurization of intact SGs stops Steam flow indication from faulted SG decreases to zero RCS cooldown stops Main feedwater flow rate indication of zero AFW flow rate indication to faulted SG of zero
Justification for the chosen performance limit	
PWR Owners Group Appendix	CT-17, Isolate faulted SG

	CT-12
Critical Tasks	Manually complete main steamline isolation for MSIV 1B Acceptance Criteria: Close MSIV 1B before a severe (orange path) challenge develops to either the subcriticality or the integrity CSF or before transition to EC21, whichever happens first.
EVENT	7
Safety significance	Failure to close the MSIVs under the postulated plant conditions causes challenges to CSFs beyond those irreparably introduced by the postulated conditions. Additionally, such an omission constitutes a failure by the crew to “demonstrate (the ability to) recognize a failure or an incorrect automatic actuation of an ESF system or component,” and to “take one or more actions that would prevent a challenge to plant safety.” In the typical FSAR, the analysis for a large steamline break assumes steamline isolation within a short time frame, on the order of seconds. The analysis typically assumes a steam system piping failure in which a single SG blows down completely. That is, the analysis assumes a fault that can be isolated from all but one SG. However, in the plant conditions postulated for this critical task, the break is located downstream of the MSIVs. Thus, closure of all MSIVs would terminate all uncontrolled blowdown. In this case, there is no reason for even a single SG to completely depressurize. If the crew allows all MSIVs to remain open, then all SGs depressurize uncontrollably and unnecessarily. Uncontrolled depressurization of all SGs causes an excessive rate of RCS cooldown, well beyond the conditions typically analyzed in the FSAR. The excessive cooldown rate creates large thermal stresses in the reactor pressure vessel and causes rapid insertion of a large amount of positive reactivity. Thus, failure to close the MSIVs under the postulated conditions can result in challenges to the following CSFs: • Integrity • Subcriticality Additionally, the ERG Background Document for ECA-2.1 specifically states the following: It should be noted that this event (with an extensive cooldown and subsequent repressurization) may result in a challenge to the Integrity Critical Safety Function. In this case the Integrity Critical Safety Function Status Tree may direct the operator to FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION, for further actions. For some plants, the analysis for a large steamline break shows a return to reactor criticality caused by the large and rapid RCS cooldown, even though only a single SG is assumed to blowdown completely. Failure to isolate the SGs from the steamline break such that all SGs are allowed to blow down uncontrollably significantly worsens the power excursion. This worsening of the power excursion is unnecessary; it could be prevented simply by closing the MSIVs.
Cueing	Indication that main steamline isolation is required AND Indication that main steamline isolation has not actuated automatically – MSIVs indicate open – Indication of uncontrolled depressurization of SG 1B
Performance indicator	Manipulation of controls as required to manually actuate steamline isolation Directing plant operator to locally close MSIV 1B by securing air per Addendum 6 of EO00.
Performance feedback	MSIVs indicate closed Steam flow indication from all SGs decreases to zero All SGs stop depressurizing RCS cooldown stops
Justification for the chosen performance limit	
PWROG Appendix	CT-12, Manually actuate main steamline isolation for 1B

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

Facility: South Texas Project Date of Exam: 7-12-21 Operating Test No.: LOT 25

APPLICANT	EVENT TYPE	Scenario 1 SRO	Scenario 1 ATC	Scenario 1 BOP	Scenario 3 SRO	Scenario 3 ATC	Scenario 3 BOP	Scenario 4 SRO	Scenario 4 ATC	Scenario 4 BOP	Scenario SRO	Scenario ATC	Scenario BOP	TOTAL	MINIMUM(*) R	MINIMUM(*) I	MINIMUM(*) U
Crew A		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
RO ☐ SRO-I8 ■ SRO-U ☐	RX	1					1		**1**					**3**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C	2,3, 4,6, 7,8					2,4, 6,7		**2,3**					**12**	4	**4**	2
	MAJ	5					5		**6**					**3**	2	**2**	1
	TS	2,3												**2**	0	**2**	2
RO ☐ SRO-I9 ■ SRO-U ☐	RX			1		1		**1**						**3**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C			3,4, 6,7		3,4, 8		**2,3, 4,5, 7,8**						**13**	4	**4**	2
	MAJ			5		5		**6**						**3**	2	**2**	1
	TS							**2,4**						**2**	0	**2**	2
RO ☐ SRO-I11 ■ SRO-U ☐	RX		1		1									**2**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C		2,4, 8		2,3, 4,5, 7,8					**4,5, 7,8**				**13**	4	**4**	2
	MAJ		5		5					**6**				**3**	2	**2**	1
	TS				3,4									**2**	0	**2**	2

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the “at-the-controls” (ATC) and “balance-of-plant” (BOP) positions. Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-I *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a one-for-one basis.
3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant’s competence count toward the minimum requirements specified for the applicant’s license level in the right-hand columns.
4. For licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

Facility: South Texas Project	Date of Exam: 7-12-21	Operating Test No.: LOT 25

APPLICANT	EVENT TYPE	Scenario 1 CREW POSITION SRO	Scenario 1 CREW POSITION ATC	Scenario 1 CREW POSITION BOP	Scenario 4 CREW POSITION SRO	Scenario 4 CREW POSITION ATC	Scenario 4 CREW POSITION BOP	Scenario 5 CREW POSITION SRO	Scenario 5 CREW POSITION ATC	Scenario 5 CREW POSITION BOP	CREW POSITION SRO	CREW POSITION ATC	CREW POSITION BOP	TOTAL	MINIMUM(*) R	MINIMUM(*) I	MINIMUM(*) U
Crew B																	
RO ☐ SRO-I7 ■ SRO-U ☐	RX			1		**1**		1						**3**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C			**3,4, 6,7**		**2,3**		2,3, 4,5, 7,8						**12**	4	**4**	2
	MAJ			5		**6**		6						**3**	2	**2**	1
	TS							2,3, 5						**3**	0	**2**	2
RO ☐ SRO-I12 ■ SRO-U ☐	RX	1							1					**2**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C	**2,3, 4,6, 7,8**					**4,5, 7,8**		2,5					**12**	4	**4**	2
	MAJ	5					**6**		6					**3**	2	**2**	1
	TS	2,3												**2**	0	**2**	2
RO ☐ SRO-I3 ■ SRO ☐	RX		1		**1**					1				**3**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C		**2,4, 8**		**2,3, 4,5, 7,8**					3,4, 7,8				**13**	4	**4**	2
	MAJ		5		**6**					6				**3**	2	**2**	1
	TS				**2,4**									**2**	0	**2**	2

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the “at-the-controls” (ATC) and “balance-of-plant” (BOP) positions. Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-I *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a one-for-one basis.
3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant’s competence count toward the minimum requirements specified for the applicant’s license level in the right-hand columns.
4. For licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

Facility: South Texas Project | **Date of Exam: 7-12-21** | **Operating Test No.: LOT 25**

APPLICANT	EVENT TYPE	Scenarios 1			3			5						TOTAL	MINIMUM(*)		
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
Crew C		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
RO-1 ■ SRO-I □ SRO-U □	RX		1				1			1				3	1	1	0
	NOR													0	1	1	1
	I/C		2,4,8				2,4,6,7			3,4,7,8				11	4	4	2
	MAJ		5				5			6				3	2	2	1
	TS													0	0	2	2
RO-2 ■ SRO-I □ SRO-U □	RX			1		1								2	1	1	0
	NOR													0	1	1	1
	I/C			3,4,6,7		3,4,8								7	4	4	2
	MAJ			5		5								2	2	2	1
	TS													0	0	2	2
RO □ SRO-I3 ■ SRO-U □	RX	1			1				1					3	1	1	0
	NOR													0	1	1	1
	I/C	2,3,4,6,7,8			2,3,4,5,7,8				2,5					14	4	4	2
	MAJ	5			5				6					3	2	2	1
	TS	2,3			3,4									4	0	2	2
RO □ SRO-I □ SRO-U1 ■	RX							1						1	1	1	0
	NOR													0	1	1	1
	I/C							2,3,4,5,7,8						6	4	4	2
	MAJ							6						1	2	2	1
	TS							2,3,5						3	0	2	2

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls" (ATC) and "balance-of-plant" (BOP) positions. Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-I *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a one-for-one basis.
3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.
4. For licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

Facility: South Texas Project **Date of Exam: 7-12-21** **Operating Test No.: LOT 25**

APPLICANT	EVENT TYPE	Scenarios 3			4			5						TOTAL	MINIMUM(*)		
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
Crew D		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
RO ☐ SRO-I1 ■ SRO-U ☐	RX			1		**1**		1						**3**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C			**2,4, 6,7**		**2,3**		2,3, 4,5, 7,8						**12**	4	**4**	2
	MAJ			5		**6**		6						**3**	2	**2**	1
	TS							2,3, 5						**3**	0	**2**	2
RO ☐ SRO-I5 ■ SRO-U ☐	RX	1							1					**2**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C	**2,3,4 ,5,7, 8**					**4,5, 7,8**		2,5					**12**	4	**4**	2
	MAJ	5					**6**		6					**3**	2	**2**	1
	TS	3,4												**2**	0	**2**	2
RO ☐ SRO-I10 ■ SRO ☐	RX		1		**1**					1				**3**	1	**1**	0
	NOR													**0**	1	**1**	1
	I/C		3,4, 8		**2,3, 4,5, 7,8**					3,4 ,7, 8				**13**	4	**4**	2
	MAJ		5		**6**					6				**3**	2	**2**	1
	TS				**2,4**									**2**	0	**2**	2

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls" (ATC) and "balance-of-plant" (BOP) positions. Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-I *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a one-for-one basis.
3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.
4. For licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

Rev. 1

Facility: South Texas Project		Date of Exam: 7-12-21												Operating Test No.: LOT 25			
APPLICANT	EVENT TYPE	Scenarios												TOTAL	MINIMUM(*)		
		1			3			4									
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
Crew E		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
RO ☐ SRO-I2 ■ SRO-U ☐	RX	1					1		**1**					3	1	**1**	0
	NOR													0	1	**1**	1
	I/C	2,3, 4,6, 7,8					2,4, 6,7		**2,3**					12	4	**4**	2
	MAJ	5					5		**6**					3	2	**2**	1
	TS	2,3												2	0	**2**	2
RO ☐ SRO-I4 ■ SRO-U ☐	RX		1		1									2	1	**1**	0
	NOR													0	1	**1**	1
	I/C		**2,4, 8**		**2,3,4 ,5,7, 8**					**4,5 ,7, 8**				13	4	**4**	2
	MAJ		5		5					**6**				3	2	**2**	1
	TS				3,4									2	0	**2**	2
RO ☐ SRO-I6 ■ SRO ☐	RX			1		1		**1**						3	1	**1**	0
	NOR													0	1	**1**	1
	I/C			3,4, 6,7		3,4, 8		**2,3, 4,5, 7,8**						13	4	**4**	2
	MAJ			5		5		**6**						3	2	**2**	1
	TS							**2,4**						2	0	**2**	2

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls" (ATC) and "balance-of-plant" (BOP) positions. Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-I *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
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3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.
4. For licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.