Facility: Columbia Generating Station Date of Exam: 2/27/17

Tier	Group	RO K/A Category Points K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	SRO-Only Points A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			4	3	N/A		4	20			
	2	2	1	1				1	1			1	7			
	Tier Totals	5	4	4				5	4			5	27			
2. Plant Systems	1	2	2	2	2	3	3	3	3	2	2	2	26			
	2	1	1	1	2	1	1	1	1	1	1	1	12			
	Tier Totals	3	3	3	4	4	4	4	4	3	3	3	38			

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

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 outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the “Tier Totals” in each K/A category shall not be less than two). (One Tier 3 Radiation Control K/A is allowed if the K/A 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 associated 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’ importance ratings (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

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4				X			AA1.01 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Recirculation system (CFR: 41.7 / 45.6)	3.5	1
295003 Partial or Complete Loss of AC / 6					X		AA2.01 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Cause of partial or complete loss of A.C. power (CFR: 41.10 / 43.5 / 45.13)	3.4	2
295004 Partial or Total Loss of DC Pwr / 6						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	3
295005 Main Turbine Generator Trip / 3	X						AK1.03 Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP : Pressure effects on reactor level (CFR: 41.8 to 41.10)	3.5	4
295006 SCRAM / 1		X					AK2.04 Knowledge of the interrelations between SCRAM and the following: Turbine trip logic (CFR: 41.7 / 45.8)	3.6	5
295016 Control Room Abandonment / 7				X			AA1.06 Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Reactor water level (CFR: 41.7 / 45.6)	4.0	6
295018 Partial or Total Loss of CCW / 8			X				AK3.01 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Isolation of non-essential heat loads (CFR: 41.5 / 45.6)	2.9	7
295019 Partial or Total Loss of Inst. Air / 8					X		AA2.01 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure (CFR: 41.10 / 43.5 / 45.13)	3.5	8
295021 Loss of Shutdown Cooling / 4						X	2.4.11 Knowledge of abnormal condition procedures: Loss of Shutdown Cooling (CFR: 41.10 / 43.5 / 45.13)	4.0	9
295023 Refueling Acc / 8	X						AK1.03 Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: Inadvertent criticality (CFR: 41.8 to 41.10)	3.7	10

295024 High Drywell Pressure / 5		X					EK2.08 Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: ADS (CFR: 41.7 / 45.8)	4.0	11
295025 High Reactor Pressure / 3			X				EK3.02 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : Recirculation pump trip (CFR: 41.5 / 45.6)	3.9	12
295026 Suppression Pool High Water Temp. / 5				X			EA1.01 Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling (CFR: 41.7 / 45.6)	4.1	13
295027 High Containment Temperature / 5									
295028 High Drywell Temperature / 5						X	2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	3.7	14
295030 Low Suppression Pool Wtr Lvl / 5	X						EK1.01 Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Steam condensation (CFR: 41.8 to 41.10)	3.8	15
295031 Reactor Low Water Level / 2				X			EA1.05 Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Reactor core isolation system (CFR: 41.7 / 45.6)	4.3	16
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1			X				EK3.05 Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Cold shutdown boron weight (CFR: 41.5 / 45.6)	3.2	17
295038 High Off-site Release Rate / 9		X					EK2.07 Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Control room ventilation (CFR: 41.7 / 45.8)	3.5	18
600000 Plant Fire On Site / 8					X		AA2.14 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Equipment that will be affected by fire suppression activities in each zone	3.0	19
700000 Generator Voltage and Electric Grid Disturbances / 6						X	2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6)	4.5	20
K/A Category Totals:	3	3	3	4	3	4	Group Point Total:		20

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

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3					X		AA2.04 Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Offgas system flow (CFR: 41.10 / 43.5 / 45.13)	2.8	21
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2		X					AK2.01 Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: Reactor water level indication (CFR: 41.7 / 45.8)	3.9	22
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5			X				AK3.01 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: Increased drywell cooling (CFR: 41.5 / 45.6)	3.5	23
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1				X			AA1.02 Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: RPS (CFR: 41.7 / 45.6)	4.0	24
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Wtr Lvl / 5	X						EK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Containment integrity (CFR: 41.8 to 41.10)	3.4	25
295032 High Secondary Containment Area Temperature / 5									
295033 High Secondary Containment Area Radiation Levels / 9									
295034 Secondary Containment Ventilation High Radiation / 9						X	2.2.40 Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)	3.4	26
295035 Secondary Containment High Differential Pressure / 5									

295036 Secondary Containment High Sump/Area Water Level / 5	X						EK1.01 Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Radiation releases (CFR: 41.8 to 41.10)	2.9	27
500000 High CTMT Hydrogen Conc. / 5									
K/A Category Point Totals:	2	1	1	1	1	1	Group Point Total:		7

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode					X							K5.02 Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): Core cooling methods (CFR: 41.5 / 45.3)	3.5	28
203000 RHR/LPCI: Injection Mode							X					A1.03 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: System flow (CFR: 41.5 / 45.5)	3.8	29
205000 Shutdown Cooling						X						K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): A.C. electrical power (CFR: 41.7 / 45.7)	3.3	30
206000 HPCI														
207000 Isolation (Emergency) Condenser														
209001 LPCS							X					A1.01 Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Core spray flow (CFR: 41.5 / 45.5)	3.4	31
209002 HPCS				X								K4.07 Knowledge of HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) design feature(s) and/or interlocks which provide for the following: Override of reactor water level interlock (CFR: 41.7)	3.5	32
211000 SLC										X		A4.08 Ability to manually operate and/or monitor in the control room: System initiation (CFR: 41.7 / 45.5 to 45.8)	4.2	33
212000 RPS									X			A3.01 Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: Reactor power (CFR: 41.7 / 45.7)	4.4	34
215003 IRM											X	2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (CFR: 41.10 / 45.12)	4.6	35

215003 IRM								X				A2.06 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty range switch (CFR: 41.5 / 45.6)	3.0	36
215004 Source Range Monitor	X											K1.01 Knowledge of the physical connections and/or cause-effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: Reactor protection system (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.6	37
215005 APRM / LPRM		X										K2.02 Knowledge of electrical power supplies to the following: APRM channels (CFR: 41.7)	2.6	38
217000 RCIC			X									K3.04 Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Adequate core cooling (CFR: 41.7 / 45.4)	3.6	39
217000 RCIC								X				A2.02 Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine trips (CFR: 41.5 / 45.6)	3.8	40
218000 ADS				X								K4.03 Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: ADS logic control (CFR: 41.7)	3.8	41
223002 PCIS/Nuclear Steam Supply Shutoff						X						K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF: Nuclear boiler instrumentation (CFR: 41.7 / 45.7)	3.3	42
239002 SRVs					X							K5.05 Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES: Discharge line quencher operation (CFR: 41.5 / 45.3)	2.6	43

259002 Reactor Water Level Control							X					A1.04 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor water level control controller indications (CFR: 41.5 / 45.5)	3.6	44
261000 SGTS								X				A2.03 Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High train temperature (CFR: 41.5 / 45.6)	2.9	45
262001 AC Electrical Distribution									X			A3.01 Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Breaker tripping (CFR: 41.7 / 45.7)	3.1	46
262002 UPS (AC/DC)										X		A4.01 Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source (CFR: 41.7 / 45.5 to 45.8)	2.8	47
263000 DC Electrical Distribution											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)	4.0	48
263000 DC Electrical Distribution					X							K5.01 Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION: Hydrogen generation during battery charging (CFR: 41.5 / 45.3)	2.6	49
264000 EDGs	X											K1.05 Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency generator fuel oil supply system (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.2	50
300000 Instrument Air		X										K2.01 Knowledge of electrical power supplies to the following: Instrument air compressor (CFR: 41.7)	2.8	51
300000 Instrument Air						X						K6.07 Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: Valves. (CFR: 41.7 / 45.7)	2.5	52

400000 Component Cooling Water			X									K3.01 Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS (CFR: 41.7 / 45.6)	2.9	53
K/A Category Point Totals:	2	2	2	2	3	3	3	3	2	2	2	Group Point Total:		26

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic														
201002 RMCS							X					A1.03 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: Rod movement sequence lights (CFR: 41.5 / 45.5)	3.0	54
201003 Control Rod and Drive Mechanism					X							K5.08 Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM: How control rods affect shutdown margin (CFR: 41.5 / 45.3)	3.1	55
201004 RSCS														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control						X						K6.05 Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM: Reactor water level (CFR: 41.7 / 45.7)	3.1	56
204000 RWCU				X								K4.02 Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: Piping over-pressurization protection (CFR: 41.7)	2.7	57
214000 RPIS														
215001 Traversing In-Core Probe								X				A2.01 Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level: Mark-I&II(Not-BWR1) (CFR: 41.5 / 45.6)	2.7	58
215002 RBM									X			A3.04 Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: Verification or proper functioning/ operability: BWR-3,4,5 (CFR: 41.7 / 45.7)	3.6	59
216000 Nuclear Boiler Inst.														

219000 RHR/LPCI: Torus/Pool Cooling Mode														
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode										X		A4.19 Ability to manually operate and/or monitor in the control room: Drywell temperature (CFR: 41.7 / 45.5 to 45.8)	3.4	60
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam											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	61
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														
245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate														
259001 Reactor Feedwater	X											K1.11 Knowledge of the physical connections and/or cause-effect relationships between REACTOR FEEDWATER SYSTEM and the following: RFP lube oil system (CFR: 41.2 to 41.9 / 45.7 to 45.8)	2.7	62
268000 Radwaste			X									K3.04 Knowledge of the effect that a loss or malfunction of the RADWASTE will have on following: Drain sumps (CFR: 41.5 / 45.3)	2.7	63
271000 Offgas														
272000 Radiation Monitoring				X								K4.02 Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for the following: Automatic actions to contain the radioactive release in the event that the predetermined release rates are exceeded (CFR: 41.7)	3.7	64
286000 Fire Protection		X										K2.02 Knowledge of electrical power supplies to the following: Pumps (CFR: 41.7)	2.9	65
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC														
290002 Reactor Vessel Internals														
K/A Category Point Totals:	1	1	1	2	1	1	1	1	1	1	1	Group Point Total:		12

ES-401 Generic Knowledge and Abilities Outline (Tier 3) (Rev 3 - 2/8/17) Form ES-401-3

Facility: Columbia Generating Station Date of Exam: 2/27/17						
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.2	Knowledge of operator responsibilities during all modes of plant operation. (CFR: 41.10 / 45.13)	4.1	66		
	2.1.20	Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)	4.6	67		
	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)	4.3	68		
	Subtotal			3		
2. Equipment Control	2.2.6	Knowledge of the process for making changes to procedures. (CFR: 41.10 / 43.3 / 45.13)	3.0	69		
	2.2.35	Ability to determine Technical Specification Mode of Operation. (CFR: 41.7 / 41.10 / 43.2 / 45.13)	3.6	70		
	2.2.12	Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)	3.7	71		
	Subtotal			3		
3. Radiation Control	2.3.12	Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)	3.2	72		
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)	3.4	73		
	Subtotal			2		
4. Emergency Procedures / Plan	2.4.4	Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6)	4.5	74		
	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. (CFR: 41.10 / 43.5 / 45.13)	3.7	75		
	Subtotal			2		
Tier 3 Point Total				10		

Facility: Columbia Generating Station Date of Exam: 2/27/17

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 & Abnormal Plant Evolutions	1				N/A					N/A				4	3	7
	2													1	2	3
	Tier Totals													5	5	10
2. Plant Systems	1													3	2	5
	2													1 1	1	3
	Tier Totals													5	3	8

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

Note:
1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the “Tier Totals” in each K/A category shall not be less than two). (One Tier 3 Radiation Control K/A is allowed if the K/A 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 associated 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 Secti on D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
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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’ importance ratings (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

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X		AA2.03 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Actual core flow (CFR: 41.10/43.5/45.13)	3.3	76
295003 Partial or Complete Loss of AC / 6						X	2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13)	4.2	77
295004 Partial or Total Loss of DC Pwr / 6					X		AA2.01 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Cause of partial or complete loss of D.C. power (CFR: 41.10 / 43.5 / 45.13)	3.6	78
295005 Main Turbine Generator Trip / 3									
295006 SCRAM / 1									
295016 Control Room Abandonment / 7						X	2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)	4.0	79
295018 Partial or Total Loss of CCW / 8									
295019 Partial or Total Loss of Inst. Air / 8					X		AA2.02 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads (CFR: 41.10 / 43.5 / 45.13)	3.7	80
295021 Loss of Shutdown Cooling / 4						X	2.1.19 Ability to use plant computers to evaluate system or component status. (CFR: 41.10 / 45.12)	3.8	81
295023 Refueling Acc / 8									
295024 High Drywell Pressure / 5									
295025 High Reactor Pressure / 3									
295026 Suppression Pool High Water Temp. / 5									
295027 High Containment Temperature / 5									
295028 High Drywell Temperature / 5									
295030 Low Suppression Pool Wtr Lvl / 5									
295031 Reactor Low Water Level / 2									
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1									
295038 High Off-site Release Rate / 9									
600000 Plant Fire On Site / 8									

700000 Generator Voltage and Electric Grid Disturbances / 6					X		AA2.08 Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Criteria to trip the turbine or reactor (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)	4.4	82
K/A Category Totals:					4	3	Group Point Total:		7

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

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3						X	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. (CFR: 41.7 / 43.5 / 45.12)	4.6	83
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5					X		AA2.03 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell radiation levels (CFR: 41.10 / 43.5 / 45.13)	3.6	84
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Wtr Lvl / 5									
295032 High Secondary Containment Area Temperature / 5									
295033 High Secondary Containment Area Radiation Levels / 9						X	2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)	4.6	85
295034 Secondary Containment Ventilation High Radiation / 9									
295035 Secondary Containment High Differential Pressure / 5									
295036 Secondary Containment High Sump/Area Water Level / 5									
500000 High CTMT Hydrogen Conc. / 5									
K/A Category Point Totals:					1	2	Group Point Total:		3

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode														
205000 Shutdown Cooling								X				A2.06 Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SDC/RHR pump trips. (CFR: 41.5 / 45.6)	3.5	86
206000 HPCI														
207000 Isolation (Emergency) Condenser														
209001 LPCS											X	2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	4.2	87
209002 HPCS														
211000 SLC														
212000 RPS														
215003 IRM								X				A2.05 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system. (CFR: 41.5 / 45.6)	3.5	88
215004 Source Range Monitor														
215005 APRM / LPRM														
217000 RCIC														
218000 ADS											X	2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	4.7	89
223002 PCIS/Nuclear Steam Supply Shutoff														
239002 SRVs														
259002 Reactor Water Level Control														
261000 SGTS														
262001 AC Electrical Distribution														
262002 UPS (AC/DC)														

263000 DC Electrical Distribution														
264000 EDGs														
300000 Instrument Air														
400000 Component Cooling Water								X				A2.01 Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump (CFR: 41.5 / 45.6)	3.4	90
K/A Category Point Totals:								3			2	Group Point Total:		5

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic														
201002 RMCS														
201003 Control Rod and Drive Mechanism														
201004 RSCS														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control														
204000 RWCU														
214000 RPIS														
215001 Traversing In-Core Probe														
215002 RBM														
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode								X				A2.13 Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High suppression pool temperature. (CFR: 41.5 / 45.6)	3.7	91
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment									X			A3.02 Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including: Interlock operation (CFR: 41.7 / 45.7)	3.7	92
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														
245000 Main Turbine Gen. / Aux.														

256000 Reactor Condensate											X	2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12)	4.2	93
259001 Reactor Feedwater														
268000 Radwaste														
271000 Offgas														
272000 Radiation Monitoring														
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC														
290002 Reactor Vessel Internals														
K/A Category Point Totals:								1	1		1	Group Point Total:		3

Facility: Columbia Generating Station Date of Exam: 2/27/17

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.9	Ability to direct personnel activities inside the control room. (CFR: 41.10 / 45.5 / 45.12 / 45.13)			4.5	94
	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	95
	Subtotal					2
2. Equipment Control	2.2.11	Knowledge of the process for controlling temporary design changes. (CFR: 41.10 / 43.3 / 45.13)			3.3	96
	2.2.20	Knowledge of the process for managing troubleshooting activities. (CFR: 41.10 / 43.5 / 45.13)			3.8	97
	Subtotal					2
3. Radiation Control	2.3.11	Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)			4.3	98
	Subtotal					1
4. Emergency Procedures / Plan	2.4.29	2.4.29 Knowledge of the emergency plan. (CFR: 41.10 / 43.5 / 45.11)			4.4	99
	2.4.31	Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)			4.1	100
	Subtotal					2
Tier 3 Point Total						7

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	295021.2.4.39 (**Question 9** on RO outline)	Knowledge of RO responsibilities in emergency plan implementation: Loss of Shutdown Cooling. (RO-3.9). Reason for rejection: Supports testing at the SRO-only level, but NOT the RO level due to CGS RO job responsibilities. There are no ERO responsibilities assigned to RO personnel. Suggested K/A: 295021.2.4.11 - Knowledge of abnormal condition procedures: Loss of Shutdown Cooling (CFR: 41.10 / 43.5 / 45.13) Importance RO-4.0. K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	295028 2.4.45 (**Question 14** in RO outline)	Ability to prioritize and interpret the significance of each annunciator or alarm. (RO-4.1) Reason for rejection: Unable to write a suitable RO level question with three plausible distractors. Suggested K/A: 295028 2.4.6 – Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13) Importance: RO-3.7 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	700000 2.4.18 (**Question 20** on RO outline)	Generator Voltage and Electric Grid Disturbances: Knowledge of the specific bases for EOPs (RO: 3.3) Reason for rejection: Unable to write a suitable question. There are no EOP bases concerning generator voltage and electric grid disturbances. Suggested K/A: 700000 2.4.4 – Generator Voltage and Electric Grid Disturbances: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6) Importance: RO-4.5 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	203000.A1.05 (**Question 29** on RO outline)	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Suppression pool level (RO-3.8) Reason for rejection: Unable to develop a question with three plausible distractors. Suggested K/A: 203000.A1.03 - Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE controls including: System flow (CFR: 41.5 / 45.6) Importance: RO- 3.8 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	209001.A1.05 (**Question 31** on RO outline)	Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Torus/suppression pool water level (RO-3.5) Reason for rejection: Unable develop a question with three plausible distractors. Suggested K/A: 209001.A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Core spray flow (CFR: 41.5 / 45.5) Importance: RO- 3.4 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	211000.A4.05 (**Question 33** on RO outline)	Ability to manually operate and/or monitor in the control room: Flow indication (RO-4.1) Reason for rejection: Overlap with Operating Test dynamic scenario event in which a degraded SLC flow condition exists. Simulator dynamic event better suited to evaluate K/A. Suggested K/A: 211000.A4.08 - Ability to manually operate and/or monitor in the control room: System initiation (CFR: 41.7 / 45.5 to 45.8) Importance: RO- 4.2 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	212000.A3.07 (**Question 34** on RO outline)	Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: SCRAM air header pressure (RO-3.6) Reason for rejection: Overlap with Operating Test dynamic scenario event involving hydraulic ATWS where RPS trip signals have to be bypassed to reset the scram for a re-scram attempt. Simulator dynamic event better suited to evaluate K/A. Suggested K/A: 212000.A3.01 - Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: Reactor power (CFR: 41.7 / 45.7) Importance: RO- 4.4 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	217000.A2.14 (**Question 40** on RO outline)	Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rupture disc failure: Exhaust-Diaphragm (RO-3.3) Reason for rejection: Unable to write a satisfactory question. There is no procedural guidance governing actions for a failed RCIC exhaust rupture disc. Suggested K/A:217000.A2.02 -Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine Trips (CFR: 41.5 / 45.6) Importance: RO- 3.8 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	300000.K6.13 (**Question 52** in RO outline)	2.2.44 Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: Filters. (RO-2.8) Reason for rejection: Unable to write a satisfactory RO question. There are no filters normally in service in the instrument air system. Suggested K/A: 300000.K6.07 – Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: Valves. (CFR: 41.7 / 45.7) Importance: RO 2.5 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/2	226001.A4.08 (**Question 60** on RO outline)	Ability to manually operate and/or monitor in the control room: Containment spray system flow (RO-3.2) Reason for rejection: Overlap with Operating Test dynamic scenario event involving placing containment sprays in service and monitoring effectiveness. Simulator dynamic event better suited to evaluate K/A. Suggested K/A: 226001.A4.19 - Ability to manually operate and/or monitor in the control room: Drywell temperature (CFR: 41.7 / 45.5 to 45.8) Importance: RO- 3.4 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
3/2	2.2.20 (**Question 70** on RO outline)	Knowledge of the process for managing troubleshooting activities. (RO-2.6) Reason for rejection: Overlap with NRC exam item SRO-97. This K/A has a SRO importance factor of 3.8 vice 2.6 for RO and therefore supports questioning at the SRO level. Suggested K/A: 2.2.35 - Ability to determine Technical Specification Mode of Operation. (CFR: 41.7 / 41.10 / 43.2 / 45.13) Importance: RO-3.6 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
3/2	295016.2.2.21 (**Question 71** on RO outline)	Knowledge of pre- and post-maintenance operability requirements. (RO-2.9) Reason for rejection: Unable to write RO level question that can evaluate BOTH pre and post maintenance operability. Also, operability determination is generally SRO function. Suggested K/A 2.2.12 - Knowledge of surveillance procedures (CFR 41.10 / 45.13). Importance: RO - 3.7 K/A was randomly selected using methodology contained in ES-401, Attachment 1. RO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	2.2.50 (**Question 79** on SRO outline)	Control Room Abandonment: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (SRO – 4.0) Reason for rejection: Unable to write an SRO level question. There are limited alarm responses (2) from the Remote Shutdown Panel. Suggested K/A 2.4.35 - Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR 41.10/43.5/45.13). Importance: SRO – 4.0 K/A was randomly selected using methodology contained in ES-401, Attachment 1. SRO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	205000.A2.10 (**Question 86** on SRO outline)	Rejected K/A: 215000.A2.10 - Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve Operation. (CFR: 41.5 / 45.6) Importance: (SRO- 2.9) Reason for rejection: Unable to write a satisfactory SRO question. Suggested K/A: 215000.A2.06 - Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SDC/RHR pump trips. (CFR: 41.5 / 45.6) Importance: (SRO- 3.5) K/A was randomly selected using methodology contained in ES-401, Attachment 1. SRO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	215003.A2.04 (**Question 88** on SRO outline)	Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips (SRO-3.8) Reason for rejection: Overlap with Operating Test dynamic scenario event in which an IRM fails upscale during startup. Simulator dynamic event better suited to evaluate K/A. Suggested K/A: 215003.A2.05 - Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system (CFR: 41.5 / 45.6) Importance: SRO- 3.5 K/A was randomly selected using methodology contained in ES-401, Attachment 1. SRO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/2	219000.A2.10 (**Question 91** on SRO outline)	Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Nuclear boiler instrument failures (SRO-3.2) Reason for rejection: Unable to write a satisfactory SRO level question with 3 plausible distractors. Suggested K/A:219000.A2.13 - Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High suppression pool temperature (CFR: 41.5 / 45.6) Importance: SRO- 3.7 K/A was randomly selected using methodology contained in ES-401, Attachment 1. SRO outline meets requirements of NUREG 1021 (Rev 10) with this change.

Facility: Columbia Generating Station   Date of Examination: 2/27/17

Examination Level: RO ☒ SRO ☐   Operating Test Number: 1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
A-1 Conduct of Operations K/A: 2.1.20 (4.6 / 4.6)	(N)(R)	PERFORM ALTERNATE POWER CALCULATION WORKSHEET Description: Determines that the Core Thermal Power Validation is satisfactory by performing PPM 9.3.1 Attachment 7.4 (Alternate Power Calculation Worksheet).
A-2 Conduct of Operations K/A: 2.1.25 (3.9 / 4.2)	(M)(P)(R)	MAIN TURBINE CHANGE OF LOAD RATE DETERMINATION Description: Determine the Main Turbine Load Change Recommendation when raising Main Turbine load from 12% to 70%.
A-3 Equipment Control K/A: 2.2.41 (3.5 / 3.9) **OPEX AR 00314141**	(N)(R)	VALIDATE FUSE INSTALLATION PER PPM 1.3.47 (RO) Description: For the RO Candidate, given circumstance requiring fuse replacement and an electrical print, determine correct replacement fuse and provide justification.
A-4 Radiation Control K/A: 2.3.7 (3.5 / 3.6)	(N)(R)	DETERMINE RWP/ALARA TASK TO USE FOR CLEARANCE TASK Description: When hanging a Clearance Order tag, determination the proper RWP and ALARA Task to sign on to to accomplish the task.

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

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

Facility: Columbia Generating Station	Date of Examination: 2/27/17
Examination Level: RO ☐ SRO ☒	Operating Test Number: 1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
A-5 Conduct of Operations K/A: 2.1.7 (4.4 / 4.7)	(D)(R)	DETERMINE ACTION BASED ON PLANT CONDITIONS AND PROCEDURAL GUIDANCE Description: Given equipment status and an electrical bus lockout, determine required operator action based on existing plant conditions.
A-6 Conduct of Operations K/A: 2.1.25 (3.9 / 4.2)	(M)(R)	DETERMINE THE OPERABILITY OF THE SLC SYSTEM Description: Given OSP-INST-H101 (Shift and Daily Instrument Checks for Modes 1, 2 & 3), and CSP-SLC-M101 (Chemistry SLC Surveillance), determine the operability status of the Standby Liquid Control (SLC) System.
A-7 Equipment Control K/A: 2.2.41 (3.5 / 3.9) **OPEX AR 00314141**	(N)(R)	VALIDATE FUSE INSTALLATION PER PPM 1.3.47 (SRO) Description: For the SRO Candidate, given circumstance requiring fuse replacement and an electrical print, validate that the correct replacement fuse was chosen and provide justification.
A-8 Radiation Control K/A: 2.3.11 (3.8 / 4.3)	(D)(P)(R)	ESTIMATE MAIN CONDENSER AIR EJECTOR GROSS GAMMA ACTIVITY RATE AND DETERMINE ACTIONS Description: Estimate Main Condenser air ejector Gross gamma activity rate and determine that a reactor power reduction is required to maintain Main Condenser Gross activity LT the LCO 3.7.5 limit.
A-9 Emergency Plan K/A: 2.4.41 (2.9 / 4.6)	(M)(R)	COMPLETE CLASSIFICATION NOTIFICATION FORM (CNF) FOR SAE Description: Given a dose projection printout, classify the event and complete Classification Notification Form. **(Time Critical)**

NOTE:	**All items (five total) are required for SROs. RO applicants require only four items unless they are retaking only the administrative topics (which would require all five items).**
* Type Codes & Criteria:	(C)ontrol room, (S)imulator, or Class(R)oom (D)irect from bank (≤ 4 for SROs) **(2)** (N)ew or (M)odified from bank (≥ 1) **(3)** (P)revious 2 exams (≤ 1; randomly selected) **(1)**

Facility: Columbia Generating Station		Date of Examination: 2/27/17
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐		Operating Test No.: 1

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

System / JPM Title	Type Code*	Safety Function
S-1: TRANSFER BUS SM-3 FROM TR-S TO TR-N & TRANSFER SM-8 FROM TR-B TO SM-3 Description: Transfer 4160 VAC Bus SM-3 from the Startup Transformer to the Normal Transformer and then transfer 4160 VAC Bus SM-8 from the Backup Transformer to SM-3 K/A: 262001.A4.04 (3.6 / 3.7)	(A)(M)(S)	6
S-2: RESPOND TO LOSS OF SHUTDOWN COOLING Description: Restore Residual Heat Removal (RHR) Loop B shutdown cooling per SOP-RHR-SDC (RHR Loop B Shutdown Cooling Quick Restart). K/A: 205000.A2.06 (3.4 / 3.5)	(D)(L)(S)	4
S-3: HPCS SYSTEM INITIATION Description: Initiate High Pressure Core Spray (HPCS) system per SOP-HPCS-INJECTION and restore RPV level back to directed band. Following start of the HPCS pump its minimum flow valve will fail to automatically close once RPV injection has occurred (resulting in a lower injection rate into the RPV). Valve must be manually closed to maximize injection. K/A: 209002.A4.04 (3.1 / 3.1)	(A)(N)(EN)(L)(S)	2
S-4: INITIATE CR HVAC IN MANUAL PRESSURIZATION MODE Description: Place both trains of Control Room Ventilation in the Manual Pressurization Mode of operation per SOP-HVAC/CR-OPS (inlet damper for one of the Control Room Emergency Filter Units fail to auto open and must be opened manually). K/A: 290003.A4.03 (2.8 / 2.8)	(A)(M)(EN)(S)	9
S-5: RE-ESTABLISH SECONDARY CONTAINMENT/START RB HVAC Description: Restart Reactor Building (RB) HVAC using RB Outside Air Fan 1A and RB Exhaust Air Fan 1A per SOP-HVAC RB-RESTART-QC to re-establish Secondary Containment integrity. K/A: 290001.A4.01 (3.3 / 3.4)	(D)(P)(S)	5
S-6: LOWER RPV PRESSURE USING DEH Description: Recognize that auto control of bypass valves to lower RPV pressure to a target of 550 psig does not work and that the manual lowering of RPV pressure at a rate LE 50 psig per minute through manual control of Bypass Valves would be required. K/A: 241000.A4.02 (4.1 / 4.1)	(A)(D)(L)(P)(S)	3
S-7: (RPS) RESTORE RPS A FROM ALTERNATE POWER SOURCE Description: Transfer RPS A to its Alternate power supply by performing subsequent steps in ABN-RPS. K/A: 212000.A2.01 (3.7 / 3.9)	(D)(P)(S)	7

S-8: Swap RCC Heat Exchangers Description: Direction is provided to swap RCC Heat Exchangers from RCC-HX-1C in service to RCC-HX-1A in service. K/A: 400000.A4.01 (3.1 / 3.0)	(N)(S)	8
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
P-1: RESTART RPS-MG-1 AND REPOWER RPS BUS Description: Direction is provided to restart the RPS Motor Generator (RPS-MG-1) which supplies power to RPS Bus 'A' using SOP-RPS-START. During the start the expected voltage indication is not present requiring manual reset of the MG overvoltage trip. The Underfrequency indicator remains lit and must manually be reset. K/A: 212000.A2.01 (3.7 / 3.9)	(A)(D)(R)	6
P-2: INSERT CONTROL RODS BY VENTING SCRAM AIR HEADER Description: Based on initial conditions provided, recognize that manually venting the scram air header is the next action to take in an attempt to insert control rods. K/A: 295037.EA1.05 (3.9 / 4.0)	(D)(E)(R)	1
P-3: REMOTE SHUTDOWN PANEL ACTIVATION DURING A CONTROL ROOM EVACUATION **(Time Critical)**** Description: Based on a Main Control Room evacuation due to fire, and from a designated starting point, transit to the Remote Shutdown Panel and activate panel within required time using ABN-CR-EVAC Attachment 7.2. K/A: 295016 AA1.07 (4.2 / 4.3) **** Ref: OI-69, TCOA-3/TCOA-4**	(D)(E)(R)	7

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

Facility: Columbia Generating Station		Date of Examination: 2/27/17
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐		Operating Test No.: 1

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

System / JPM Title	Type Code*	Safety Function
S-1: TRANSFER BUS SM-3 FROM TR-S TO TR-N & TRANSFER SM-8 FROM TR-B TO SM-3 Description: Transfer 4160 VAC Bus SM-3 from the Startup Transformer to the Normal Transformer and then transfer 4160 VAC Bus SM-8 from the Backup Transformer to SM-3 K/A: 262001.A4.04 (3.6 / 3.7)	(A)(M)(S)	6
S-2: RESPOND TO LOSS OF SHUTDOWN COOLING Description: Restore Residual Heat Removal (RHR) Loop B shutdown cooling per SOP-RHR-SDC (RHR Loop B Shutdown Cooling Quick Restart). K/A: 205000.A2.06 (3.4 / 3.5)	(D)(L)(S)	4
S-3: HPCS SYSTEM INITIATION Description: Initiate High Pressure Core Spray (HPCS) system per SOP-HPCS-INJECTION and restore RPV level back to directed band. Following start of the HPCS pump its minimum flow valve will fail to automatically close once RPV injection has occurred (resulting in a lower injection rate into the RPV). Valve must be manually closed to maximize injection. K/A: 209002.A4.04 (3.1 / 3.1)	(A)(N)(EN)(L)(S)	2
S-4: INITIATE CR HVAC IN MANUAL PRESSURIZATION MODE Description: Place both trains of Control Room Ventilation in the Manual Pressurization Mode of operation per SOP-HVAC/CR-OPS (inlet damper for one of the Control Room Emergency Filter Units fail to auto open and must be opened manually). K/A: 290003.A4.03 (2.8 / 2.8)	(A)(M)(EN)(S)	9
S-5: RE-ESTABLISH SECONDARY CONTAINMENT/START RB HVAC Description: Restart Reactor Building (RB) HVAC using RB Outside Air Fan 1A and RB Exhaust Air Fan 1A per SOP-HVAC RB-RESTART-QC to re-establish Secondary Containment integrity. K/A: 290001.A4.01 (3.3 / 3.4)	(D)(P)(S)	5
S-6: LOWER RPV PRESSURE USING DEH Description: Recognize that auto control of bypass valves to lower RPV pressure to a target of 550 psig does not work and that the manual lowering of RPV pressure at a rate LE 50 psig per minute through manual control of Bypass Valves would be required. K/A: 241000.A4.02 (4.1 / 4.1)	(A)(D)(L)(P)(S)	3
S-8: Swap RCC Heat Exchangers Description: Direction is provided to swap RCC Heat Exchangers from RCC-HX-1C in service to RCC-HX-1A in service. K/A: 400000.A4.01 (3.1 / 3.0)	(N)(S)	8

In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
P-1: RESTART RPS-MG-1 AND REPOWER RPS BUS Description: Direction is provided to restart the RPS Motor Generator (RPS-MG-1) which supplies power to RPS Bus 'A' using SOP-RPS-START. During the start the expected voltage indication is not present requiring manual reset of the MG overvoltage trip. The Underfrequency indicator remains lit and must manually be reset. K/A: 212000.A2.01 (3.7 / 3.9)	(A)(D)(R)	6
P-2: INSERT CONTROL RODS BY VENTING SCRAM AIR HEADER Description: Based on initial conditions provided, recognize that manually venting the scram air header is the next action to take in an attempt to insert control rods. K/A: 295037.EA1.05 (3.9 / 4.0)	(D)(E)(R)	1
P-3: REMOTE SHUTDOWN PANEL ACTIVATION DURING A CONTROL ROOM EVACUATION **(Time Critical)**** Description: Based on a Main Control Room evacuation due to fire, and from a designated starting point, transit to the Remote Shutdown Panel and activate panel within required time using ABN-CR-EVAC Attachment 7.2. K/A: 295016 AA1.07 (4.2 / 4.3) **** Ref: OI-69, TCOA-3/TCOA-4**	(D)(E)(R)	7

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

Facility: Columbia Generating Station		Date of Examination: 2/27/17
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒		Operating Test No.: 1
Control Room Systems:* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U		
System / JPM Title	**Type Code***	**Safety Function**
S-5: RE-ESTABLISH SECONDARY CONTAINMENT/START RB HVAC Description: Restart Reactor Building (RB) HVAC using RB Outside Air Fan 1A and RB Exhaust Air Fan 1A per SOP-HVAC RB-RESTART-QC to re-establish Secondary Containment integrity. K/A: 290001.A4.01 (3.3 / 3.4)	(D)(P)(S)	5
S-2: RESPOND TO LOSS OF SHUTDOWN COOLING Description: Restore Residual Heat Removal (RHR) Loop B shutdown cooling per SOP-RHR-SDC (RHR Loop B Shutdown Cooling Quick Restart). K/A: 205000.A2.06 (3.4 / 3.5)	(D)(L)(S)	4
S-3: HPCS SYSTEM INITIATION Description: Initiate High Pressure Core Spray (HPCS) system per SOP-HPCS-INJECTION and restore RPV level back to directed band. Following start of the HPCS pump its minimum flow valve will fail to automatically close once RPV injection has occurred (resulting in a lower injection rate into the RPV). Valve must be manually closed to maximize injection. K/A: 209002.A4.04 (3.1 / 3.1)	(A)(N)(EN)(L)(S)	2
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
P-1: RESTART RPS-MG-1 AND REPOWER RPS BUS Description: Direction is provided to restart the RPS Motor Generator (RPS-MG-1) which supplies power to RPS Bus ‘A’ using SOP-RPS-START. During the start the expected voltage indication is not present requiring manual reset of the MG overvoltage trip. The Underfrequency indicator remains lit and must manually be reset. K/A: 212000.A2.01 (3.7 / 3.9)	(A)(D)(R)	6
P-3: REMOTE SHUTDOWN PANEL ACTIVATION DURING A CONTROL ROOM EVACUATION **(Time Critical)**** Description: Based on a Main Control Room evacuation due to fire, and from a designated starting point, transit to the Remote Shutdown Panel and activate panel within required time using ABN-CR-EVAC Attachment 7.2. K/A: 295016 AA1.07 (4.2 / 4.3) **** Ref: OI-69, TCOA-3/TCOA-4**	(D)(E)(R)	7

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

SC-1

INSTRUCTIONAL COVER SHEET

PROGRAM TITLE	OPERATIONS TRAINING
COURSE TITLE	COLUMBIA GENERATING STATION SIMULATOR EXAMINATION
LESSON TITLE	RWCU NRHX fouling causes high temperature isolation on RWCU-V-4; CRD-P-1A trips requiring CRD-P-1B to be started; HPCS-P-1 control power failure (Tech Spec); RRC-FT-14A fails low causing APRM-CHS-1 to trip; SRV MS-RV-2B inadvertently opens (will close upon fuse removal); LOCA from RRC-P-1B suction line requiring manual scram; Spray Wetwell and Drywell; RFW-FIC-620 controller failure with RFW-V-109 failing to open and RFW-V-112A & B failing to open once closed; RCIC–FIC-600 fails low on startup requiring manual trip of RCIC turbine; Initiate Emergency Depressurization (ED) on low RPV level and restore RPV level to above TAF

LENGTH OF LESSON 1.5 Hours

Lesson Plan PQD Code		Rev. No.	
Simulator Guide PQD Code	SC-1	Rev. No.	1
JPM PQD Code		Rev. No.	
Exam PQD Code		Rev. No.	

DIVISION TITLE	Nuclear Training		
DEPARTMENT	Operations Training		
PREPARED BY	Dave E. Crawford	DATE	12/22/16
REVISED BY	Dave E. Crawford	DATE	02/09/17
VALIDATED BY		DATE	
TECHNICAL REVIEW		DATE	
INSTRUCTIONAL REVIEW		DATE	
APPROVED		DATE	

Facility:	Columbia Generating Station	Scenario No.:	1	Op Test No.:	1
Examiners:	____________		Operators:	____________	
	____________			____________	
	____________			____________	

Initial Conditions:	The reactor is in Mode 1 at 100% power. RCIC Operability Test surveillance was just completed to satisfy Post Maintenance Testing (PMT) requirements and has been returned to a Standby status and declared operable. RHR-SYS-B was placed in Suppression Pool Cooling three (3) hours ago to restore Suppression Pool temperature following the testing and to satisfy RHR-P-2B PMT requirements. LCO 3.5.1 A.1, LCO 3.6.1.5 A.1, LCO 3.6.2.3 A.1, and RFO 1.6.1.5 A.1 have been entered for RHR-SYS-B being inoperable.
Turnover:	Maintain RHR-P-2B in operation for the next three (3) hours to satisfy pump PMT requirements for operability.

Critical Tasks:

CT-1	Initiate Drywell sprays when Wetwell pressure exceeds 12 psig but prior to exceeding PSP, after verifying Drywell parameters are within DSIL and RHR is NOT required for adequate core cooling.
CT-2	Initiate Emergency Depressurization (ED) by opening seven (7) Safety Relief Valves (ADS preferred) after RPV water level reaches TAF (-161 inches) and within 10 minutes of level dropping below TAF. CT considered met if any combination of 7 Safety Relief Valves are opened.
CT-3	After ED, and within 10 minutes of RPV pressure lowering to 200 psig, restore and maintain RPV water level above TAF (-161 inches) using Low Pressure ECCS systems.

NOTE: An unintentional or unnecessary RPS or ESF actuation may result in the creation of a post-scenario Critical Task, if that actuation results in a significant plant degradation or significantly alters a mitigation strategy.

Event No.	Malf No.	Event Type*	Event Description
1	TRG-2	C (BOP,SRO) TS (SRO)	RWCU NRHX fouling causes high temperature isolation signal to RWCU system. RWCU-V-4 will not close requiring manual closure of RWCU-V-1 (Tech Spec)
2	TRG-3	C (ATC,SRO)	CRD-P-1A trips requiring CRD-P-1B to be started
3	TRG-4	TS (SRO)	HPCS-P-1 control power failure (Tech Spec)
4	TRG-5	I (ATC,SRO)	RRC-FT-14A fails low causing APRM-CHS-1 to trip
5	TRG-6	C (BOP,SRO) R (ATC,SRO)	SRV MS-RV-2B inadvertently opens (will close upon fuse removal)
6	TRG-7	M (ALL)	LOCA from RRC-P-1B suction line requiring manual scram Spray Wetwell and Drywell **(CT #1)**
7	N/A	C (ATC,SRO)	RFW-FIC-620 controller failure with RFW-V-109 failing to open and RFW-V-112A & B failing to open once closed
8	N/A	C (BOP)	RCIC–FIC-600 fails low on startup requiring manual trip of RCIC turbine
9	N/A	---	Initiate Emergency Depressurization (ED) on low RPV level and restore RPV level to above TAF **(CT #2) (CT #3)**

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

Target Quantitative Attributes	Actual	Description
Malfunctions after EOP entry (1-2)	**2**	Inability to inject with feedwater; RCIC-FIC-600 output fails low
Abnormal events (2-4)	**5**	RWCU NRHX fouling; RWCU-V-4 will not close; CRD-P-1A trip; HPCS control power failure; SRV MS-RV-2B opens
Major transients (1-2)	**1**	LOCA from RRC-P-1B suction line
EOPs entered/requiring substantive actions (1-2)	**2**	PPM 5.1.1 (RPV Control); PPM 5.2.1 (Primary Containment Control)
EOP contingencies requiring substantive actions (0-2)	**1**	PPM 5.1.3 (Emergency RPV Depressurization)
Critical tasks (2-3)	**3**	See Critical Task Determination table

Trigger (TRG-x)	Evaluator Directed	How Triggered	Purpose	Malfunction Numbers
TRG-2	YES	Manually	Event Initiator	HTX-RCC010F
TRG-3	YES	Manually	Event Initiator	BKR-CRD001
TRG-4	YES	Manually	Event Initiator	BKR-CSS001
TRG-5	YES	Manually	Event Initiator	XMT-RRS036A
TRG-6	YES	Manually	Event Initiator	OVR-RRS022D
TRG-7	YES	Manually	Event Initiator	MAL-RRS004B
TRG-8		Automatically	Malf Trigger	MAL-RRS004D
TRG-9		Automatically	Malf Trigger	MOV-CFW044F
TRG-10		Automatically	Malf Trigger	MOV-CFW045F
TRG-11	**YES**(2)	Manually	Malf Trigger	BKR-CFW004; BKR-CFW005; BKR-CFW006
			Initial Condition	MAL-FWC011
			Initial Condition	MOV-RWC010F
			Initial Condition	MOV-CFW043F
			Initial Condition	CNH-RCI002B

(2) **Contingency action (see Event 7 description).**

SCENARIO 1 SUMMARY

Event 1

(TRG-2) Reactor Water Cleanup Non-Regenerative Heat Exchanger (RWCU-HX-2A/2B NRHX) fouling causes a rising temperature at the outlet of the NRHX leading to the RWCU filter demineralizers. The crew takes actions per ARP 4.602.A5 6-8 (CLEANUP FLTR INLET TEMP HI) to include monitoring temperature, verifying system lineup, and ensuring proper Reactor Closed Cooling (RCC) flow to the RWCU NRHX exist. When the crew recognizes that the RWCU NRHX outlet temperature is approaching 140°F, and isolation appears imminent, CRO-2 will stop the running RWCU pump (RWCU-P-1A) and attempt to close the RWCU Suction Outboard Isolation Motor-Operated Valve (MOV) (RWCU-V-4), which will not close. The crew will close the Inboard Isolation MOV (RWCU-V-1) to isolate RWCU.

The CRS refers to Technical Specifications and determines the following actions apply:

- LCO 3.6.1.3 A.1 – Isolate the affected penetration flow path (within 4 hours) by use of at least **one closed and de-activated automatic valve**, closed manual valve, blind flange or check valve with flow through the valve secured.
- LCO 3.6.1.3 A.2 – Verify the affected penetration flow path is isolated once per 31 days for isolation devices outside primary containment.

Event 2

(TRG-3) Control Rod Drive Pump 1A (CRD-P-1A) inadvertently trips requiring the ATC operator to start CRD-P-1B per ARP H13-P603 A-7 3-8 (CRD CHARGE WATER PRESS LOW). Actions include placing the CRD Flow Controller in Manual, zeroing the output and then starting CRD-P-1B. The controller is then nulled and placed back in Auto and CRD system parameters restored.

Event 3

(TRG-4) High Pressure Core Spray (HPCS-P-1) control power fails (fuses blow) due to electrical fault. The BOP operator refers to ARP 601.A1 6-8 (HIGH PRESSURE CORE SPRAY SYSTEM OUT OF SERVICE). If directed to investigate, the HPCS pump control power fuses are reported as blown. Any attempt to replace fuses will result in fuses again blowing.

With both RHR-SYS-B and HPCS inoperable, the CRS refers to Technical Specifications and determines the following additional actions apply:

- LCO 3.5.1 B.1 – Immediately verify by administrative means that RCIC is operable
- LCO 3.5.1 B.2 – Restore HPCS system to operable status within 14 days
- LCO 3.5.1 C.1 – Restore RHR-SYS-B or HPCS system to operable status within 72 hours

Event 4

(TRG-5) A downscale failure of Reactor Recirculation Flow Transmitter 14A (RRC-FT-14A) occurs causing Channel 1 of the Average Power Range Monitor (APRM-CHS-1) to trip. With only one (1) “vote” sent to the 2-out-of-4 voter logic no half-scram or reactor trip signals are generated. The crew takes actions per annunciator 603.A8 3-6 (FLOW REFERENCE OFF NORMAL). The CRS directs the ATC operator to bypass APRM-CHS-1.

With APRM-CHS-1 inoperable (and bypassed), the CRS refers to Technical Specifications and determines that only three (3) APRM channels are required to be operable and that no Technical Specification actions are required.

Event 5

(TRG-6) Non-ADS Safety Relief Valve (SRV) MS-RV-2B inadvertently opens. The crew confirms this by observing at least one of the following: 1) Rise on MS-RV-2B tailpipe temperature on MS-TR-614; 2) Rising Suppression Pool temperature or level; or 3) Reduction in Main Generator output of ~70 MWe. The CRS enters ABN-SRV and directs the ATC operator to reduce reactor power to < 90% using Reactor Recirculation (RRC) flow. The BOP attempts to close the SRV using the control switch. The valve will not close requiring the BOP to remove solenoid fuses per Attachment 7.1. Once fuses are removed the SRV closes. Entry into PPM 5.2.1 (Primary Containment Control) will be required if Suppression Pool level exceeds +2 inches or wetwell temperature exceeds 90°F.

Event 6

(TRG-7) A primary leak from the RRC-P-1B suction line occurs. The crew takes actions to identify and isolate the leak per ABN-LEAK which will not be successful. The leak continues to increase until degrading plant parameters require a manual reactor scram. The crew takes actions per PPM 3.3.1 (Reactor Scram), PPM 5.1.1 (RPV Control), and PPM 5.2.1 (Primary Containment Control). The crew initiates Wetwell sprays when Wetwell pressure reaches 2 psig and initiates Drywell sprays when Wetwell pressure exceeds 12 psig but prior to exceeding the Pressure Suppression Pressure (PSP) limit (PPM 5.2.1 Figure F) and after verifying Drywell parameters are within the Drywell Spray Initiation Limit (DSIL) (PPM 5.2.1 Figure E) and RHR is NOT required for adequate core cooling **(CT #1)**. RHR will be re-aligned from Drywell spray to LPCI injection after emergency depressurization is initiated. Due to a loss of sufficient RPV injection, RPV level continues to lower requiring the crew to emergency depressurize the RPV because sufficient high pressure injections system are not available.

Event 7

Total loss of feedwater injection occurs: Reactor Feedwater Flow Indicating Controller (RFW-FIC-620) output fails low and FWH 6A/6B Bypass Valve (RFW-V-109) fails to open preventing RFW injection into the RPV. RFW-HX-6A & B Discharge to Rx Discharge MOVs (RFW-V-112A & B) fail to open (if attempted) after being initially closed to support feeding with the RFW Flow Control Valves (RFW-FCV-10A/B).

Examiner Note: If the ATC operator fails to close either RFW-V-112A or RFW-V-112B then with specific Examiner direction, Trigger 11 will be entered to cause a trip of all running Condensate Booster pumps to ensure a total loss of feedwater injection occurs which is needed to support Critical Tasks.

Event 8

Reactor Core Isolation Cooling Flow Indicating Controller (RCIC-FIC-600) fails low on RCIC system startup requiring a manual trip of the RCIC turbine.

Event 9

With insufficient high pressure injection sources available, and with RPV level continuing to lower, the CRS enters PPM 5.1.3 (Emergency RPV Depressurization) and initiates emergency depressurization by opening seven (7) Safety Relief Valves (ADS preferred) after RPV water level reaches TAF (-161 inches) and within 10 minutes of level dropping below TAF. **(CT #2)** After ED, and within 10 minutes of RPV pressure lowering to 200 psig, the crew will restore and maintain RPV water level above TAF (-161 inches) using Low Pressure ECCS systems. **(CT #3)** Wetwell and Drywell sprays can be reinitiated per PPM 5.2.1 when not needed for adequate core cooling.

TERMINATION CRITERIA: The scenario will be terminated when Emergency Depressurization has been performed and RPV level is being controlled in the prescribed band OR as directed by the Examination Team.

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT #1 - Initiate Drywell sprays when Wetwell pressure exceeds 12 psig but prior to exceeding PSP, after verifying Drywell parameters are within DSIL and RHR is NOT required for adequate core cooling.	Primary containment pressures at or above specified limits pose a direct threat to primary containment integrity and the pressure suppression function. (Ref: PPM 13.1.1A (Classifying the Emergency – Technical Bases) Attachment 4.1 section 3)	Procedural direction in PPM 5.2.1 (Primary Containment Control - step P-7) when Wetwell pressure exceeds 12 psig.	The operator will manually open Drywell spray isolation valves.	Valve position will change and Drywell spray flow will increase.
CT #2 - Initiate Emergency Depressurization (ED) by opening seven (7) Safety Relief Valves (ADS preferred) after RPV water level reaches TAF (-161 inches) and within 10 minutes of level dropping below TAF.	Preclude core damage by establishing conditions that allow low pressure ECCS systems to restore water level above TAF (Safety Limit) (Ref: CGS Technical Specifications - 2.1.1.3)	Procedural direction in PPM 5.1.1 (RPV Control - step L-15) when RPV Level cannot be restored and maintained above -186 inches.	The operator will manually open 7 Safety Relief Valves (ADS preferred) to emergency depressurize the RPV.	The valve light indications for each of the 7 Safety Relief Valves will change from Green lit to Red lit when control switch is taken to Open. Reactor pressure will lower in response.
CT #3 - After ED, and within 10 minutes of RPV pressure lowering to 200 psig, restore and maintain RPV water level above TAF (-161 inches) using Low Pressure ECCS systems.	Preclude core damage by establishing conditions that allow low pressure ECCS systems to restore water level above TAF (Safety Limit) (Ref: CGS Technical Specifications - 2.1.1.3)	Procedural direction in PPM 5.1.1 (RPV Control - step L-16) which directs restoring and maintaining RPV level above -186 inches and ultimately above TAF.	All available low pressure ECCS systems are aligned to restore RPV level.	Indication of applicable ECCS system flow. RPV level rises to greater than TAF.

TURNOVER

Initial Conditions:

- Columbia is operating at 100% power
- RCIC Operability Test surveillance was just completed to satisfy Post Maintenance Testing (PMT) requirements and has been returned to a Standby status and declared operable
- RHR-SYS-B was placed in Suppression Pool Cooling three (3) hours ago to restore Suppression Pool temperature following the testing and to satisfy RHR-P-2B PMT requirements (see marked up procedure)
- LCO 3.5.1 A.1, LCO 3.6.1.5 A.1, LCO 3.6.2.3 A.1, and RFO 1.6.1.5 A.1 have been entered for RHR-SYS-B being inoperable

Shift Turnover:

- Maintain RHR-P-2B in operation for the next three (3) hours to satisfy pump PMT requirements for operability.

ENERGY NORTHWEST

SC-2

INSTRUCTIONAL COVER SHEET

PROGRAM TITLE	OPERATIONS TRAINING
COURSE TITLE	COLUMBIA GENERATING STATION SIMULATOR EXAMINATION
LESSON TITLE	Lower RRC Flow to 90% using Flow (enter GV Sequential Mode); CRD-FC-600 Fails High; LPCS-P-2 Trips (TS); MS-PS-23D Fails causing Half Scram (2 Rods Scram but 1 does not Fully Insert – Can Manually Insert)(TS); FPC-P-1B Trip (FPC-P-1A Fails to Auto Start); Trip of E-CB-1/7 with Scram (ATWS) occurring on Auto-Shift to TR-B; Hydraulic ATWS (Lower Level to -140” to -80”); Reduced SLC Injection Flow; RWCU-V-4 Fails to Auto Close; Scram-Reset-Scram not Effective in Inserting Rods (Manual Insertion Permitted)

LENGTH OF LESSON 1.5 Hours

Lesson Plan PQD Code		Rev. No.	
Simulator Guide PQD Code	SC-2	Rev. No.	1
JPM PQD Code		Rev. No.	
Exam PQD Code		Rev. No.	

DIVISION TITLE	Nuclear Training		
DEPARTMENT	Operations Training		
PREPARED BY	Dave E. Crawford	DATE	12/22/16
REVISED BY	Dave E. Crawford	DATE	02/08/17
VALIDATED BY		DATE	

TECHNICAL REVIEW		DATE	
INSTRUCTIONAL REVIEW		DATE	
APPROVED		DATE	

Operations Training Manager

Facility:	Columbia Generating Station	Scenario No.:	2	Op Test No.:	1
Examiners:	____________		Operators:	____________	
	____________			____________	
	____________			____________	

Initial Conditions:	Columbia is operating at 100% power. Control Rod Drive (CRD) Pump 1B (CRD-P-1B) is out of service for extended Maintenance. CRD-P-1A is Protected.
Turnover:	Lower reactor power to 90% using Reactor Recirculation flow per PPM 3.2.6 (Power Maneuvering) after assuming the shift based on BPA Load Following request. Steps 5.1.1 thru 5.1.6 of PPM 3.2.6 are complete. Proper margin to Pre-Conditioned Status (PCS) exists per PPM 9.3.18. The Reactivity brief has been performed.

Critical Tasks:

CT-1	During ATWS with power > 5%, terminate and prevent injection with exception of SLC, RCIC, and CRD, into the RPV until RPV level is -65 inches to establish a Lowered Level (LL). -AND- Maintain RPV level above -186 inches. Short excursions below -186 inches does not constitute failure of CT provided level restored and maintained above -186 inches within 10 minutes of going below -186 inches.
CT-2	With reactor scram required and the reactor not shutdown, commence inserting control rods per PPM 5.5.11 Attachment 6.1 Tab B prior to transitioning to Tab E.

NOTE: An unintentional or unnecessary RPS or ESF actuation may result in the creation of a post-scenario Critical Task, if that actuation results in a significant plant degradation or significantly alters a mitigation strategy.

Event No.	Malf.	Event Type*	Event Description
1	N/A	R (ATC) N (BOP)	Lower reactor power with Reactor Recirculation (RRC) flow to 90% for load following per PPM 3.2.6 (which includes placing Main Turbine into Governor Valve Sequential Valve Mode)
2	TRG-2	I (ATC)	CRD Drive Header Flow Control Valve controller (CRD-FC-600) output fails high while in automatic
3	TRG-3	C (BOP,SRO) TS (SRO)	RHR-SYS-A/LPCS Keep Fill Pump (LPCS-P-2) trips (Tech Spec)
4	TRG-4	C (ATC,SRO) TS (SRO)	Failure of MS-PS-23D which causes a half scram on RPS “B” side. Two control rods scram but one does not go full in (must be manually inserted) (Tech Spec)
5	TRG-5	C (BOP)	Ground causes FPC-P-1B to spuriously trip (FPC-P-1A fails to auto start)
6	TRG-6	M (ALL)**	Trip of E-CB-1/7 with transfer of SM-7 to Backup Transformer resulting in reactor trip signal Hydraulic ATWS - Lower RPV Level -80 inches to -140 inches **(CT #1) (CT #2)**
7	N/A	N/A	SLC-P-1A shaft shears when pump starts and SLC-P-1B develops a discharge flow blockage which limits SLC injection flow.
8	N/A	C (ATC)	RWCU-V-4 does not auto close on SLC initiation but can be closed manually.
9	N/A	C (BOP)	Scram/Reset/Scram not effective in inserting control rods - Control rods can be manually driven in

*	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS) Technical Specifications
**	**Event forms a portion of significant CGS PSA Accident Sequence (TTC044) (Ref: PSA-1-SM-0001 (Rev 7))**

Target Quantitative Attributes	Actual	Description
Malfunctions after EOP entry (1-2)	**3**	Reduced SLC injection capability; RWCU-V-4 fails to auto close; Scram-reset-scram ineffective
Abnormal events (2-4)	**4**	CRD-FC-600 failure; LPCS-P-2 shaft seizure; RPS “B” half scram (2 control rods inadvertently scram); FPC-P-1B trip
Major transients (1-2)	**1**	E-CB-1/7 breaker trip leading to hydraulic ATWS
EOPs entered/requiring substantive actions (1-2)	**1**	PPM 5.1.1 (RPV Control)
EOP contingencies requiring substantive actions (0-2)	**1**	PPM 5.1.2 (RPV Control – ATWS)
EOP based Critical tasks (2-3)	**2**	See Critical Task Determination table

Trigger (TRG-x)	Evaluator Directed	How Triggered	Purpose	Malfunction Numbers
TRG-2	YES	Manually	Event Initiator	CNH-CRD001E; BST-CRD001F
TRG-3	YES	Manually	Event Initiator	PMP-CSS004S
TRG-4	YES	Manually	Event Initiator	BST-RRS067F; MAL-RMC007-3835; MAL-RMC007-1815; MAL-RMC005-1815
TRG-5	YES	Manually	Event Initiator	MOT-FPC002G
TRG-6	YES	Manually	Event Initiator	BKR-EPS003; MAL-CRD007A1; MAL-CRD007A2; MAL-CRD007B1; MAL-CRD007B2
TRG-7		Manually	Field Action	BKR-RHR001
TRG-8		Manually	Field Action	BKR-CSS002
TRG-9		Automatically	Malf Trigger	BST-CRD001F
TRG-10		Automatically	Malf Trigger	MAL-RMC007-1815; MAL-RMC005-1815
			Initial Condition	BST-FPC020F
			Initial Condition	PMP-SLC001B
			Initial Condition	BKR-CRD002
			Initial Condition	PMP-SLC002F
			Initial Condition	MOV-RWU010F

Event 1

The Scenario starts from 100% power with Control Rod Drive (CRD) Pump 1B (CRD-P-1B) out of service for extended maintenance. Once the crew has the shift, the ATC operator lowers reactor power (for load following) using Reactor Recirculation (RRC) flow to 90% per PPM 3.2.6 (Power Maneuvering). The BOP operator takes the Main Turbine out of Governor Valve Optimization mode per SOP-MT-GV/OPTIMIZATION (Section 5.2) prior to the RRC flow reduction.

Event 2

(TRG-2) CRD Drive Header Flow Control Valve controller (CRD-FC-600) output fails high while in automatic which causes 603.A7 5-8 (CRD PUMP SUCTION FLTR D HIGH) annunciator to come in caused by abnormally high system flow. Upon finding the CRD-FC-600 controller output failed high, the ATC operator informs the CRS and shifts the controller to manual and restores CRD system parameters to normal. Annunciator will clear once system parameters restored to normal.

Event 3

(TRG-3) The shaft on RHR-SYS-A/LPCS Keep Fill Pump (LPCS-P-2) seizes causing a trip of the pump. The RHR A PUMP DISCH PRESS HIGH/LOW annunciator alarms shortly after LPCS-P-2 trips. The LPCS PUMP DISCH PRESS HIGH/LOW annunciator will alarm ~13 minutes after LPCS-P-2 trips (unless LPCS pump started before then). Based on system status and ARP direction, the CRS will direct the BOP operator to start the Low Pressure Core Spray (LPCS) Pump (and place into Suppression Pool Mixing per SOP-LPCS-SP) to maintain system availability provided the LPCS PUMP DISCH PRESS HIGH/LOW annunciator is not in alarm. To prevent an inadvertent start of Residual Heat Removal (RHR) Pump 2A (RHR-P-2A) and therefore a potential for water hammer, the CRS will direct control power fuses removed (TRG-7) from the RHR-P-2A starting circuit. If LPCS pump is not started and the LPCS PUMP DISCH PRESS HIGH/LOW annunciator is received, LPCS Pump control power fuses will also be removed (TRG-8). The CRS will refer to ABN-RHR-DEPRESS as time permits to determine system recovery actions.

With RHR-P-2A and LPCS inoperable, the CRS refers to Technical Specifications and Licensee Controlled Specifications and determines the following actions are applicable:

- LCO 3.5.1 A.1 (RHR-SYS-A & LPCS are both tracked as inoperable) - Restore respective subsystem to operable status within 7 days
- LCO 3.5.1 C.1 - Restore either RHR-SYS-A or LPCS subsystem to operable status within 72 hours
- LCO 3.6.1.5 A.1 - Restore RHR-SYS-A Drywell Spray subsystem to operable status within 7 days
- LCO 3.6.2.3 A.1 - Restore RHR-SYS-A Suppression Pool Cooling subsystem to operable status within 7 days
- RFO 1.6.1.5 A.1 - Restore RHR-SYS-A Suppression Pool Spray subsystem to operable status within 7 days

Note that LCOs 3.4.6, 3.4.9, and 3.6.1.3 are considered but not applicable with the plant in Mode 1.

Event 4

(TRG-4) Main Steam pressure switch 23D (MS-PS-23D) fails high causing Reactor Protection System (RPS) relay K5D (RPS-RLY-K5D) to actuate a RPV Pressure High Trip Scram relay (as evidenced by annunciator 603.A8 2-2 (RPV PRESS HIGH TRIP)). This actuation causes a half scram on the RPS "B" side with all RPS "B" white RPS scram lights de-energized. The ATC operator will determine that two control rods (38-35 and 18-15) inadvertently scrammed during the half scram and that control rod 18-15 only partially inserted. The CRS enters ABN-ROD, section 4.2, for inadvertently scrammed rods. The ATC operator reduces RRC flow to 74 Mlbm/hr at 5% per minute. Following flow reduction, an attempt is made to fully insert control rod 18-15 using the CONTINUOUS INSERT pushbutton (which will be successful). The crew diagnoses the instrument failure and determines the half scram cannot be reset.

The CRS refers to Technical Specifications and determines that TS 3.3.1.1 (RPS Instrumentation) Action A.1 or A.2 requires affected channel or affected trip system, respectively, to be placed in TRIP within 12 hours. In addition, control rod 18-15 is considered inoperable for not fully inserting when inadvertently scrammed. LCO 3.1.3 (Control Rod Operability) Action C.1 requires rod 18-15 to be fully inserted within 3 hours and its associated CRD (HCU) disarmed within four hours.

Event 5

(TRG-5) Bus 81 ground as sensed on MC-8BB which powers Fuel Pool Cooling Pump 1B (FPC-P-1B) causes FPC-P-1B to trip when power fuses blow. With this power loss, the standby Fuel Pooling Cooling pump (FPC-P-1A) will not auto start. ARP 4.627.FPC2.3-1 (CIRCULATION PUMP B DISCHARGE PRESSURE LOW) directs entry into ABN-FPC-LOSS. The BOP operator will manually start FPC-P-1A to re-establish fuel pool cooling. Resetting the Bus 81 ground annunciator (TRG-1) will be successful, if attempted, since ground cleared upon the FPC-P-1B power fuses blowing. Since the status of the FPC-P-1B thermal overloads are unknown at this point the BOP operator may place the FPC-P-1B control switch in the IR-69 position to allow reset of associated overloads.

Event 6

(TRG-6) Trip of CB-1/7 (4160V feed from SM-1 to SM-7) results in an automatic transfer of Division 1 AC safety bus (SM-7) to the Backup Transformer (TR-B). The transient results in a trip of the LPCS Pump (previously started) and a loss of RPS Motor Generator "A" power to RPS "A". With a RPS "B" half scram signal already present, a full scram signal now exists. The ATC operator recognizes a scram should have occurred and that an ATWS condition exists. The ATC operator takes scram actions including pressing all Manual scram pushbuttons and initiating ARI logic. Both trains of SLC are started due to reactor power being > 5%.

The CRS enters PPM 5.1.1 (RPV Control) and transitions into PPM 5.1.2 (RPV Control – ATWS) and directs the BOP operator to inhibit ADS and to take manual control of HPCS. The CRS addresses the level leg first and directs the BOP operator to perform PPM 5.5.6 (Bypassing MSIV Low RPV Level and High Steam Tunnel Temperature interlocks) to allow MSIVs to stay open on subsequent RPV level reduction. PPM 5.5.1 (Overriding ECCS Valve Logic To Allow Throttling ECCS Injection) is also performed. The CRS then directs stopping and preventing all injection into the RPV except for SLC, CRD and RCIC. When level reaches -65 inches, the ATC operator will restart injection into the RPV through the RFW Startup flow control valve to maintain a RPV Level band of -80 to -140 inches. **(CT #1)** The CRS directs an RPV pressure band of 800 to 1050 psig with the Digital Electro-Hydraulic (DEH) system in automatic. If reactor power is above 25%, the capacity of the RFW Start-up flow line will be exceeded and the ATC operator will have to augment flow by opening RFW-V-109 (Bypass valve for Feedwater Heaters 6A and 6B). The BOP operator performs PPM 5.5.11 (Alternate Control Rod Insertions) in an attempt to insert control rods.

This event forms a portion of significant CGS PSA Accident Sequence (TTC044) (Ref: PSA-1-SM-0001 (Rev 7))

Event 7

Standby Liquid Control (SLC) Pump 1A fails due to a sheared shaft and SLC Pump 1B discharge is partially blocked resulting in a reduced SLC injection flow in the RPV at approximately 18 gpm. This injection rate will cause reactor power to drop slowly but not prior to the crew lowering RPV level to -80 to -140 inches. Reactor Water Cleanup Valve 4 (RWCU-V-4) does not auto close on the SLC initiation but will be closed manually.

Event 8

Reactor Water Cleanup Valve 4 (RWCU-V-4) does not auto close on the SLC initiation but will be closed manually.

Event 9

Control rods insertion will be attempted per PPM 5.5.11 (Alternate Control Rod Insertions). **(CT #2)** Since hydraulic ATWS occurred (no white RPS scram lights lit), the BOP operator will remove two (2) ARI fuses and bypass (via switch) the Scram Discharge Volume (SDV) High Level trip. CRD-P-1A will be found tripped and will have to be restarted before a re-scram is attempted. The Scram – Reset – Scram method of control rod insertion is not effective requiring the BOP operator to bypass the Rod Worth Minimizer (RWM) and manually insert control rods individually using CRD drive pressure.

TERMINATION CRITERIA: The scenario will be terminated when RPV level is being maintained between -80 inches to -140 inches, one attempt at scram-reset-scram has been completed, and manual insertion of control rods has commenced OR as directed by the Examination Team.

Critical Task Determination Table

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT #1 - During ATWS with power > 5%, terminate and prevent injection with exception of SLC, RCIC, and CRD, into the RPV until RPV level is -65 inches to establish a Lowered Level (LL). -AND- Maintain RPV level above -186 inches. Short excursions below -186 inches does not constitute failure of CT provided level restored and maintained above -186 inches within 10 minutes of going below -186 inches	This is a procedural requirement of PPM 5.1.2 (RPV Control – ATWS). Allowing SLC, RCIC and CRD injection avoids conflicts with other instructions in the EOPs such as injecting SLC and inserting control rods. Stopping other injection sources prevents potential fuel damage due to cold water injection. (Ref: PPM 5.0.10 Rev 21, section 8.3.4.) -AND- Prevent unnecessary significant challenge to containment or the RPV.	Procedural direction by PPM 5.1.2 Step L-6 directs lowering RPV level to < -65 inches by stopping and preventing all injection into RPV except from boron injections systems, RCIC and CRD, defeating interlocks if necessary. -AND- Procedural direction by PPM 5.1.2 Step L-12 directs maintaining RPV level from -140 inches to -80 inches (best practice band) with outside shroud injection systems (Table 5). OI-15 (EOP and EAL Clarifications), Section 4.3.2.b.)	Crew stops and prevents injection with the exception of SLC, RCIC, and CRD. -AND- Crew uses Reactor Feedwater system to maintain RPV level above -186 inches. (ED required if level cannot be restored and maintained above -186 inches)	RPV level and reactor power start lowering. -AND- RPV level indication.
CT #2 - With reactor scram required and the reactor not shutdown, commence inserting control rods per PPM 5.5.11 Attachment 6.1 Tab B prior to transitioning to Tab E.	This is a procedural requirement of PPM 5.1.2 (RPV Control – ATWS). Provides a method to shutdown the reactor when required, lowering the reactors energy state, to prevent exceeding primary containment design limits and minimize the potential consequences of power oscillations. (Ref: PPM 5.0.10 Rev 21, section 8.3.6.)	Reactor scram required and reactor not shutdown.	Crew uses alternate methods to insert control rods per PPM 5.5.11 Attachment 6.1 Tab B	Reactor power is decreasing. Control rod Full-In lights as rods are fully inserted.

Initial Conditions:

- Columbia is operating at 100% power
- CRD-P-1B is out of service for extended Maintenance
- CRD-P-1A is Protected

Shift Turnover:

- Lower power to 90% using Reactor Recirculation flow per PPM 3.2.6 (Power Maneuvering) after assuming the shift based on BPA Load Following request
- Steps 5.1.1 thru 5.1.6 of PPM 3.2.6 are complete
- Proper margin to Pre-Conditioned Status (PCS) exists per PPM 9.3.18
- The Reactivity brief has been performed

SC-3

INSTRUCTIONAL COVER SHEET

PROGRAM TITLE	OPERATIONS TRAINING
COURSE TITLE	COLUMBIA GENERATING STATION SIMULATOR EXAMINATION
LESSON TITLE	Place RHR-SYS-A in SP Cooling (LPCS/RHR "A" ADS Permissive fails to annunciate) (Tech Spec); Rod (26-19) drifts out. Once inserted, control rod to drift out again (Tech Spec); SW-P-1A trips which requires RHR-P-2A to be secured; RFP "B" vibrations rise requiring RRC Flow reduction and manual trip of RFP "B"; OBE causes steam leak in RCIC Pump Room with Failure of RCIC-V-8 and RCIC-V-63 to fully close; Manual scram inserted; Steam leak develops in the Main Steam Tunnel; MS-V-22A and MS-V-28A through D fail to automatically close (MS-V-28A through D can be closed manually but does not isolate leak); Emergency Depressurization required on two Max Safes

LENGTH OF LESSON 1.5 Hours

Lesson Plan PQD Code		Rev. No.	
Simulator Guide PQD Code	SC-3	Rev. No.	1
JPM PQD Code		Rev. No.	
Exam PQD Code		Rev. No.	

DIVISION TITLE	Nuclear Training		
DEPARTMENT	Operations Training		
PREPARED BY	Dave E. Crawford	DATE	12/22/16
REVISED BY	Dave E. Crawford	DATE	02/08/17
VALIDATED BY		DATE	
TECHNICAL REVIEW		DATE	
INSTRUCTIONAL REVIEW		DATE	
APPROVED		DATE	

Facility:	Columbia Generating Station	Scenario No.:	3	Op Test No.:	1
Examiners:	____________		Operators:	____________	
	____________			____________	
	____________			____________	

Initial Conditions:	Columbia is operating at 85% power due to economic dispatch. Safety Relief Valve 2C (MS-RV-2C) is known to be leaking. Suppression Pool high temperature alarms (601.A11.1-3 and 601.A12.1-3) have just annunciated. Reactor Closed Cooling (RCC) Pump 1B is tagged out for planned maintenance. RCC-P-1A and RCC-P-1C are protected.
Turnover:	After shift turnover place RHR-P-2A in Suppression Pool Cooling (using maximum cooling) and allow SW-P-1A to auto start per SOP-RHR-SPC (section 5.1) – Steps 5.1.1 through 5.1.4 are complete. Associate Tech Specs and LCS action statements have been entered for RHR-SYS-A being inoperable but available. • LCO 3.5.1 Action A.1 which requires restoring RHR-SYS-A to operable status within 7 days • LCO 3.6.1.5 Action A.1 which requires restoring RHR-SYS-A drywell spray subsystem to operable status within 7 days • LCO 3.6.2.3 Action A.1 which requires restoring RHR-SYS-A suppression pool cooling subsystem to operable status within 7 days • RFO 1.6.1.5 Action A.1 which requires restoring RHR-SYS-A suppression pool spray subsystem to operable status within 7 days The pre-evolution brief has been completed and operators are stationed near both pumps.

Critical Tasks:

CT-1	With reactor at power and with primary system discharging into secondary containment, manually scram reactor before any area exceeds its maximum safe operating temperature.
CT-2	With a primary system discharging into secondary containment and area temperature exceeding maximum safe operating level in more than one area, initiate Emergency Depressurization (ED) by opening seven (7) Safety Relief Valves (ADS preferred) within 10 minutes of second MSOT being exceeded. CT considered met if any combination of 7 Safety Relief Valves are opened. Note: If the crew properly elects to invoke the "EMERG DEPRESS is anticipated" override in ppm 5.1.1 (RPV Control) and in doing so, the second maximum safe operating level is not exceeded, this Critical Task is considered to be met.

NOTE: An unintentional or unnecessary RPS or ESF actuation may result in the creation of a post-scenario Critical Task, if that actuation results in a significant plant degradation or significantly alters a mitigation strategy.

Event No.	Malf.	Event Type*	Event Description
1	N/A	N (BOP) TS (SRO)	Place RHR-SYS-A in Suppression Pool Cooling (LPCS/RHR "A" ADS Permissive fails to annunciate during pump start) (Tech Spec) **
2	TRG-2	C (ATC,SRO) TS (SRO)	Control rod (26-19) drifts out. Once inserted, releasing the continuous insert pushbutton allows the control rod to drift out again, requiring the control rod to be isolated (Tech Spec)
3	TRG-3	C (BOP,SRO) TS (SRO)	Standby Service Water Pump 1A (SW-P-1A) trips which requires Residual Heat Removal Pump 2A (RHR-P-2A) (currently in Suppression Pool Cooling) to be manually secured (Tech Spec)

4	TRG-4	C (ATC,BOP,SRO)	Reactor Feed Pump (RFP) “B” vibrations rise requiring RRC Flow reduction and manual trip of the “B” RFP
5	TRG-5	M (ALL)	Operating Bases Earthquake causes a steam leak in the RCIC Pump Room with Failure of RCIC-V-8 and RCIC-V-63 to fully close (preventing RCIC leak isolation). Manual scram inserted before first secondary containment max safe operating temperature is reached **(CT #1)**
6	N/A	M (ALL)	Steam leak develops in the Main Steam Tunnel
7	N/A	C (BOP)	MS-V-22A and MS-V-28A through D fail to automatically close (MS-V-28A through D can be closed manually but does not isolate leak)
8	N/A	---	Emergency Depressurization (PPM 5.1.3) is performed when two areas exceed their max safe operating temperature **(CT #2)**
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS) Technical Specifications			
**** Ref: Columbia OE (AR-00049685 – Root Cause Analysis of RHR-PS-19A Isolation Mispositioning Event)**			

Target Quantitative Attributes	Actual	Description
Malfunctions after EOP entry (1-2)	**1**	Several MSIVs fail to automatically close and one cannot be closed
Abnormal events (2-4)	**3**	ADS Permissive fails on RHR pump “A” start; Rod 26-19 drifts out; RFB “B” high vibrations
Major transients (1-2)	**2**	RCIC steam leak requiring scram; Main steam line break
EOPs entered/requiring substantive actions (1-2)	**2**	PPM 5.1.1 (RPV Control); PPM 5.3.1 (Secondary Containment Control)
EOP contingencies requiring substantive actions (0-2)	**1**	PPM 5.1.3 (Emergency RPV Depressurization)
EOP based Critical tasks (2-3)	**2**	See Critical Task Determination table

Trigger (TRG-x)	Evaluator Directed	How Triggered	Purpose	Malfunction Numbers
TRG-2	YES	Manually	Event Initiator	MAL-RMC004-2619
TRG-3	YES	Manually	Event Initiator	BKR-SSW001
TRG-4	YES	Manually	Event Initiator	ANN-840A1G05; MAL-FPT005B
TRG-5	YES	Manually	Event Initiator	MAL-RWB001; MAL-RCI004
TRG-6		Manually	Event Initiator	MAL-RMC004-2619
TRG-7		Manually	Field Action	ANN-840A1G05
TRG-8		Automatically	Malf Trigger	MAL-RRS006A; MAL-RCI004
TRG-9		Automatically	Malf Trigger	MAL-RRS006A
			Initial Condition	BST-RHR014F
			Initial Condition	AOV-RRS003F
			Initial Condition	MOV-RCI012F
			Initial Condition	MOV-RCI016F
			Initial Condition	RLY-NSF097F
			Initial Condition	BKR-RCC002

SCENARIO 3 SUMMARY

Event 1

As part of the turnover, and with annunciators for Suppression Pool high temperature in alarm (601.A11 1-3 and 601.A12 1-3), CRO-2 will place Residual Heat Removal Loop “A” (RHR-SYS-A) into Suppression Pool Cooling mode per SOP-RHR-SPC (Suppression Pool Cooling/Spray/Discharge /Mixing). Standby Service Water Pump (SW-P-1A) will be allowed to auto start as permitted by procedure.

During RHR-P-2A pump start for entering Suppression Pool cooling mode, an isolated pressure switch (RHR-PS-19A) prevents the LPCS/RHR “A” ADS Permissive alarm from annunciating on P601. The CRS refers to Technical Specifications and determines that LCO 3.3.5.1 (Emergency Core Cooling System (ECCS) Instrumentation) Action A.1 applies which directs entry into the Condition referenced in Table 3.3.5.1-1 for the channel (Function 4.e) immediately (Condition G). ACTION G.2 directs restoring channel to operable status within 8 days.

Previous Columbia OE (Ref: AR-00049685 – Root Cause Analysis of RHR-PS-19A Isolation Mispositioning Event dated 4/1/2007) involved isolation of same pressure switch which was not discovered until RHR-P-2A was started and the ADS Permissive annunciator did not come in as expected.

Event 2

(TRG-2) Control rod 26-19 drifts out of the core. CRO-1 recognizes the rod drift and takes Immediate Actions to fully insert the control rod using the Continuous Insert pushbutton. The CRS enters ABN-ROD. When the Insert pushbutton is released, the control rod begins again to drift out of the core. CRO-1 re-inserts the control rod full-in (and keeps the Continuous Inset pushbutton pressed) while the crew takes action to isolate the HCU for control rod 26-19 (TRG-6). The CRS declares control rod 26-19 inoperable. The CRS refers to Technical Specifications and determines that LCO 3.1.3 (Control Rod Operability) Action C.1 applies which requires rod 26-19 to be fully inserted within 3 hours and Action C.2 which requires associated CRD (HCU) disarmed within four hours.

Event 3

(TRG-3) Standby Service Water Pump 1A (SW-P-1A) trips on motor winding overcurrent which requires Residual Heat Removal Pump 2A (RHR-P-2A) (currently in Suppression Pool Cooling) to be manually secured per ABN-SW. Standby Service Water System “A” (SW-SYS-A) not being available requires that the DG1 Diesel Engine Mode Selector be placed MAINT (Maintenance) effectively making DG1 inoperable.

The CRS declares SW-SYS-A and DG1 inoperable and refers to Technical Specifications and determines that the following applies:

- LCO 3.7.1 Action B.1 which requires restoring SW-SYS-A to operable status within 72 hours
- LCO 3.8.1 Action B.1 which requires performing SR 3.8.1.1 for operable offsite circuits (OSP-ELEC-W101 (Offsite Station Power Alignment Check)) within 1 hour and every 8 hours thereafter
- LCO 3.8.1 Action B.2 which requires declaring required feature(s) supported by DG 1, inoperable when the redundant required feature(s) are inoperable within 4 hours of DG1 going inoperable concurrent with the inoperability of the redundant required feature(s)
- LCO 3.8.1 Action B.3.1 which requires determining operable DGs are not inoperable due to common cause failure within 24 hours **- OR -** LCO 3.8.1 Action B.3.2 which requires performance of SR 3.8.1.2 for operable DGs within 24 hours (if not performed in the past 24 hours)
- LCO 3.8.1 Action B.4.1 which requires restoring DG1 to operable status within 72 hours of DG1 becoming inoperable AND within 6 days of failure to meet LCO (the 72 hours is more restrictive in this case) **- OR -** LCO 3.8.1 Action B.4.2.1 which requires establishing risk management actions for the alternate AC sources within 72 hours AND LCO 3.8.1 Action B.4.2.2 which requires DG1 to be restored to operable status within 14 days after being declared inoperable but in no case longer than 17 days from failure to meet LCO

Evaluator note: Although several Technical Specification actions are involved, the CRS will only have to refer to LCO 3.7.1 Condition B and LCO 3.8.1 Condition B to find them.

Event 4

(TRG-4) Vibrations start to rise above the ALERT setpoint on Reactor Feed Pump (RFP) "B" as indicated by annunciator P840.A1.7-5 (Turbine B Vibration Trouble) and validated on (local) vibration instrument RFW-VBI-1B/XS/T1BXY (Turbine Radial Inboard Bearing Vibration). Feed pump bias is adjusted to minimize load on RFP "B" in an attempt to reduce vibration (which is unsuccessful). Vibration level will exceed the DANGER setpoint requiring Reactor Recirculation flow to be incrementally reduced in 1% to 5% step changes while monitoring vibration level. Vibration level remains above the DANGER setpoint even after Reactor Recirculation (RRC) flow has been reduced to 74 Mlbm/hr. RFP "B" is manually tripped per ARP direction. The CRS may direct tripping of RFP "B" before the flow reduction is complete if equipment damage is a concern. Following the trip, the high vibration annunciator will clear if crew attempts a local reset (TRG-7). As RPV level lowers due to the feed pump trip, both Reactor Recirculation (RRC) Pumps will runback to 30 Hz causing reactor power to stabilize at a lower level of ~68% power.

Event 5

(TRG-5) An earthquake (OBE) causes annunciator 851.S-1 5-1 (Operating Basis Earthquake Exceeded) to alarm. ABN-EARTHQUAKE is entered. Concurrently, a steam leak in the RCIC Pump Room develops resulting in RCIC Equipment Area high temperature alarms. PPM 5.3.1 (Secondary Containment Control) and ABN-HELB (Line Break) are entered on Reactor Building (RB) area high temperature. Crew attempts to isolate steam leak as directed by PPM 5.3.1 (Secondary Containment Control). Control Room notifies plant personnel of safety hazard and directs evacuation of affected areas. Neither RCIC-V-63 (RCIC Steam Supply Inboard Isolation) nor RCIC-V-8 (RCIC Turbine Steam Supply Isolation) will automatically close. Manual attempts to shut RCIC-V-63 and RCIC-V-8 are unsuccessful.

CRS enters PPM 5.1.1 (RPV Control) and directs a manual reactor scram before reaching the max safe operating temperature for the RCIC Pump room **(CT #1)**. All control rods fully insert. The CRS WILL direct a reactor pressure reduction to 500 to 600 psig to reduce leak rate.

Event 6

Three (3) minutes after the scram, Main Steam Line "A" piping ruptures causing an unisolable steam leak. The CRS re-enters PPM 5.3.1 (Secondary Containment Control) based on a second unisolable steam leak in Secondary Containment resulting in high Main Steam Tunnel temperature.

Event 7

Following the Main Steam Line "A" rupture, the outboard MSIVs fail to AUTO close due to failure of a logic relay but can be manually closed. MSIV 22A (MS-V-22A) fails to AUTO close due to mechanical failure. Inability to manually close MS-V-22A results in an unisolable leak into secondary containment.

Event 8

The CRS directs entry into PPM 5.1.3 (Emergency RPV Depressurization) once Main Steam Tunnel Temperature exceeds its max safe operating value of 330°F based on two secondary containment areas greater than max safe operating value. With a primary system discharging into secondary containment and area temperature exceeding maximum safe operating level in more than one area, Emergency Depressurization (ED) is initiated by opening seven (7) Safety Relief Valves (ADS preferred) within 10 minutes of second MSOT being exceeded. **(CT #2)** RPV level will be restored using Condensate Booster Pumps following Emergency Depressurization.

TERMINATION CRITERIA: The scenario will be terminated when an Emergency Depressurization has been performed and RPV level is being controlled in the prescribed band OR as directed by the Examination Team.

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT #1 - With reactor at power and with primary system discharging into secondary containment, manually scram reactor before any area exceeds its maximum safe operating temperature.	If secondary containment temperature exceeds its maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EOP flowchart actions can no longer be assured. (Ref: PPM 5.0.10 Rev 21, section 8.9.3 k.1))	Procedural direction by PPM 5.3.1 (EOP for Secondary Containment Control) Step SC-14 directs entering PPM 5.1.1 (which requires placing Reactor Mode Switch in Shutdown) before any area exceeds its maximum safe operating temperature.	The operator will manually scram reactor by placing Reactor Mode Switch in Shutdown.	All control rods will fully insert.
CT #2 - With a primary system discharging into secondary containment and area temperature exceeding maximum safe operating level in more than one area, initiate Emergency Depressurization (ED) by opening seven (7) Safety Relief Valves (ADS preferred) within 10 minutes of second MSOT being exceeded. Note: If the crew properly elects to invoke the "EMERG DEPRESS is anticipated" override in ppm 5.1.1 (RPV Control) and in doing so, the second maximum safe operating level is not exceeded, this Critical Task is considered to be met.	The criteria of "2 or more areas" identifies the increase in parameter trend as a wide spread problem which may pose a direct and immediate threat to secondary containment integrity, equipment located in the secondary containment, continued safe operation of the plant, and personnel both on and off site. (Ref: PPM 5.0.10 Rev 21, section 8.9.3 k.3))	Procedural direction by PPM 5.3.1 (EOP for Secondary Containment Control) Step SC-15 directs Emergency Depressurizing reactor when a primary system (RCIC) is discharging into secondary containment and two or more area temperatures are exceeding their maximum safe operating level.	The operator will manually open 7 Safety Relief Valves (ADS preferred) to emergency depressurize the RPV.	The valve light indications for each of the 7 Safety Relief Valves will change from Green lit to Red lit when control switch is taken to Open. Reactor pressure will lower in response.

TURNOVER

Initial Conditions:

- Columbia is operating at 85% power due to economic dispatch
- Safety Relief Valve 2C (MS-RV-2C) is known to be leaking
- Suppression Pool high temperature alarms (601.A11.1-3 and 601.A12.1-3) have just annunciated
- Reactor Closed Cooling (RCC) Pump 1B is tagged out for planned maintenance
- RCC-P-1A and RCC-P-1C are protected

Shift Turnover:

- After shift turnover place RHR-P-2A in Suppression Pool Cooling (using maximum cooling) and allow SW-P-1A to auto start per SOP-RHR-SPC (section 5.1) – Steps 5.1.1 through 5.1.4 are complete.
- Associate Tech Specs and LCS action statements have been entered for RHR-SYS-A being inoperable but available
    - LCO 3.5.1 Action A.1 which requires restoring RHR-SYS-A to operable status within 7 days
    - LCO 3.6.1.5 Action A.1 which requires restoring RHR-SYS-A drywell spray subsystem to operable status within 7 days
    - LCO 3.6.2.3 Action A.1 which requires restoring RHR-SYS-A suppression pool cooling subsystem to operable status within 7 days
    - RFO 1.6.1.5 Action A.1 which requires restoring RHR-SYS-A suppression pool spray subsystem to operable status within 7 days
- The pre-evolution brief has been completed and operators are stationed near both pumps

INSTRUCTIONAL COVER SHEET

SC-4

PROGRAM TITLE	OPERATIONS TRAINING
COURSE TITLE	COLUMBIA GENERATING STATION SIMULATOR EXAMINATION
LESSON TITLE	Withdraw Control Rods during Startup; REA-FN-1B Trip requiring PPM 5.3.1 Entry and SGTS Start (TS); IRM "A" Upscale Failure with Half Scram; Loss of SL-11 (Re-energized from Alternate Source); RCIC-P-1 Coupling Found Broken; RHR-P-2B Suction Rupture (Lowering WW Level); SW-V-29 Fails to Auto Open; FDR-V-607 Fails to Close; Manual Scram on Low WW Level (Mode Switch Failure – Scram Pushbuttons Successful); ED performed on Low WW Level (One ADS Valve Fails to Open)

LENGTH OF LESSON 1 Hour

Lesson Plan PQD Code		Rev. No.	
Simulator Guide PQD Code	SC-4	Rev. No.	1
JPM PQD Code		Rev. No.	
Exam PQD Code		Rev. No.	

DIVISION TITLE	Nuclear Training		
DEPARTMENT	Operations Training		
PREPARED BY	Dave E. Crawford	DATE	01/20/17
REVISED BY	Dave E. Crawford	DATE	02/08/17
VALIDATED BY		DATE	

TECHNICAL REVIEW		DATE	
INSTRUCTIONAL REVIEW		DATE	
APPROVED		DATE	

Operations Training Manager

Facility:	Columbia Generating Station	Scenario No.:	4	Op Test No.:	1
Examiners:	____________________		Operators:	____________________	
	____________________			____________________	
	____________________			____________________	

Initial Conditions:	The reactor is in Mode 2 (Reactor startup). Reactor is critical at 5% power with RPV pressure at 500 psig. DEH is in Auto with Bypass Valves at 19.5% open. DEH pressure setpoint is 600 psig with pressurization rate set to 6 psig/minute but will remain in Hold until rods are withdrawn to establish Bypass Valves approximately 30% open. Reactor Building Exhaust Fan 1A (REA-FN-1A) is out of service for extended maintenance.
Turnover:	Withdraw control rods as required to establish and maintain Bypass Valves approximately 30% open in preparation for the SJAE second stage steam supply shift per ppm 3.1.2 step Q34. Next in-sequence rod is from Group 35, Step 08 (rod 06-39). Continue RPV pressure rise to 600 psig at 6 psig/minute when Bypass Valves are approximately 30% open.

Critical Tasks:

CT-1	Manually scram the reactor before wetwell level drops below 19 feet 2 inches (as read on CMS-LR-3 or 4 on H13-P601).
CT-2	When wetwell level cannot be maintained above 19 feet 2 inches (as read on CMS-LR-3 or 4 on H13-P601), initiate emergency depressurization by opening seven (7) Safety Relief Valves (ADS preferred) within 10 minutes of wetwell level lowering to 19 feet 2 inches. CT considered met if any combination of 7 Safety Relief Valves are opened.

<u>NOTE</u>: An unintentional or unnecessary RPS or ESF actuation may result in the creation of a post-scenario Critical Task, if that actuation results in a significant plant degradation or significantly alters a mitigation strategy.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R (ATC)	Withdraw control rods as required to establish and maintain the bypass valves approximately 30% open
2	TRG-2	C (BOP,SRO) TS (SRO)	Trip of REA-FN-1B results in a high reactor building pressure and entry into PPM 5.3.1 (EOP - Secondary Containment Control) (Tech Spec)
3	TRG-3	I (ATC,SRO)	IRM “A” fails upscale resulting in a half scram
4	TRG-4	C (BOP,SRO)	Differential current lockout of transformer (TR-1/11) results in a loss of SL-11 (due to the failure to automatically transfer to SL-21) which requires bus to be manually transferred to SL-21
5	N/A	C (ATC**,SRO) TS (SRO)	RCIC-P-1 coupling discovered broken (Tech Spec)
6	TRG-6	M (ALL)	Failure of the RHR-P-2A suction line results in lowering wetwell level (RHR-V-4A fails to close preventing isolation of leak)
			FDR-V-607 fails to auto close due to a failed level switch (which allows flooding to continue into RCIC Pump Room). Cannot be closed manually
7	N/A	C (BOP)	SW-V-29 fails to auto open when HPCS-P-2 is started for wetwell makeup

8	N/A	C (ATC)	Reactor mode switch fails to scram reactor, requiring use of manual scram pushbuttons to scram reactor prior to wetwell level lowering to 19 feet 2 inches **(CT #1)**
9	N/A	---	Prior to wetwell level going below 19 feet 2 inches, the crew determines that wetwell level cannot be maintained ≥ 19 feet 2 inches and initiates RPV Emergency Depressurization (ED) with 7 SRVs opened **(CT #2)**
		C (BOP)	One ADS SRV (MS-RV-4D) fails to open requiring manually opening one non-ADS SRV **(CT #2)**
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications			
** Normally assigned to BOP. NRC Evaluator will have to direct CRS to use ATC.			

Target Quantitative Attributes	Actual	Description
Malfunctions after EOP entry (1-2)	**3**	SW-V-29 fails to auto open; Mode switch failure; ADS SRV fails to open
Abnormal events (2-4)	**3**	Fan REA-FN-1B trip; IRM "A" trip with half scram; Loss of SL-11
Major transients (1-2)	**2**	Primary containment failure; Manual scram
EOPs entered/requiring substantive actions (1-2)	**3**	PPM 5.1.1 (RPV Control); PPM 5.2.1 (Primary Containment Control); PPM 5.3.1 (Secondary Containment Control);
EOP contingencies requiring substantive actions (0-2)	**1**	PPM 5.1.3 (Emergency RPV Depressurization)
EOP-based Critical Tasks (2-3)	**2**	See Critical Task Determination table

Trigger (TRG-x)	Evaluator Directed	How Triggered	Purpose	Malfunction Numbers
TRG-2	YES	Manually	Event Initiator	PMP-SCN010S
TRG-3	YES	Manually	Event Initiator	MAL-NIS002A
TRG-4	YES	Manually	Event Initiator	ANN-800C3A02; BKR-EPS001; BKR-EPS004
TRG-6	YES	Manually	Event Initiator	MAL-RHR001; XMT-PCN006A; XMT-PCN007A; XMT-PCN003A; XMT-PCN004A
TRG-7		Manually	Event Initiator	BKR-RHR001
			Initial Condition	BKR-SCN001
			Initial Condition	MOV-SSW009F
			Initial Condition	MOV-RHR029F
			Initial Condition	SRV-RRS016C
			Initial Condition	AOV-SCN014F
			Initial Condition	OVR-RPS001A

SCENARIO 4 SUMMARY

Event 1

With reactor power at ~5% and reactor pressure at ~500 psig during reactor startup, RO1 withdraws control rods per SOP-CR-MOVEMENT (Control Rod Movement) to establish and maintain Main Turbine Bypass Valves (BPVs) approximately 30% open as directed by PPM 3.1.2 (Startup Flowchart), Attachment 7.3, step Q34.

Event 2

(TRG-2) Trip of Reactor Building Exhaust Fan 1B (REA-FN-1B) results in a high Reactor Building pressure and entry into PPM 5.3.1 (EOP - Secondary Containment Control). Secondary containment becomes inoperable. ARP 4.812.R2 9-1 (REACTOR BUILDING EXHAUST FAN B TRIP) directs starting REA-FN-1A which cannot be started (out-of-service). Subsequent ARP direction requires CRO2 to isolate Reactor Building HVAC and starting the Standby Gas Treatment system to return Reactor Building pressure to within the TS limit (≥ 0.25 inch of vacuum water gauge). The CRS refers to Technical Specifications and determines that TS 3.6.4.1 (Secondary Containment), Action A.1 applies which requires restoring secondary containment to operable status within 4 hours.

Event 3

(TRG-3) IRM “A” fails upscale resulting in an IRM upscale trip and Neutron Monitor System trip annunciators and a half scram. Per the ARP and when directed by the CRS, CRO1 bypasses IRM “A” and resets the half-scram. The CRS refers to Technical Specifications 3.3.1.1 (RPS Instrumentation) and determines that the minimum number of IRM instruments required remains operable and that no TS actions are required.

Event 4

(TRG-4) Differential current lockout of transformer (TR-1/11) supplying 480V Bus SL-11 occurs which de-energizes the bus due to CB-21/11 failing to auto close. After accessing what caused the lockout, and when directed, CRO2 repowers SL-11 from SL-21 using the Quick Card (SOP-ELEC-480V-OPS-QC).

Event 5

Call comes into the Control Room reporting RCIC turbine coupling to the RCIC pump was found broken. CRS will direct the RCIC turbine to be tripped. The CRS refers to Technical Specifications and determines that TS 3.5.3 (RCIC System), Action A.1 applies which immediately requires verifying that HPCS is operable by administrative means AND Action A.2 which requires restoring RCIC system to operable status within 14 days.

Event 6

(TRG-6) A break on the Residual Heat Removal Pump 2A (RHR-P-2A) suction line causes wetwell level to lower. ABN-FLOODING is entered. When attempting to close the RHR-P-2A Motor-Operated suction valve (RHR-V-4A), the valve fails open. The CRS enters PPM 5.2.1 (EOP - Primary Containment Control) on Suppression Pool low level. Crew should direct removal of control power fuses (TRG-7) for RHR-P-2A as time permits.

FDR-V-607, the cross-connect valve between the RHR-SYS-A and Reactor Core Isolation Cooling (RCIC) rooms fails to auto close due to a failed level switch (which allows flooding to continue into RCIC Pump Room). The valve cannot be manually closed. The CRS re-enters PPM 5.3.1 (EOP - Secondary Containment Control) due high RHR-SYS-A and RCIC room levels. The leak from the Suppression Pool is not considered a “Primary System discharging into Secondary Containment” and therefore a controlled reactor shutdown is required for high RCIC room water level (6 inches above floor).

Event 7

The crew takes actions to restore wetwell level using the High Pressure Core Spray (HPCS) pump (HPCS-P-1) per PPM 5.5.23 (Emergency Suppression Pool Makeup). During this lineup, the HPCS Standby Service Water Pump (HPCS-P-2) discharge valve (SW-V-29) fails to auto open when HPCS-P-2 is started, requiring CRO2 to manually open the valve. HPCS is ineffective is restoring Suppression Pool level.

Event 8

The CRS enters PPM 5.1.1 (EOP - RPV Control) and directs manually scramming the reactor prior to wetwell level reaching 19 feet 2 inches. **(CT #1)** The reactor will not scram when the mode switch is taken to SHUTDOWN. CRO1 identifies the failure to scram and takes actions per PPM 3.3.1 (Reactor Scram) to scram the reactor. The Manual Scram Pushbuttons are effective in inserting all control rods.

Event 9

Prior to wetwell level going below 19 feet 2 inches, the CRS determines that wetwell level cannot be maintained ≥ 19 feet 2 inches and directs Emergency Depressurization (ED) per PPM 5.1.3 by opening seven (7) Safety Relief Valves (ADS preferred) within 10 minutes of wetwell level lowering to 19 feet 2 inches. **(CT #2)**

One Automatic Depressurization System (ADS) Safety Relief Valve (MS-RV-4D) fails to open during the ED requiring CRO2 to manually open a non-ADS SRV. **(CT #2)**

TERMINATION CRITERIA: The scenario will be terminated when emergency depressurization has commenced (7 SRVs open) and RPV level is being controlled in the prescribed band OR as directed by the Examination Team.

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT #1 – Manually scram the reactor before wetwell level drops below 19 feet 2 inches (as read on CMS-LR-3 or 4 on H13-P601).	Ensures reactor is scrammed and shutdown before requirement to Emergency Depressurize (ED) is reached. If ED is anticipated (see PPM 5.1.1 P-1 override), dumping steam to main condenser via Main Turbine bypass valves may be used to reduce reactor pressure before the requirement to ED occurs. ED would still be performed if required by EOPs. (Ref: PPM 5.0.10 Rev 21, 8.8.2 f))	Procedural direction by PPM 5.2.1 (EOP for Primary Containment Control) Step L-5 directs entering PPM 5.1.1 (which requires placing Reactor Mode Switch in Shutdown) once it is determined that wetwell level cannot be maintained above 19 feet 2 inches.	The operator will manually scram reactor by placing Reactor Mode Switch in Shutdown (and follow up with all Manual Scram pushbuttons when RMS fails to scram the reactor).	All control rods fully insert.
CT #2 - When wetwell level cannot be maintained above 19 feet 2 inches (as read on CMS-LR-3 or 4 on H13-P601), initiate emergency depressurization by opening seven (7) Safety Relief Valves (ADS preferred) within 10 minutes of wetwell level lowering to 19 feet 2 inches. .	Suppression of pressure from blowdown (Emergency Depressurization) through the downcomers cannot be assured for water levels below 19 feet 2 inches. (Ref: PPM 5.0.10 Rev 21, 7.12.3)	Procedural direction by PPM 5.2.1 (EOP for Primary Containment Control) Step L-6 directs Emergency Depressurizing reactor when Wetwell water level cannot be maintained above 19 feet 2 inches.	The operator will manually open 7 Safety Relief Valves (ADS preferred) to emergency depressurize the RPV.	The valve light indications for each of the 7 Safety Relief Valves will change from Green lit to Red lit when control switch is taken to Open. Reactor pressure will lower in response.

TURNOVER

Initial Conditions:

- The reactor is in Mode 2 (Reactor startup).
- Reactor is critical at 5% power with RPV pressure at 500 psig.
- DEH is in Auto with Bypass Valves at 19.5% open.
- DEH pressure setpoint is 600 psig with pressurization rate set to 6 psig/minute but will remain in Hold until rods are withdrawn to establish Bypass Valves approximately 30% open.
- Reactor Building Exhaust Fan 1A (REA-FN-1A) is out of service for extended maintenance.

Shift Turnover:

- Withdraw control rods as required to establish and maintain Bypass Valves approximately 30% open in preparation for the SJAE second stage steam supply shift per ppm 3.1.2 (Startup Flowchart) step Q34.
- Next in-sequence rod is from Group 35, Step 08 (rod 06-39).
- Continue RPV pressure rise to 600 psig at 6 psig/minute when Bypass Valves are approximately 30% open.