Facility: River Bend Station Date of Exam:

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1	3	3	4	N/A			4	3	N/A		3	20			7
	2	1	1	1				2	1			1	7			3
	Tier Totals	4	4	5				6	4			4	27			10
2. Plant Systems	1	2	2	3	3	2	3	2	3	2	2	2	26			5
	2	1	1	1	1	1	1	1	1	1	1	2	12			3
	Tier Totals	3	3	4	4	4	4	3	3	3	3	4	38			8

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

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

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

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

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

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

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

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

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

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

G* Generic K/As

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

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

ES-401 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 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4			X				Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.5 / 45.6) AK3.01 Reactor water level response	3.4	39
295003 (APE 3) Partial or Complete Loss of AC Power / 6	X						Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.8 to 41.10) AK1.03 Under voltage/degraded voltage effects on electrical loads	2.9	40
295004 (APE 4) Partial or Total Loss of DC Power / 6						X	2.2.22 Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)	4.0	41
295005 (APE 5) Main Turbine Generator Trip / 3			X				Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.5 / 45.6) AK3.04 Main generator trip	3.2	42
295006 (APE 6) Scram / 1				X			Ability to operate and/or monitor the following as they apply to SCRAM: (CFR: 41.7 / 45.6) AA1.07 Control rod position	4.1	43
295016 (APE 16) Control Room Abandonment / 7					X		Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.10 / 43.5 / 45.13) AA2.05 Drywell pressure	3.8	44
295018 (APE 18) Partial or Complete Loss of CCW / 8	X						Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.8 to 41.10) AK1.01 Effects on component/system operations	3.5	45
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8		X					Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: (CFR: 41.7 / 45.8) AK2.07 Condensate system	3.2	46
295021 (APE 21) Loss of Shutdown Cooling / 4		X					Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.7 / 45.6) AA1.01 Reactor water cleanup system	3.4	47
295023 (APE 23) Refueling Accidents / 8				X			Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6) AA1.06 Neutron monitoring	3.3	48

295024 High Drywell Pressure / 5	X						Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.8 to 41.10) EK1.02 Containment building integrity	3.9	49
295025 (EPE 2) High Reactor Pressure / 3						X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.)	4.4	50
295026 (EPE 3) Suppression Pool High Water Temperature / 5					X		Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) EA2.02 Suppression pool level	3.9	51
295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5					X		Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): (CFR: 41.10 / 43.5 / 45.13) EA2.01 Containment temperature: Mark-III	3.7	52
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5									
295030 (EPE 7) Low Suppression Pool Water Level / 5			X				Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) EK3.03 RCIC operation	3.6	53
295031 (EPE 8) Reactor Low Water Level / 2		X					Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: (CFR: 41.7 / 45.8) EK2.06 High pressure (feedwater) coolant injection (FWCI/HPCI)	4.1	54
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1						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)	3.8	55
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9				X			Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6) EA1.01 Stack-gas monitoring	3.9	56
600000 (APE 24) Plant Fire On Site / 8				X			Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: AA1.08 Fire fighting equipment used on each class of fire	2.6	57
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6			X				Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8) AK3.02 Actions contained in abnormal operating procedure for voltage and grid disturbances	3.6	58
K/A Category Totals:	3	3	4	4	3	3	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	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295002 (APE 2) Loss of Main Condenser Vacuum / 3					X		Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) AK2.07 Offgas system	3.1	59
295007 (APE 7) High Reactor Pressure / 3									
295008 (APE 8) High Reactor Water Level / 2									
295009 (APE 9) Low Reactor Water Level / 2									
295010 (APE 10) High Drywell Pressure / 5	X						Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.8 to 41.10) AK1.03 Temperature increases	3.2	60
295011 (APE 11) High Containment Temperature (Mark III Containment only) / 5									
295012 (APE 12) High Drywell Temperature / 5				X			Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.7 / 45.6) AA1.02 Drywell cooling system	3.8	61
295013 (APE 13) High Suppression Pool Temperature. / 5				X			Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.7 / 45.6) AA1.01 Suppression pool cooling	3.9	62
295014 (APE 14) Inadvertent Reactivity Addition / 1									
295015 (APE 15) Incomplete Scram / 1									
295017 (APE 17) Abnormal Offsite Release Rate / 9									
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7									
295022 (APE 22) Loss of Control Rod Drive Pumps / 1									
295029 (EPE 6) High Suppression Pool Water Level / 5			X				Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) EK3.01 Emergency depressurization	3.5	63
295032 (EPE 9) High Secondary Containment Area Temperature / 5									
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9						X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)	4.4	64
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									

295035 (EPE 12) Secondary Containment High Differential Pressure / 5									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5		X					Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following: (CFR: 41.7 / 45.8) EK2.01 Secondary containment equipment and floor drain system	3.1	65
500000 (EPE 16) High Containment Hydrogen Concentration / 5									
K/A Category Point Totals:	1	1	1	2	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 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 Pumps	3.5	1
205000 (SF4 SCS) Shutdown Cooling						X						Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.7 / 45.7) K6.04 Abnormal Reactor water level	3.6	2
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection														
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray	X											Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.05 Automatic depressurization system	3.7	3
209002 (SF2, SF4 HPCS) High-Pressure Core Spray						X						Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): (CFR: 41.7 / 45.7) K6.02 Condensate storage tank water level	3.4	4
211000 (SF1 SLCS) Standby Liquid Control										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.02 SBLC control switch	4.2	5
212000 (SF7 RPS) Reactor Protection				X								Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.12 Bypassing of selected SCRAM signals (manually and automatically)	3.9	6
215003 (SF7 IRM) Intermediate-Range Monitor							X					Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: (CFR: 41.5 / 45.5) A1.02 Reactor power indication response to rod position changes	3.7	7
215004 (SF7 SRMS) Source-Range Monitor		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 SRM channels/detectors	2.6	8
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor											X	2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)	4.6	9

217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling						X						Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): (CFR: 41.7 / 45.7) K6.03 Suppression pool water supply	3.5	10
218000 (SF3 ADS) Automatic Depressurization									X			Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: (CFR: 41.7 / 45.7) A3.09 Reactor vessel water level	4.1	11
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff								X				Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.05 Nuclear boiler instrumentation failures	3.3	12
239002 (SF3 SRV) Safety Relief Valves					X							Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES: (CFR: 41.5 / 45.3) K5.04 Tail pipe temperature monitoring	3.3	13
259002 (SF2 RWLCS) Reactor Water Level Control					X							Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: (CFR: 41.5 / 45.3) K5.03 Water level measurement	3.1	14
261000 (SF9 SGTS) Standby Gas Treatment			X									Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: (CFR: 41.7 /45.6) K3.05 Secondary containment radiation/ contamination levels	3.2	15
262001 (SF6 AC) AC Electrical Distribution							X					Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: (CFR: 41.5 / 45.5) A1.04 Load currents	2.7	16
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.01 Transfer from alternative source to preferred source	2.8	17
263000 (SF6 DC) DC Electrical Distribution			X									Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: (CFR: 41.7 / 45.4) K3.02 Components using D.C. control power (i.e. breakers)	3.5	18

264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG								X				Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.01 Parallel operation of emergency generator	3.5	19
300000 (SF8 IA) Instrument Air	X											Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.03 Containment air	2.8	20
400000 (SF8 CCS) Component Cooling Water			X									Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: (CFR: 41.7 / 45.6) K3.01 Loads cooled by CCWS	2.9	21
510000 (SF4 SWS*) Service Water (Normal and Emergency)														
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode								X				Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.16 Loss of coolant accident	4.4	22
205000 (SF4 SCS) Shutdown Cooling									X			Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) including: (CFR: 41.7 / 45.7) A3.02 Pump trips	3.2	23
211000 (SF1 SLCS) Standby Liquid Control											X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)	3.2	24
215004 (SF7 SRMS) Source-Range Monitor				X								Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.01 Rod withdrawal blocks	3.7	25
263000 (SF6 DC) DC Electrical Distribution				X								Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.02 Breaker interlocks, permissives, bypasses and cross ties	3.1	26
K/A Category Point Totals:	2	2	3	3	2	3	2	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	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 (SF1 CRDH) CRD Hydraulic														
201002 (SF1 RMCS) Reactor Manual Control														
201003 (SF1 CRDM) Control Rod and Drive Mechanism											X	2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)	3.9	27
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer														
202001 (SF1, SF4 RS) Recirculation														
202002 (SF1 RSCTL) Recirculation Flow Control														
204000 (SF2 RWCU) Reactor Water Cleanup														
214000 (SF7 RPIS) Rod Position Information														
215001 (SF7 TIP) Traversing In-Core Probe														
215002 (SF7 RBMS) Rod Block Monitor														
216000 (SF7 NBI) Nuclear Boiler Instrumentation				X								Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.07 Recirculation pump protection	2.9	28
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode								X				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: (CFR: 41.5 / 45.6) A2.01 Inadequate net positive suction head	3.0	29
223001 (SF5 PCS) Primary Containment and Auxiliaries														
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup														
234000 (SF8 FH) Fuel-Handling Equipment			X									Knowledge of the effect that a loss or malfunction of the FUEL HANDLING EQUIPMENT will have on following: K3.03 Fuel handling operations	3.1	30
239001 (SF3, SF4 MRSS) Main and Reheat Steam		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 Main steam isolation valve solenoids	3.2	31
239003 (SF9 MSVLCS) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating														
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary						X						Knowledge of the effect that a loss or malfunction of the following will have on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: (CFR: 41.7 / 45.7) K6.08 Main steam	3.0	32

256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater														
268000 (SF9 RW) Radwaste														
271000 (SF9 OG) Offgas														
272000 (SF7, SF9 RMS) Radiation Monitoring									X			Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including: (CFR: 41.7 / 45.7) A3.03 Liquid radwaste isolation indications	3.1	33
286000 (SF8 FPS) Fire Protection											X	2.1.28 Knowledge of the purpose and function of major system components and controls(CFR 41.7 / 45.7)	4.1	34
288000 (SF9 PVS) Plant Ventilation								X				Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.02 Low reactor water level	3.4	35
290001 (SF5 SC) Secondary Containment										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.05 Fuel building differential pressure	3.3	36
290003 (SF9 CRV) Control Room Ventilation	X											Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROOM HVAC and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.04 Nuclear steam supply shut off system (NSSSS/PCIS)	3.2	37
290002 (SF4 RVI) Reactor Vessel Internals					X							Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS: (CFR: 41.5 / 45.3) K5.07 Safety limits	3.9	38
51001 (SF8 CWS*) Circulating Water														
K/A Category Point Totals:	1	1	1	1	1	1	1	1	1	1	2	Group Point Total:		12

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.18	Ability to make accurate, clear, and concise logs, records, status boards, and reports. (CFR: 41.10 / 45.12 / 45.13)	3.6	66		
	2.1.30	Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	4.4	67		
	Subtotal			2		
2. Equipment Control	2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc. (CFR: 41.10 / 43.5 / 45.13)	2.6	68		
	2.2.38	Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)	3.6	69		
	2.2.40	Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)	3.9	70		
	Subtotal			3		
3. Radiation Control	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9)	2.9	71		
	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10)	3.5	72		
	Subtotal			2		
4. Emergency Procedures/Plan	2.4.20	Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)	3.8	73		
	2.4.27	Knowledge of “fire in the plant” procedures. (CFR: 41.10 / 43.5 / 45.13)	3.4	74		
	2.4.39	Knowledge of RO responsibilities in emergency plan implementation. (CFR: 41.10 / 45.11)	3.9	75		
	Subtotal			3		
Tier 3 Point Total				10		

<table>
<tr><td colspan="19">Facility: River Bend Station Date of Exam:</td></tr>
<tr><td rowspan="2">Tier</td><td rowspan="2">Group</td><td colspan="12">RO K/A Category Points</td><td colspan="5">SRO-Only Points</td></tr>
<tr><td>K1</td><td>K2</td><td>K3</td><td>K4</td><td>K5</td><td>K6</td><td>A1</td><td>A2</td><td>A3</td><td>A4</td><td>G*</td><td>Total</td><td colspan="2">A2</td><td colspan="2">G*</td><td>Total</td></tr>
<tr><td rowspan="3">1. Emergency and Abnormal Plant Evolutions</td><td>1</td><td></td><td></td><td></td><td colspan="3" rowspan="3">N/A</td><td></td><td></td><td colspan="2" rowspan="3">N/A</td><td></td><td>20</td><td colspan="2">4</td><td colspan="2">3</td><td>7</td></tr>
<tr><td>2</td><td></td><td></td><td></td><td></td><td></td><td></td><td>7</td><td colspan="2">2</td><td colspan="2">1</td><td>3</td></tr>
<tr><td>Tier Totals</td><td></td><td></td><td></td><td></td><td></td><td></td><td>27</td><td colspan="2">6</td><td colspan="2">4</td><td>10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td>26</td><td colspan="2">2</td><td colspan="2">3</td><td>5</td></tr>
<tr><td>2</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td>12</td><td></td><td>2</td><td colspan="2">1</td><td>3</td></tr>
<tr><td>Tier Totals</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td>38</td><td colspan="2">4</td><td colspan="2">4</td><td>8</td></tr>
<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2">10</td><td>1</td><td>2</td><td>3</td><td>4</td><td rowspan="2">7</td></tr>
<tr><td colspan="2"></td><td colspan="2"></td><td colspan="2"></td><td colspan="2"></td><td>1</td><td>2</td><td>2</td><td>2</td></tr>
</table>

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

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

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

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

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

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

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

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

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

G* Generic K/As

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

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

ES-401 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 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4									
295003 (APE 3) Partial or Complete Loss of AC Power / 6									
295004 (APE 4) Partial or Total Loss of DC Power / 6									
295005 (APE 5) Main Turbine Generator Trip / 3									
295006 (APE 6) Scram / 1					X		Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) AA2.05 Whether a reactor SCRAM has occurred	4.6	84
295016 (APE 16) Control Room Abandonment / 7									
295018 (APE 18) Partial or Complete Loss of CCW / 8									
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8					X		Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.10 / 43.5 / 45.13) AA2.01 Instrument air system pressure	3.6	85
295021 (APE 21) Loss of Shutdown Cooling / 4						X	2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	4.0	86
295023 (APE 23) Refueling Accidents / 8									
295024 High Drywell Pressure / 5									
295025 (EPE 2) High Reactor Pressure / 3									
295026 (EPE 3) Suppression Pool High Water Temperature / 5									
295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5						X	2.4.18 Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)	4.0	87
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5									
295030 (EPE 7) Low Suppression Pool Water Level / 5					X		Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) EA2.01 Suppression pool level	4.2	88
295031 (EPE 8) Reactor Low Water Level / 2									
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1						X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)	4.0	89
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9									

600000 (APE 24) Plant Fire On Site / 8					X		Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: AA2.13 Need for emergency plant shutdown	3.0	90
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6									
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	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295002 (APE 2) Loss of Main Condenser Vacuum / 3									
295007 (APE 7) High Reactor Pressure / 3					X		Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Reactor power	4.1	91
295008 (APE 8) High Reactor Water Level / 2									
295009 (APE 9) Low Reactor Water Level / 2									
295010 (APE 10) High Drywell Pressure / 5									
295011 (APE 11) High Containment Temperature (Mark III Containment only) / 5									
295012 (APE 12) High Drywell Temperature / 5									
295013 (APE 13) High Suppression Pool Temperature. / 5									
295014 (APE 14) Inadvertent Reactivity Addition / 1									
295015 (APE 15) Incomplete Scram / 1									
295017 (APE 17) Abnormal Offsite Release Rate / 9									
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7						X	2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.3	92
295022 (APE 22) Loss of Control Rod Drive Pumps / 1									
295029 (EPE 6) High Suppression Pool Water Level / 5									
295032 (EPE 9) High Secondary Containment Area Temperature / 5					X		Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) EA2.03 Cause of high area temperature	4.0	93
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure / 5									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5									
500000 (EPE 16) High Containment Hydrogen Concentration / 5									
K/A Category Point Totals:					2	1	Group Point Total:		3

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 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode														
205000 (SF4 SCS) Shutdown Cooling														
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection														
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray														
209002 (SF2, SF4 HPCS) High-Pressure Core Spray														
211000 (SF1 SLCS) Standby Liquid Control														
212000 (SF7 RPS) Reactor Protection														
215003 (SF7 IRM) Intermediate-Range Monitor														
215004 (SF7 SRMS) Source-Range Monitor														
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor											X	2.2.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)	4.5	76
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling								X				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: (CFR: 41.5 / 45.6) A2.03 Valve closures	3.3	77
218000 (SF3 ADS) Automatic Depressurization											X	2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)	4.1	78
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff														
239002 (SF3 SRV) Safety Relief Valves											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	79
259002 (SF2 RWLCS) Reactor Water Level Control														
261000 (SF9 SGTS) Standby Gas Treatment														
262001 (SF6 AC) AC Electrical Distribution														
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)														
263000 (SF6 DC) DC Electrical Distribution														

264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG								X				Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.03 Operating unloaded, lightly loaded, and highly loaded	3.4	80
300000 (SF8 IA) Instrument Air														
400000 (SF8 CCS) Component Cooling Water														
510000 (SF4 SWS*) Service Water (Normal and Emergency)														
K/A Category Point Totals:								2			3	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 (SF1 CRDH) CRD Hydraulic											X	2.1.28 Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)	4.1	81
201002 (SF1 RMCS) Reactor Manual Control														
201003 (SF1 CRDM) Control Rod and Drive Mechanism														
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer														
202001 (SF1, SF4 RS) Recirculation														
202002 (SF1 RSCTL) Recirculation Flow Control														
204000 (SF2 RWCU) Reactor Water Cleanup														
214000 (SF7 RPIS) Rod Position Information														
215001 (SF7 TIP) Traversing In-Core Probe														
215002 (SF7 RBMS) Rod Block Monitor														
216000 (SF7 NBI) Nuclear Boiler Instrumentation														
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
223001 (SF5 PCS) Primary Containment and Auxiliaries								X				Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.06 High containment pressure	4.1	82
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup														
234000 (SF8 FH) Fuel-Handling Equipment														
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSVLCS) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating														
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary								X				Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.02 Loss of lube oil	3.5	83
256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater														
268000 (SF9 RW) Radwaste														
271000 (SF9 OG) Offgas														
272000 (SF7, SF9 RMS) Radiation Monitoring														

286000 (SF8 FPS) Fire Protection														
288000 (SF9 PVS) Plant Ventilation														
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation														
290002 (SF4 RVI) Reactor Vessel Internals														
51001 (SF8 CWS*) Circulating Water														
K/A Category Point Totals:								2			1	Group Point Total:		3

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
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
	Subtotal					1
2. Equipment Control	2.2.11	Knowledge of the process for controlling temporary design changes. (CFR: 41.10 / 43.3 / 45.13)			3.3	95
	2.2.21	Knowledge of pre- and post-maintenance operability requirements. (CFR: 41.10 / 43.2)			4.1	96
	Subtotal					2
3. Radiation Control	2.3.6	Ability to approve release permits. (CFR: 41.13 / 43.4 / 45.10)			3.8	97
	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)			3.7	98
	Subtotal					2
4. Emergency Procedures/Plan	2.4.13	Knowledge of crew roles and responsibilities during EOP usage. (CFR: 41.10 / 45.12)			4.6	99
	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	100
	Subtotal					2
Tier 3 Point Total						7

Tier / Group (**Original**)	Randomly Selected K/A (**New**)	Reason for Rejection
RO T2G1 205000 K6.05 Component cooling water systems	205000 K6.04 Abnormal Reactor Water Level	Original K/A: 205000 Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.7 / 45.7) K6.05 Component cooling water systems Reason for Rejection: Already have multiple questions about component cooling water on exam testing this K/A. Randomly Selected K/A: 205000 K6.04 Abnormal Reactor Water Level Page 1 point totals not affected by this change.
RO T2G1 259002 K5.07 Turbine speed control mechanisms: TDRFP	259002 K5.03 Water level measurement	Original K/A: 259002 Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: (CFR: 41.5 / 45.3) K5.07 Turbine speed control mechanisms: TDRFP Reason for Rejection: Original K/A is not applicable to RBS. Randomly selected K/A: 259002 K5.03 Water Level Measurement Page 1 point totals not affected by this change.
RO T2G1 264000 A2.02 Unloading prior to securing emergency generator	264000 A2.01 Parallel operation of emergency generator	Original K/A: 264000 Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.02 Unloading prior to securing emergency generator Reason for Rejection: inability to develop a good A2 level question to match the K/A Randomly Selected K/A: 201005 A2.01 Parallel operation of emergency generator Page 1 point totals not affected by this change.

RO T2G1 300000 K1.04 Cooling water to compressor	300000 K1.03 Containment Air	Original K/A: 300000 Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.04 Cooling water to compressor Reason for Rejection: Original K/A is not applicable to RBS. Randomly Selected K/A: 300000 K1.03 Containment air. Page 1 point totals not affected by this change.
RO T2G2 290001 A4.06 Fuel building area temperature	A4.05 Fuel building differential pressure	Original K/A: 290001 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.06 Fuel building area temperature Reason for Rejection: No procedural guidance to implement for fuel building high temperature Randomly Selected K/A: 290001 A4.05 Fuel building differential pressure Page 1 point totals not affected by this change.
RO T1G1 295025 2.4.41 Knowledge of the emergency action level thresholds and classifications	295025 2.1.7 Ability to evaluate plant performance and make operational judgments | based on operating characteristics, reactor behavior, and instrument | interpretation.	Original K/A: 295025 2.4.41Knowledge of the emergency action level thresholds and classifications. Reason for Rejection: Knowledge of emergency action levels is a SRO task. Randomly Selected K/A: 295025 2.1.7 Ability to evaluate plant performance and make operational judgments | based on operating characteristics, reactor behavior, and instrument | interpretation. Page 1 point totals not affected by this change.
RO T1G1 295038 EA1.05 Post accident sample system (PASS)	295038 EA1.01 Stack-gas monitoring system	Original K/A: 295038 Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6) EA1.05 Post accident sample system (PASS) Reason for Rejection: The PASS system is operated by the chemistry department. Randomly Selected K/A: 295038 Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6) EA1.01 Stack-gas monitoring system Page 1 point totals not affected by this change.

RO T2G2 286000 2.4.30 Knowledge of events related to system operation/status that must be | reported to internal organizations or external agencies, such as the State, | the NRC, or the transmission system operator.	286000 2.1.28 Knowledge of the purpose and function of major system components and controls(CFR 41.7 / 45.7)	Original K/A: 286000 2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. Reason for Rejection: Difficulty in writing an RO level question to this K/A. Randomly Selected K/A: 286000 2.1.28 Knowledge of the purpose and function of major system components and controls(CFR 41.7 / 45.7) Page 1 point totals not affected by this change. Page 1 point totals not affected by this change.
SRO T2G1 217000 A2.09 Loss of vacuum pump	217000 A2.03 Valve closures	Original K/A: 217000 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: (CFR: 41.5 / 45.6) A2.09 Loss of vacuum pump Reason for Rejection: Original K/A is not applicable to RBS. Randomly Selected K/A: 217000 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: (CFR: 41.5 / 45.6) A2.03 Valve closures Page 1 point totals not affected by this change.
SRO T2G1 215005 2.2.19 Knowledge of maintenance work order requirements(CFR: 41.10 / 43.5 / 45.13)	215005 2.2.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)	Original K/A: 2150052.2.19 Knowledge of maintenance work order requirements. (CFR: 41.10 / 43.5 / 45.13) Reason for Rejection: Difficulty developing a SRO level question to meet the intent of this K/A. Randomly Selected K/A: 215005 2.2.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13) Page 1 point totals not affected by this change.

SRO T2G2 223001 A2.05 High containment/drywell oxygen concentration	223001 A2.06 High containment pressure	Original K/A: 223001 Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions A2.05 High containment/drywell oxygen concentration Reason for Rejection: Original K/A is not applicable to RBS. Randomly Selected K/A: 223001 Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions A2.06 High containment pressure Page 1 point totals not affected by this change.
SRO T1G1 600000 AA2.07 Whether malfunction is due to common-mode electrical failures	600000 AA2.13 Need for emergency plant shutdown	Original K/A: 600000 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: AA2.07 Whether malfunction is due to common-mode electrical failures Reason for Rejection: Unable to write an SRO level question to meet the K/A. Lead examiner suggested a new K/A. Randomly Selected K/A: 60000 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: AA2.13 Need for emergency plant shutdown Page 1 point totals not affected by this change.

Facility: River Bend	Date of Examination: 11/18/19
Examination Level: RO ☒ SRO ☐	Operating Test Number: 2019-11

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M R	Perform Jet Pump Operability RB-2019-11-A1 (Modified from RJPM-NRC-M14-A1) KA: 2.1.7
Conduct of Operations	N R	P808 DAILY OPERATING LOGS RB-20190-11-A2 KA: 2.1.18
Equipment Control	N R	Determination of Alternate Decay Heat Removal Method RB-20190-11-A3 KA: 2.2.44
Radiation Control	N R	Determine Radiation Release RB-20190-11-A4 KA: 2.3.15
Emergency Plan		

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

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

RB-2019-11-A1, Perform Jet Pump Operability
Given photos of plant status, the applicant will complete STP-053-3001, Jet Pump Operability Test. The applicant will perform all of Attachment 1 except Jet Pumps #6-20.

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Importance Rating 4.4

RB-20190-11-A2, P808 DAILY OPERATING LOGS
Applicant determines the required log entries on the P808 and calculates required values based on log readings.

2.1.18 Ability to make accurate, clear, and concise logs, records, status boards,| and reports.
Importance Rating 3.6

RB-20190-11-A3, Determination of Alternate Decay Heat Removal Method
Applicant determines the Alternate Shutdown Cooling Methods available to restore and maintain reactor coolant temperature less than 160F.

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.
Importance Rating 4.2

RB-20190-11-A4, Determine Radiation Release
Given a photo from DRMS computer screens, the applicant determines main steam tunnel is the source of the leak. Based on plant knowledge, the applicant will determine that the result of the location of the leak is an unfiltered monitored release.

2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
Importance Rating 2.9

Facility: River Bend	Date of Examination: 11/18/19
Examination Level: RO ☐ SRO ☒	Operating Test Number: 2019-11

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D R	Review GOP-0004 Calculations RB-2019-11-A5 (RJPM-NRC-M14-A1) KA: 2.1.7
Conduct of Operations	N R	DETERMINE THE TIME TO 200°F AND IF FORCED CIRCULATION IS REQUIRED RB-2019-11-A6 KA: 2.1.25
Equipment Control	N S	Risk Evaluation RB-2019-11-A7 KA: 2.2.18
Radiation Control	D R	Determine Required Actions upon Radioactive Effluent Monitor Failure RB-2019-11-A7 (12/18/14 NRC A8) KA: 2.3.11
Emergency Plan	N R	Determine the PAR Scenario Number for PAR Recommendation RB-2019-11-A9 KA: 2.4.44

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

* Type Codes and Criteria:

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

RB-2019-11-A5, Review GOP-0004 Calculations
The applicant will complete the review of the GOP to determine errors. Upon identifying errors, applicant must determine Tech Spec required actions.

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Importance Rating 4.7

RB-2019-11-A6, DETERMINE THE TIME TO 200°F AND IF FORCED CIRCULATION IS REQUIRED
The applicant will calculate the time to 200°F as a result of cooling capability. The applicant will also determine if plant conditions support maintenance activities to continue.

2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.
Importance Rating 4.2

RB-2019-11-A7, Risk Evaluation
The applicant will evaluate plant conditions (severe weather and INOP equipment) to determine risk number and color.

2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.
Importance Rating 3.9

RB-2019-11-A8, Determine Required Actions upon Radioactive Effluent Monitor Failure
The applicant will determine operability of RMS-RE107 and what actions if any are required to continue with the liquid radiological release.

2.3.11 Ability to control radiation releases.
Importance Rating 4.3

RB-2019-11-A9, Determine the PAR Scenario Number for PAR Recommendation
The applicant will declare a PAR based on a rapidly progressing severe accident given plant conditions.

2.4.44 Knowledge of emergency plan protective action recommendations.
Importance Rating 4.4

Facility: River Bend	Date of Examination: 11/18/19
Exam Level: **RO** ☐ SRO-I ☐ ***SRO-U*** ☐	Operating Test Number: 2019-11

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

System/JPM Title	Type Code*	Safety Function
a. Reset A Recirc FCV runback RB-2019-11-S1	D-S	1
b. RWCU Reject to main condenser with controller failure RB-2019-11-S2	A-D-L-S	2
c. Roll turbine with high vibrations RB-2019-11-S3	A-D-S	3
d. RCIC Initiation with controller failure. RB-2019-11-S4	A-EN-L-N-S	4
e. Place low volume purge in service RB-2019-11-S5	N-S	5
f. Parallel and load HPCS EDG. RB-2019-11-S6	A- EN-M-S	6
g. Reset Reactor Scram RB-2019-11-S7	D-L-S	7
h. Place Fuel Building Ventilation in refuel mode RB-2019-11-S8	D-L-S	9

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

System/JPM Title	Type Code*	Safety Function
i. LINE UP FIRE WATER PROTECTION SYSTEM FOR RPV INJECTION RB-2019-11-P1	D-E-L-R	2
j. RESTORE RPS B NORMAL POWER SUPPLY RB-2019-11-P2	A-D	7
k. EMERGENCY START OF THE DIESEL AIR COMPRESSOR RB-2019-11-P3	A-N	8

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

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

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

	RO		SRO-I		*SRO-U*	
	ALL (h RO only)		a-g & i-k		*d, e, i, j, & k*	
(A)lternate path	4–6	6	4–6	6	2–3	3
(C)ontrol room		0		0		0
(D)irect from bank	≤ 9	7	≤ 8	6	≤ 4	2
(E)mergency or abnormal in-plant	≥ 1	1	≥ 1	1	≥ 1	1
(EN)gineered safety feature	≥ 1 (control room system)	2	≥ 1 (control room system)	2	≥ 1 (control room system)	1
(L)ow-Power/Shutdown	≥ 1	4	≥ 1	3	≥ 1	2
(N)ew or (M)odified from bank including 1(A)	≥ 2	4	≥ 2	4	≥ 1	3
(P)revious 2 exams	≤ 3 (randomly selected)	0	≤ 3 (randomly selected)	0	≤ 2 (randomly selected)	0
(R)CA	≥ 1	1	≥ 1	1	≥ 1	1
(S)imulator		8		7		2

Schedule in parallel
S1/S5
S2/S8(RO Only)
S3/S6
S4/S7

Revision 1: Added JPM designation for each candidate.
Revision 2: Changed e to Place low volume purge in service per SOP-59.
Revision 3: Changed k to air compressor due to more clearly directed alternate path.

2019 NRC Scenario 1

Facility: River Bend Nuclear Station Scenario No.: 1 Op-Test No.: 2019-11

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions: 100% Power. 36 hours ago RCIC was tagged out for governor repair and repair parts will be on site in approximately 4 hours.
LCO 3.5.3 RCIC System entered. Condition A. RCIC System inoperable. A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE within 1 hour (DONE). A.2 Restore RCIC System to OPERABLE status in 14 days.
Turnover: AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event No.	Malf. No.	Event Type †	Event Description
1	NMS011E	C (ATC/CRS) A (CREW)	APRM F failed upscale ARP, APRM B OR F UPSCALE TRIP OR INOP, H13-P680/06A/A03
2	DI_EGS_SDEBS (B Start SW) P877_31A:H_3	C (BOP/CRS) TS (CRS) A (CREW)	Division 2 Diesel Generator auto start and oil leak ARP, H13-P877/32A/H03, ENS-SWG1B SUSTAINED OR DEGRADED UV LCO 3.8.1 Condition C LCO 3.3.8.1 Condition A
3	CCP001B CCP004c	C (BOP/CRS) A (CREW)	CCP P1B pump trip, CCP P1A fails to auto start AOP-11, Loss of Reactor Plant Component Cooling Water
4	NMS011A	TS (CRS)	APRM B failed upscale LCO 3.3.1.1 Condition A
5	WCS007 (100)	C (ATC/CRS) A (CREW)	RWCU Valve Nest Room steam leak with A Pump fails to trip. RWCU HIGH DIFF FLOW, H13-P680/01A/B06, AOP-3 Automatic Isolations
6	ED001 RCS007 RCS001A	M (CREW)	Station Blackout/LOCA EOP-1, RPV Control AOP-50, Station Blackout
7	ED003J	C (ATC/CRS)	Division 3 Bus Fault
8	DI_ENS-ACB07L	C (BOP/CRS)	Division 1 Diesel Generator output breaker fails to auto close

† (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec

* **Critical Task** (As defined in NUREG 1021 Appendix D)

Quantitative Attributes Table			
Attribute	**E3-301-4 Target**	**Actual**	**Description**
Malfunctions after EOP entry	1-2	2	• Division 3 Bus Fault • Division 1 Diesel Generator output breaker fails to auto close
Abnormal Events	2-4	4	• APRM F failed upscale • Division 2 Diesel Generator auto start and oil leak • CCP pump trip • RWCU Valve Nest Room leak
Major Transients	1-2	1	• Station Blackout/LOCA
EOP entries requiring substantive action	1-2	1	• EOP-1, RPV Control
Entry into a contingency EOP with substantive actions	≥1	1	• Emergency Depressurization
Preidentified critical tasks	≥2	2	• Manually close ENS-ACB07, STBY D/G A OUTPUT BRKR, prior to exiting EOP-1 and entering the SAPs for reactor water level. • Emergency Depressurize with at least one injection source lined up for injection when level cannot be restored and maintained above -187”, within 15 minutes. (15 minutes start when RPV level reaches -187 inches).

SCENARIO ACTIVITIES:

Initial Conditions:
100% Power. 36 hours ago RCIC was tagged out for governor repair.
LCO 3.5.3 RCIC System entered. Condition A. RCIC System inoperable. A.2 Restore RCIC System to OPERABLE status in 14 days.

Turnover:
AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event 1 – (Triggered by Lead Examiner)
APRM F fails upscale per ARP, APRM B OR F UPSCALE TRIP OR INOP, H13-P680/06A/A03. ATC bypasses APRM F and resets the half scram.

Event 2 – (Triggered by Lead Examiner)
After APRM F is bypassed and half scram is reset, the Division 2 Diesel Generator will automatically start due to a relay failure (2018 Robinson OE) and an oil leak will require crew to secure the diesel generator (2018 Fermi OE). Division 2 Diesel Generator will not be able to restart after it is secured. Division 2 Diesel Generator is not available for remainder of scenario. CRS entered LCO 3.8.1 AC Sources – Operating Condition C One required DG inoperable and 3.3.8.1 Condition A 3.3.8.1 Loss of Power (LOP) Instrumentation Condition A.

Event 3 – (Triggered by Lead Examiner)
CCP P1B pump trips and CCP P1A fails to auto start. BOP manually starts CCP P1A and restores normal CCP flow and pressure. CRS enters AOP-11, Loss of Reactor Plant Component Cooling Water, and directs manual start of CCP P1C.

Event 4 – (Triggered by Lead Examiner)
APRM B fails upscale per ARP, APRM B OR F UPSCALE TRIP OR INOP, H13-P680/06A/A03. CRS enters LCO 3.3.1.1 RPS Instrumentation Condition A One or more required channels inoperable.

Event 5 – (Triggered by Lead Examiner)
RWCU Valve Nest Room steam leak per RWCU HIGH DIFF FLOW, H13-P680/01A/B06. All isolation valves automatically close. A RWCU pump fails to trip. ATC manually secures the A RWCU pump.

NOTE: The 2018 NRC Scenario 3 had a RWCU leak as major (unisolable). This malfunction has the leak auto isolate, but the pump fails to stop.

Event 6 – (Triggered by Lead Examiner)
Station Blackout occurs with a LOCA. The crew does not have any high pressure feed available. When the Division 1 Diesel Generator output breaker is closed, the division 1 ECCS systems will be available to restore reactor water level once the crew emergency depressurizes.

<u>Event 7 – </u>(Initial Setup - Automatic)
Division 3 bus fault prevents Division 3 equipment from being energized. HPCS is inoperable.

NOTE: This is similar to a malfunction used in 2016 NRC Scenario 4.

<u>Event 8 – </u>(Initial Setup - Automatic)
Division 1 Diesel Generator output breaker fails to auto close. The BOP operator manually closes the diesel output breaker, allowing the diesel to automatically energize the Division 1 bus.

	CT-1	CT-2
Critical Task	Emergency Depressurize with at least one injection source lined up for injection when level cannot be restored and maintained above -187”, within 15 minutes. (15 minutes start when RPV level reaches -187 inches).	Manually close ENS-ACB07, STBY D/G A OUTPUT BRKR, prior to exiting EOP-1 and entering the SAPs for reactor water level.
EVENT	6	8
Safety Significance	Per EOP-1, RPV Control, ALC-12 bases, if an injection source is available, emergency depressurization should be delayed at least until RPV level reaches the top of the active fuel, but may be performed anytime RPV level is between the top of the active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). If it is believed that available injection sources are capable of restoring and maintaining RPV water level above the Minimum Steam Cooling RPV Water Level following RPV depressurization, the blowdown may be performed as soon as RPV level reaches the top of the active fuel. (For example, all low pressure ECCS are running but cannot inject until RPV pressure decreases below their shutoff heads.)	Per EOP-1, RPV Control, ALC-13 and ALC-14, when adequate core cooling cannot be restored and maintained after maximizing injection with all available sources, exit all EOPs and enter all SAPs. When ENS-ACB07 is manually closed, the diesel will energize the division 1 bus. This will energize the division 1 ECCS systems. LPCS and RHR A will be available for injection to restore reactor water level.
Cueing	Reactor water level lowering on wide range and fuel zone indications.	Diesel voltage and frequency indications are normal with the output breaker open and ENS-ACB06 closed. All division 1 equipment not energized.

* If an operator or the crew significantly deviates from, or fails to, follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review (NUREG 1021, Appendix D). An unintentional or unnecessary RPS or ESF actuation may result in the creation of a post-scenario Critical Task, if that actuation results in significant plant degradation or significantly alters a mitigation strategy

2019 NRC Scenario 2

Facility: River Bend Nuclear Station Scenario No.: 2 Op-Test No.: 2019-11

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions: 100% Power. 36 hours ago RCIC was tagged out for governor repair.
LCO 3.5.3 RCIC System entered. Condition A. RCIC System inoperable. A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE within 1 hour (DONE). A.2 Restore RCIC System to OPERABLE status in 14 days.

Turnover: AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event No.	Malf. No.	Event Type †	Event Description
1	DI-HVR-UC1A LO_HVR-UC1A-A	C (BOP/CRS) A (CREW) TS (CRS)	HVR UC1A trip ARP CONTAINMENT UNIT COOLERS OUTLET FLOW LOW, H13-P863/71A/F03 LCO 3.6.1.7 Condition A.
2	RPS003A	C (CREW) A (CREW) TS (CRS)	Loss of RPS A AOP-10, Loss of one RPS Bus LCO 3.3.8.2 Condition A
3	DI_CNM-HA114-CAM DI_CNM-HA114-COS	C (ATC/CRS) A (CREW)	CNM-F114, CNDS MIN RECIRC fails open RX FW PUMPS LOW SUCTION PRESS, H13-P680/03A/B03 AOP-6, Condensate/Feedwater Failures
4	TMS002	C (BOP/CRS) A (CREW)	MSTR SEP DRAIN TANK HIGH WATER LEVEL, H13-P870/53A/H03 DSM-LV78A, SHELL RCVR TK A HIGH DUMP manually operation
5	DI_DSM-LV78A after 240 to NTEST	M (CREW)	DSM-LV78A, SHELL RCVR TK A HIGH DUMP fails to manually open, Scram, Turbine Trip EOP-1, RPV Control
6	DI_C51-RSMS to RUN DI_C11_S1AP (N_INIT) DI_C11_S1BP (N_INIT)	M (CREW)	Mode switch/ARI not working, Scram with RPS pushbuttons within 15 minutes AOP-1, Reactor Scram
7	ED017A ED017B	C (ATC/CRS)	NPS B fast transfer fails and A Main Feed Pump trips, manually restore feed (B or C) and condensate (maintain A & C Cond Pumps running)
8	E22MOVF004P to 0	C (BOP/CRS)	HPCS injection valve stuck closed

† (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec
* **Critical Task** (As defined in NUREG 1021 Appendix D)

Quantitative Attributes Table			
Attribute	**E3-301-4 Target**	**Actual**	**Description**
Malfunctions after EOP entry	1-2	2	• Fast transfer fails • HPCS injection valve stuck closed
Abnormal Events	2-4	4	• HVR UC1A trip • Loss of RPS A • CNM-F114, CNDS MIN RECIRC fails open • MSTR SEP DRAIN TANK HIGH WATER LEVEL
Major Transients	1-2	2	• DSM-LV78A, SHELL RCVR TK A HIGH DUMP fails to manually open • Mode switch/ARI not working
EOP entries requiring substantive action	1-2	1	• EOP-1, RPV Control
Entry into a contingency EOP with substantive actions	≥1	1	• Alternate Level Control
Preidentified critical tasks	≥2	2	• Reactor must be scrammed using RPS manual pushbuttons prior to tripping both recirc pumps per EOP-1A, RPV CONTROL ATWS. • Manually restore feed prior to RPV water level reaching the top of active fuel.

SCENARIO ACTIVITIES:

Initial Conditions:
100% Power. 36 hours ago RCIC was tagged out for governor repair.
LCO 3.5.3 RCIC System entered. Condition A. RCIC System inoperable. A.2 Restore RCIC System to OPERABLE status in 14 days.

Turnover:
AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event 1 – (Triggered by Lead Examiner)
Containment Unit Cooler A trips requiring start of the non-safety related unit cooler C. Entry is made into Tech Spec 3.6.1.7 Condition A.

Event 2 – (Triggered by Lead Examiner)
Loss of RPS A occurs. Building Operator reports EPA breakers 1st one is tripped on undervoltage and the second one is still closed. CRS declares the second EPA breaker INOP. LCO 3.3.8.2 RPS Electric Power Monitoring Condition A. One or both in-service power supplies with one electric power monitoring assembly inoperable.

NOTE: The crew will exit the LCO when RPS is transferred to the alternate power supply. A follow-up question may be required to ask what tech specs applied prior to transferring RPS to the alternate power supply.

NOTE: A loss of RPS B was used in 2016 Scenario 3. There are different priorities and restoration with RPS A. The most limiting restoration and difference is the loss of instrument air with the loss of RPS A.

Event 3 – (Triggered by Lead Examiner)
CNM-F114, CNDS MIN RECIRC fails open and crew manually closes it per RX FW PUMPS LOW SUCTION PRESS, H13-P680/03A/B03. AOP-6, Condensate/Feedwater Failures, is also entered and directs manual control.

Event 4 – (Triggered by Lead Examiner)
MSTR SEP DRAIN TANK HIGH WATER LEVEL, H13-P870/53A/H03, alarms. BOP manually operates DSM-LV78A, SHELL RCVR TK A HIGH DUMP valve to maintain drain tank water level in band.

NOTE: If the crew fails to recognize and correct the issue in 60 seconds the turbine will trip.

Event 5 – (Triggered by Lead Examiner)
After several manipulations, DSM-LV78A, SHELL RCVR TK A HIGH DUMP fails to manually re-open. The ATC manually scrams the reactor and trips the turbine.

Event 6 – (Initial Setup - Automatic)

The mode switch and ARI don't work. The ATC must manually scram with RPS pushbuttons per AOP-1, Reactor Scram.

NOTE: 2018 NRC Scenario 4 used ARI pushbuttons, which is similar to RPS pushbuttons.

<u>Event 7 – </u>(Initial Setup - Automatic)
After the turbine trip, the fast transfer fails. Feed is lost and ATC must manually restore feed and condensate.

<u>Event 8 – </u>(Initial Setup - Automatic)
HPCS injection valve stuck closed. This removes all high pressure feed. Alternate level control will be entered until feed and condensate is restored.

The reactor mode switch will be stuck in the RUN position. At 849 psig reactor pressure, the MSIV will close resulting in a transition to SRVs for pressure control.

	CT-1	CT-2
Critical Task	Reactor must be scrammed using RPS manual pushbuttons prior to tripping both recirc pumps per EOP-1A, RPV CONTROL ATWS.	Manually restore feed prior to RPV water level reaching the top of active fuel.
EVENT	6	7
Safety Significance	Per AOP-1, Reactor Scram, Immediate Operator Actions, 4.3 If all control rods are not fully inserted, then perform one or both of the following steps to insert control rods: 4.3.1 Arm and depress C71A-S3A, B, C, and D, MANUAL SCRAM Pushbuttons. Per EOP-1, Step RC-1 bases when an automatic scram signal is received or a manual scram inserted, operators perform a series of immediate scram actions in accordance with the scram procedure (AOP-0001) that include verification of the scram signal and placement of the reactor mode switch in SHUTDOWN. These immediate actions normally occur before the operator responsible for controlling the plant with the EOP flowcharts enters (or reenters) EOP-1. Once the reactor is successfully shut down (which may be the condition which exists when the RPV Control flowchart is entered), there are no further reactor power control actions required by the EOPs.	When RPV level cannot be restored and maintained above the top of active fuel, the Level flowpath continues at transition point 2, Alternate Level Control. In this flowpath, direction is given to restore core submergence by more extreme measures that include emergency depressurization of the RPV.
Cueing	No rod motion observed. All RPS lights on P680 are still energized.	Reactor Water Level lowering on wide range level indication.

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

Facility: River Bend Nuclear Station Scenario No.: 3 Op-Test No.: 2019-11

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions: 100% Power. 36 hours ago RCIC was tagged out for governor repair.
LCO 3.5.3 RCIC System entered. Condition A. RCIC System inoperable. A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE within 1 hour (DONE). A.2 Restore RCIC System to OPERABLE status in 14 days.
Turnover: AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event No.	Malf. No.	Event Type †	Event Description
1	HDL001A	C (ATC/CRS) A (CREW)	Heater drain pump A trip per H13-P680 / 02A / A08, HTR DR PUMP BREAKERS AUTO TRIP Per OSP-53, Lower power to 93%. AOP-7, Loss of Feedwater Heating AOP-24, Thermal Hydraulics Stability Controls
2	P808_81a:e_4	C (BOP/CRS) A (CREW) TS (CRS)	Failure of PVLC Compressor A to auto start per H13-P808/81A/E04, DIV I PVLCS ISOL ACCUMULATOR 6A LOW PRESSURE. LCO 3.6.1.9 Condition A
3	MSC011	TS (CRS)	171’ airlock inner door seal failure LCO 3.6.1.2 Condition A
4	GMC002B GMC001A TGS009	C (BOP/CRS) A (CREW)	Trip of Stator Water Cooling Pump B, Standby pump fails to auto start STATOR COOLING WATER PUMPS AUTO TRIP, H13-P870/54A/D01
5	FWS017A	I (ATC/CRS) A (CREW)	Steam Flow Transmitter, C33-N003A fail downscale AOP-6, CONDENSATE/FEEDWATER FAILURES
6	FWS009, 100%	M (CREW)	B feedwater line break in the Drywell, SCRAM, Isolate Feedwater line with rupture AOP-1, REACTOR SCRAM AOP-6, CONDENSATE/FEEDWATER FAILURES EOP-1, RPV CONTROL
7	LPCS005 RHR009A	C (BOP/CRS)	LPCS and RHR A pumps fail to auto start on high drywell pressure signal.
8	CNM001	C (ATC/CRS)	Complete loss of vacuum
9	HPCS001	C (BOP/CRS)	HPCS pump trips immediately upon start.

† (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec
* **Critical Task** (As defined in NUREG 1021 Appendix D)

Quantitative Attributes Table			
Attribute	**E3-301-4 Target**	**Actual**	**Description**
Malfunctions after EOP entry	1-2	3	• LPCS and RHR A Pumps fail to auto start on high drywell pressure signal. • Complete loss of vacuum • HPCS pump trips immediately upon start.
Abnormal Events	2-4	4	• HDL Pump A Trip • Trip of PVLC Compressor • Trip of Stator Water Cooling Pump B, Standby pump fails to auto start • Steam Flow Transmitter, C33-N003A fail downscale
Major Transients	1-2	1	• B feedwater line break in the Drywell, SCRAM, Isolate Feedwater line with rupture
EOP entries requiring substantive action	1-2	1	• EOP-1, RPV CONTROL
Entry into a contingency EOP with substantive actions	≥1	1	• Alternate Level Control per EOP-1, RPV Control
Preidentified critical tasks	≥2	2	• Close FWS-MOV7B, REACTOR FEEDWATER INLET, and/or B21-MOV-F065B, FDW TO REAC OUTBD ISOL VALVE, within 15 minutes of alarm REACTOR COOLANT SYS HIGH LEAKAGE RATE, H13-P601/19A/A05. • Restore reactor water level via the A feedwater line prior to reaching -160 inches wide range.

SCENARIO ACTIVITIES:

Initial Conditions:
100% Power. 36 hours ago RCIC was tagged out for governor repair.
LCO 3.5.3 RCIC System entered. Condition A. RCIC System inoperable. A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE within 1 hour (DONE). A.2 Restore RCIC System to OPERABLE status in 14 days.

Turnover:
AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event 1 – (Triggered by Lead Examiner)
Heater drain pump A trips per H13-P680 / 02A / A08, HTR DR PUMP BREAKERS AUTO TRIP. ATC lowers power to 93%, per OSP-53, EMERGENCY AND TRANSIENT RESPONSE SUPPORT PROCEDURE. AOP-7, Loss of Feedwater Heating, and AOP-24, Thermal Hydraulics Stability Controls, entered.

Event 2 – (Triggered by Lead Examiner)
Failure of PVLC Compressor A to auto start per H13-P808/81A/E04, DIV I PVLCS ISOL ACCUMULATOR 6A LOW PRESSURE. The CRS recognizes and enters LCO 3.6.1.9 Condition A.

Event 3 – (Triggered by Lead Examiner)

The BOP recognizes and reports H13-P863/71A/E05, AUX BLDG/CTMT AL INBOARD DOOR SEAL FAILURE and H13-P863/71A/H05, AUX BLDG/CTMT AL INBD DR AIR SPLY PRESSURE LOW, 171' airlock inner door seal failure. The CRS recognizes and enters LCO 3.6.1.2 Condition A.

Event 4 – (Triggered by Lead Examiner)
The B stator cooling water pump trips next with failure of the A standby pump to auto start. The resulting turbine-generator run back is stopped when the operators manually start the A pump.

Event 5 – (Triggered by Lead Examiner)
Steam Flow Transmitter, C33-N003A, fails downscale causing reactor water to slowly lower. Per AOP-6, CONDENSATE/FEEDWATER FAILURES, ATC takes manual control of the master level controller and restores reactor water level to approximately 36 inches narrow range.

NOTE: If no action is taken by the operators, reactor water level stabilizes at approximately 18 inches Narrow Range and 1.5 inches Wide Range.

Event 6 – (Initial Setup - Automatic)
The B feedwater line breaks in the Drywell. The crew SCRAMs and isolates Feedwater line B. The CRS enters AOP-1, REACTOR SCRAM; AOP-6, CONDENSATE / FEEDWATER FAILURES; and EOP-1, RPV CONTROL.

<u>Event 7 –</u> (Initial Setup - Automatic)
LPCS and RHR A Pumps fail to auto start on high drywell pressure signal due to feedwater leak. The pumps must be manually started per OSP-53, EMERGENCY AND TRANSIENT RESPONSE SUPPORT PROCEDURE.

<u>Event 8 –</u> (Initial Setup - Automatic)
The crew recognizes and reports complete loss of vacuum. MSIVs close resulting in manual SRV operation to maintain pressure control. This causes a loss of inventory to assist diagnosing the feedwater rupture.

<u>Event 9 –</u> (Initial Setup - Automatic)
The crew recognizes and reports HPCS pump trip immediately upon manual start. This causes a loss of all high pressure injection sources and forces the crew to isolate and restore reactor water level using feedwater system.

	CT-1	CT-2
Critical Task	Close FWS-MOV7B, REACTOR FEEDWATER INLET, and/or B21-MOV-F065B, FDW TO REAC OUTBD ISOL VALVE, within 15 minutes of alarm REACTOR COOLANT SYS HIGH LEAKAGE RATE, H13-P601/19A/A05.	Restore reactor water level via the A feedwater line prior to reaching -160 inches wide range.
EVENT	6	6/7
Safety Significance	Per ARP REACTOR COOLANT SYS HIGH LEAKAGE RATE, H13-P601/19A/A05 the possible cause is excessive inleakage to the Drywell Floor or Pedestal Drain Sump due to steam or coolant leak in the Drywell. The high leakage contributes to rising drywell pressure. The high drywell pressure condition is indicative of a line break occurring in the drywell and thus relates to RPV water level control. The feedwater line break directly affects reactor water level. The ruptured feedwater line must be isolated to minimize the energy put in the drywell as well as allowing reactor water level to be restored using feedwater.	Per EPSTG*002, EOP-1 Transition Point 2 for alternatel level control states, the Alternate Level Control flowpath begins at transition point 2 and provides instructions for controlling RPV level that are more detailed and explicit than those in Step RL-3. Its primary objective is to restore and maintain RPV level above the top of the active fuel. This goal is achieved through use of available injection sources and, if necessary, emergency depressurization of the RPV. If core cooling by submergence cannot be reestablished, spray cooling or steam cooling is employed. The limit of -160 inches reactor water level is to prevent reaching the top of active fuel at -162 inches. Also, wide range indication stops at -160 inches.
Cueing	A feed flow mismatch will cue the applicant of B feed line rupture. Drywell pressure rise will correspond to feed flow.	Reactor water level will slowly lower until feed flow is restored or HPCS is manually started.

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

2019 NRC Scenario 4

Facility: River Bend Nuclear Station   Scenario No.: 4   Op-Test No.: 2019-11

Examiners: ______________________   Operators: ______________________

______________________   ______________________

______________________   ______________________

Initial Conditions: Mode 2, 4% Power. Continue with startup per GOP-1, Plant Startup.

Turnover: AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event No.	Malf. No.	Event Type †	Event Description
1		R (ATC/CRS)	Raise power with rods per the reactivity maneuvering plan to continue startup per GOP-1, Reactor Startup.
2	CRDM1621	C (ATC/CRS) A (CREW) TS (CRS)	Rod not coupled, H13-P680/07A/C02, CONTROL ROD OVERTRAVEL LCO 3.1.3 Condition C
3	P601_22A:C_1	C (BOP/CRS) A (CREW)	CRD Pump A Seal Leak per H13-P601/22A/C01, CRD PMP A HIGH SEAL LEAKAGE.
4	Remote DI_ENS_ACB06 (TRIP) DG002A	C (BOP/CRS) A (CREW) TS (CRS)	ENS-A feeder breaker trip, Div 1 DG fail to auto start H13-P877/31A/H03, ENS-SWG1A SUSTAINED OR DEGRADED UV LCO 3.3.8.1, Condition A LCO 3.5.1, Condition C LCO 3.6.2.3, Condition A LCO 3.8.1, Condition A, C, E LCO 3.8.9, Condition A
5	CNM004A	C (CREW) A (CREW)	Condensate Pump C Trip per AOP 6 (Condensate / Feedwater Failures), ARP H13-P680 / 02A / D03, CONDENSATE PUMP P1C OVERLOAD and H13-P680 / 02A / A03, CONDENSATE PUMP AUTO TRIP.
6	MSC018	M (CREW)	Suppression pool leak EOP-1, RPV Control EOP-2, Primary Containment Control EOP-3, Secondary Containment and Ratioactivity Release Control
7		C (ATC/CRS)	Failure of suppression pool makeup (HPCS test return fails to open & RCIC E51-F045 valve fails to open).
8	DI_E12-AS21DA (ARM) DI_E12-AS21MNI (M_INIT) P601_17a:a_3 (Fail_off)	C (BOP/CRS)	Division 2 RHR Initiation

† (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec
* **Critical Task** (As defined in NUREG 1021 Appendix D)

Quantitative Attributes Table			
Attribute	**E3-301-4 Target**	**Actual**	**Description**
Malfunctions after EOP entry	1-2	2	• Failure of suppression pool makeup • Division 2 RHR Initiation
Abnormal Events	2-4	4	• Rod not coupled • CRD pump seal leak • ENS-A feeder breaker trip • Condensate Pump C Trip
Major Transients	1-2	1	• Suppression pool leak
EOP entries requiring substantive action	1-2	2	• EOP-1, RPV Control • EOP-2, Primary Containment Control
Entry into a contingency EOP with substantive actions	≥1	1	• Anticipate/Emergency Depressurization
Preidentified critical tasks	≥2	2	• Manually scram prior to suppression pool level reaching 15 feet 5 inches. • Anticipate emergency depressurization per RP-1 or open at least 5 SRVs prior to suppression pool level reaching 13 feet 5 inches (top of the first row of horizontal vents).

SCENARIO ACTIVITIES:

Initial Conditions
4% Power. Mode 2. Startup in progress.
Inoperable Equipment: None

Turnover:
Continue startup. Continue rod withdrawal to enter Mode 1.
AOP-29, Severe Weather entered for Severe Thunderstorm Warning.

Event 1 – (Reactivity Manipulation)
The ATC raises power with rods per the reactivity maneuvering plan to continue startup per GOP-1, Plant Startup. The ATC withdraws rod 16-37 from position 12 to 48.

Event 2 – (Initial Setup - Automatic)
On the second rod manipulation, the ATC discovers and reports control rod 16-21 not coupled, H13-P680/07A/C02, CONTROL ROD OVERTRAVEL. ATC recouples the control rod per ARP Operator Actions. CRS enters LCO 3.1.3 Control Rod OPERABILITY Condition C One or more control rods inoperable for reasons other than Condition A or B.

Event 3 – (Triggered by Lead Examiner)
The CRD Pump A will develop a seal leak per H13-P601/22A/C01, CRD PMP A HIGH SEAL LEAKAGE. The BOP will swap CRD pumps.

Event 4 – (Triggered by Lead Examiner)
After CRD pump swap, ENS-A feeder breaker trips. The Division 1 Diesel Generator fails to auto start and the BOP manually starts the diesel generator per H13-P877/32A/H03, ENS-SWG1B SUSTAINED OR DEGRADED UV. The CRS will recognize and enter LCO 3.3.8.1 Condition A; LCO 3.5.1 Condition C; LCO 3.6.2.3 Condition A; LCO 3.8.1 Condition A, C, & E; and LCO 3.8.9 Condition A.

Event 5 – (Triggered by Lead Examiner)
The ATC will recognize the Condensate Pump C trip per AOP 6 (Condensate / Feedwater Failures), ARP H13-P680 / 02A / D03, CONDENSATE PUMP P1C OVERLOAD and H13-P680 / 02A / A03, CONDENSATE PUMP AUTO TRIP.

The ATC and BOP will start A Condensate pump per SOP-7 Condensate System, Section 5.1.

<u>Event 6 – </u>(Triggered by Lead Examiner)
The crew will recognize and report a Suppression pool leak into crescent area through RHR A. Per EOP-2, the crew will manually scram prior to 15 feet 5 inches and emergency depressurize prior to 13 feet 5 inches. EOP-1, RPV Control
EOP-2, Primary Containment Control. EOP-3, Secondary Containment and Radioactivity Release Control. If the crew attempts to start HPCS to makeup to the suppression pool from the CST, the test return valve is stuck closed.

<u>Event 7 – </u>(Initial Setup - Automatic)
The crew will recognize and report the failure of suppression pool makeup (HPCS test return fails to open & RCIC E51-F045 valve fails to open). The crew will not be able to isolate the leak or provide makeup water to the suppression pool. As a result of the lowering suppression pool water level, the crew will be required to emergency depressurize.

<u>Event 8 – </u>(Initial Setup - Automatic)
During Emergency Depressurization (RPV pressure <100 psig or at the direction of the lead evaluator), Division 2 RHR Initiation will start and inject RHR B and C. Both will not be needed for adequate core cooling, so the BOP will be required to terminate and prevent injection from RHR B and C as needed to prevent a high reactor water level trip of the feed pumps.

NOTE: Scenario 3 has division 1 ECCS initiation before the major. At rated power the low pressure systems never inject. BOP is required to override LPCS and RHR A pumps off. The CRS is required to enter Tech Spec for inoperable systems.

.

	CT-1	CT-2
Critical Task	Manually scram prior to suppression pool level reaching 15 feet 5 inches.	Anticipate emergency depressurization per RP-1 or open at least 5 SRVs prior to suppression pool level reaching 13 feet 5 inches (top of the first row of horizontal vents).
EVENT	6	6
Safety Significance	When suppression pool level decreases to two feet above the top of the Mark III horizontal vents, any further drop in water level could result in direct exposure of the drywell atmosphere to the containment airspace thus compromising the pressure suppression function of the containment. Suppression pool level must therefore be maintained above this elevation. The systems listed in Step SPL-10 are those that can be used to add water to the suppression pool. Entering EOP-1 at Step RC-1 “before” suppression pool water level drops to the limiting elevation ensures that, if possible, the reactor is scrammed before RPV depressurization is initiated.	The RPV is not permitted to remain at pressure if suppression of steam discharged from the RPV into the drywell cannot be assured. When the horizontal vents are not adequately submerged, any steam discharged from the RPV into the drywell may not condense in the suppression pool before containment pressure reaches unacceptable levels. Emergency depressurization of the RPV is required if this low water level condition cannot be avoided.
Cueing	Suppression pool lowering.	Suppression pool lowering.

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